Docket File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 27, 1995

Mr. George A. Hunger, Jr.
Director-Licensing, MC 62A-1
PECO Energy Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, PA 19087-0195

SUBJECT: TECHNICAL SPECIFICATIONS CHANGE, ARTS/MELLLA IMPLEMENTATION, LIMERICK GENERATING STATION, UNIT 2 (TAC NO. M87309)

Dear Mr. Hunger:

The Commission has issued the enclosed Amendment No. 48 to Facility Operating License No. NPF-85 for the Limerick Generating Station, Unit 2. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated August 27, 1993, and supplemented by your submittal of November 17, 1993.

The amendment revises the TS, contained in Appendix A of the Operating License, to allow an expanded power-to-flow operating domain supported by Average Power Range Monitor - Rod Block Monitor Technical Specifications/ Maximum Extended Load Line Limit Analyses (ARTS/MELLLA). The design basis Loss-of-Coolant Accident (LOCA) has been analyzed using the SAFER/GESTR-LOCA methodology which is generically approved by the staff when plant-specific criteria are met. The staff finds the implementation of the ARTS/MELLLA program and SAFER/GESTR-LOCA methods acceptable.

The amendment is effective as of its date of issuance. You are to inform the staff when you have implemented the provisions of this amendment. In your application, you proposed that the ARTS/MELLLA amendments apply to both Units 1 and 2, however, you noted that the ARTS/MELLLA modifications would not be made on Unit 2 until its third refueling outage, which is currently scheduled for January 1995. In order to preclude confusion between the effective date for the Unit 2 ARTS/MELLLA amendment and any amendment requests that might affect the same TS pages, the staff issued the ARTS/MELLLA amendment for Unit 1 on February 10, 1994, and delayed the issuance of the TS amendment for Unit 2 until the Unit 2 third refueling outage. The enclosed safety evaluation applies to both units. However, the enclosed amendment applies only to Unit 2.

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

Sincerely,

/S/

Frank Rinaldi, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-353

Enclosures:
1. Amendment No. 48 to License No. NPF-85
2. Safety Evaluation

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

Sincerely,

Frank Rinaldi

Frank Rinaldi, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-353

Enclosures:
1. Amendment No. 48 to License No. NPF-85
2. Safety Evaluation

cc w/encls:
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-353

LIMERICK GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 48
License No. NPF-85

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

   A. The application for amendment by Philadelphia Electric Company (the licensee) dated August 27, 1993, and supplemented by letter dated November 17, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-85 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 48 , are hereby incorporated into this license. Philadelphia Electric Company 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 shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

for

John F. Stolz, 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: January 27, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 48

FACILITY OPERATING LICENSE NO. NPF-85

DOCKET NO. 50-353

Replace the following pages of the Appendix A Technical Specifications with the attached page. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
i	i
ii	ii
iii	iii
vi	vi
xviii	xviii
1-2	1-2
1-3	1-3
1-4	1-4
1-5	1-5
1-6	1-6
1-7	1-7
2-4	2-4
B 2-7	B 2-7
3/4 1-18	3/4 1-18
3/4 1-19	3/4 1-19
3/4 1-20	3/4 1-20
3/4 2-1	3/4 2-1
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8
3/4 2-9	3/4 2-9

ATTACHMENT TO LICENSE AMENDMENT NO. 48

FACILITY OPERATING LICENSE NO. NPF-85

DOCKET NO. 50-353

Replace the following pages of the Appendix A Technical Specifications with the attached page. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
3/4 3-8	3/4 3-8
3/4 3-59	3/4 3-59
3/4 3-60	3/4 3-60
3/4 3-60a	3/4 3-60a
3/4 3-61	3/4 3-61
3/4 3-62	3/4 3-62
3/4 4-1	3/4 4-1
3/4 4-1a	3/4 4-1a
B 3/4 1-3	B 3/4 1-3
B 3/4 1-4	B 3/4 1-4
-	B 3/4 1-5
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-4	B 3/4 2-4
B 3/4 2-5	B 3/4 2-5
B 3/4 4-1	B 3/4 4-1
B 3/4 6-3	B 3/4 6-3
6-18a	6-18a

DEFINITIONS

SECTION

DEFINITIONS

SECTION

DEFINITIONS (Continued) PAGE

SECTION

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

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 (COLR) is the unit-specific document that provides the 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 Specifications 6.9.1.9 thru 6.9.12. Plant operation within these 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 (GEXL) correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

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

DOWNSCALE TRIP SETPOINT (DTSP)

1.9a The downscale trip setpoint associated with the Rod Block Monitor (RBM) rod block trip setting.

$\bar{E}$-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 noniodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

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

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.12 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to complete 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.

1.13 (Deleted)
1.14 (Deleted)

FREQUENCY NOTATION

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

HIGH (POWER) TRIP SETPOINT (HTSP)

1.15a The high power trip setpoint associated with the Rod Block Monitor (RBM) rod block trip setting applicable above 85% reactor thermal power.

IDENTIFIED LEAKAGE

1.16 IDENTIFIED LEAKAGE shall be:

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

INTERMEDIATE (POWER) TRIP SETPOINT (ITSP)

1.16a The intermediate power trip setpoint associated with the Rod Block Monitor (RBM) rod block trip setting applicable between 65% and 85% reactor thermal power.

ISOLATION SYSTEM RESPONSE TIME

1.17 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

1.18 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

1.19 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

1.20 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e., 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.

LOW (POWER) TRIP SETPOINT (LTSP)

1.20a The low power trip setpoint associated with the Rod Block Monitor (RBM) rod block trip setting applicable between 30% and 65% reactor thermal power.

1.21 (Deleted)

MEMBER(S) OF THE PUBLIC

1.22 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.

MAPFAC(F)-(MAPLHGR FLOW FACTOR)

1.22a A core flow dependent multiplication factor used to flow bias the standard Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit.

MAPFAC(P)-(POWER DEPENDENT MAPLHGR MULTIPLIER)

1.22b A core power dependent multiplication factor used to power bias the standard Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit.

MINIMUM CRITICAL POWER RATIO (MCPR)

1.23 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core (for each class of fuel). Associated with the minimum critical power ratio is a core flow dependent (MCPR(F)) and core power dependent (MCPR(P)) minimum critical power ratio.

OFFSITE DOSE CALCULATION MANUAL

1.24 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.7 and 6.9.1.8.

OPERABLE - OPERABILITY

1.25 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

1.26 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.

PHYSICS TESTS

1.27 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.28 PRESSURE BOUNDARY LEAKAGE shall be leakage through a nonisolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.29 PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 of Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. The primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows, or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM

1.30 The PROCESS CONTROL PROGRAM (PCP) shall contain the provisions to assure that the SOLIDIFICATION or dewatering and packaging of radioactive wastes results in a waste package with properties that meet the minimum and stability requirements of 10 CFR Part 61 and other requirements for transportation to the disposal site and receipt at the disposal site. With SOLIDIFICATION or dewatering, the PCP shall identify the process parameters influencing SOLIDIFICATION or dewatering based on laboratory scale and full scale testing or experience.

PURGE - PURGING

1.31 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.32 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3293 MWt.

REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY

1.33 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All reactor enclosure secondary containment penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, slide gate damper or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.5.2.1-1 of Specification 3.6.5.2.1.

b. All reactor enclosure secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

d. The reactor enclosure recirculation system is in compliance with the requirements of Specification 3.6.5.4.

e. At least one door in each access to the reactor enclosure secondary containment is closed.

f. The sealing mechanism associated with each reactor enclosure secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

g. The pressure within the reactor enclosure secondary containment is less than or equal to the value required by Specification 4.6.5.1.1a.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.34 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY

1.35 REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All refueling floor secondary containment penetrations required to be closed during accident conditions are either:

REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY (Continued)

1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, slide gate damper or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.5.2.2-1 of Specification 3.6.5.2.2.

b. All refueling floor secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

d. At least one door in each access to the refueling floor secondary containment is closed.

e. The sealing mechanism associated with each refueling floor secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

f. The pressure within the refueling floor secondary containment is less than or equal to the value required by Specification 4.6.5.1.2a.

REPORTABLE EVENT

1.36 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.37 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SHUTDOWN MARGIN

1.38 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

SITE BOUNDARY

1.39 The SITE BOUNDARY shall be that line as defined in Figure 5.1.3-1a.

1.40 (Deleted)

SOURCE CHECK

1.41 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Neutron Flux-Upscale		
1) During two recirculation loop operation:		
a) Flow Biased	≤ 0.66 W+ 66%, with a maximum of	≤ 0.66 W+ 68%, with a maximum of
b) High Flow Clamped	≤ 115% of RATED THERMAL POWER	≤ 117% of RATED THERMAL POWER
2) During single recirculation loop operation:		
a) Flow Biased	≤ 0.66 W+ 61%,	≤ 0.66 W+ 63%,
b) High Flow Clamped	Not Required OPERABLE	Not Required OPERABLE
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 4% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High	≤ 1037 psig	≤ 1057 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches above instrument zero*	≥ 11.0 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed
6. Main Steam Line Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full power background
7. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Level Transmitter	≤ 261' 1 1/4" elevation**	≤ 261' 9 1/4" elevation
b. Float Switch	≤ 261' 1 1/4" elevation**	≤ 261' 9 1/4" elevation

* See Bases Figure B 3/4.3-1.
** Equivalent to 25.58 gallons/scram discharge volume.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks and because several rods must be moved to change power by a significant amount, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the trip level, the rate of power rise is not more than 5% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the Run position.

The APRM trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation for the case of the Neutron Flux-Upscale flow bias setpoint; i.e, for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer associated with the fuel.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown.

3. Reactor Vessel Steam Dome Pressure-High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This trip setpoint is effective at low power/flow conditions when the turbine stop valve and control fast closure trips are bypassed. For a turbine trip or load rejection under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both rod block monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER and less than 90% of RATED THERMAL POWER with MCPR less than 1.70, or THERMAL POWER greater than or equal to 90% of rated with MCPR less than 1.40.

ACTION:

a. With one RBM channel inoperable:

   1. Verify that the reactor is not operating on a LIMITING CONTROL ROD PATTERN, and

   2. Restore the inoperable RBM channel to OPERABLE status within 24 hours.

   Otherwise, place the inoperable rod block monitor channel in the tripped condition within the next hour.

b. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within 1 hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 Each of the above required RBM channels shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies and for the OPERATIONAL CONDITIONS specified in Table 4.3.6-1.

b. CHANNEL FUNCTIONAL TEST prior to control rod withdrawal when the reactor is operating on a LIMITING CONTROL ROD PATTERN.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system, consisting of a minimum of two pumps and corresponding flow paths, shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 5*

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

1. With only one pump and corresponding explosive valve OPERABLE, restore one inoperable pump and corresponding explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

2. With standby liquid control system otherwise inoperable, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5*:

1. With only one pump and corresponding explosive valve OPERABLE, restore one inoperable pump and corresponding explosive valve to OPERABLE status within 30 days or insert all insertable control rods within the next hour.

2. With the standby liquid control system otherwise inoperable, insert all insertable control rods within 1 hour.

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that:

1. The temperature of the sodium pentaborate solution is within the limits of Figure 3.1.5-1.

2. The available volume of sodium pentaborate solution is at least 3160 gallons.

3. The temperature of the pump suction piping is within the limits of Figure 3.1.5-1 for the most recent concentration analysis.

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

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by:

1. Verifying the continuity of the explosive charge.

2. Determining by chemical analysis and calculation* that the available weight of sodium pentaborate is greater than or equal to 3754 lbs; the concentration of sodium pentaborate in solution is less than or equal to 13.8% and within the limits of Figure 3.1.5-1 and; the following equation is satisfied:

$$\frac{C}{13\%\ wt.} \times \frac{E}{29\ atom\ \%} \times \frac{Q}{86\ gpm} \geq 1$$

where

C = Sodium pentaborate solution (% by weight)
Q = Two pump flowrate, as determined per surveillance requirement 4.1.5.c.
E = Boron 10 enrichment (atom % Boron 10)

3. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm per pump at a pressure of greater than or equal to 1190 psig is met.

d. At least once per 24 months during shutdown by:

1. Initiating at least one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of the batch successfully fired. All injection loops shall be tested in 3 operating cycles.

2. Verify all heat-treated piping between storage tank and pump suction is unblocked.**

e. Prior to addition of Boron to storage tank verify sodium pentaborate enrichment to be added is $\geq$ 29 atom % Boron 10.

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* This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after water or boron addition or solution temperature is restored.

** This test shall also be performed whenever suction piping temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after solution temperature is restored.

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 each type of fuel as a function of axial location and AVERAGE PLANAR EXPOSURE shall be within limits based on applicable APLHGR limit values which have been determined by approved methodology for the respective fuel and lattice types. When hand calculations are required, the APLHGR for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) as shown in the CORE OPERATING LIMITS REPORT (COLR). During operation, the APLHGR for each fuel type shall not exceed the above values multiplied by the appropriate reduction factors for power and flow as defined in the COLR.

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

ACTION:

With an APLHGR exceeding the limiting value, 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 limiting value:

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.

Section 3/4.2.2 (DELETED)

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3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the rated MCPR limit adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT, provided that the end-of-cycle recirculation pump trip (EOC-RPT) system is OPERABLE per Specification 3.3.4.2 and the main turbine bypass system is OPERABLE per Specification 3.7.8, with:

$$\tau = \frac{(\tau_{ave} - \tau_B)}{\tau_A - \tau_B}$$

where:

τ_A = 0.86 seconds, control rod average scram insertion time limit to notch 39 per Specification 3.1.3.3,

$$\tau_B = 0.672 + 1.65 \left(\frac{N_1}{\sum_{i=1}^{n} N_i} \right)^{1/2} (0.016),$$

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i},$$

n = number of surveillance tests performed to date in cycle,

N_i = number of active control rods measured in the ith surveillance test,

τ_i = average scram time to notch 39 of all rods measured in the ith surveillance test, and

N_1 = total number of active rods measured in Specification 4.1.3.2.a.

APPLICABILITY:

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

LIMITING CONDITION FOR OPERATION (Continued)

ACTION

a. With the end-of-cycle recirculation pump trip system inoperable per Specification 3.3.4.2, operation may continue provided that, within 1 hour, MCPR is determined to be greater than or equal to the rated MCPR limit as a function of the average scram time (shown in the CORE OPERATING LIMITS REPORT) EOC-RPT inoperable curve, adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT.

b. With MCPR less than the applicable MCPR limit adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

c. With the main turbine bypass system inoperable per Specification 3.7.8, operation may continue provided that, within 1 hour, MCPR is determined to be greater than or equal to the rated MCPR limit as a function of the average scram time (shown in the CORE OPERATING LIMITS REPORT) main turbine bypass valve inoperable curve, adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT.

