August 4, 1993

Docket Nos. 50-387
and 50-388

Mr. Robert G. Byram
Senior Vice President-Nuclear
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Dear Mr. Byram:

SUBJECT: REMOVAL OF CYCLE-SPECIFIC PARAMETER LIMITS FROM THE TECHNICAL SPECIFICATIONS, SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (MPA D021 AND D027) (PLA-3892) (TAC NOS. M85486 AND M85487)

The Commission has issued the enclosed Amendment No. 126 to Facility Operating License No. NPF-14 and Amendment No. 95 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Units 1 and 2. These amendments are in response to your letter dated December 18, 1992, as supplemented by your telecopy dated January 28, 1993 and your letters dated March 25, and May 20, 1993.

These amendments change the Technical Specifications (TSs) to remove cycle-specific parameter limits in accordance with NRC Generic Letter 88-16 and modify Section 5.3.1 in accordance with NRC Generic Letter 90-02, Supplement 1.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Biweekly Federal Register Notice.

Sincerely, Original signed by
Richard J. Clark
Richard J. Clark, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 126 to License No. NPF-14
2. Amendment No. 95 to License No. NPF-22
3. Safety Evaluation

cc w/enclosures:
See next page

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CP-1

OFC	:PDI-2/LA	:PDI-2/PM	:C/SRXB	:OGC	:PDI-2/D
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Senior Vice President-Nuclear
Pennsylvania Power and Light Company
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1. Amendment No. 126 to
   License No. NPF-14
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   License No. NPF-22
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See next page

Mr. Robert G. Byram
Pennsylvania Power & Light Company

Susquehanna Steam Electric Station,
Units 1 & 2

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

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-387

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 126
License No. NPF-14

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

   A. The application for the amendment filed by the Pennsylvania Power & Light Company, dated December 18, 1992, as supplemented by telecopy dated January 28, 1993, and by letters dated March 25, and May 20, 1993, 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-14 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. 126 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 license amendment is effective as of its date of issuance and is to be implemented within 30 days after its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael L. Boyle, Acting Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 4, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 126

FACILITY OPERATING LICENSE NO. NPF-14

DOCKET NO. 50-387

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

REMOVE	INSERT
i	i
ii	ii*
iii	iii*
iv	iv
v	v*
vi	vi
xix	xix*
xx	xx
xxi	xxi
xxii	xxii*
1-1	1-1*
1-2	1-2
1-3	1-3
1-4	1-4*
3/4 2-1	3/4 2-1
3/4 2-2	-
3/4 2-3	-
3/4 2-4	-
3/4 2-4a	-
-	-
3/4 2-5	3/4 2-2
3/4 2-5a	3/4 2-3
3/4 2-6	-
3/4 2-7	3/4 2-4

3/4 2-8	-
-	-
3/4 2-9	-
3/4 2-9a	-
3/4 2-9b	-
3/4 2-9c	-
3/4 2-10	-
-	-
3/4 2-10a	3/4 2-5
3/4 2-10b	-
3/4 4-1b	3/4 4-1b
3/4 4-1c	3/4 4-1c
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 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 4-1	B 3/4 4-1
B 3/4 4-1a	B 3/4 4-1a*
5-5	5-5*
5-6	5-6
-	6-20a
-	6-20b

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1.0 DEFINITIONS

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

ACTION

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

AVERAGE 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.

CHANNEL CALIBRATION

1.4 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.

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

CORE ALTERATION

1.7 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.

CORE OPERATING LIMITS REPORT

1.7A The CORE OPERATING LIMITS REPORT is the Susquehanna SES Unit 1 specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.3. Plant operation within these operating limits is addressed in individual specifications.

CRITICAL POWER RATIO

1.8 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 1-135 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 functions, 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 suppression 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 is measured.

DEFINITIONS

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 applicable LHGR for APRM Setpoint limit specified in the CORE OPERATING LIMITS REPORT 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.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for all fuel shall not exceed the limit specified in the CORE OPERATING LIMITS REPORT.

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

ACTION:

With an APLHGR exceeding the limit, 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 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 with a LIMITING CONTROL ROD PATTERN for APLHGR.

d. The provisions of Specification 4.0.4 are not applicable.

POWER DISTRIBUTION LIMITS

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-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 #
$S \leq (0.58W + 59\%)\ T$ $S_{RB} \leq (0.58W + 50\%)\ T$	$S \leq (0.58W + 62\%)\ T$ $S_{RB} \leq (0.58W + 53\%)\ T$

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. The FLPD for SNP fuel is the actual LHGR divided by the LINEAR HEAT GENERATION RATE for APRM Setpoints limit specified in the CORE OPERATING LIMITS REPORT.

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 determined above, 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.

---

* 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.

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:

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 rector is operating with MFLPD greater than or equal to FRTP.

d. The provisions of Specification 4.0.4 are not applicable.

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 applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT.

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

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.

POWER DISTRIBUTION LIMITS

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed the applicable LHGR limit specified in the CORE OPERATING LIMITS REPORT.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.4 LHGRs shall be determined to be equal to or less than the limit:

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.

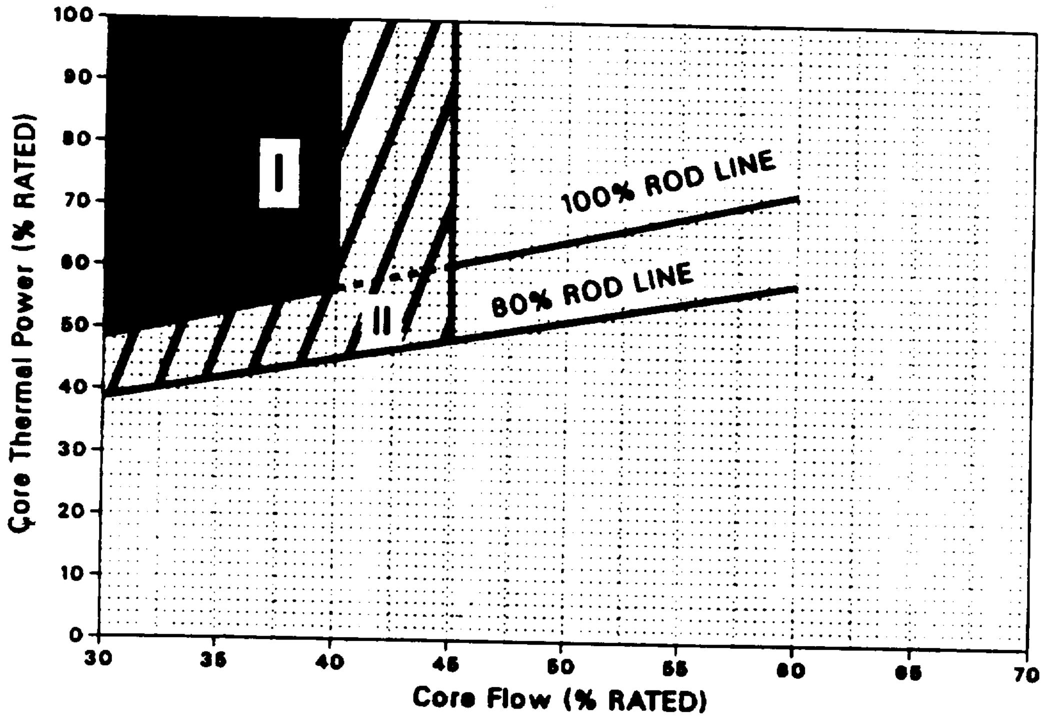

Figure 3.4.1.1.1-1
THERMAL POWER RESTRICTIONS

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 ≤ 80% of the rated pump speed and the reactor at a THERMAL POWER/core flow condition outside of Regions I and II of Figure 3.4.1.1.1-1, 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
≤ 0.58W + 54%	≤ 0.58W + 57%

