Docket File

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

June 13, 1995

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 OPUS Place, Suite 500
Downers Grove, IL 60515

SUBJECT: ISSUANCE OF AMENDMENTS RELATED TO TSUP SECTIONS 2.0, 3/4.11, AND 3/4.12 (TAC NOS. M86774, M86775, M86772 AND M86773)

Dear Mr. Farrar:

The Commission has issued the enclosed Amendment No. 134 to Facility Operating License No. DPR-19 and Amendment No. 128 to Facility Operating License No. DPR-25 for the Dresden Nuclear Power Station, Units 2 and 3, respectively; and Amendment No. 155 to Facility Operating License No. DPR-29 and Amendment No. 151 to Facility Operating License No. DPR-30 for the Quad Cities Nuclear Power Station, Units 1 and 2, respectively. The amendments are in response to your application dated September 15, 1992, as supplemented by letter dated April 21, 1995.

As a result of findings by a Diagnostic Evaluation Team inspection performed by the NRC staff at the Dresden Nuclear Power Station in 1987, Commonwealth Edison Company (ComEd, the licensee) made a decision that both the Dresden Nuclear Power Station and sister site Quad Cities Nuclear Power Station, needed attention focused on the existing custom Technical Specifications (TS) being used at both sites.

The licensee made the decision to initiate a Technical Specification Upgrade Program (TSUP) for both Dresden and Quad Cities. The licensee evaluated the current TS for both Dresden and Quad Cities against the Standard Technical Specifications (STS) contained in NUREG-0123, "Standard Technical Specification General Electric Plants BWR/4." The licensee's evaluation identified numerous potential improvements such as clarifying requirements, changing the TS to make them more understandable and to eliminate interpretation, and deleting requirements that are no longer considered current with industry practice. As a result of the evaluation, ComEd has elected to upgrade both the Dresden and Quad Cities TS to the STS contained in NUREG-0123.

The TSUP for Dresden and Quad Cities is not a complete adoption of the STS. The TSUP focuses on (1) integrating additional information such as equipment operability requirements during shutdown conditions, (2) clarifying requirements such as limiting conditions for operations and action statements utilizing STS terminology, (3) deleting superseded requirements and modifications to the TS based on the licensee's responses to Generic Letters (GL), and (4) relocating specific items to more appropriate TS locations.

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The application dated September 15, 1992, as supplemented April 21, 1995, contains the proposed upgrade of Sections 2.0 (Safety Limits and Limiting Safety System Settings), 3/4.11 (Power Distribution Limits), and 3/4.12 (Special Test Exceptions) of the Dresden and Quad Cities TS.

The review guidance to be used by the NRC staff in the review of the TSUP is described in Section 2.0 of the enclosed Safety Evaluation (SE). The staff reviewed the proposed changes and evaluated all deviations and changes between the proposed TS, the STS, and the current TS.

Based on discussions between ComEd and the staff, it has been mutually agreed upon that the NRC will review the sections of TSUP as they are submitted and provide ComEd an amendment for each submittal. Once all of the TSUP sections have been reviewed and the amendments issued, it is our understanding that ComEd will make one final submittal addressing any changes that may be required as a result of problems uncovered during the course of this effort. Upon receipt and review of this final submittal, the staff will issue a final amendment which addresses any remaining open items and any changes or corrections to the previous amendments. The applicable TSUP TS will be issued with each amendment and will become effective no later than December 31, 1995, for Dresden and June 30, 1996, for Quad Cities.

The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by:

John F. Stang, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-237, 50-249, 50-254, 50-265

Enclosures:
1. Amendment No. 134 to DPR-19
2. Amendment No. 128 to DPR-25
3. Amendment No. 155 to DPR-29
4. Amendment No. 151 to DPR-30
5. Safety Evaluation

cc w/encls: see next page

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D. L. Farrar
Commonwealth Edison Company

Dresden Nuclear Power Station
Unit Nos. 2 and 3
Quad Cities Nuclear Power Station
Unit Nos. 1 and 2

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 134
License No. DPR-19

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated September 15, 1992, as supplemented by letter dated April 21, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-19 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 134 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented no later than December 31, 1995.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stang, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 13, 1995

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 128
License No. DPR-25

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated September 15, 1992, as supplemented by letter dated April 21, 1995, 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 3.B. of Facility Operating License No. DPR-25 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 128 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented no later than December 31, 1995.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stang

John F. Stang, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 13, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 134 and 128

FACILITY OPERATING LICENSE NOS. DPR-19 AND DPR-25

DOCKET NOS. 50-237 AND 50-249

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number.

UNIT 2 REMOVE	UNIT 3 REMOVE	INSERT
1/2.1-1	1/2.1-1	2-1
1/2.1-2	1/2.1-2	2-2
1/2.1-3	1/2.1-3	2-3
1/2.1-4	1/2.1-4	2-4
1/2.1-5	1/2.1-5	2-5
B 1/2.1-6	B 1/2.1-6	B 2-1
B 1/2.1-7	B 1/2.1-7	B 2-2
B 1/2.1-8	B 1/2.1-8	B 2-3
B 1/2.1-9	B 1/2.1-9	B 2-4
B 1/2.1-10	B 1/2.1-10	B 2-5
B 1/2.1-11	B 1/2.1-11	B 2-6
B 1/2.1-12	B 1/2.1-12	B 2-7
B 1/2.1-13	B 1/2.1-13	B 2-8
B 1/2.1-14	B 1/2.1-14	B 2-9
B 1/2.1-15	B 1/2.1-15	B 2-10
B 1/2.1-16	B 1/2.1-16	B 2-11
B 1/2.1-17	B 1/2.1-17	---
1/2.2-1	1/2.2-1	---
B 1/2.2-2	B 1/2.2-2	---
B 1/2.2-3	B 1/2.2-3	---
B 1/2.2-4	B 1/2.2-4	---
---	---	3/4.11-1
---	---	3/4.11-2
---	---	3/4.11-3
---	---	3/4.11-4
---	---	3/4.11-5
---	---	B 3/4.11-1
---	---	B 3/4.11-2
---	---	B 3/4.11-3
---	---	3/4.12-1
---	---	3/4.12-2
---	---	B 3/4.12-1

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.A THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.4.

THERMAL POWER, High Pressure and High Flow

2.1.B The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.08 with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow. During single recirculation loop operation, this MCPR limit shall be increased by 0.01.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With MCPR less than the above applicable limit and the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.4.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

Reactor Coolant System Pressure

2.1.C The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1345 psig.

APPLICABILITY: OPERATIONAL MODE(s) 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1345 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1345 psig within 2 hours and comply with the requirements of Specification 6.4.

Reactor Vessel Water Level

2.1.D The reactor vessel water level shall be greater than or equal to twelve inches above the top of the active irradiated fuel.

APPLICABILITY: OPERATIONAL MODE(s) 3, 4 and 5.

ACTION:

With the reactor vessel water level at or below twelve inches above the top of the active irradiated fuel, manually initiate the ECCS to restore the water level, after depressurizing the reactor vessel, if required, and comply with the requirements of Specification 6.4.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.2 LIMITING SAFETY SYSTEM SETTINGS

Reactor Protection System (RPS) Instrumentation Setpoints

2.2.A The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.A-1.

APPLICABILITY: As shown in Table 3.1.A-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Trip Setpoint column of Table 2.2.A-1, declare the CHANNEL inoperable and apply the applicable ACTION statement requirement of Specification 3.1.A until the CHANNEL is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

Functional Unit	Trip Setpoint
1. Intermediate Range Monitor:	
a. Neutron Flux - High	≤120/125 divisions of full scale
b. Inoperative	NA
2. Average Power Range Monitor:	
a. Setdown Neutron Flux - High	≤15% of RATED THERMAL POWER
b. Flow Biased Neutron Flux - High	
1) Dual Recirculation Loop Operation	
a) Flow Biased	≤0.58W[a] + 62%, with a maximum of
b) High Flow Maximum	≤120% of RATED THERMAL POWER
2) Single Recirculation Loop Operation	
a) Flow Biased	≤0.58W[a] + 58.5%, with a maximum of
b) High Flow Maximum	≤116.5% of RATED THERMAL POWER
c. Fixed Neutron Flux - High	≤120% of RATED THERMAL POWER
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	≤1060 psig
4. Reactor Vessel Water Level - Low	≥144 inches above top of active fuel
5. Main Steam Line Isolation Valve - Closure	≤10% closed
6. Main Steam Line Radiation - High	≤3[b] x normal full power background (without hydrogen addition)

a W shall be the recirculation loop flow expressed as a percentage of the recirculation loop flow which produces a rated core flow of 98 million lbs/hr.

b With Unit 2 operating above 20% RATED THERMAL POWER and hydrogen being injected into the primary coolant, this Unit 2 setting may be increased to "≤3 x full power background (with hydrogen addition)."