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR, with:

a. τ = 1.0 prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.1.3.2, or

b. τ as defined in Specification 3.2.3 used to determine the limit within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2,

shall be determined to be equal to or greater than the applicable MCPR limit including application of the MCPR(P) and MCPR(F) factors as determined from the CORE OPERATING LIMITS REPORT.

a. At least once per 24 hours,

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

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

d. The provisions of Specification 4.0.4 are not applicable.

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
9. Turbine Stop Valve - Closure	N.A.	Q	R	1
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	N.A.	Q	R	1
11. Reactor Mode Switch Shutdown Position	N.A.	R	N.A.	1, 2, 3, 4, 5
12. Manual Scram	N.A.	W	N.A.	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for a least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.
(c) Within 24 hours prior to startup, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.
(g) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing loop flow (APRM % flow). During the startup test program, data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon the conclusion of the startup test program.
(h) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.
(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
(j) If the RPS shorting links are required to be removed per Specification 3.9.2, they may be reinstalled for up to 2 hours for required surveillance. During this time, CORE ALTERATIONS shall be suspended, and no control rod shall be moved from its existing position.
(k) Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3.

TABLE 3.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

ACTION STATEMENTS

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

ACTION 61 - With the number of OPERABLE Channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 12 hours or place the inoperable channel in the tripped condition.

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

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within 12 hours.

ACTION 63 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, initiate a rod block.

NOTES

* For OPERATIONAL CONDITION of Specification 3.1.4.3.

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

*** These channels are not required when sixteen or fewer fuel assemblies, adjacent to the SRMs, are in the core.

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

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

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

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

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

(f) Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale[a]		
1) Low Trip Setpoint (LTSP)	*	*
2) Intermediate Trip Setpoint (ITSP)	*	*
3) High Trip Setpoint (HTSP)	*	*
b. Inoperative	N/A	N/A
c. Downscale (DTSP)	*	*
d. Power Range Setpoint[b]		
1) Low Power Setpoint (LPSP)	23% RATED THERMAL POWER	26% RATED THERMAL POWER
2) Intermediate Power Setpoint (IPSP)	58% RATED THERMAL POWER	61% RATED THERMAL POWER
3) High Power Setpoint (HPSP)	78% RATED THERMAL POWER	81% RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux - Upscale		
1) During two recirculation loop operation	≤ 0.66 W + 59%	≤ 0.66 W + 63%
2) During single recirculation loop operation	≤ 0.66 W + 54%	≤ 0.66 W + 58%
b. Inoperative	N.A.	N.A.
c. Downscale	≥ 4% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 1×10^5 cps	≤ 1.6×10^5 cps
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 3 cps**	≥ 1.8 cps**

TABLE 3.3.6-2 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 108/125 divisions of full scale	≤ 110/125 divisions of full scale
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High a. Float Switch	≤ 257' 7 3/8" elevation***	≤ 257' 9 3/8" elevation
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	*	*
b. Inoperative	N.A.	N.A.
c. Comparator	≤ 10% flow deviation	≤ 11% flow deviation
7. REACTOR MODE SWITCH SHUTDOWN POSITION	N.A.	N.A.

* Refer to the COLR for these setpoints.

** May be reduced, provided the source range monitor has an observed count rate and signal-to-noise ratio on or above the curve shown in Figure 3.3.6-1.

*** Equivalent to 13.56 gallons/scram discharge volume.

(a) There are three upscale trip levels. Each is applicable only over its specified operating core thermal power range. All RBM trips are automatically bypassed below the low power setpoint (LPSP). The upscale LTSP is applied between the low power setpoint (LPSP) and the intermediate power setpoint (IPSP). The upscale ITSP is applied between the intermediate power setpoint and the high power setpoint (HPSP). The HTSP is applied above the high power setpoint.

(b) Power range setpoints control enforcement of appropriate upscale trips over the proper core thermal power ranges. The power signal to the RBM is provided by the APRM.

TABLE 4.3.6-1
CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	N.A.	S/U[b][c],Q[c]	R	1*
b. Inoperative	N.A.	S/U[b][c],Q[c]	N.A.	1*
c. Downscale	N.A.	S/U[b][c],Q[c]	R	1*
2. APRM				
a. Flow Biased Neutron Flux-Upscale	N.A.	S/U[b],Q	SA	1
b. Inoperative	N.A.	S/U[b],Q	N.A.	1, 2, 5***
c. Downscale	N.A.	S/U[b],Q	SA	1
d. Neutron Flux - Upscale, Startup	N.A.	S/U[b],Q	SA	2, 5***
3. SOURCE RANGE MONITORS				
a. Detector not full in	N.A.	S/U[b],W	N.A.	2, 5
b. Upscale	N.A.	S/U[b],W	SA	2, 5
c. Inoperative	N.A.	S/U[b],W	N.A.	2, 5
d. Downscale	N.A.	S/U[b],W	SA	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	N.A.	S/U[b],W	N.A.	2, 5
b. Upscale	N.A.	S/U[b],W	SA	2, 5
c. Inoperative	N.A.	S/U[b],W	N.A.	2, 5
d. Downscale	N.A.	S/U[b],W	SA	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level - High	N.A.	Q	R	1, 2, 5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	N.A.	S/U[b],Q	SA	1
b. Inoperative	N.A.	S/U[b],Q	N.A.	1
c. Comparator	N.A.	S/U[b],Q	SA	1
7. REACTOR MODE SWITCH SHUTDOWN POSITION	N.A.	R	N.A.	3, 4

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) Within 24 hours prior to startup, if not performed within the previous 7 days.

(c) Includes reactor manual control multiplexing system input.

* For OPERATIONAL CONDITION of Specification 3.1.4.3.

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

*** Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation with:

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

b. THERMAL POWER within the unrestricted zone of Figure 3.4.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

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

1. Within 4 hours:

a. Place the recirculation flow control system in the Local Manual mode, and

b. Reduce THERMAL POWER to ≤ 70% of RATED THERMAL POWER, and,

c. Limit the speed of the operating recirculation pump to less than or equal to 90% of rated pump speed, and

d. Verify that the differential temperature requirements of Surveillance Requirement 4.4.1.1.5 are met if THERMAL POWER is ≤ 30% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is ≤ 50% of rated loop flow, or suspend the THERMAL POWER or recirculation loop flow increase.

*See Special Test Exception 3.10.4.

ACTION: (Continued)

2. Within 6 hours:

   Reduce the Average Power Range Monitor (APRM) Scram and Rod Block Trip Setpoints and Allowable Values, to those applicable for single recirculation loop operation per Specifications 2.2.1 and 3.3.6, or declare the associated channel(s) inoperable and take the actions required by the referenced specifications, and,

3. The provisions of Specification 3.0.4 are not applicable.

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

b. With no reactor coolant system recirculation loops in operation, immediately initiate action to reduce THERMAL POWER such that it is not within the restricted zone of Figure 3.4.1.1-1 within 2 hours, and initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.

c. With one or two reactor coolant system recirculation loops in operation and total core flow less than 45% but greater than 39% of rated core flow and THERMAL POWER within the restricted zone of Figure 3.4.1.1-1:

   1. Determine the APRM and LPRM** noise levels (Surveillance 4.4.1.1.3):

      a. At least once per 8 hours, and

      b. Within 30 minutes after the completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER.