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

Trip Setpoint	Allowable Value
$S \leq (0.58W + 54\%)\ T$ $S_{RB} \leq (0.58W + 45\%)\ T$	$S \leq (0.58W + 57\%)\ T$ $S_{RB} \leq (0.58W + 48\%)\ T$

4. Specification 3.2.3: The MINIMUM CRITICAL POWER RATIO (MCPR) shall be greater than or equal to the applicable Single Loop Operation MCPR limit as specified in the CORE OPERATING LIMITS REPORT.

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

	Trip Setpoint	Allowable Value
a. RBM - Upscale	≤ 0.66W + 36%	≤ 0.66W + 39%
	Trip Setpoint	**Allowable Value**
b. APRM - Flow Biased	≤ 0.58W + 45%	≤ 0.58W + 48%

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

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 calculated beginning of cycle shutdown margin minus the calculated 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.

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 throughly 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 for the specific reload cycle. The MCPR operating limits as specified in the CORE OPERATING LIMITS REPORT may be a function of average scram speed. In such a case, the results of the required scram time testing (Specification 4.1.3.3) are used to adjust the MCPR operating limits to assure the validity of the cycle specific transient analyses. This ultimately assures that MCPR remains greater than the limit specified in Specification 2.1.2. The occurrence of scram times longer than 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 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 RODS (Continued)

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.

The RSCS and RWM logic automatically initiates at the low power setpoint (20% of RATED THERMAL POWER) to 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

3/4.1.4 CONTROL ROD PROGRAM CONTROLS (Continued)

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 referenced in Specification 6.9.3.

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 pentaborate solution is necessary to ensure that the sodium pentaborate 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.

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

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. The Technical Specification APLHGR for SNP fuel is specified to assure the PCT following a postulated LOCA will not exceed the 2200°F limit. The limiting value for APLHGR is specified in the CORE OPERATING LIMITS REPORT.

The calculational procedure used to establish the APLHGR specified in the CORE OPERATING LIMITS REPORT is based on a loss-of-coolant accident analysis. The analysis was performed using calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. These models are part of the approved methodology referenced in Specification 6.9.3.

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 ≥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 SNP 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 the LHGR for APRM Setpoints Curve specified in the CORE OPERATING LIMITS REPORT. The LHGR for APRM Setpoints Curve specified in the CORE OPERATING LIMITS REPORT is based on SNP's Protection Against Fuel Failure (PAFF) line which was developed using the approved methodology referenced in Specification 6.9.3. The LHGR for APRM Setpoints Curve specified in the CORE OPERATING LIMITS REPORT corresponds to the ratio of PAFF/1.2 under which cladding and fuel integrity is protected during AOOs.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

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

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR specified in the CORE OPERATING LIMITS REPORT is obtained. The required MCPR operating limits as functions core power, core flow, and plant equipment availability condition are specified in the CORE OPERATING LIMITS REPORT.

The cycle specific transient analyses to determine the MCPR operating limits were performed using the NRC approved methods referenced in Specification 6.9.3. The MCPR operating limits as specified in the CORE OPERATING LIMITS REPORT may be specified as a function of average scram speed. In such a case, the results of the required scram time testing (Specification 4.1.3.3) are used to adjust the MCPR operating limits to assure the validity of the cycle specific transient analyses. This ultimately assures that MCPR remains greater than the limit specified in Specification 2.1.2 for all anticipated operational occurrences.

The CORE OPERATING LIMITS REPORT specifies core flow dependent MCPR operating limits which assure that the Safety Limit MCPR will not be exceeded 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. The CORE OPERATING LIMITS REPORT specifies the power dependent MCPR operating limits which assure that the Safety Limit MCPR will not be exceeded in the event of a Feedwater Controller Failure, Rod Withdrawal Error, or Load Reject without Main Turbine Bypass Operable initiated from a full power or reduced power condition.

Cycle specific analyses are performed for the most limiting local and 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.

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

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.

LOCA analyses for two loop operating conditions, which result in Peak Cladding Temperatures (PCTs) below 2200°F, bound single loop operating conditions. Single loop operation LOCA analyses using two-loop MAPLHGR limits result in lower PCTs. Therefore, the use of two-loop MAPLHGR limits during single loop operation assures that the PCT during a LOCA event remains below 2200°F.

The MINIMUM CRITICAL POWER RATIO (MCPR) limits for single loop operation assure that the Safety Limit MCPR is not exceeded for any Anticipated Operational Occurrence (AOO). In addition, the MCPR limits for single-loop operation protect against the effects of the Recirculation Pump Seizure Accident. That is, for operation in single-loop with an operating MCPR limit ≥ 1.30, the radiological consequences of a pump seizure accident from single-loop operating conditions are but a small fraction of 10 CFR 100 guidelines.

For single loop operation, the RBM and APRM setpoints are adjusted by a 8.5% 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 on 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.

Specifications have been provided to prevent, detect, and mitigate core thermal hydraulic instability events. These specifications are prescribed in accordance with NRC Bulletin 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors (BWRs)," dated December 30, 1988.

LPRM upscale alarms are required to detect reactor core thermal hydraulic instability events. The criteria for determining which LPRM upscale alarms are required is based on assignment of these alarms to designated core zones. These core zones consist of the level A, B and C alarms in 4 or 5 adjacent LPRM strings. The number and location of LPRM strings in each zone assure that with 50% or more of the associated LPRM upscale alarms OPERABLE sufficient monitoring capability is available to detect core wide and regional oscillations. Operating plant instability data is used to determine the specific LPRM strings assigned to each zone. The core zones and required LPRM upscale alarms in each zone are specified in appropriate procedures.

3/4.4.1 RECIRCULATION SYSTEM (Continued)

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.