TABLE 2.2.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

Functional Unit	Trip Setpoint
7. Drywell Pressure - High	≤2 psig
8. Scram Discharge Volume Water Level - High	≤40.4 gallons (Unit 2) ≤41 gallons (Unit 3)
9. Turbine Stop Valve - Closure	≤10% closed
10. Turbine EHC Control Oil Pressure - Low	≥900 psig
11. Turbine Control Valve Fast Closure	≥460 psig EHC fluid pressure
12. Turbine Condenser Vacuum - Low	≥21 inches Hg vacuum
13. Reactor Mode Switch Shutdown Position	NA
14. Manual Scram	NA

BASES

2.1 SAFETY LIMITS

The Specifications in Section 2.1 establish operating parameters to assure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). These parameters are based on the Safety Limits requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(1):

> "Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity."

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an AOO. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit for the MINIMUM CRITICAL POWER RATIO (MCPR) that represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical boundaries which separate radioactive materials from the environs. The integrity of the fuel cladding is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforations is just as measurable as that from use-related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding integrity Safety Limit is defined with margin to the conditions which would produce onset of transition boiling (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation. Therefore, the fuel cladding integrity Safety Limit is established such that no calculated fuel damage shall result from an abnormal operational transient. This is accomplished by selecting a MCPR fuel cladding integrity Safety Limit which assures that during normal operation and AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.

Exceeding a Safety Limit is cause for unit shutdown and review by the Nuclear Regulatory Commission (NRC) before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

BASES

2.1.A THERMAL POWER, Low Pressure or Low Flow

This fuel cladding integrity Safety Limit is established by establishing a limiting condition on core THERMAL POWER developed in the following method. At pressures below 800 psia (~785 psig), the core elevation pressure drop (0% power, 0% flow) is greater than 4.56 psi. At low powers and flows, this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a bundle flow of 28×10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. At 25% of RATED THERMAL POWER, the peak powered bundle would have to be operating at 3.86 times the average powered bundle in order to achieve this bundle power. Thus, a core thermal power limit of 25% for reactor pressures below 785 psig is conservative.

2.1.B THERMAL POWER, High Pressure and High Flow

This fuel cladding integrity Safety Limit is set such that no (mechanistic) fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power ratio (CPR) at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The margin between a MCPR of 1.0 (onset of transition boiling) and the Safety Limit, is derived from a detailed statistical analysis which considers the uncertainties in monitoring the core operating state, including uncertainty in the critical power correlation. Because the transition boiling correlation is based on a significant quantity of practical test data, there is a very high confidence that operation of a fuel assembly at the condition where MCPR is equal to the fuel cladding integrity Safety Limit would not produce transition boiling. In addition, during single recirculation loop operation, the MCPR Safety Limit is increased by 0.01 to conservatively account for increased uncertainties in the core flow and TIP measurements.

However, if transition boiling were to occur, cladding perforation would not necessarily be expected. Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative

approach. Much of the data indicates that BWR fuel can survive for an extended period in an environment of transition boiling.

2.1.C Reactor Coolant System Pressure

The Safety Limit for the reactor coolant system pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The reactor coolant system pressure Safety Limit of 1345 psig, as measured by the vessel steam space pressure indicator, is equivalent to 1375 psig at the lowest elevation of the reactor vessel. The 1375 psig value is derived from the design pressures of the reactor pressure vessel and coolant system piping. The respective design pressures are 1250 psig at 575°F and 1175 psig at 560°F. The pressure Safety Limit was chosen as the lower of the pressure transients permitted by the applicable design codes, ASME Boiler and Pressure Vessel Code Section III for the pressure vessel, and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% x 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over design pressure (120% x 1175 = 1410 psig). The Safety Limit pressure of 1375 psig is referenced to the lowest elevation of the reactor vessel. The design pressure for the recirculation suction line piping (1175 psig) was chosen relative to the reactor vessel design pressure. Demonstrating compliance of peak vessel pressure with the ASME overpressure protection limit (1375 psig) assures compliance of the suction piping with the USASI limit (1410 psig). Evaluation methodology to assure that this Safety Limit pressure is not exceeded for any reload is documented by the specific fuel vendor. The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig; this is a factor of 1.5 below the yield strength of 40,100 psi at 575°F. At the pressure limit of 1375 psig, the general membrane stress will only be 29,400 psi, still safely below the yield strength.

The relationships of stress levels to yield strength are comparable for the primary system piping and provides similar margin of protection at the established pressure Safety Limit.

The normal operating pressure of the reactor coolant system is nominally 1000 psig. Both pressure relief and safety relief valves have been installed to keep the reactor vessel peak pressure below 1375 psig. However no credit is taken for relief valves during the postulated full closure of all MSIVs without a direct (valve position switch) scram. Credit, however, is taken for the neutron flux scram. The indirect flux scram and safety valve actuation provide adequate margin below the allowable peak vessel pressure of 1375 psig.

2.1.D Reactor Vessel Water Level

With fuel in the reactor vessel during periods when the reactor is shutdown, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and cladding perforation. The core will be cooled sufficiently to prevent cladding melting should the water level be reduced to two-thirds of the core height. The Safety Limit has been established at 12 inches above the top of the active irradiated fuel to provide a point which can be monitored and also provide adequate margin for effective action. The top of active fuel is 360 inches above vessel zero.

2.2 LIMITING SAFETY SYSTEM SETTINGS

The Specifications in Section 2.2 establish operational settings for the reactor protection system instrumentation which initiates the automatic protective action at a level such that the Safety Limits will not be exceeded. These settings are based on the Limiting Safety System Settings requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(1):

> "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. "

2.2.A Reactor Protection System Instrumentation Setpoints

The Reactor Protection System (RPS) instrumentation setpoints specified in the table are the values at which the reactor scrams are set for each parameter. The scram settings have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and assist in mitigating the consequences of accidents. Conservatism incorporated into the transient analysis is documented by each approved fuel vendor. The bases for individual scram settings are discussed in the following paragraphs.

1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of eight chambers, four in each of the reactor protection system logic CHANNELs. The IRM is a 5 decade, 10 range, instrument which covers the range of power level between that covered by the SRM and the APRM. The IRM scram setting at 120 of 125 divisions is active in each range of the IRM. For example, if the instrument were on Range 1, the scram setting would be 120 divisions for that range; likewise, if the instrument were on Range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up.

The most significant sources of reactivity change during the power increase are due to control rod withdrawal. In order to ensure that the IRM provides adequate protection against the single rod withdrawal error, a range of rod withdrawal events has been analyzed. This analysis included starting the event at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale.

Additional conservatism was taken in this analysis by assuming that the IRM CHANNEL closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power is limited to 1% of rated power, thus maintaining MCPR above the fuel cladding integrity Safety Limit. Based on the above analysis, the IRM provides protection against local

control rod withdrawal errors and continuous withdrawal of control rods in the sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

For operation at low pressure and low flow during Startup, a reduced power level, i.e., setdown, APRM scram setting of 15% of RATED THERMAL POWER provides adequate thermal margin between the setting and the Safety Limit. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor; cold water from sources available during startup are not much colder than that already in the system; temperature coefficients are small; and, control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. 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 percentage of rated power, 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 scram setting, the rate of power rise is no more than 5% of RATED THERMAL POWER per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the Safety Limit. The 15% APRM setdown scram setting remains active until the mode switch is placed in the Run position.

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, also provides a flow biased neutron flux which reads in percent of RATED THERMAL POWER. Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. During abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that, with a 120% scram setting for dual recirculation loop operation, or with a 116.5% scram setting for single recirculation loop operation, none of the abnormal operational transients analyzed violates the fuel cladding integrity Safety Limit, and there is a substantial margin from fuel damage. One of the neutron flux scrams is flow dependent until it reaches the applicable setting where it is "clamped" at its maximum allowed value. The use of the flow referenced neutron flux scram setting provides additional margin beyond the use of a the fixed high flux scram setting alone.

An increase in the APRM scram setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams, which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit, yet allows operating margin that reduces the possibility of unnecessary scrams.

During single recirculation loop operation, the normal drive flow relationship is altered as a result of reverse flow through the idle loop jet pumps when the active loop recirculation pump speed is above approximately 40% of rated. The core receives less flow than would be predicted based upon the dual recirculation loop drive flow to core flow relationship, and the APRM flow biased scram settings must be altered to continue to provide a reactor scram at a conservative neutron flux.