   2. With the APRM or LPRM** neutron flux noise levels greater than three times their established baseline noise levels, within 15 minutes initiate corrective action to restore the noise levels within the required limits within 2 hours by increasing core flow or by reducing THERMAL POWER.

d. With one or two reactor coolant system recirculation loops in operation and total core flow less than or equal to 39% and THERMAL POWER within the restricted zone of Figure 3.4.1.1-1, within 15 minutes initiate corrective action to reduce THERMAL POWER to within the unrestricted zone of Figure 3.4.1.1-1 or increase core flow to greater than 39% within 4 hours.

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

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 10% 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 RWM to be OPERABLE when THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER provides adequate control.

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

The analysis of the rod drop accident is presented in Section 15.4.9 of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1, and two supplements, References 2 and 3. Additional pertinent analysis is also contained in Amendment 17 to the Reference 4 Topical Report.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density over the range of power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods. RBM OPERABILITY is required when the limiting condition described in Specification 3.1.4.3 exists.

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 the withdrawn control rods remain fixed in the rated power pattern. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core and other piping systems connected to the reactor vessel. To allow for potential leakage and improper mixing, this concentration is increased by 25%. The required concentration is achieved by having available a minimum quantity of 3,160 gallons of sodium pentaborate solution containng a minimum of 3,754 lbs of sodium pentaborate having the requisite B 10 atom % enrichment of 29% as determined from Reference 5. This quantity of solution is a net amount which is above the pump suction shutoff level setpoint thus allowing for the portion which cannot be injected. The pumping rate of 41.2 gpm provides a negative reactivity insertion rate over the permissible solution volume range, which adequately compensates for the positive reactivity effects due to elimination of steam voids, increased water density from hot to cold, reduced doppler effect in uranium, reduced neutron leakage from boiling to cold, decreased control rod worth as the moderator cools, and xenon decay. The temperature requirement ensures that the sodium pentaborate always 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.

The SLCS system consists of three separate and independent pumps and explosive valves. Two of the separate and independent pumps and explosive valves are required to meet the minimum requirements of this technical specification and, where applicable, satisfy the single failure criterion.

The SLCS must have an equivalent control capacity of 86 gpm of 13% weight sodium pentaborate in order to satisfy 10 CFR 50.62 (Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants). As part of the ARTS/MELLL program the ATWS analysis was updated to reflect the new rod line. As a result of this it was determined that the Boron 10 enrichment was required to be increased to 29% to prevent exceeding a suppression pool temperature of 190°F. This equivalency requirement is fulfilled by having a system which satisfies the equation given in 4.1.5.b.2.

The upper limit concentration of 13.8% has been established as a reasonable limit to prevent precipitation of sodium pentaborate in the event of a loss of tank heating, which allow the solution to cool.

STANDBY LIQUID CONTROL SYSTEM (Continued)

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

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

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1. C. J. Paone, R. C. Stirn and J. A. Woolley, "Rod Drop Accident Analysis for Large BWR's," G. E. Topical Report NEDO-10527, March 1972.

2. C. J. Paone, R. C. Stirn, and R. M. Young, Supplement 1 to NEDO-10527, July 1972.

3. J. M. Haun, C. J. Paone, and R. C. Stirn, Addendum 2, "Exposed Cores," Supplement 2 to NEDO-10527, January 1973.

4. Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel".

5. "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Limerick Generating Station Units 1 and 2," NEDC-32193P, Revision 2, October 1993.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature (PCT) following the postulated design basis Loss-of-Coolant Accident (LOCA) will not exceed the limits specified in 10 CFR 50.46 and that the fuel design analysis limits specified in NEDE-24011-P-A (Reference 2) will not be exceeded.

Mechanical Design Analysis: NRC approved methods (specified in Reference 2) are used to demonstrate that all fuel rods in a lattice operating at the bounding power history, meet the fuel design limits specified in Reference 2. No single fuel rod follows, or is capable of following, this bounding power history. This bounding power history is used as the basis for the fuel design analysis MAPLHGR limit.

LOCA Analysis: A LOCA analysis is performed in accordance with 10CFR50 Appendix K to demonstrate that the permissible planar power (MAPLHGR) limits comply with the ECCS limits specified in 10 CFR 50.46. The analysis is performed for the most limiting break size, break location, and single failure combination for the plant, using the evaluation model described in Reference 9.

The MAPLHGR limit as showm in the CORE OPERATING LIMITS REPORT is the most limiting composite of the fuel mechanical design anaylsis MAPLHGR and the ECCS MAPLHGR limit.

Only the most limiting MAPLHGR values are shown in the CORE OPERATING LIMITS REPORT for multiple lattice fuel. Compliance with the specific lattice MAPLHGR operating limits, which are available in Reference 3, is ensured by use of the process computer.

As a result of no longer utilizing an APRM trip setdown requirement, additional constraints are placed on the MAPLHGR limits to assure adherence to the fuel-mechanical design bases. These constraints are introduced through the MAPFAC(P) and MAPFAC(F) factors as defined in the COLR.

3/4.2.2 (DELETED)

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3/4.2.3 MINIMUM CRITICAL POWER RATIO

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

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-2 that are input to a GE-core dynamic behavior transient computer program. The codes used to evaluate transients are discussed in Reference 2.

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state (MCPR(F), and MCPR(P), respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Ref. 6). Flow dependent MCPR limits (MCPR(F)) are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 7) to analyze slow flow runout transients. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

Power dependent MCPR limits (MCPR(P)) are determined mainly by the one dimensional transient code (Ref. 8). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scrams are bypassed, high and low flow MCPR(P), operating limits are provided for operating between 25% RTP and 30% RTP.

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the MCPR(F), and MCPR(P) limits.

MINIMUM CRITICAL POWER RATIO (Continued)

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

3/4.2.4 LINEAR HEAT GENERATION RATE

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

Reference:

1. Deleted.

2. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (latest approved revision).

3. "Basis of MAPLHGR Technical Specifications for Limerick Unit 2," NEDC-31930P (as amended).

4. Deleted

5. Increased Core Flow and Partial Feedwater Heating Analysis for Limerick Generating Station Unit 2 Cycle 1, NEDC-31578P, March 1989 including Errata and Addenda Sheet No. 1 dated May 31, 1989.

6. NEDC-32193P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Limerick Generating Station Units 1 and 2," Revision 2, October 1993.

7. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.

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

9. NEDC-32170P, "Limerick Generating Station Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," June 1993.

3/4.4.1 RECIRCULATION SYSTEM

The impact of single recirculation loop operation upon plant safety is assessed and shows that single-loop operation is permitted if the MCPR fuel cladding safety limit is increased as noted by Specification 2.1.2, APRM scram and control rod block setpoints are adjusted as noted in Tables 2.2.1-1 and 3.3.6-2, respectively.

Additionally, surveillance on the pump speed of the operating recirculation loop is imposed to exclude the possibility of excessive internals vibration. The surveillance on differential temperatures below 30% RATED THERMAL POWER or 50% rated recirculation loop flow is to mitigate the undue thermal stress on vessel nozzles, recirculation pump and vessel bottom head during the extended operation of the single recirculation loop mode.