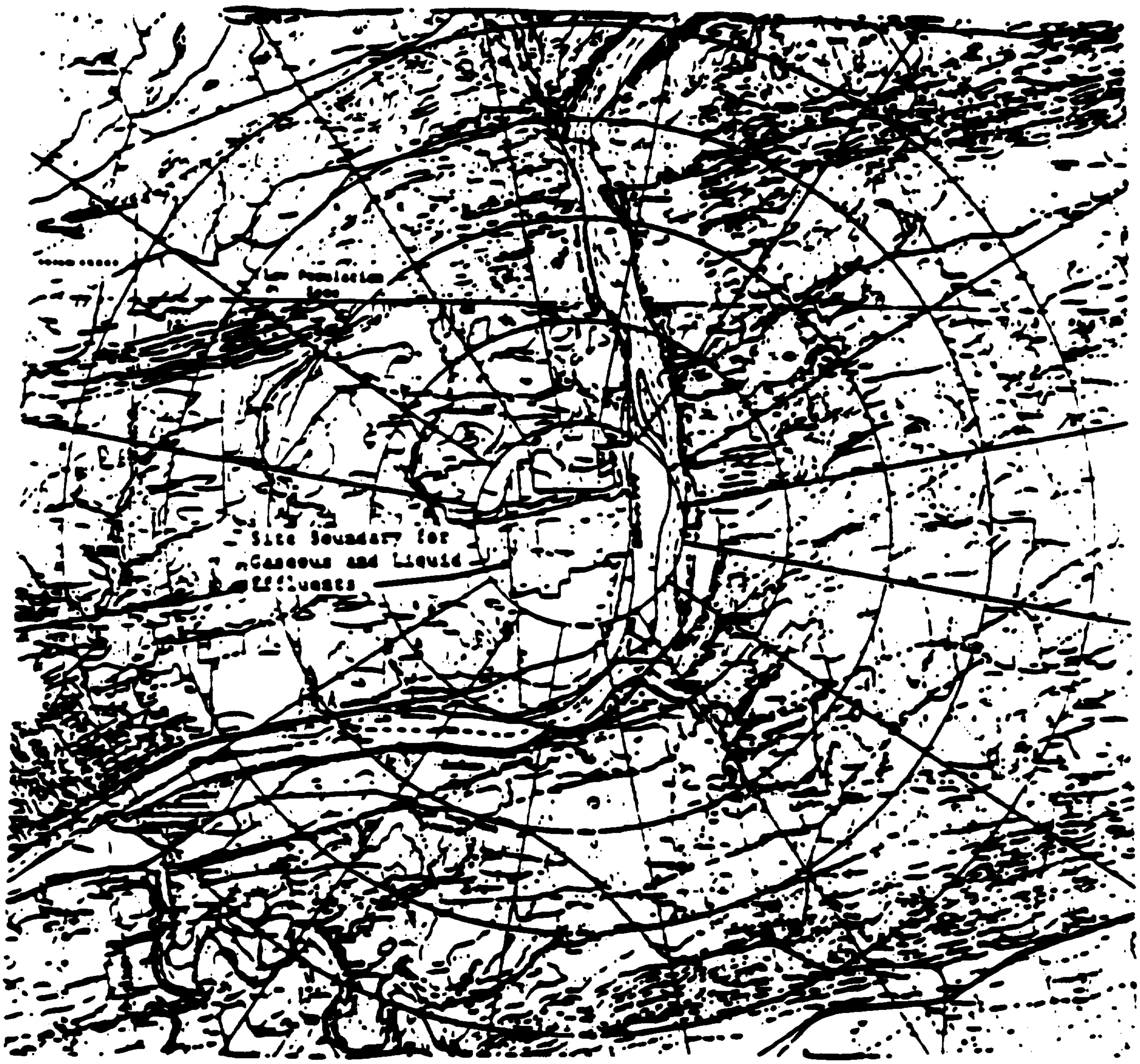

FIGURE 5.1.3-1b

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. Each assembly consists of a matrix of Zircaloy clad fuel rods with an initial composition of non-enriched or slightly enriched uranium dioxide as fuel material and water rods. Limited substitutions of Zirconium alloy filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by test or analyses to comply with all fuel safety design bases. A limited number of lead use assemblies that have not completed representative testing may be placed in non-limiting core regions. Each fuel rod shall have a nominal active fuel length of 150 inches. Reload fuel 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 cruciform shaped control rod assemblies. The control material shall be boron carbide powder (B_4C), and/or Hafnium metal. The control rod shall have a nominal axial absorber length of 143 inches. Control rod assemblies shall be limited to those control rod designs approved by the NRC for use in BWRS.

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.

CORE OPERATING LIMITS REPORT

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

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

a. The Average Planar Linear Heat Generation Rate (APLHGR) for Specification 3.2.1.

b. The Linear Heat Generation Rate for Average Power Range Monitor (APRM) Setpoints for Specification 3.2.2.

c. The Minimum Critical Power Ratio (MCPR) for Specification 3.2.3 and 3.4.1.1.2.

d. The Linear Heat Generation Rate (LHGR) for Specification 3.2.4.

e. The Thermal Power Restrictions for Specification 3.4.1.1.1 and 3.4.1.1.2.

And shall be documented in the CORE OPERATING LIMITS REPORT.

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

1. PL-NF-87-001-A, "Qualification of Steady State Core Physics Methods for BWR Design and Analysis," July, 1988.

2. PL-NF-89-005-A, "Qualification of Transient Analysis Methods for BWR Design and Analysis," July, 1992.

3. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," July, 1992.

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

5. XN-NF-85-67(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, Inc., September 1986.

6. PLA-3407, "Proposed Amendment 132 to License No. NPF-14: Unit 1 Cycle 6 Reload," Letter from H. W. Keiser (PP&L) to W. R. Butler (NRC), July 2, 1990.

7. Letter from Elinor G. Adensam (NRC) to H. W. Keiser (PP&L), "Issuance of Amendment No. 31 to Facility Operating License No. NPF-22 - Susquehanna Steam Electric Station, Unit 2," October 3, 1986.

8. PLA-3533, Revised Proposed Amendment 67 to License No. NPF-22: Unit 2 Cycle 5 Reload," Letter from H. W. Keiser (PP&L) to W. R. Butler (NRC), March 7, 1991.

CORE OPERATING LIMITS REPORT (Continued)

9. XN-NF-84-97, Revision 0, "LOCA-Seismic Structural Response of an ENC 9x9 Jet Pump Fuel Assembly," Exxon Nuclear Company, Inc., December 1984.

10. PLA-2728, "Response to NRC Question: Seismic/LOCA Analysis of U2C2 Reload," Letter from H. W. Keiser (PP&L) to E. Adensam (NRC), September 25, 1986.

11. XN-NF-82-06(P)(A), Supplement 1, Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup Supplement 1 Extended Burnup Qualification of ENC 9x9 Fuel," May 1988.