The scram setting must also be adjusted to ensure that the LHGR transient limit is not violated for any power distribution. The scram setting is adjusted in accordance with Specification 3/4.11.B in order to maintain adequate margin for the Safety Limit and yet allow operating margin sufficient to reduce the possibility of an unnecessary shutdown. The adjustment may also be accomplished by increasing the APRM gain. This provides the same degree of protection as reducing the scram settings by raising the initial APRM readings closer to the scram settings such that a scram would be received at the same point in a transient as if the scram settings had been reduced.

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 scram will quickly reduce the neutron flux, counteracting the pressure increase. The scram setting is slightly higher than the operating pressure to permit normal operation without spurious scrams. The scram setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement (reactor vessel steam space) compared to the highest pressure that occurs in the system during a transient.

In compliance with Section III of the ASME Code, the safety valves must be set to open at no higher than 103% of design pressure, and they must limit the reactor pressure to no more than 110% of design pressure. Both the high neutron flux scram and safety valve actuation are required to prevent overpressurizing the reactor pressure vessel and thus, exceeding the pressure Safety Limit. The pressure scram is available as backup protection to the high flux scram. Analyses are performed for each reload to assure that the pressure Safety Limit is not exceeded.

4. Reactor Vessel Water Level - Low

The reactor vessel water level scram setting was chosen far enough below the normal operating level to avoid spurious scrams but high enough above the fuel to assure that there is adequate protection for the fuel cladding integrity and reactor coolant system pressure Safety Limits. The scram setting is based on normal operating temperature and pressure conditions because the level instrumentation is density compensated.

The scram setting provided is the actual water level which may be different than the water level as measured by the instrumentation outside the shroud. The water level inside the shroud will

decrease as power is increased to 100% in comparison to the level outside the shroud, to a maximum of seven inches, due to the pressure drop across the steam dryer. Therefore, at 100% power, an indicated water level of +8 inches water level may be as low as +1 inches inside the shroud which corresponds to 144 inches above the top of active fuel and 504 inches above vessel zero.

5. Main Steam Line Isolation Valve - Closure

Automatic isolation of the main steam lines is provided to give protection against rapid reactor depressurization and cooldown of the vessel. When the main steam line isolation valves begin to close, a scram signal provides for reactor shutdown so that high power operation at low reactor pressures does not occur. With the scram setting at 10% valve closure (from full open), there is no appreciable increase in neutron flux during normal or inadvertent isolation valve closure, thus providing protection for the fuel cladding integrity Safety Limit. Operation of the reactor at pressures lower than the MSIV closure setting requires the reactor mode switch to be in the Startup/Hot Standby position, where protection of the fuel cladding integrity Safety Limit is provided by the IRM and APRM high neutron flux scram signals. Thus, the combination of main steam line low pressure isolation and the isolation valve closure scram with the mode switch in the Run position assures the availability of the neutron flux scram protection over the entire range of applicability of fuel cladding integrity Safety Limit.

6. Main Steam Line Radiation - High

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity are an indication of leaking fuel. When high radiation is detected, a scram is initiated to mitigate the failure of fuel cladding. The scram setting is high enough above background radiation levels to prevent spurious scrams yet low enough to promptly detect gross failures in the fuel cladding. This setting is determined based on normal full power background (NFPB) radiation levels without hydrogen addition. With the injection of hydrogen into the feedwater for mitigation of intergranular stress corrosion cracking, the full power background levels may be significantly increased. The setting is increased based on the new background levels to allow for the injection of hydrogen. This trip function provides an anticipatory scram to limit offsite dose consequences, but is not assumed to occur in the analysis of any design basis event.

7. Drywell Pressure - High

High pressure in the drywell could indicate a break in the primary pressure boundary systems or a loss of drywell cooling. Therefore, pressure sensing instrumentation is provided as a backup to the water level instrumentation. The reactor is scrammed on high pressure in order to minimize the possibility of fuel damage and reduce the amount of energy being added to the coolant and the primary containment. The scram setting was selected as low as possible without causing spurious scrams.

8. Scram Discharge Volume Water Level - High

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. A part of this system is an individual instrument volume for each of the scram discharge volumes. These two instrument volumes and their piping can hold in excess of 90 gallons of water and are the low point in the piping. No credit was taken for the instrument volumes in the design of the discharge piping relative to the amount of water which must be accommodated during a scram. During normal operations, the scram discharge volumes are empty; however, should either scram discharge volume accumulate water, the water discharged to the piping from the reactor during a scram may not be accommodated which could result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level switches have been installed in both instrument volumes which will alarm and scram the reactor while sufficient volume remains to accommodate the discharged water. Diverse level sensing methods have been incorporated into the design and logic of the system to prevent common mode failure. The setting for this anticipatory scram signal has been chosen on the basis of providing sufficient volume remaining to accommodate a scram, even with 5 gpm leakage per drive into the scram discharge volume. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or the amount of insertion of the control rods.

9. Turbine Stop Valve - Closure

The turbine stop valve closure scram setting anticipates the pressure, neutron flux, and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram setting of 10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the fuel cladding integrity Safety Limit, even during the worst-case transient that assumes the turbine bypass fails to operate.

10. Turbine EHC Control Oil Pressure - Low

The turbine EHC control system operates using high pressure oil. There are several points in this oil system where a loss of oil pressure could result in a fast closure of the turbine control valves. This fast closure of the turbine control valves is not protected by the turbine control valve fast

closure scram since failure of the oil system would not result in the fast closure solenoid valves being actuated. For a turbine control valve fast closure, the core would be protected by the APRM and reactor high pressure scrams. However, to provide the same margins as provided for the generator load rejection on fast closure of the turbine control valves, a scram has been added to the reactor protection system which senses failure of control oil pressure to the turbine control system. This scram anticipates the pressure transient which would be caused by imminent control valve closure and results in reactor shutdown before any significant increase in neutron flux occurs. The transient response is very similar to that resulting from the turbine control valve fast closure scram. However, since the control valves will not start to close until the fluid pressure is approximately 600 psig, the scram on low turbine EHC control oil pressure occurs well before turbine control valve closure begins. The scram setting is high enough to provide the necessary anticipatory function and low enough to minimize the number of spurious scrams.

11. Turbine Control Valve Fast Closure

The turbine control valve fast closure scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass valves; i.e., MCPR remains above the fuel cladding integrity Safety Limit for this transient. For the load rejection without bypass transient from 100% power, the peak heat flux (and therefore LHGR) increases on the order of 15% which provides a wide margin to the value corresponding to 1% plastic strain of the cladding.

The scram setting based on EHC fluid pressure was developed to ensure that the pressure switch is actuated prior to the closure of the turbine control valves (at approximately 400 psig EHC fluid pressure), yet assure that the system is not actuated unnecessarily due to EHC system pressure transients which may cause EHC system pressure to momentarily decrease.

12. Turbine Condenser Vacuum - Low

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise and an increase in surface heat flux. To prevent the fuel cladding integrity Safety Limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the fuel cladding integrity Safety Limit from being exceeded, in the event of a turbine trip transient with bypass closure. The condenser low vacuum scram is anticipatory to the stop valve closure scram and causes a scram before the stop valves (and bypass valves) are closed and thus, the resulting transient is less severe.

13. Reactor Mode Switch Shutdown Position

The reactor mode switch Shutdown position is a redundant CHANNEL to the automatic protective instrumentation CHANNEL(s) and provides additional manual reactor scram capability.

14. Manual Scram

The manual scram is a redundant CHANNEL to the automatic protective instrumentation CHANNEL(s) and provides manual reactor scram capability.

3.11 - LIMITING CONDITIONS FOR OPERATION

A. AVERAGE PLANAR LINEAR HEAT GENERATION RATE

All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR) for each type of fuel as a function of bundle average exposure shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

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

ACTION:

With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective action within 15 minutes, and

2. Restore APLHGR to within the required limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

A. AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT.

1. At least once per 24 hours,

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

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

4. The provisions of Specification 4.0.D are not applicable.

3.11 - LIMITING CONDITIONS FOR OPERATION

B. Average Power Range Monitor Setpoints

The Average Power Range Monitor (APRM) gain or setpoints shall be set such that the FUEL DESIGN LIMITING RATIO FOR CENTERLINE MELT (FDLRC) shall be less than or equal to 1.0.