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 recirculation 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 two loop operation, continued operation is permitted in a single recirculation 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. Sudden equalization of a temperature difference > 145°F between the reactor vessel bottom head coolant and the coolant in the upper region of the reactor vessel by increasing core flow rate would cause undue stress in the reactor vessel bottom head.

The objective of GE BWR plant and fuel design is to provide stable operation with margin over the normal operating domain. However, at the high power/low flow corner of the operating domain, a small probability of limit cycle neutron flux oscillations exists depending on combinations of operating conditions (e.g., rod pattern, power shape). To provide assurance that neutron flux limit cycle oscillations are detected and suppressed, APRM and LPRM neutron flux noise levels should be monitored while operating in this region.

Stability tests at operating BWRs were reviewed to determine a generic region of the power/flow map in which surveillance of neutron flux noise levels should be performed. A conservative decay ratio of 0.6 was chosen as the bases for determining the generic region for surveillance to account for the plant to plant variability of decay ratio with core and fuel designs. This generic region has been determined to correspond to a core flow of less than or equal to 45% of rated core flow and a THERMAL POWER greater than that specified in Figure 3.4.1.1-1.

3/4.6.2 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of 55 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from rated conditions. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss-of-coolant accident, the pressure of the suppression chamber air space must not exceed 55 psig. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in this specification, suppression pool pressure during the design basis accident is below the design pressure. Maximum water volume of 134,600 ft^3 results in a downcomer submergence of 12'3" and the minimum volume of 122,120 ft^3 results in a submergence approximately 2'3" less. The majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate. The maximum temperature at the end of the blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperature above 170°F.

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operating conditions, blowdown through safety/relief valves assuming an initial suppression chamber water temperature of 95°F results in a bulk water temperature of approximately 136°F immediately following blowdown which is below the 190°F bulk temperature limit used for complete condensation via T-quencher devices. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependency on containment overpressure for post-LOCA operations.

CORE OPERATING LIMITS REPORT

6.9.1.9 Core Operating Limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the CORE OPERATING LIMITS REPORT for the following:

a. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1,

b. MAPFAC(P) and MAPFAC(F) factors for Specification 3.2.1,

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

d. The MCPR(P) and MCPR(F) adjustment factor for specification 3.2.3,

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

f. The power biased Rod Block Monitor setpoints and the Rod Block Monitor MCPR OPERABILITY limits of Specification 3.3.6.

g. The Reactor Coolant System Recirculation Flow upscale trip setpoint and allowable value for Specification 3.3.6,

h. The Recirculation MG set mechanical and electrical overspeed stop setpoints for Specification 4.4.1.1.2.

6.9.1.10 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

a. NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel" (Latest approved revision).

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

6.9.1.12 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.

SPECIAL REPORTS

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED AMENDMENT NO. 48 TO FACILITY OPERATING LICENSE NO. NPF-85

PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION, UNIT 2

DOCKET NO. 50-353

1.0 INTRODUCTION

By letter dated August 27, 1993 (Reference 1), and supplemented by letter dated November 17, 1993 (Reference 2), the Philadelphia Electric Company (PECO or the licensee) submitted a request for changes to the Limerick Generating Station (LGS), Units 1 and 2 Technical Specifications (TS). The requested changes implement an expanded power-to-flow operating domain supported by the Average Power Range Monitor - Rod Block Monitor Technical Specifications/ Maximum Extended Load Line Limit Analyses (ARTS/MELLLA) program. The November 17, 1993, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The request proposed four fundamental changes: (1) Deletion of the flow-biased APRM scram and rod block trip setpoint setdown requirements; (2) Modification of the flow-biased APRM scram and rod block trip equations to expand the power-to-flow operating domain; (3) Replacement of the flow-biased Rod Block Monitor (RBM) trip setpoints with power-dependent trips; and (4) Revision of the Standby Liquid Control System (SLCS) Boron-10 enrichment percentage to accommodate operation in the MELLL region. Additionally, a proposal is made to reset the recirculation pump runback intermediate speed to accommodate a feedwater pump trip while operating in the MELLL region.

The first change, eliminating the APRM scram and rod block trip setpoint setdown, is a result of ARTS updates to the thermal limits requirements. These updates are made so that the safety limit Minimum Critical Power Ratio (MCPR) and fuel thermal-mechanical design bases are not violated during postulated transient events initiated from other than rated power or flow conditions. These updates include:

a. Elimination of the use of and reference to the K_f MCPR flow multiplication factor.

b. Introduction of power and flow dependent adjustment factors for the Maximum Average Planar Linear Generation Heat Rate (MAPLHGR) and MCPR limits.

c. Revision of the Core Operating Limits Report (COLR) documentation requirements to include parameters used to determine the power and flow dependent MCPR and MAPLHGR limits for each cycle.

d. Removal of the Fraction of Rated Power (FRP) and the Maximum Fraction of Limiting Power Density (MFLPD) definitions and requirements since they are used only in the determination of the required setdown of the APRM and rod block setpoints.

The APRM and RBM equations, setpoints, operability requirements, and hardware are modified to implement the thermal limit changes of the ARTS/MELLLA program. The RBM trip setpoint changes include alterations to the RBM input and trip logic. The SLCS Boron-10 enrichment is increased in order to maintain the suppression pool temperatures below the design limit of 190 °F during an ATWS while operating in the MELLL region. The recirculation pump runback intermediate speed setting is reduced to bring the power sufficiently low to be within the normal capacity of the feedwater system in the event of a feedwater pump trip.

In support of the requested changes, the licensee has submitted the proposed TS changes, a brief explanation of the changes, and a General Electric (GE) topical report (Reference 3) describing in detail the ARTS/MELLLA program for LGS.

Also as a part of the submittal, the licensee adopted the SAFER/GESTR-LOCA methodology for the analysis of the design basis Loss of Coolant Accident (LOCA). The safety analyses prepared by General Electric to support the change to this LOCA evaluation model was presented in a GE topical report included in the original submittal (Reference 4). In anticipation of implementing the ARTS/MELLLA at LGS, Units 1 and 2, the SAFER/GESTR model was also used to calculate the fuel rod peak cladding temperature during a LOCA with the ARTS/MELLLA improvements.

2.0 EVALUATION

The proposed ARTS/MELLL changes for LGS are common for GE Boiling Water Reactors (BWRs). They have become part of standard operating flexibility options as described in the GE standard application for reactor fuel (Reference 5). These options have been approved for several BWRs, including ARTS/MELLL upgrades on plants such as Hatch and Monticello in 1984, as well as Fermi 2 and Pilgrim in 1991. The methodologies used for the safety analyses justifying the changes and establishment of new operating limits have been previously reviewed and approved by the staff. The proposed new operating region and the APRM and RBM changes are similar to equivalent changes approved for other reactors.

Since the submittal for LGS included changes which have become standard and have been well considered for other plants, only a brief description of them is included here. More detailed information is available in the first such reviews performed for Hatch and Monticello. Aspects of changes or analyses

specific to LGS are discussed in more depth, although all of the analyses were examined for this review. The changes to the SLCS Boron-10 enrichment and recirculation pump runback setting are required as a result of the findings of the MELLL analyses during particular Anticipated Transients Without Scram (ATWS).