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

13. XN-NF-524(A), Revision 1, "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors," Exxon Nuclear Company, Inc., November 1983.

14. XN-NF-512-P-A, Revision 1 and Supplement 1, Revision 1, "XN-3 Critical Power Correlation," October, 1982.

15. XN-NF-80-19(A), Volumes 2, 2A, 2B, and 2C, "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," Exxon Nuclear Company, Inc., September 1982.

16. XN-NF-CC-33(A), Revision 1, "HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option," Exxon Nuclear Company, Inc., November 1975.

17. XN-NF-82-07(A), Revision 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, Inc., November 1982.

18. XN-NF-84-117(P), "Generic LOCA Break Spectrum Analysis: BWR 3 and 4 with Modified Low Pressure Coolant Injection Logic," Exxon Nuclear Company, Inc., December 1984.

19. XN-NF-86-65, "Susquehanna LOCA-ECCS Analysis MAPLHGR Results for 9x9 Fuel," Exxon Nuclear Company, Inc., May 1986.

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

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. 95
License No. NPF-22

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

   A. The application for the amendment filed by the Pennsylvania Power & Light Company, dated December 18, 1992, as supplemented by telecopy dated January 28, 1993, and by letters dated March 25, and May 20, 1993, 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. 95 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 license amendment is effective as of its date of issuance and is to be implemented within 30 days after its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael L. Boyle

Michael L. Boyle, Acting Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: August 4, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 95

FACILITY OPERATING LICENSE NO. NPF-22

DOCKET NO. 50-388

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

REMOVE	INSERT
i	i
ii	ii*
iii	iii*
iv	iv
v	v*
vi	vi
xix	xix*
xx	xx
xxi	xxi*
xxii	xxii
1-1	1-1*
1-2	1-2
1-3	1-3
1-4	1-4*
3/4 2-1	3/4 2-1
3/4 2-2	-
3/4 2-3	3/4 2-2
3/4 2-4	3/4 2-3
3/4 2-5	-
3/4 2-5a	-
3/4 2-6	3/4 2-4
3/4 2-7	-
3/4 2-8	-
3/4 2-8a	-
3/4 2-8b	-
3/4 2-9	3/4 2-5

3/4 2-10	-
3/4 2-10a	-
3/4 4-1b	3/4 4-1b*
3/4 4-1c	3/4 4-1c
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 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
5-5	5-5*
5-6	5-6
6-19	6-19*
6-20	6-20
-	6-20a
-	6-20b

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ADMINISTRATIVE CONTROLS

SECTION PAGE

SUSQUEHANNA REVIEW COMMITTEE (SRC) (Continued)

ADMINISTRATIVE CONTROLS

LIST OF FIGURES

1.0 DEFINITIONS

ıe 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.

IANNEL CALIBRATION

1.4 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.

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

CORE ALTERATION

1.7 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.

CORE OPERATING LIMITS REPORT

1.7A The CORE OPERATING LIMITS REPORT is the Susquehanna SES Unit 2 specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.3. Plant operation within these operating limits is addressed in individual specifications.

CRITICAL POWER RATIO

1.8 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 1-135 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 functions, 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 suppression 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 applicable LHGR for APRM Setpoint limit specified in the CORE OPERATING LIMITS REPORT 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.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for all fuel shall not exceed the limit specified in the CORE OPERATING LIMITS REPORT.

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

ACTION:

With an APLHGR exceeding the limit, 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 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 with a LIMITING CONTROL ROD PATTERN for APLHGR.

d. The provisions of Specification 4.0.4 are not applicable.

POWER DISTRIBUTION LIMITS

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-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 #
$S \leq (0.58W + 59\%)\ T$ $S_{RB} \leq (0.58W + 50\%)\ T$	$S \leq (0.58W + 62\%)\ T$ $S_{RB} \leq (0.58W + 53\%)\ T$

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 (FRTP) divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY. The FRACTION OF LIMITING POWER DENSITY (FLPD) for SNP fuel is the actual LHGR divided by the LINEAR HEAT GENERATION RATE for APRM Setpoints limit specified in the CORE OPERATING LIMITS REPORT.

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 determined above, 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:

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* 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.

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 rector is operating with MFLPD greater than or equal to FRTP.

d. The provisions of Specification 4.0.4 are not applicable.

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 applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT.

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

ACTION:

With MCPR less than the applicable MCPR limit, 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:

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.

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed the applicable LHGR limit specified in the CORE OPERATING LIMITS REPORT.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.4 LHGRs shall be determined to be equal to or less than the limit:

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.

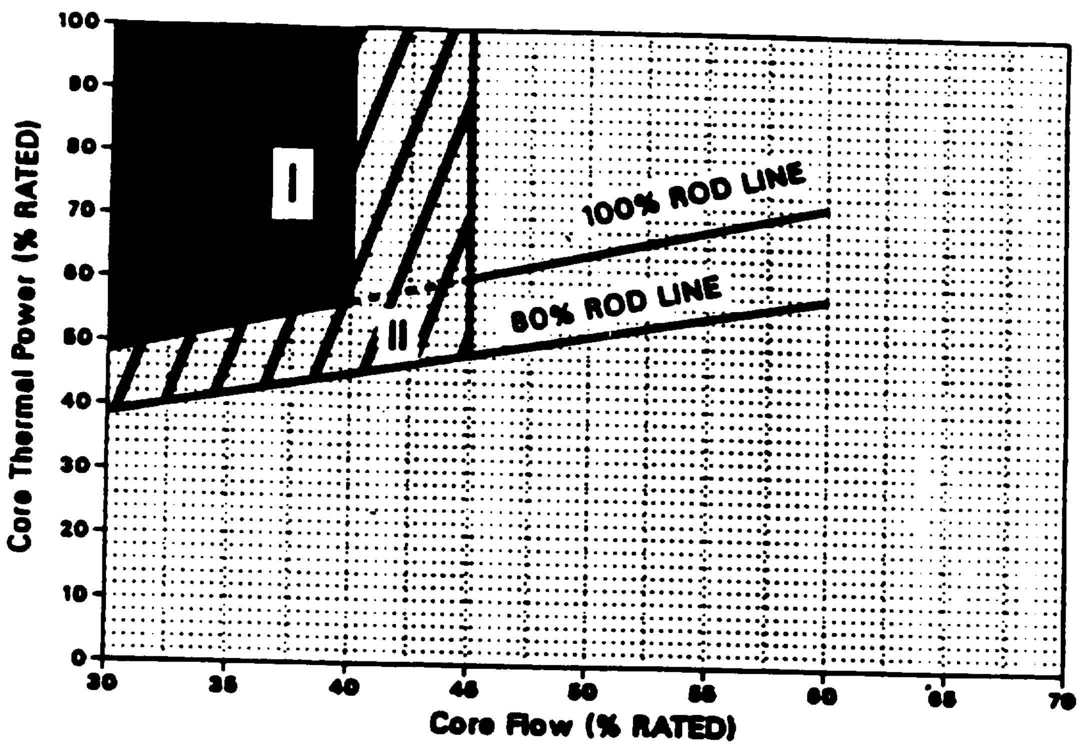

Figure 3.4.1.1.1-1
THERMAL POWER RESTRICTIONS

REACTOR COOLANT SYSTEM

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 ≤ 80% of the rated pump speed and the reactor at a THERMAL POWER/core flow condition outside of Regions I and II of Figure 3.4.1.1.1-1, 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
≤ 0.58W + 54%	≤ 0.58W + 57%