APPLICABILITY:

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

ACTION:

With FDLRC greater 1.0, initiate corrective ACTION within 15 minutes and within 6 hours either:

1. Restore FDLRC to within its limit, or

2. Adjust the flow biased APRM setpoints specified in Specifications 2.2.A and 3.2.E by 1/FDLRC, or

3. Adjust[a] the APRM gain such that the APRM readings are ≥100% of the FRACTION OF RATED THERMAL POWER (FRTP) times FDLRC.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

B. Average Power Range Monitor Setpoints

The value of FDLRC shall be verified:

1. At least once per 24 hours,

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

3. Initially and at least once per 12 hours when the reactor is operating with FDLRC greater than or equal to 1.0.

4. The provisions of Specification 4.0.D are not applicable.

a Provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

3.11 - LIMITING CONDITIONS FOR OPERATION

C. MINIMUM CRITICAL POWER RATIO

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR operating limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

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

ACTION:

With MCPR less than the applicable MCPR limit as determined for one of the conditions specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective ACTION within 15 minutes, and

2. Restore MCPR to within the required limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

C. MINIMUM CRITICAL POWER RATIO

MCPR, with:

1. t_{ave} = 3.50 prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.3.D, or

2. t_{ave} determined within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.3.D,

shall be determined to be equal to or greater than the applicable MCPR operating limit specified in the CORE OPERATING LIMITS REPORT.

1. At least once per 24 hours,

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

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

4. The provisions of Specification 4.0.D are not applicable.

3.11 - LIMITING CONDITIONS FOR OPERATION

D. STEADY STATE LINEAR HEAT GENERATION RATE

The STEADY STATE LINEAR HEAT GENERATION RATE (SLHGR) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

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

ACTION:

With a SLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective ACTION within 15 minutes, and

2. Restore the SLHGR to within the required limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

D. STEADY STATE LINEAR HEAT GENERATION RATE

The SLHGR shall be determined to be equal to or less than the limit:

1. At least once per 24 hours,

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

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

4. The provisions of Specification 4.0.D are not applicable.

3.11 - LIMITING CONDITIONS FOR OPERATION

E. TRANSIENT LINEAR HEAT GENERATION RATE

The TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

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

ACTION:

With a TLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective ACTION within 15 minutes, and

2. Restore the TLHGR to within the required limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

E. TRANSIENT LINEAR HEAT GENERATION RATE

The TLHGR shall be determined to be equal to or less than the limit:

1. At least once per 24 hours,

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

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

4. The provisions of Specification 4.0.D are not applicable.

BASES

3/4.11.A AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50.46. The specification also assures that fuel rod mechanical integrity is maintained during normal and transient operations.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod-to-rod power distribution within an assembly. The peak clad temperature is calculated assuming a LINEAR HEAT GENERATION RATE (LHGR) for the highest powered rod which is equal to or less than the design LHGR corrected for densification. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

The calculational procedure used to establish the maximum APLHGR values uses NRC approved calculational models which are consistent with the requirements of Appendix K of 10 CFR Part 50. The approved calculational models are listed in Specification 6.6.A.4.

The daily requirement for calculating APLHGR 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 to calculate APLHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating APLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that APLHGR will be known following a change in THERMAL POWER or power shape, that could place operation above a thermal limit.

3/4.11.B APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased neutron flux - high scram setting and control rod block functions of the APRM instruments for both two recirculation loop operation and single recirculation loop operation must be adjusted to ensure that the MCPR does not become less than the fuel cladding safety limit or that ≥1% plastic strain does not occur in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the value of FDLRC indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

3/4.11.C MINIMUM CRITICAL POWER RATIO

The required operating limit MCPR at steady state operating conditions as specified in Specification 3.11.C 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 the CRITICAL POWER RATIO (CPR). The type of transients evaluated were change of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.11.C is obtained and presented in the CORE OPERATING LIMITS REPORT.

The steady state values for MCPR specified were determined using NRC-approved methodology listed in specification 6.6.A.4.

The purpose of the reduced flow MCPR curves specified in the CORE OPERATING LIMITS REPORT are to define MCPR operating limits at other than rated core flow conditions. The reduced flow MCPR curves assure that the Safety Limit MCPR will not be violated.

Since the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. The 72 hour completion time is acceptable due to the relatively minor changes in t_{ave} expected during the fuel cycle.

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 has considerable margin. Thus, the demonstration of MCPR below this power level is 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 after initially determining that a LIMITING CONTROL ROD PATTERN exists ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation above a thermal limit.

3/4.11.D STEADY STATE LINEAR HEAT GENERATION RATE

This specification assures that the maximum STEADY STATE LINEAR HEAT GENERATION RATE in any fuel rod is less than the design STEADY STATE LINEAR HEAT GENERATION RATE even if fuel pellet densification is postulated. This provides assurance that the fuel end-of-life steady state criteria are met. The daily requirement for calculating SLHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distributions shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate SLHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating SLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that SLHGR will be known following a change in THERMAL POWER or power shape that could place operation above a thermal limit.

3/4.11.E TRANSIENT LINEAR HEAT GENERATION RATE

This specification provides assurance that the fuel will neither experience centerline melt nor exceed 1% plastic strain for transient overpower events beginning at any power and terminating at 120% of RATED THERMAL POWER. The daily requirement for calculating TLHGR 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 to calculate TLHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating TLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that TLHGR will be known following a change in THERMAL POWER or power shape that could place operation above a thermal limit.

3.12 - LIMITING CONDITIONS FOR OPERATION

A. PRIMARY CONTAINMENT INTEGRITY

The provisions of Specifications 3.7.A, 3.7.E and 3.10.A and Table 1-2 may be suspended to permit the reactor pressure vessel closure head and the drywell head to be removed and the primary containment air lock doors to be open when the reactor mode switch is in the Startup position during low power PHYSICS TESTS with THERMAL POWER less than 1% of RATED THERMAL POWER and reactor coolant temperature less than 212°F.

APPLICABILITY:

OPERATIONAL MODE 2, during low power PHYSICS TESTS.

ACTION:

With THERMAL POWER greater than or equal to 1% of RATED THERMAL POWER or with the reactor coolant temperature greater than or equal to 212°F, immediately place the reactor mode switch in the Shutdown position.

4.12 - SURVEILLANCE REQUIREMENTS

A. PRIMARY CONTAINMENT INTEGRITY

The THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during low power PHYSICS TESTS.

3.12 - LIMITING CONDITIONS FOR OPERATION

B. SHUTDOWN MARGIN Demonstrations

The provisions of Specifications 3.10.A and 3.10.C and Table 1-2 may be suspended to permit the reactor mode switch to be in the Startup position and to allow more than one control rod to be withdrawn for SHUTDOWN MARGIN demonstration, provided that at least the following requirements are satisfied.

1. The source range monitors are OPERABLE with the RPS circuitry "shorting links" removed per Specification 3.10.B.

2. The rod worth minimizer is OPERABLE per Specification 3.3.L and is programmed for the SHUTDOWN MARGIN demonstration, or conformance with the SHUTDOWN MARGIN demonstration procedure is verified by a second licensed operator or other technically qualified individual.

3. The "rod-out-notch-override" control shall not be used during out-of-sequence movement of the control rods.

4. No other CORE ALTERATION(s) are in progress.

APPLICABILITY:

OPERATIONAL MODE 5, during SHUTDOWN MARGIN demonstrations.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown or Refuel position.

4.12 - SURVEILLANCE REQUIREMENTS

B. SHUTDOWN MARGIN Demonstrations

Within 30 minutes prior to and at least once per 12 hours during the performance of a SHUTDOWN MARGIN demonstration, verify that;

1. The source range monitors are OPERABLE with the RPS circuitry "shorting links" removed per Specification 3.10.B,

2. The rod worth minimizer is OPERABLE with the required program per Specification 3.3.L or a second licensed operator or other technically qualified individual is present and verifies compliance with the SHUTDOWN MARGIN demonstration procedures, and

3. No other CORE ALTERATION(s) are in progress.

BASES

3/4.12.A PRIMARY CONTAINMENT INTEGRITY

The requirement for PRIMARY CONTAINMENT INTEGRITY is not applicable during the period when open vessel tests are being performed during the low power PHYSICS TESTS. Low power PHYSICS TESTS during OPERATIONAL MODE 2 may be required to be performed while still maintaining access to the primary containment and reactor pressure vessel. Additional requirements during these tests to restrict reactor power and reactor coolant temperature provide protection against potential conditions which could require primary containment or reactor coolant pressure boundary integrity.