2.1 ARTS/MELLL Analyses

Program Description

The MELLL mode of operation extends the current operating envelope to the region bounded by the rod line that passes through the 100% power/75% core flow point(i.e. approximately the 121% rod line), the rated power line, and the 100% rated load line. This region allows for more flexibility with power ascensions and allows other fuel cycle efficiency strategies to be utilized. In addition, the ARTS program is developed to increase plant efficiency while in the MELLL region by updating the thermal limit requirements and improving plant instrumentation responses and accuracies.

The changes associated with the MELLL mode of operation and the ARTS program include the following:

a. A power dependent MCPR thermal limit similar to that used by BWR type 6 plants is implemented to complement the new power biased RBM system.

b. The APRM trip setdown requirement is replaced by power and flow dependent thermal limits to reduce the need for manual setpoint adjustments and to allow more direct thermal limits administration. These limits are specified through the use of MAPLHGR and MCPR adjustment factors: MAPFAC(P), MAPFAC(F), MCPR(P), and MCPR(F).

c. The flow biased RBM trips are replaced with power dependent trips. As part of the RBM modification, the Local Power Range Monitor (LPRM) inputs will be reassigned to improve the response characteristics of the system, to improve the response predictability, and to reduce the frequency of nonessential alarms. In addition, the RBM electronics are modified to produce a trip signal as a function of the percentage increase from an initial reference signal.

d. The Rod Withdrawal Error (RWE) analysis is updated utilizing statistical methods that more accurately reflect actual plant operating conditions and is consistent with the system changes.

e. The Standby Liquid Control System (SLCS) Boron-10 enrichment percentage is revised to accommodate operation in the MELLL region as a result of ATWS studies.

Results of analyses that justify the above changes and which determine instrument setpoints and operating limits consistent with their implementation are included with the submittal. These analyses include fuel performance

transient analyses, mechanical evaluations of the reactor internals and structural vibrations, LOCA analyses, containment load evaluations, and rod withdrawal error analyses. The thermal limits developed through ARTS/MELLLA are specified for fuel protection during Anticipated Operational Occurrences (AOOs) and portions are intended to be applicable to future fuel cycles utilizing GE fuel designs up to GE11 and also for Asea Brown Boveri (ABB) and Siemens Nuclear Power (SNP) qualification bundles. The specific operating limits and validation of the multipliers are to be updated for each reload and reported in the COLR. Changes in fuel designs, analytical methods, or plant configurations may require confirmatory verification. Plant-specific portions of the generic ARTS limits for LGS were developed based on the LGS Unit 1 Cycle 5 configuration. Similarity of plant configuration and fuel types also allow these ARTS plant-specific limits to be applicable to LGS Unit 2.

MELLL Analyses

LGS is currently licensed to operate in the Extended Load Line Limit (ELLL) and Increased Core Flow (ICF) regions, above the rated rod line along the 108% APRM rod block line, up to the 100% power, between 87% and 105% core flow and/or Partial Feedwater Heating (PFH). The MELLL analysis further expands the operating domain along the 121% rod line to 100% power at 75% core flow. The APRM scram trip setpoints will insert clamp values for core flows greater than 75% rated core flow.

The core wide AOOs included current LGS Unit 1 Cycle 5 reload licensing analyses and were expanded to justify operation in MELLL domain. These also included relaxed assumptions of Recirculation Pump Trip Out-of-Service (RPTOOS) and Turbine Bypass Valve Out-of-Service (TBVOOS). The power and flow dependent MCPR and MAPLHGR limits are derived from evaluations of the most limiting of these transients. The limiting occurrences studied in detail included the Turbine Trip with No Bypass (TTNBP), Feedwater Controller Failure (FWCF), Inadvertent High Pressure Coolant Injection (IHPCI), and Feedwater Heater Failure (FWHF) of 100 °F. The analysis input assumptions, such as Reactor Protection System setpoints and plant configurations, are based on LGS Unit 1 Cycle 5 information and also used the 100% power at 75% core flow operating point for reanalysis. ICF and Feedwater Temperature Reduction (FWTR) operating points were also considered as starting points for some of the analyses. These transients determined the power dependent MCPR(P) and MAPFAC(P) that bound the initial MCPR and MAPLHGR to assure that the fuel safety limits will not be violated for each transient. The flow dependent MCPR(F) and MAPFAC(F) were derived from the results of slow-flow recirculation pump run out events with the corresponding power rise. These factors are derived so that the fuel MAPLHGR will not increase above the fuel thermal mechanical design basis values and so that the MCPR values remain within the generic bounding values. The analyses show that the generic multipliers are conservative when applied to the rated MCPR and MAPLHGR operating limit for nominal assumptions and are, therefore, applicable to LGS. The COLR will include MCPR(P) curves for both conditions of operable and inoperable recirculation pump trip and turbine bypass valves.

Reactor Vessel overpressure protection was demonstrated by evaluation of the MSIV closure with neutron flux scram from the 102% power/75% flow point using the End of Cycle 5 (EOC5) target exposure shape. The results show the peak vessel pressure is below the ASME Code limit.

Even though the operating region has been expanded with MELLL, compliance with interim measure of USNRC Bulletin 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors," has been maintained through operating procedures and/or Technical Specifications. The Loss of Feedwater pump transient was evaluated to determine the recirculation pump speed setting that corresponds to a power level low enough to be within the capacity of the remaining feedwater pumps. In order to meet this requirement when operating along the MELLL rod line, the new setting is determined to be 42% speed, which is lower than the current 47%. The stability requirements are still met because this speed corresponds to approximately 54% flow and if flow were to drop below 45%, TS requirements would then ensure proper stability actions are taken.

Results of the Anticipated Transient Without Scram (ATWS) analysis conducted for operation in the MELLL domain showed maximum values of the key performance parameters (i.e., fuel cladding temperature and reactor vessel bottom pressure) were within generic limits. The suppression pool temperature did show an increase above the 190 °F limit due to operation in the MELLL domain with elevated values of 110% power and 87% core flow. As a result, to maintain the suppression pool temperature less than 190 °F beginning from this limiting operation point, the required Boron-10 enrichment in the sodium pentaborate solution is raised so that the SLCS can sooner reduce the reactivity and the reactor heat load.

Subsequent reload licensing reviews will include examination of cycle-specific data in the MELLL operating region and the reference operating limits will be reported in the COLR. Except for minor differences included for sensitivity study and future consideration, the analyses presented for LGS operation in the MELLL region conform to those previously evaluated by the staff and yield acceptable results.

ARTS Analyses

The ARTS improvement program provides changes to both the APRM and RBM systems. The removal of the APRM trip setdown requirement is justified by showing certain criteria are met while operating in the MELLL region. These criteria include ensuring the MCPR Safety Limit is not violated as a result of any AOO; fuel thermal-mechanical design bases remain within licensing limits; and PCT, maximum hydrogen generation, and maximum cladding oxidation fraction following a LOCA remain within the limits of 10 CFR 50.46. The LOCA analysis is discussed in Section 2.2 of this evaluation. With the requirements met, the setdown factor on the APRM flow biased trip is replaced with a set of power and flow dependent MAPLHGR and MCPR adjustment factors, as verified with the MELLL analysis.