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

Trip Setpoint	Allowable Value
S ≤ (0.58W + 54%) T S_{RB} ≤ (0.58W + 45%) T	S ≤ (0.58W + 57%) T S_{RB} ≤ (0.58W + 48%) T

4. Specification 3.2.3: The MINIMUM CRITICAL POWER RATIO (MCPR) shall be greater than or equal to the applicable Single Loop Operation MCPR limit as specified in the CORE OPERATING LIMITS REPORT.
5. Table 3.3.6-2: the RBM/APRM Control Rod Block Setpoints shall be as follows:

	Trip Setpoint	Allowable Value
a. RBM - Upscale	≤ 0.66W + 36%	≤ 0.66W + 39%
	Trip Setpoint	**Allowable Value**
b. APRM - Flow Biased	≤ 0.58W + 45%	≤ 0.58W + 48%

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

ACTION:

a. In OPERATIONAL CONDITION 1:
   1. With
      a) no reactor coolant system recirculation loops in operation, or
      b) Region I of Figure 3.4.1.1.1-1 entered, or
      c) Region II of Figure 3.4.1.1.1-1 entered and core thermal hydraulic instability occurring as evidenced by:

3/4.1 REACTIVITY CONTROL SYSTEMS

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 for the specific reload cycle. The MCPR operating limits as specified in the CORE OPERATING LIMITS REPORT may be a function of average scram speed. In such a case, the results of the required scram time testing (Specification 4.1.3.3) are used to adjust the MCPR operating limits to assure the validity of the cycle specific transient analyses. This ultimately assures that MCPR remains greater than the limit specified in Specification 2.1.2. The occurrence of scram times longer than 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 logic automatically initiates at the low power setpoint (20% of RATED THERMAL POWER) to 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 referenced in Specification 6.9.3.

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 pentaborate solution is necessary to ensure that the sodium pentaborate 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.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

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. The Technical Specification APLHGR for SNP fuel is specified to assure the PCT following a postulated LOCA will not exceed the 2200°F limit. The limiting value for APLHGR is specified in the CORE OPERATING LIMITS REPORT.

The calculational procedure used to establish the APLHGR specified in the CORE OPERATING LIMITS REPORT is based on a loss-of-coolant accident analysis. The analysis was performed using calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. These models are part of the approved methodology referenced in Specification 6.9.3.

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 ≥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 SNP 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 the LHGR for APRM Setpoints Curve specified in the CORE OPERATING LIMITS REPORT. The LHGR for APRM Setpoints Curve specified in the CORE OPERATING LIMITS REPORT is based on SNP's Protection Against Fuel Failure (PAFF) line which was developed using the approved methodology referenced in Specification 6.9.3. The LHGR for APRM Setpoints Curve specified in the CORE OPERATING LIMITS REPORT corresponds to the ratio of PAFF/1.2 under which cladding and fuel integrity is protected during AOOs.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

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

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR specified in the CORE OPERATING LIMITS REPORT is obtained. The required MCPR Operating Limits as a function of core power, core flow, and plant equipment availability condition are specified in the CORE OPERATING LIMITS REPORT.

The cycle specific transient analyses to determine the MCPR operating limits were performed using the NRC approved methods referenced in Specification 6.9.3. The MCPR operating limits as specified in the CORE OPERATING LIMITS REPORT may be specified as a function of average scram speed. In such a case, the results of the required scram time testing (Specification 4.1.3.3) are used to adjust the MCPR operating limits to assure the validity of the cycle specific transient analyses. This ultimately assures that MCPR remains greater than the limit specified in Specification 2.1.2 for all anticipated operational occurrences.

The CORE OPERATING LIMITS REPORT specifies 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. The CORE OPERATING LIMITS REPORT specifies the power dependent MCPR operating limits which assure that the Safety Limit MCPR will not be violated in the event of a Feedwater Controller Failure, Rod Withdrawal Error, or Load Reject without Main Turbine Bypass operable initiated from a full power or reduced power condition.

Cycle specific analyses are performed for the most limiting local and 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.