3/4.12.B SHUTDOWN MARGIN Demonstrations

Performance of SHUTDOWN MARGIN demonstrations with the vessel head removed requires additional restrictions in order to ensure that criticality does not occur. These additional restrictions are specified in this LCO. SHUTDOWN MARGIN tests may be performed while in OPERATIONAL MODE 2 in accordance with Table 1-2 without meeting this Special Test Exception. For SHUTDOWN MARGIN demonstrations performed while in OPERATIONAL MODE 5, additional requirements must be met to ensure that adequate protection against potential reactivity excursions is available. Because multiple control rods will be withdrawn and the reactor will potentially become critical, the approved control rod withdrawal sequence must be enforced by the RWM, or must be verified by a second licensed operator or other technically qualified individual. To provide additional protection against inadvertent criticality, control rod withdrawals that are "out-of-sequence", i.e., do not conform to the Banked Position Withdrawal Sequence, must be made in individual notched withdrawal mode to minimize the potential reactivity insertion associated with each movement. Because the reactor vessel head may be removed during these tests, no other CORE ALTERATION(s) may be in progress. This Special Test Exception then allows changing the Table 1-2 reactor mode switch position requirements to include the Startup or Hot Standby position such that the SHUTDOWN MARGIN demonstrations may be performed while in OPERATIONAL MODE 5.

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

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 155
License No. DPR-29

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated September 15, 1992, as supplemented by letter dated April 21, 1995, 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 3.B. of Facility Operating License No. DPR-29 is hereby amended to read as follows:

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B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 155 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented no later than June 30, 1996.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert M. Pulsifer, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 13, 1995

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

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 151
License No. DPR-30

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated September 15, 1992, as supplemented by letter dated April 21, 1995, 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 3.B. of Facility Operating License No. DPR-30 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 151 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented no later than June 30, 1996.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert M. Pulsifer, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 13, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 155 AND 151

FACILITY OPERATING LICENSE NOS. DPR-29 AND DPR-30

DOCKET NOS. 50-254 AND 50-265

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number.

UNIT 1 REMOVE	UNIT 2 REMOVE	INSERT
1.1/2.1-1	1.1/2.1-1	2-1
1.1/2.1-2	1.1/2.1-2	2-2
1.1/2.1-3	1.1/2.1-2a	2-3
1.1/2.1-4	1.1/2.1-3	2-4
1.1/2.1-5	1.1/2.1-4	2-5
1.1/2.1-6	1.1/2.1-5	B 2-1
1.1/2.1-7	1.1/2.1-6	B 2-2
1.1/2.1-8	1.1/2.1-7	B 2-3
1.1/2.1-9	1.1/2.1-7a	B 2-4
1.1/2.1-10	1.1/2.1-8	B 2-5
1.1/2.1-11	1.1/2.1-9	B 2-6
1.1/2.1-12	1.1/2.1-10	B 2-7
1.1/2.1-13	1.1/2.1-11	B 2-8
1.1/2.1-14	1.1/2.1-12	B 2-9
1.1/2.1-15	---	B 2-10
1.1/2.1-16	---	B 2-11
1.1/2.1-17	---	---
Figure 2.1-1	Figure 2.1-1	---
Figure 2.1-2	Figure 2.1-2	---
Figure 2.1-3	Figure 2.1-3	---
1.2/2.2-1	1.2/2.2-1	---
1.2/2.2-2	1.2/2.2-2	---
1.2/2.2-3	1.2/2.2-2a	---
1.2/2.2-4	1.2/2.2-3	---
---	---	3/4.11-1
---	---	3/4.11-2
---	---	3/4.11-3
---	---	3/4.11-4
---	---	B 3/4.11-1
---	---	B 3/4.11-2
---	---	B 3/4.11-3
---	---	3/4.12-1
---	---	3/4.12-2
---	---	B 3/4.12-1

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.A THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.4.

THERMAL POWER, High Pressure and High Flow

2.1.B The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.07 with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow. During single recirculation loop operation, this MCPR limit shall be increased by 0.01.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With MCPR less than the above applicable limit and the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.4.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

Reactor Coolant System Pressure

2.1.C The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1345 psig.

APPLICABILITY: OPERATIONAL MODE(s) 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1345 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1345 psig within 2 hours and comply with the requirements of Specification 6.4.

Reactor Vessel Water Level

2.1.D The reactor vessel water level shall be greater than twelve inches above the top of the active irradiated fuel.

APPLICABILITY: OPERATIONAL MODE(s) 3, 4 and 5.

ACTION:

With the reactor vessel water level at or below twelve inches above the top of the active irradiated fuel, manually initiate the ECCS to restore the water level, after depressurizing the reactor vessel, if required, and comply with the requirements of Specification 6.4.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.2 LIMITING SAFETY SYSTEM SETTINGS

Reactor Protection System (RPS) Instrumentation Setpoints

2.2.A The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.A-1.

APPLICABILITY: As shown in Table 3.1.A-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Trip Setpoint column of Table 2.2.A-1, declare the CHANNEL inoperable and apply the applicable ACTION statement requirement of Specification 3.1.A until the CHANNEL is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

Functional Unit	Trip Setpoint
1. Intermediate Range Monitor:	
a. Neutron Flux - High	≤120/125 divisions of full scale
b. Inoperative	NA
2. Average Power Range Monitor:	
a. Setdown Neutron Flux - High	≤15% of RATED THERMAL POWER
b. Flow Biased Neutron Flux - High	
1) Dual Recirculation Loop Operation	
a) Flow Biased	≤0.58W[a] + 62%, with a maximum of
b) High Flow Clamped	≤120% of RATED THERMAL POWER
2) Single Recirculation Loop Operation	
a) Flow Biased	≤0.58W[a] + 58.5%, with a maximum of
b) High Flow Clamped	≤116.5% of RATED THERMAL POWER
c. Fixed Neutron Flux - High	≤120% of RATED THERMAL POWER
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	≤1060 psig
4. Reactor Vessel Water Level - Low	≥144 inches above top of active fuel
5. Main Steam Line Isolation Valve - Closure	≤10% closed
6. Main Steam Line Radiation - High	≤15 x normal full power background (without hydrogen addition)

a W shall be the recirculation loop flow expressed as a percentage of the recirculation loop flow which produces a rated core flow of 98 million lbs/hr.

TABLE 2.2.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

Functional Unit	Trip Setpoint
7. Drywell Pressure - High	≤2.5 psig
8. Scram Discharge Volume Water Level - High:	≤40 gallons
9. Turbine Stop Valve - Closure	≤10% closed
10. Turbine EHC Control Oil Pressure - Low	≥900 psig
11. Turbine Control Valve Fast Closure	≥460 psig EHC fluid pressure
12. Turbine Condenser Vacuum - Low	≥21 inches Hg vacuum
13. Reactor Mode Switch Shutdown Position	NA
14. Manual Scram	NA

BASES

2.1 SAFETY LIMITS

The Specifications in Section 2.1 establish operating parameters to assure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). These parameters are based on the Safety Limits requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(1):

> "Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity."

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an AOO. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit for the MINIMUM CRITICAL POWER RATIO (MCPR) that represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical boundaries which separate radioactive materials from the environs. The integrity of the fuel cladding is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforations is just as measurable as that from use-related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding integrity Safety Limit is defined with margin to the conditions which would produce onset of transition boiling (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation. Therefore, the fuel cladding integrity Safety Limit is established such that no calculated fuel damage shall result from an abnormal operational transient. This is accomplished by selecting a MCPR fuel cladding integrity Safety Limit which assures that during normal operation and AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.

Exceeding a Safety Limit is cause for unit shutdown and review by the Nuclear Regulatory Commission (NRC) before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

BASES

2.1.A THERMAL POWER, Low Pressure or Low Flow

This fuel cladding integrity Safety Limit is established by establishing a limiting condition on core THERMAL POWER developed in the following method. At pressures below 800 psia (~785 psig), the core elevation pressure drop (0% power, 0% flow) is greater than 4.56 psi. At low powers and flows, this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a bundle flow of 28×10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. At 25% of RATED THERMAL POWER, the peak powered bundle would have to be operating at 3.86 times the average powered bundle in order to achieve this bundle power. Thus, a core thermal power limit of 25% for reactor pressures below 785 psig is conservative.