The RBM changes take advantage of the new MAPLHGR and MCPR limits and advances in electronic circuitry. The MELLL analyses were used to justify these changes. In addition, new RWE analyses were conducted to establish the CPR limit and trip setpoints for each power level. There are three power level ranges (low, intermediate, and high) and each range has a corresponding RBM trip setpoint. The RBM was statistically evaluated with the reconfigured LPRM inputs and APRM initial reference signal with consideration given to low MCPR and high MAPLHGR conditions. The many RWEs required for the statistical approach were generated by randomly varying the initial position of the error rod and varying the location and number of failed LPRMs. This analysis was shown to be valid for all GE fuel types including GE11 and is also applicable to Asea Brown Boveri (ABB) and Siemens Nuclear Power (SNP) qualification bundles.

Specific LGS analyses were performed to confirm the applicability of generic power and flow dependent MCPR and MAPLHGR limits (in terms of multiplication factors on plant rated operating limits) taken from the ARTS data base. The plant limits were selected to remain valid through future reloads using up to GE11 fuel and currently approved analysis methods. The ARTS analyses used the LGS Unit 1, Cycle 5 inputs along with bounding values for core power, maximum core flow, and reduced feedwater temperature (for feedwater controller failure transient). The cycle-specific MCPR and MAPLHGR limits for rated conditions and for relaxed conditions of RPTOOS and TBVOOS and the curves for the power and flow dependent factors will be required and referenced in the COLR.

Overall, the ARTS analyses and improvements to the APRM and RBM systems parallel ARTS submittals for other BWRs which were accepted by the staff. The power setpoints and RBM setpoints requested in the TS change are within the range of generic settings presented in Reference 3 and are acceptable.

The ARTS changes to the APRM and RBM systems and the supporting analyses are similar to submittals for other BWRs which were accepted by the staff. The adoption of the SAFER/GESTR-LOCA methodology was used to support these changes and is further discussed in Section 2.2. The ARTS hardware updates proposed for LGS are the same as others evaluated by the staff. Of note however, discussion on the use of the adjustable trip time delay option t_{d2} above the minimum setting is also included in the analysis report. The option is included with the hardware, although it is acknowledged that sufficient RWE analysis was not performed to allow its use. The suggestion made by the GE report that the t_{d2} setting could be used to bypass the RBM system when permitted is counter to previous staff findings (i.e. the Hatch ARTS Instrument and Control review). Manual adjustment of the t_{d2} setpoint as a means of bypassing a RBM channel in lieu of using the existing RBM channel bypass switch (which provides automatic indication of the bypass condition) is not acceptable and is not to be permitted. Any future use of this time delay setting will require the evaluation of further analysis, as discussed in the GE report. Beside this single exclusion, the analyses and system changes associated with the ARTS updates, including the hardware modifications and proposed analytical limits, are acceptable.

2.2 SAFER/GESTR-LOCA Analyses

As part of their submittal, the licensee adopted the SAFER/GESTR-LOCA methodology for LOCA analysis. Application and validation of this approach was detailed in Reference 4 and was evaluated in conjunction with the LGS ARTS/MELLLA implementation. LOCA analysis was also performed using the SAFER/GESTR-LOCA methodology to ensure that the 10 CFR 50.46 and Appendix K LOCA criteria would be met when using the new thermal limits and the MELLL operating region. The results of this analysis were submitted with the application and were evaluated by the staff as part of the ARTS/MELLL application.

Requirements for the use of SAFER/GESTR-LOCA were established in the Topical Report Evaluation contained in Reference 6. The methodology includes the stipulation that a sufficient number of plant specific Peak Cladding Temperature (PCT) points based on both nominal input values and Appendix K values are calculated so that the shape of the PCT versus break size can be verified. The conditions for demonstrating applicability of the SAFER/GESTR analysis to a particular plant also includes confirming that plant-specific operating parameters have been bounded by the models and inputs used in the generic calculations and confirming that the plant-specific ECCS configuration is consistent with the referenced plant class ECCS configuration. The plant operating conditions and model inputs have been reviewed and found to be bounding and/or consistent with the generic analysis of Reference 7 and, therefore, the licensee meets the latter two criteria for acceptability. The applicability of the PCT values will be discussed in the next paragraphs.

The nominal PCT (PCT_{NOM}) curve is determined using best estimate values of plant response and a representative number of break sizes. The analysis included break sizes ranging from 0.05 ft^2 to the design basis accident (DBA) recirculation suction line break (4.16 ft^2). The curve generated is used to determine the limiting LOCA (highest PCT) which is then used for subsequent calculations. Another curve is generated using the Appendix K conservative assumptions and resultant PCT_{AppK} values. A Licensing Basis PCT (PCT_{LIC}) is determined from the limiting PCT_{NOM}, PCT_{AppK}, and plant uncertainty terms. The limiting PCT_{NOM} must also pass another criterion for its statistical upper bound value to be less than the PCT_{LIC}. The Upper Bound PCT (PCT_{UB}) is a function of the limiting PCT_{NOM}, modeling bias, and plant variable uncertainty. The analysis presented in the generic report uses assumptions arising from conditions based on the large break event. The requirements of the Topical Report Evaluation ensure that specific plant LOCA response does not significantly diverge from the generic LOCA response and possibly invalidate application of SAFER/GESTR-LOCA analysis.

LGS 1 and 2 are BWR-4s with Low Pressure Coolant Injection (LPCI) introduction into the bypass region of the core, therefore, LGS must be compared to the generic conformance calculation for the BWR-5/6 and some BWR-4 type plants. Results of break calculations presented in the LGS PCT vs break size plot in Figure 5-1 of Reference 4 are consistent with the curves in Figure 3-4 in Reference 7. These studies were performed with a power level of 110% to

conservatively bound the currently licensed rated power. The limiting break for the nominal and Appendix K studies was found to be the DBA recirculation suction line break coincident with battery failure. Results of a sensitivity study show that the increase in PCT by using the 110% power level is less than 35 °F. In both cases, the PCT_{LIC} are below the 10 CFR 50.46 requirement of 2200 °F and the PCT_{UB} are less than the respective PCT_{LIC}. In all cases the PCT_{UB} is below the 1600 °F limit set by the bases of the SAFER/GESTR analysis. Conformance with the other 10 CFR 50.46 criteria for maximum local oxidation and hydrogen generation is also demonstrated by the analysis in Reference 4.

PCT results were obtained for several GE fuel types up to the GE11 type. Because the accident analyses have been performed using approved methods, and the results meet the staff's acceptance criteria, we conclude that these analyses are acceptable and the results may be used to provide a new LOCA licensing basis for LGS 1 & 2. Studies considering ARTS/MELLL conditions at 110% power and 75% core flow were performed and results show that increases in PCT are small (<20 °F) when compared with rated core flow cases and in no cases do the PCT_{LIC} exceed 2200 °F. Cases run for ICF, FWTR, FWHOOS, and SLO also show little change in PCT from the rated condition cases. These extra studies also show that no low flow MAPLHGR multiplier is required for ECCS considerations while in the MELLL operating domain because there is sufficient PCT margin available with respect to the 2200 °F criteria.

Coincident with the LOCA analysis, the containment responses under the revised MELLL assumptions were determined for a double-ended guillotine break of a recirculation line. The results show the peak containment drywell pressure is bounded by the LGS Updated Final Safety Analysis Report (UFSAR) values and remain well below the design value of 55 psig. The drywell deck differential pressure is above the UFSAR value but is below the design value. The containment dynamic loads analysis included loads from pool swell, condensation oscillation, and chugging. The results show that the peak containment wetwell airspace pressure during the suppression pool swell period is calculated to be 38 psig which is above the UFSAR result but below the design basis limit of 55 psig. These analyses considered the bounding short-term containment response and appear acceptable. The results of the long-term response analysis described in the UFSAR remain applicable for MELLL operation.