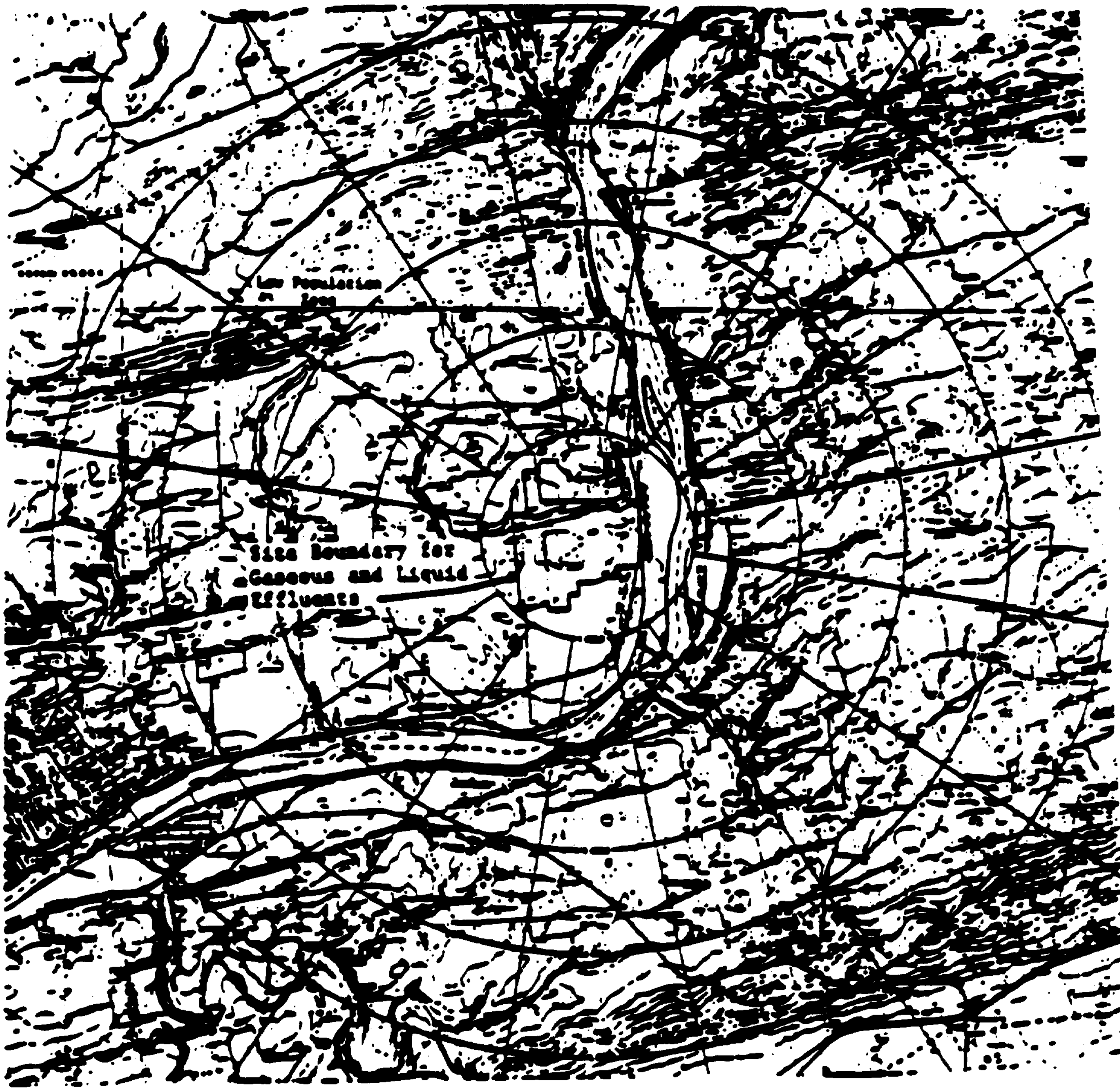

FIGURE 5.1.3-1b

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. Each assembly consists of a matrix of Zircaloy clad fuel rods with an initial composition of non-enriched or slightly enriched uranium dioxide as fuel material and water rods. Limited substitutions of Zirconium alloy filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by test or analyses to comply with all fuel safety design bases. A limited number of lead use assemblies that have not completed representative testing may be placed in non-limiting core regions. Each fuel rod shall have a nominal active fuel length of 150 inches. Reload fuel 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 cruciform shaped control rod assemblies. The control material shall be boron carbide powder (B_4C), and/or Hafnium metal. The control rod shall have a nominal axial absorber length of 143 inches. Control rod assemblies shall be limited to those control rod designs approved by the NRC for use in BWRs.

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.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

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

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

The Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction and atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, atmospheric stability.** This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figures 5.1.3-1a and 5.1.3-1b) during the report period. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. The assessment of radiation doses shall be performed in accordance with the methodology and parameters of the Offsite Dose Calculation Manual (ODCM).

The Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBERS OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

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*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

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

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

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

1. Container volume,
2. Total curie quantity (specify whether determined by measurement or estimate),
3. Principal radionuclides (specify whether determined by measurement or estimate),
4. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),
5. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and
6. Solidification agent or absorbent (e.g., cement; urea formaldehyde).

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents on a quarterly basis.

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

SPECIAL REPORTS

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

CORE OPERATING LIMITS REPORT

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

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

a. The Average Planar Linear Heat Generation Rate (APLHGR) for Specification 3.2.1.
b. The Linear Heat Generation Rate for Average Power Range Monitor (APRM) Setpoints for Specification 3.2.2.
c. The Minimum Critical Power Ratio (MCPR) for Specification 3.2.3 and 3.4.1.1.2.
d. The Linear Heat Generation Rate (LHGR) for Specification 3.2.4.
e. The Thermal Power Restrictions for Specification 3.4.1.1.1 and 3.4.1.1.2.

And shall be documented in the CORE OPERATING LIMITS REPORT.

CORE OPERATING LIMITS REPORT (Continued)

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

1. PL-NF-87-001-A, "Qualification of Steady State Core Physics Methods for BWR Design and Analysis," July, 1988.

2. PL-NF-89-005-A, "Qualification of Transient Analysis Methods for BWR Design and Analysis," July, 1992.

3. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," July, 1992.

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

5. XN-NF-85-67(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, Inc., September 1986.

6. PLA-3407, "Proposed Amendment 132 to License No. NPF-14: Unit 1 Cycle 6 Reload," Letter from H. W. Keiser (PP&L) to W. R. Butler (NRC), July 2, 1990.

7. Letter from Elinor G. Adensam (NRC) to H. W. Keiser (PP&L), "Issuance of Amendment No. 31 to Facility Operating License No. NPF-22 - Susquehanna Steam Electric Station, Unit 2," October 3, 1986.

8. PLA-3533, Revised Proposed Amendment 67 to License No. NPF-22: Unit 2 Cycle 5 Reload," Letter from H. W. Keiser (PP&L) to W. R. Butler (NRC), March 7, 1991.

9. XN-NF-84-97, Revision 0, "LOCA-Seismic Structural Response of an ENC 9x9 Jet Pump Fuel Assembly," Exxon Nuclear Company, Inc., December 1984.

10. PLA-2728, "Response to NRC Question: Seismic/LOCA Analysis of U2C2 Reload," Letter from H. W. Keiser (PP&L) to E. Adensam (NRC), September 25, 1986.

11. XN-NF-82-06(P)(A), Supplement 1, Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup Supplement 1 Extended Burnup Qualification of ENC 9x9 Fuel," May 1988.

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

13. XN-NF-524(A), Revision 1, "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors," Exxon Nuclear Company, Inc., November 1983.

CORE OPERATING LIMITS REPORT (Continued)

14. XN-NF-512-P-A, Revision 1 and Supplement 1, Revision 1, "XN-3 Critical Power Correlation," October, 1982.

15. XN-NF-80-19(A), Volumes 2, 2A, 2B, and 2C, "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," Exxon Nuclear Company, Inc., September 1982.

16. XN-NF-CC-33(A), Revision 1, "HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option," Exxon Nuclear Company, Inc., November 1975.

17. XN-NF-82-07(A), Revision 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, Inc., November 1982.

18. XN-NF-84-117(P), "Generic LOCA Break Spectrum Analysis: BWR 3 and 4 with Modified Low Pressure Coolant Injection Logic," Exxon Nuclear Company, Inc., December 1984.