2.1.B THERMAL POWER, High Pressure and High Flow

This fuel cladding integrity Safety Limit is set such that no (mechanistic) fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power ratio (CPR) at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The margin between a MCPR of 1.0 (onset of transition boiling) and the Safety Limit, is derived from a detailed statistical analysis which considers the uncertainties in monitoring the core operating state, including uncertainty in the critical power correlation. Because the transition boiling correlation is based on a significant quantity of practical test data, there is a very high confidence that operation of a fuel assembly at the condition where MCPR is equal to the fuel cladding integrity Safety Limit would not produce transition boiling. In addition, during single recirculation loop operation, the MCPR Safety Limit is increased by 0.01 to conservatively account for increased uncertainties in the core flow and TIP measurements.

However, if transition boiling were to occur, cladding perforation would not necessarily be expected. Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative

approach. Much of the data indicates that BWR fuel can survive for an extended period in an environment of transition boiling.

2.1.C Reactor Coolant System Pressure

The Safety Limit for the reactor coolant system pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The reactor coolant system pressure Safety Limit of 1345 psig, as measured by the vessel steam space pressure indicator, is equivalent to 1375 psig at the lowest elevation of the reactor vessel. The 1375 psig value is derived from the design pressures of the reactor pressure vessel and coolant system piping. The respective design pressures are 1250 psig at 575°F and 1175 psig at 560°F. The pressure Safety Limit was chosen as the lower of the pressure transients permitted by the applicable design codes, ASME Boiler and Pressure Vessel Code Section III for the pressure vessel, and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% x 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over design pressure (120% x 1175 = 1410 psig). The Safety Limit pressure of 1375 psig is referenced to the lowest elevation of the reactor vessel. The design pressure for the recirculation suction line piping (1175 psig) was chosen relative to the reactor vessel design pressure. Demonstrating compliance of peak vessel pressure with the ASME overpressure protection limit (1375 psig) assures compliance of the suction piping with the USASI limit (1410 psig). Evaluation methodology to assure that this Safety Limit pressure is not exceeded for any reload is documented by the specific fuel vendor. The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig; this is a factor of 1.5 below the yield strength of 40,100 psi at 575°F. At the pressure limit of 1375 psig, the general membrane stress will only be 29,400 psi, still safely below the yield strength.

The relationships of stress levels to yield strength are comparable for the primary system piping and provides similar margin of protection at the established pressure Safety Limit.

The normal operating pressure of the reactor coolant system is nominally 1000 psig. Both pressure relief and safety relief valves have been installed to keep the reactor vessel peak pressure below 1375 psig. However no credit is taken for relief valves during the postulated full closure of all MSIVs without a direct (valve position switch) scram. Credit, however, is taken for the neutron flux scram. The indirect flux scram and safety valve actuation provide adequate margin below the allowable peak vessel pressure of 1375 psig.

BASES

2.1.D Reactor Vessel Water Level

With fuel in the reactor vessel during periods when the reactor is shutdown, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and cladding perforation. The core will be cooled sufficiently to prevent cladding melting should the water level be reduced to two-thirds of the core height. The Safety Limit has been established at 12 inches above the top of the active irradiated fuel to provide a point which can be monitored and also provide adequate margin for effective action. The top of active fuel is 360 inches above vessel zero.

2.2 LIMITING SAFETY SYSTEM SETTINGS

The Specifications in Section 2.2 establish operational settings for the reactor protection system instrumentation which initiates the automatic protective action at a level such that the Safety Limits will not be exceeded. These settings are based on the Limiting Safety System Settings requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(1):

> "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. "

2.2.A Reactor Protection System Instrumentation Setpoints

The Reactor Protection System (RPS) instrumentation setpoints specified in the table are the values at which the reactor scrams are set for each parameter. The scram settings have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and assist in mitigating the consequences of accidents. Conservatism incorporated into the transient analysis is documented by each approved fuel vendor. The bases for individual scram settings are discussed in the following paragraphs.

1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of eight chambers, four in each of the reactor protection system logic CHANNELs. The IRM is a 5 decade, 10 range, instrument which covers the range of power level between that covered by the SRM and the APRM. The IRM scram setting at 120 of 125 divisions is active in each range of the IRM. For example, if the instrument were on Range 1, the scram setting would be 120 divisions for that range; likewise, if the instrument were on Range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up.

The most significant sources of reactivity change during the power increase are due to control rod withdrawal. In order to ensure that the IRM provides adequate protection against the single rod withdrawal error, a range of rod withdrawal events has been analyzed. This analysis included starting the event at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale.

Additional conservatism was taken in this analysis by assuming that the IRM CHANNEL closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power is limited to 1% of rated power, thus maintaining MCPR above the fuel cladding integrity Safety Limit. Based on the above analysis, the IRM provides protection against local

control rod withdrawal errors and continuous withdrawal of control rods in the sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

For operation at low pressure and low flow during Startup, a reduced power level, i.e., setdown, APRM scram setting of 15% of RATED THERMAL POWER provides adequate thermal margin between the setting and the Safety Limit. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor; cold water from sources available during startup are not much colder than that already in the system; temperature coefficients are small; and, control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. 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 percentage of rated power, 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 scram setting, the rate of power rise is no more than 5% of RATED THERMAL POWER per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the Safety Limit. The 15% APRM setdown scram setting remains active until the mode switch is placed in the Run position.

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, also provides a flow biased neutron flux which reads in percent of RATED THERMAL POWER. Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. During abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that, with a 120% scram setting for dual recirculation loop operation, or with a 116.5% scram setting for single recirculation loop operation, none of the abnormal operational transients analyzed violates the fuel cladding integrity Safety Limit, and there is a substantial margin from fuel damage. One of the neutron flux scrams is flow dependent until it reaches the applicable setting where it is "clamped" at its maximum allowed value. The use of the flow referenced neutron flux scram setting provides additional margin beyond the use of a the fixed high flux scram setting alone.

An increase in the APRM scram setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams, which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit, yet allows operating margin that reduces the possibility of unnecessary scrams.

During single recirculation loop operation, the normal drive flow relationship is altered as a result of reverse flow through the idle loop jet pumps when the active loop recirculation pump speed is above approximately 40% of rated. The core receives less flow than would be predicted based upon the dual recirculation loop drive flow to core flow relationship, and the APRM flow biased scram settings must be altered to continue to provide a reactor scram at a conservative neutron flux.

The scram setting must also be adjusted to ensure that the LHGR transient limit is not violated for any power distribution. The scram setting is adjusted in accordance with Specification 3/4.11.B in order to maintain adequate margin for the Safety Limit and yet allow operating margin sufficient to reduce the possibility of an unnecessary shutdown. The adjustment may also be accomplished by increasing the APRM gain. This provides the same degree of protection as reducing the scram settings by raising the initial APRM readings closer to the scram settings such that a scram would be received at the same point in a transient as if the scram settings had been reduced.

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 scram will quickly reduce the neutron flux, counteracting the pressure increase. The scram setting is slightly higher than the operating pressure to permit normal operation without spurious scrams. The scram setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement (reactor vessel steam space) compared to the highest pressure that occurs in the system during a transient.

In compliance with Section III of the ASME Code, the safety valves must be set to open at no higher than 103% of design pressure, and they must limit the reactor pressure to no more than 110% of design pressure. Both the high neutron flux scram and safety valve actuation are required to prevent overpressurizing the reactor pressure vessel and thus, exceeding the pressure Safety Limit. The pressure scram is available as backup protection to the high flux scram. Analyses are performed for each reload to assure that the pressure Safety Limit is not exceeded.

4. Reactor Vessel Water Level - Low

The reactor vessel water level scram setting was chosen far enough below the normal operating level to avoid spurious scrams but high enough above the fuel to assure that there is adequate protection for the fuel cladding integrity and reactor coolant system pressure Safety Limits. The scram setting is based on normal operating temperature and pressure conditions because the level instrumentation is density compensated.

The scram setting provided is the actual water level which may be different than the water level as measured by the instrumentation outside the shroud. The water level inside the shroud will

decrease as power is increased to 100% in comparison to the level outside the shroud, to a maximum of seven inches, due to the pressure drop across the steam dryer. Therefore, at 100% power, an indicated water level of +8 inches water level may be as low as +1 inches inside the shroud which corresponds to 144 inches above the top of active fuel and 504 inches above vessel zero.

5. Main Steam Line Isolation Valve - Closure

Automatic isolation of the main steam lines is provided to give protection against rapid reactor depressurization and cooldown of the vessel. When the main steam line isolation valves begin to close, a scram signal provides for reactor shutdown so that high power operation at low reactor pressures does not occur. With the scram setting at 10% valve closure (from full open), there is no appreciable increase in neutron flux during normal or inadvertent isolation valve closure, thus providing protection for the fuel cladding integrity Safety Limit. Operation of the reactor at pressures lower than the MSIV closure setting requires the reactor mode switch to be in the Startup/Hot Standby position, where protection of the fuel cladding integrity Safety Limit is provided by the IRM and APRM high neutron flux scram signals. Thus, the combination of main steam line low pressure isolation and the isolation valve closure scram with the mode switch in the Run position assures the availability of the neutron flux scram protection over the entire range of applicability of fuel cladding integrity Safety Limit.