In summary, the licensee demonstrated conformance to 10 CFR 50.46 and Appendix K with the submitted LOCA analyses and based on the review described above, the SAFER/GESTR methodology is found to be acceptable and results may be used to provide a new LOCA licensing basis for LGS Units 1 & 2.

3.0 TECHNICAL SPECIFICATIONS

Changes to LGS limits and operability requirements in the TS are necessary to implement ARTS/MELLL. The proposed TS changes are as follows:

a. Definitions are added to TS Section 1.0 for the Downscale Trip Setpoint (DTSP), High Trip Setpoint (HTSP), Intermediate Trip Setpoint (ITSP), Low

Trip Setpoint (LTSP), flow dependent MAPLHGR factor (MAPFAC(F)), and power dependent MAPLHGR factor (MAPFAC(P)). The definition for MCPR is revised to include the flow and power dependent MCPR limit factors (MCPR(F) and MCPR(P), respectively. The definitions for the Fraction of Limiting Power Density (FLPD), Fraction of Rated Thermal Power (FRP), and Maximum Fraction of Limiting Power Density (MFLPD) are deleted.

b. Current flow referenced RBM setpoint TS are replaced with the RBM power referenced setpoints as described below:

   i. Revise Limiting Condition for Operation (LCO) "Rod Block Monitor," TS Section 3.1.4.3, to update the operability requirements.

   ii. Revise "Control Rod Withdrawal Block Instrumentation," TS Table 3.3.6-1, to reference the RBM LCO.

   iii. Revise "Control Rod Block Instrumentation Setpoints," TS Table 3.3.6-2, to reference the COLR requirements, implement the new Power Range Setpoints, and update the flow referenced APRM rod block equations.

   iv. Revise "Control Rod Block Instrumentation Surveillance Requirements," TS Table 4.3.6-1, to reference to the RBM LCO.

   v. Revise "Control Rod Program Controls," TS Bases 3/4.1.4, to include power reference and operability requirements.

c. TS are changed as follows, to reflect the implementation of power and flow dependent fuel thermal limits in order to eliminate APRM trip setdown requirements and to support the power dependent RBM trips:

   i. Revise "APRM Setpoints," TS Table 2.2.1-1, to update the flow referenced APRM scram setpoints.

   ii. Delete the flow referenced trip setpoint discussion from "Reactor Protection System Instrumentation Setpoints for the Average Power Range Monitor," TS Bases, page B 2-7, because it is no longer required with the new setpoints.

   iii. Revise LCO "Average Planar Linear Heat Generation Rate," TS Section 3/4.2.1, to include the MAPLHGR limit adjustments defined in the COLR.

   iv. Delete LCO and Surveillance Requirements "APRM Setpoints," TS Section 3/4.2.2, because it is no longer required with the new limits.

   v. Revise LCO and Surveillance Requirements for "Minimum Critical Power Ratio," TS Section 3/4.2.3, to include the MCPR adjustment factors defined in the COLR.

vi. Revise the Bases discussion of Average Planar Linear Heat Generation Rate (APLHGR), TS Bases 3/4.2.1, to discuss the implementation of the new flow and power dependent MAPLHGR limits.

vii. Delete the Bases discussion of APRM Setpoints, TS Bases 3/4.2.2, because it is no longer required with the new limits and setpoints.

viii. Revise the Bases discussion of Minimum Critical Power Ratio (MCPR), TS Bases 3/4.2.3, to discuss the implementation of the new operating limit MCPR's dependent on core flow and power.

ix. Revise "Reactor Protection System Instrumentation Surveillance Requirements," TS Table 4.3.1.1-1, to eliminate the APRM setdown.

x. Revise Administrative Controls "Core Operating Limits Report," TS Section 6.9.1.9, to include the new flow and power dependent fuel thermal limits.

d. Revise the Reactor Recirculation System TS to incorporate Single Loop Operation (SLO) requirements as follows:

i. Delete LCO ACTION a.1.c in TS Section 3.4.1.1 and reletter subsequent sections, to remove the APRM setdown.

ii. Revise LCO ACTION a.2 in TS Section 3.4.1.1, to reference the new APRM scram and rod block setpoint equations for SLO.

iii. Revise TS Bases 3/4 4.1 to reference the new APRM scram and rod block setpoints for SLO.

e. Revise and add to the Standby Liquid Control System Surveillance Requirements and Bases (TS Sections 4.1.5 and B3/4 1.5 respectively) to incorporate the new requirements for sodium pentaborate volume and Boron-10 enrichment.

f. Revise the Depressurization System TS Bases 3/4 6.2 to account for the updated containment pressure response of the ARTS/MELLL analysis.

g. Revise the References cited in TS Bases 3/4.2.4 to include various topical reports related to the ARTS/MELLL and SAFER/GESTR-LOCA analyses.

Based upon the acceptance of the methods and results of the ARTS/MELLLA for LGS as discussed in Section 2 of this evaluation, these TS changes are acceptable.

4.0 SUMMARY

The Philadelphia Electric Company (PECO) requested changes to the Limerick Generating Station (LGS), Units 1 and 2 Technical Specifications. The changes implement an expanded power-to-flow operating domain supported by the

ARTS/MELLLA program. The application included the adoption of the SAFER/GESTR-LOCA methods as the LOCA licensing basis for LGS Units 1 & 2. The analyses presented examined the same areas as previous ARTS/MELLLA submittals reviewed by the staff. The methods used have been previously approved and the results of this study fall within accepted limits. The instrumentation modifications, operating limits, and setpoints proposed are acceptable. The staff review concludes that the results presented in the report contained in Reference 3 justify the proposed ARTS/MELLLA changes to LGS, Units 1 and 2. The SLCS Boron-10 enrichment and recirculation pump runback speed setting changes determined as an outcome of the analyses have been implemented as recommended in Reference 3 and appear acceptable. Further, the SAFER/GESTR-LOCA analysis has been reviewed, and based on submitted material and previously approved GE analytical techniques and design data, it is deemed acceptable.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes 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 amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (58 FR 52992). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

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

Principal Contributors: J. Witter
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Date: January 27, 1995

6.0 REFERENCES

1. Letter from G. A. Hunger, Jr. (PECo) to NRC dated August 27, 1993, Limerick Generating Station, Units 1 and 2, Technical Specification Change Request 92-08-0.

2. Letter from G. A. Hunger, Jr. (PECo) to NRC dated November 17, 1993, Supplemental to Limerick Generating Station, Units 1 and 2, Technical Specification Change Request 92-08-0 Additional Information.

3. NEDC-32193P, Revision 2, "Maximum Extended Load Line Limit and ARTS Improvement Program Analysis for Limerick Generating Station Units 1 and 2," October 1993, (General Electric proprietary information).

4. NEDC-32170P, Revision 1, "Limerick Generating Station Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," June 1993, (General Electric proprietary information).

5. NEDE-24011-P-A-10-US, "General Electric Standard Application for reactor Fuel," April 1991, (General Electric proprietary information).

6. Letter from C. O. Thomas (NRC) to J. F. Quirk (GE) dated June 1, 1984, Accepting GE Topical Report NEDE-23785 Rev. 1, Vol. III(P), "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident."

7. NEDE-23785-1-PA, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident," Volume III, Revision 1, October 1984, (General Electric proprietary information).