19. XN-NF-86-65, "Susquehanna LOCA-ECCS Analysis MAPLHGR Results for 9x9 Fuel," Exxon Nuclear Company, Inc., May 1986.

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

6.10 RECORD RETENTION

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

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 126 TO FACILITY OPERATING LICENSE NO. NPF-14

AMENDMENT NO. 95 TO FACILITY OPERATING LICENSE NO. NPF-22

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-387 AND 388

1.0 INTRODUCTION

By letter dated December 18, 1992, as supplemented by telecopy dated January 28, 1993, and by letters dated March 25, and May 20, 1993, the Pennsylvania Power and Light Company and Allegheny Electric Cooperative, Inc. (the licensees) submitted a request for changes to the Susquehanna Steam Electric Station, Units 1 and 2, Technical Specifications (TS). The requested changes would remove cycle-specific parameter limits from the TSs in accordance with NRC Generic Letter (GL) 88-16 and modify Section 5.3.1 in accordance with NRC GL 90-02. The January 28, 1993, telecopy provided a corrected page 3 to the application to delete two superfluous words ("are anticipated") in the last line of item 1 of the licensee's No Significant Hazards Consideration determination. The January 28, 1993, correction did not, in any way, modify the TS application but, for completeness, was referenced in the staff's initial notice published in the FEDERAL REGISTER on February 17, 1993. As discussed in more detail subsequently, the March 25, 1993, letter removed a figure that was still in the NRC's "authority" file that should have been removed by an amendment issued almost 4 years ago. Removal of the meaningless figure did not change the TSs and thus, had no effect on the staff's No Significant Hazards Consideration Determination.

As also discussed in more detail at the end of this safety evaluation, as requested by the NRC staff, the May 20, 1993 submittal retained a sentence on fuel enrichment that was in the model TSs issued with GL 90-02, but was inadvertently omitted in the model TSs issued with Supplement 1 to GL 90-02. The sentence is in the present Susquehanna, Units 1 and 2 TSs, so the effect of the May 20, 1993, submittal was to keep a present requirement. The change is thus not substantive and did not change the staff's No Significant Hazards Consideration Determination. The May 20, 1993, submittal also substituted a power/flow map figure from the Unit 2 TSs in the Unit 1 TSs since this represented the version most recently approved by the Commission for Siemens 9X9 fuel. (The figure for Unit 2 was approved by Amendment 91 issued October 28, 1992, whereas, the figure for Unit 1 was approved by Amendment No. 118, issued on May 7, 1992). The substitution was not a substantive change and did not affect the staff's proposed No Significant Hazards Consideration Determination.

As noted above, these amendments would change the TSs to remove cycle-specific parameter limits in accordance with NRC Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications" issued October 4, 1988. The proposed changes would replace the values of cycle-specific parameter limits with a reference to the Core Operating Limits Report, which contains the values of those limits. In addition, the Core Operating Limits Report (COLR) has been included in the Definitions Section of the TSs to note that it is the unit-specific document that provides these limits for the current operating reload cycle. Furthermore, the definition notes that the values of these cycle-specific parameter limits are to be determined in accordance with the Specification 6.9.3. This specification requires that the Core Operating Limits be determined for each reload cycle in accordance with the referenced NRC-approved methodology for these limits and consistent with the applicable limits of the safety analysis. Finally, this report and any mid-cycle revisions shall be provided to the NRC upon issuance.

Guidance on the proposed changes was developed by NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket that was endorsed by the Babcock and Wilcox Owners Group. This guidance was provided to all power reactor licensees and applicants by GL 88-16, dated October 4, 1988.

On February 1, 1990, the NRC staff issued GL 90-02, a line-item technical specification improvement, "Alternative Requirements for Fuel Assemblies in the Design Features Section of Technical Specifications." The change endorsed by the NRC staff provides flexibility in the repair of fuel assemblies containing damaged and leaking fuel rods by reconstituting the assemblies. Based on the NRC staff experience with implementation of GL 90-02, the staff issued "Supplement 1 to Generic Letter 90-02" on July 31, 1992. The supplement provided specific guidance for fuel reconstitution and, on page 4, provided specific wording for a specification that could be substituted for the present paragraph on "Fuel Assemblies" in Section 5 of the TSs. As part of this application, the licensee is proposing to substitute the paragraph in GL 90-02, Supplement 1, for the present Section 5.3.1, but also retain a sentence which is in the present Susquehanna TSs limiting the weight percent U-235 in reload fuel. The licensee is also proposing to revise Section 5.3.2 on Control Rod Assemblies by deleting reference to the stainless steel tubes used in the initial core and specifically stating that "control rod assemblies shall be limited to those control rod designs approved by the NRC for use in BWRs."

2.0 EVALUATION

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

1. The Definitions section of the TS (Section 1.7) was modified to include a definition of the COLR that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with an NRC-approved methodology that maintains the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

2. The following specifications were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

    a. Revise the Index.

        i. Add Section 1.7A - Core Operating Limits Report

        ii. Change current page 3/4 2-5 on APRM setpoints to 3/4 2-2

        iii. Change current page 3/4 2-5 to 3/4 2-3

        iv. Change current page 3/4 2-7 on Minimum Critical Power Ratio to 3/4 2-4

        v. Change current page 3/4 2-10a on Linear Heat Generation Rate to 3/4 2-5

        vi. Add new section 6.9.3 describing the Core Operating Limits Report on pages 6-20 and add pages 6-20a and 6-20b

    b. Section 1.13 was revised to specify that the Fraction of Limiting Power Density shall be specified in the Core Operating Limits Report.

    c. Sections 3.2.1, 3.2.2, 3.2.3 and 3.2.4 were revised to delete all figures and to state that the Average Planar Linear Heat Generation Rates, the APRM Setpoints, the Minimum Critical Power Ratio and the Linear Heat Generation Rate shall not exceed the limits specified in the Core Operating Limits Report.

    d. For Unit 1, replace the present Figure 3.4.1.1.1-1 on "Thermal Power Restrictions" with the figure that is in the Unit 2 TSs and which was approved for Unit 2 by Amendment No. 91 issued October 28, 1992.

    e. Modify Bases 3/4 1.3, 3/4 1.4, 3/4 2.1, 3/4 2.2, 3/4 2.3 and 3/4 4.1 to reference the Core Operating Limits Report.

f. Revise Section 5.3.1 on Fuel Assemblies to substitute the suggested wording in NRC GL 90-02, Supplement 1.

g. Revise Section 5.3.2 to state that "Control rod assemblies shall be limited to those control rod designs approved by the NRC for use in BWRs".

h. Add Section 6.9.3 to the Administrative Controls Section, "Special Reports" to describe the Core Operating Limits Report (COLR). This section specifies the information to be included in the COLR and the requirement to submit the COLR to the NRC. Specifically, this specification requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, this specification requires that the values of these limits be established using the NRC-approved methodology in the specific topical reports which the NRC has approved for the Susquehanna Steam Electric Station (SES) (which are listed in this section of the TSs). Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to the NRC.