6. Main Steam Line Radiation - High

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity are an indication of leaking fuel. When high radiation is detected, a scram is initiated to mitigate the failure of fuel cladding. The scram setting is high enough above background radiation levels to prevent spurious scrams yet low enough to promptly detect gross failures in the fuel cladding. This setting is determined based on normal full power background (NFPB) radiation levels without hydrogen addition. With the injection of hydrogen into the feedwater for mitigation of intergranular stress corrosion cracking, the full power background levels may be significantly increased. The setting is sufficiently high to allow the injection of hydrogen without requiring an increase in the setting. This trip function provides an anticipatory scram to limit offsite dose consequences, but is not assumed to occur in the analysis of any design basis event.

7. Drywell Pressure - High

High pressure in the drywell could indicate a break in the primary pressure boundary systems or a loss of drywell cooling. Therefore, pressure sensing instrumentation is provided as a backup to the water level instrumentation. The reactor is scrammed on high pressure in order to minimize the possibility of fuel damage and reduce the amount of energy being added to the coolant and the primary containment. The scram setting was selected as low as possible without causing spurious scrams.

8. Scram Discharge Volume Water Level - High

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. A part of this system is an individual instrument volume for each of the scram discharge volumes. These two instrument volumes and their piping can hold in excess of 90 gallons of water and are the low point in the piping. No credit was taken for the instrument volumes in the design of the discharge piping relative to the amount of water which must be accommodated during a scram. During normal operations, the scram discharge volumes are empty; however, should either scram discharge volume accumulate water, the water discharged to the piping from the reactor during a scram may not be accommodated which could result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level switches have been installed in both instrument volumes which will alarm and scram the reactor while sufficient volume remains to accommodate the discharged water. Diverse level sensing methods have been incorporated into the design and logic of the system to prevent common mode failure. The setting for this anticipatory scram signal has been chosen on the basis of providing sufficient volume remaining to accommodate a scram, even with 5 gpm leakage per drive into the scram discharge volume. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or the amount of insertion of the control rods.

9. Turbine Stop Valve - Closure

The turbine stop valve closure scram setting anticipates the pressure, neutron flux, and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram setting of 10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the fuel cladding integrity Safety Limit, even during the worst-case transient that assumes the turbine bypass fails to operate.

10. Turbine EHC Control Oil Pressure - Low

The turbine EHC control system operates using high pressure oil. There are several points in this oil system where a loss of oil pressure could result in a fast closure of the turbine control valves. This fast closure of the turbine control valves is not protected by the turbine control valve fast

closure scram since failure of the oil system would not result in the fast closure solenoid valves being actuated. For a turbine control valve fast closure, the core would be protected by the APRM and reactor high pressure scrams. However, to provide the same margins as provided for the generator load rejection on fast closure of the turbine control valves, a scram has been added to the reactor protection system which senses failure of control oil pressure to the turbine control system. This scram anticipates the pressure transient which would be caused by imminent control valve closure and results in reactor shutdown before any significant increase in neutron flux occurs. The transient response is very similar to that resulting from the turbine control valve fast closure scram. However, since the control valves will not start to close until the fluid pressure is approximately 600 psig, the scram on low turbine EHC control oil pressure occurs well before turbine control valve closure begins. The scram setting is high enough to provide the necessary anticipatory function and low enough to minimize the number of spurious scrams.

11. Turbine Control Valve Fast Closure

The turbine control valve fast closure scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass valves; i.e., MCPR remains above the fuel cladding integrity Safety Limit for this transient. For the load rejection without bypass transient from 100% power, the peak heat flux (and therefore LHGR) increases on the order of 15% which provides a wide margin to the value corresponding to 1% plastic strain of the cladding.

The scram setting based on EHC fluid pressure was developed to ensure that the pressure switch is actuated prior to the closure of the turbine control valves (at approximately 400 psig EHC fluid pressure), yet assure that the system is not actuated unnecessarily due to EHC system pressure transients which may cause EHC system pressure to momentarily decrease.

12. Turbine Condenser Vacuum - Low

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise and an increase in surface heat flux. To prevent the fuel cladding integrity Safety Limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the fuel cladding integrity Safety Limit from being exceeded, in the event of a turbine trip transient with bypass closure. The condenser low vacuum scram is anticipatory to the stop valve closure scram and causes a scram before the stop valves (and bypass valves) are closed and thus, the resulting transient is less severe.

BASES

13. Reactor Mode Switch Shutdown Position

The reactor mode switch Shutdown position is a redundant CHANNEL to the automatic protective instrumentation CHANNEL(s) and provides additional manual reactor scram capability.

14. Manual Scram

The manual scram is a redundant CHANNEL to the automatic protective instrumentation CHANNEL(s) and provides manual reactor scram capability.

3.11 - LIMITING CONDITIONS FOR OPERATION

A. AVERAGE PLANAR LINEAR HEAT GENERATION RATE

All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

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

ACTION:

With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective ACTION within 15 minutes, and

2. Restore APLHGR to within the required limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

A. AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT.

1. At least once per 24 hours,

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

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

4. The provisions of Specification 4.0.D are not applicable.

3.11 - LIMITING CONDITIONS FOR OPERATION

B. Average Power Range Monitor Setpoints

The Average Power Range Monitor (APRM) gain or setpoints shall be set such that the MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be less than or equal to the FRACTION OF RATED THERMAL POWER (FRTP).

APPLICABILITY:

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

ACTION:

With MFLPD greater than FRTP, initiate corrective ACTION within 15 minutes and within 6 hours either:

1. Restore MFLPD to within its limit, or

2. Adjust the flow biased APRM setpoints specified in Specifications 2.2.A and 3.2.E by FRTP/MFLPD, or

3. Adjust[a] the APRM gain such that the APRM readings are ≥100% of the MFLPD.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

B. Average Power Range Monitor Setpoints

The value of MFLPD shall be verified:

1. At least once per 24 hours,

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

3. Initially and at least once per 12 hours when the reactor is operating with MFLPD greater than or equal to FRTP.

4. The provisions of Specification 4.0.D are not applicable.

a Provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

3.11 - LIMITING CONDITIONS FOR OPERATION

C. MINIMUM CRITICAL POWER RATIO

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR operating limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

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

ACTION:

With MCPR less than the applicable MCPR operating limit as determined for one of the conditions specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective ACTION within 15 minutes, and

2. Restore MCPR to within the required limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

C. MINIMUM CRITICAL POWER RATIO

MCPR, with:

1. t_{ave} = 0.86 prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.3.D, or

2. t_{ave} determined within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.3.D,

shall be determined to be equal to or greater than the applicable MCPR operating limit specified in the CORE OPERATING LIMITS REPORT.

1. At least once per 24 hours,

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

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

4. The provisions of Specification 4.0.D are not applicable.

3.11 - LIMITING CONDITIONS FOR OPERATION

D. LINEAR HEAT GENERATION RATE

The LINEAR HEAT GENERATION RATE (LHGR) for each type of fuel shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

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

ACTION:

With a LHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective ACTION within 15 minutes, and

2. Restore the LHGR to within the required limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

D. LINEAR HEAT GENERATION RATE

The LHGR shall be determined to be equal to or less than the limit:

1. At least once per 24 hours,

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

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

4. The provisions of Specification 4.0.D are not applicable.

BASES

3/4.11.A AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50.46. The specification also assures that fuel rod mechanical integrity is maintained during normal and transient operations.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod-to-rod power distribution within an assembly. The peak clad temperature is calculated assuming a LINEAR HEAT GENERATION RATE (LHGR) for the highest powered rod which is equal to or less than the design LHGR corrected for densification. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

The calculational procedure used to establish the maximum APLHGR values uses NRC approved calculational models which are consistent with the requirements of Appendix K of 10 CFR Part 50. The approved calculational models are listed in Specification 6.6.A.4.

The daily requirement for calculating APLHGR 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 to calculate APLHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating APLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that APLHGR will be known following a change in THERMAL POWER or power shape, that could place operation above a thermal limit.