On the basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in GL 88-16 on modifying cycle-specific parameter limits in TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using an NRC-approved methodology, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds the proposed changes acceptable. As part of the implementation of GL 88-16, the staff has also reviewed the two sample COLRs that were provided by the licensee as part of the subject application. Both COLRs were dated November 1992. One was for Susquehanna SES Unit 1, Cycle 7 (PL-NF-92-004). The NRC approved the Unit 1 Cycle 7 core reload values by Amendment No. 118 issued May 7, 1992. Unit 1 started up in the current fuel cycle 7 on May 17, 1992, with shutdown scheduled for the seventh refueling in October 1993. The second sample COLR was for Susquehanna SES, Unit 2, Cycle 6 (PL-NR-92-008). The NRC approved the Unit 2 Cycle 6 reload values by Amendment No. 91 issued October 28, 1992. Unit 2 began the sixth fuel cycle on November 13, 1992, when the main generator was synchronized to the grid. The sixth refueling outage is scheduled to start March 12, 1994. On the basis of this review, the staff concludes that the format and content of the sample COLRs are acceptable.

As noted in the Introduction, the licensee proposes to substitute the paragraph in GL 90-02, Supplement 1, for the present wording in Section 5.3.1 on "Fuel Assemblies". The proposed revision is acceptable.

On May 15, 1989, we issued Amendment No. 90 to support Unit 1, Cycle 5 operations with Advanced Nuclear Fuel (ANF) Corporation's 9x9 reload fuel. As part of the changes, Figure 3.2.1-1 was deleted. Figure 3.2.1-2 was changed to Figure 3.2.1-1 and Figure 3.2.1-3, which was on page 3/4 2-4a, was changed to Figure 3.2.1-2 and relocated to page 3/4 2-3. However, the licensee's submittal of February 2, 1989, did not request deletion of page 3/4 2-4a so that there were two identical figures in the TSs, one on page 3/4 2-3 and one on page 3/4 2-4a. The specific figure was titled "Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) versus Average Bundle Exposure, ANF 9x9 Fuel." On November 2, 1990, we issued Amendment No. 102 to support the Unit 1, Cycle 6 reload, but did not delete the superfluous Figure 3/4 2-3 on page 3/4 2-4a, even though this figure was not mentioned in Section 3.2.1 of the TSs. On May 7, 1992, the staff issued Amendment No. 118 to support the Unit 1, Cycle 7 reload. The previous Figure 3.2.1-1 on MAPLHGR versus average bundle exposure for ANF 8x8 fuel was deleted and the previous Figure 3.2.1-2 was revised to become Figure 3.2.1-1 and to reflect that the fuel was supplied by Siemens Nuclear Power Corporation, which was formerly ANF. The text in Section 3.2.1 (page 3/4 2-1) of the present TSs refers only to the one figure - Figure 3.2.1-1. The superfluous Figure 3.2.1-3 remained on page 3/4 2-4a of the TSs. During our review of the subject application, we noted this meaningless page and discussed it with the licensee. By letter dated March 25, 1993, the licensee requested we delete page 3/4 2-4a as we suggested. This is a purely administrative action to remove a page that should have been removed by Amendment No. 90 almost 4 years ago. The deletion has no effect on the staff's No Significant Hazards Consideration Determination (NSHCD) since it is not referred to in the TSs. There was also one other minor administrative change (i.e., to change the word function to functions), with the licensee's concurrence and likewise does not change the NSHCD.

In the application of December 18, 1992, the licensee proposed to relocate Figure 3.4.1.1.1-1, "Thermal Power Restrictions," to the COLR and to revise and repaginate Sections 3.4.1.1.1 and 3.4.1.1.2 on Two-Loop and Single-Loop operation to reflect the proposed deletion of this figure from the TSs.

On March 9, 1988, a thermal hydraulic instability event occurred at LaSalle, Unit 2. The NRC discussed this event in Information Notice 88-39, "LaSalle, Unit 2 Loss of Recirculation Pumps with Power Oscillation Event," and Bulletins 88-07 and 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors." In the first bulletin, the NRC requested licensees to establish procedures and give training to reactor operators to enable them to recognize oscillations and to take appropriate actions. In the supplement, the NRC requested the licensees to implement the General Electric (GE) Interim Recommendations for Stability Actions, designated the Interim Corrective Actions (ICA) which GE issued in November 1988.

On August 15, 1992, Washington Nuclear Power, Unit 2 (WNP-2) experienced power oscillations during startup. The event occurred early in cycle 8 operation. During cycle 8, the licensee had two previous startups without incident. The reactor core consisted primarily of Siemens fuel, with about 74 percent of this fuel in 8x8 fuel assemblies and about 25% in 9x9 fuel assemblies, and with the remainder of the core consisting of various lead test assemblies. The 9x9 fuel assembly used in WNP-2, designated 9x9-9x, has a higher flow resistance than the 8x8 fuel assembly with a difference of about 10% in pressure drop. Susquehanna, Units 1 and 2, are entirely fueled by the Siemens 9x9 fuel assemblies. The WNP-2 event was discussed at length in NRC Information Notice 92-74, dated November 10, 1992.

As noted previously, on October 28, 1992, the Commission issued Amendment No. 91 for SSES, Unit 2, approving the Unit 2, Cycle 6 reload. The amendment included new thermal-hydraulic limits on the Siemens Nuclear Power Corporation on the (SNP) 9x9 fuel, including a new Figure 3.4.1.1.1-1 on Thermal Power Restrictions. The latter reflected the staff's evaluation of the August 15, 1992 incident at WNP-2. Pending resolution of the instability issue, we advised the licensee that the present figure on thermal power restrictions should remain in the TSs. We also discussed with the licensee the possibility of using the figure recently approved for Unit 2 for Unit 1 as well, since both units are fueled by the same 9x9 fuel. The licensee's letter of May 20, 1993, withdrew all changes to Sections 3.4.1.1.1 and 3.4.1.1.2 except to substitute the present Unit 2 figure 3.4.1.1.1-1 for the figure with the same title and number in the Unit 1 TS. As a result, the renumbering of pages in Sections 3.4.1.1.1 and 3.4.1.1.2 proposed in the licensee's initial application of December 18, 1992, was rescinded by the letter of May 20, 1993. Since there is no change to the present TSs in these two sections, the May 20, 1993 letter does not change the staff's initial no significant hazards consideration.

When the staff issued GL 90-02, the last sentence in the "Model Technical Specification Change" for Section 5.3.1 stated: "Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.0 weight percent U-235." This sentence was inadvertently omitted when Supplement 1 to GL 90-02 was issued. In the application of December 18, 1992, the licensee had proposed the wording for Section 5.3.1 that was in the Supplement. We requested the license to also add the sentence on enrichment that was in the initial GL 90-02, which they did with their letter of May 20, 1993. The same sentence is in the present Susquehanna TSs, so the retention in the revised Section 5.3.1 does not change the staff's no significant hazards consideration.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (58 FR 8776). Accordingly, the amendments meet 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 or environmental assessment need be prepared in connection with the issuance of the amendments.

These amendments also change recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: R. Clark, T. Dunning, H. Richings

Date: August 4, 1993