3/4.11.B APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased neutron flux - high scram setting and control rod block functions of the APRM instruments for both two recirculation loop operation and single recirculation loop operation must be adjusted to ensure that the MCPR does not become less than the fuel cladding safety limit or that ≥1% plastic strain does not occur in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the value of MFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

3/4.11.C MINIMUM CRITICAL POWER RATIO

The required operating limit MCPR at steady state operating conditions as specified in Specification 3.11.C 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 the CRITICAL POWER RATIO (CPR). The type of transients evaluated were change of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.11.C is obtained and presented in the CORE OPERATING LIMITS REPORT.

The steady state values for MCPR specified were determined using NRC-approved methodology listed in specification 6.6.A.4.

The purpose of the MCPR multiplicative factor specified in the CORE OPERATING LIMITS REPORT is to define MCPR operating limits at other than rated core flow conditions. At less than 100% of rated flow, the required MCPR is the product of the MCPR and the off rated flow MCPR multiplier factor. The MCPR multiplier assures that the Safety Limit MCPR will not be violated.

Since the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. The 72 hour completion time is acceptable due to the relatively minor changes in t_{ave} expected during the fuel cycle.

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 has considerable margin. Thus, the demonstration of MCPR below this power level is 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 after initially determining that a LIMITING CONTROL ROD PATTERN exists ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation above a thermal limit.

3/4.11.D LINEAR HEAT GENERATION RATE

This specification assures that the LINEAR HEAT GENERATION RATE (LHGR) in any fuel rod is less than the design linear heat generation even if fuel pellet densification is postulated. The daily requirement for calculating LHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distributions shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate LHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating LHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that LHGR will be known following a change in THERMAL POWER or power shape that could place operation above a thermal limit.

3.12 - LIMITING CONDITIONS FOR OPERATION

A. PRIMARY CONTAINMENT INTEGRITY

The provisions of Specifications 3.7.A, 3.7.E and 3.10.A and Table 1-2 may be suspended to permit the reactor pressure vessel closure head and the drywell head to be removed and the primary containment air lock doors to be open when the reactor mode switch is in the Startup position during low power PHYSICS TESTS with THERMAL POWER less than 1% of RATED THERMAL POWER and reactor coolant temperature less than 212°F.

APPLICABILITY:

OPERATIONAL MODE 2, during low power PHYSICS TESTS.

ACTION:

With THERMAL POWER greater than or equal to 1% of RATED THERMAL POWER or with the reactor coolant temperature greater than or equal to 212°F, immediately place the reactor mode switch in the Shutdown position.

4.12 - SURVEILLANCE REQUIREMENTS

A. PRIMARY CONTAINMENT INTEGRITY

The THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during low power PHYSICS TESTS.

3.12 - LIMITING CONDITIONS FOR OPERATION

B. SHUTDOWN MARGIN Demonstrations

The provisions of Specifications 3.10.A and 3.10.C and Table 1-2 may be suspended to permit the reactor mode switch to be in the Startup position and to allow more than one control rod to be withdrawn for SHUTDOWN MARGIN demonstration, provided that at least the following requirements are satisfied.

1. The source range monitors are OPERABLE with the RPS circuitry "shorting links" removed per Specification 3.10.B.

2. The rod worth minimizer is OPERABLE per Specification 3.3.L and is programmed for the SHUTDOWN MARGIN demonstration, or conformance with the SHUTDOWN MARGIN demonstration procedure is verified by a second licensed operator or other technically qualified individual.

3. The "rod-out-notch-override" control shall not be used during out-of-sequence movement of the control rods.

4. No other CORE ALTERATION(s) are in progress.

APPLICABILITY:

OPERATIONAL MODE 5, during SHUTDOWN MARGIN demonstrations.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown or Refuel position.

4.12 - SURVEILLANCE REQUIREMENTS

B. SHUTDOWN MARGIN Demonstrations

Within 30 minutes prior to and at least once per 12 hours during the performance of a SHUTDOWN MARGIN demonstration, verify that;

1. The source range monitors are OPERABLE with the RPS circuitry "shorting links" removed per Specification 3.10.B,

2. The rod worth minimizer is OPERABLE with the required program per Specification 3.3.L or a second licensed operator or other technically qualified individual is present and verifies compliance with the SHUTDOWN MARGIN demonstration procedures, and

3. No other CORE ALTERATION(s) are in progress.

3/4.12.A PRIMARY CONTAINMENT INTEGRITY

The requirement for PRIMARY CONTAINMENT INTEGRITY is not applicable during the period when open vessel tests are being performed during the low power PHYSICS TESTS. Low power PHYSICS TESTS during OPERATIONAL MODE 2 may be required to be performed while still maintaining access to the primary containment and reactor pressure vessel. Additional requirements during these tests to restrict reactor power and reactor coolant temperature provide protection against potential conditions which could require primary containment or reactor coolant pressure boundary integrity.

3/4.12.B SHUTDOWN MARGIN Demonstrations

Performance of SHUTDOWN MARGIN demonstrations with the vessel head removed requires additional restrictions in order to ensure that criticality does not occur. These additional restrictions are specified in this LCO. SHUTDOWN MARGIN tests may be performed while in OPERATIONAL MODE 2 in accordance with Table 1-2 without meeting this Special Test Exception. For SHUTDOWN MARGIN demonstrations performed while in OPERATIONAL MODE 5, additional requirements must be met to ensure that adequate protection against potential reactivity excursions is available. Because multiple control rods will be withdrawn and the reactor will potentially become critical, the approved control rod withdrawal sequence must be enforced by the RWM, or must be verified by a second licensed operator or other technically qualified individual. To provide additional protection against inadvertent criticality, control rod withdrawals that are "out-of-sequence", i.e., do not conform to the Banked Position Withdrawal Sequence, must be made in individual notched withdrawal mode to minimize the potential reactivity insertion associated with each movement. Because the reactor vessel head may be removed during these tests, no other CORE ALTERATION(s) may be in progress. This Special Test Exception then allows changing the Table 1-2 reactor mode switch position requirements to include the Startup or Hot Standby position such that the SHUTDOWN MARGIN demonstrations may be performed while in OPERATIONAL MODE 5.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 134 TO FACILITY OPERATING LICENSE NO. DPR-19,

AMENDMENT NO. 128 TO FACILITY OPERATING LICENSE NO. DPR-25,

AMENDMENT NO. 155 TO FACILITY OPERATING LICENSE NO. DPR-29,

AND AMENDMENT NO. 151 TO FACILITY OPERATING LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-237, 50-249, 50-254 AND 50-265

1.0 INTRODUCTION

By letter dated September 15, 1992, as supplemented by letter dated April 21, 1995, Commonwealth Edison Company (ComEd, the licensee) submitted an amendment requesting to upgrade sections of the Dresden Nuclear Power Station, Units 2 and 3, and the Quad Cities Nuclear Power Station, Units 1 and 2, Technical Specifications (TS). The changes have been requested as part of their Technical Specification Upgrade Program (TSUP).

As a result of findings by a Diagnostic Evaluation Team inspection performed by the NRC staff at the Dresden Nuclear Power Station in 1987, ComEd made a decision that both the Dresden Nuclear Power Station and sister site Quad Cities Nuclear Power Station, needed attention focused on the existing custom TS used at the sites.

The licensee made the decision to initiate a TSUP for both Dresden and Quad Cities. The licensee evaluated the current TS for both stations against the Standard Technical Specifications (STS), contained in NUREG-0123, "Standard Technical Specifications General Electric Plants BWR/4, Revision 4." Both Dresden and Quad Cities are BWR-3 designs and are nearly identical plants. The licensee's evaluation identified numerous potential improvements such as clarifying requirements, changing the TS to make them more understandable and to eliminate the need for interpretation, and deleting requirements that are no longer considered current with industry practice. As a result of the evaluation, ComEd elected to upgrade both the Dresden and Quad Cities TS to the STS contained in NUREG-0123.

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The TSUP for Dresden and Quad Cities is not a complete adaption of the STS. The TSUP focuses on (1) integrating additional information such as equipment operability requirements during shutdown conditions, (2) clarifying requirements such as limiting conditions for operations and action statements utilizing STS terminology, (3) deleting superseded requirements and modifications to the TS based on the licensee's responses to Generic Letters (GL), and (4) relocating specific items to more appropriate TS locations.

The application dated September 15, 1992, as supplemented April 21, 1995, proposed to upgrade only those sections of the TS to be included in TSUP Sections 2.0 (Safety Limit and Limiting Safety System Settings), 3/4.11 (Power Distribution Limits), and 3/4.12 (Special Test Exemptions) of the Dresden and Quad Cities TS.

The staff reviewed the proposed changes and evaluated all deviations and changes between the proposed TS, the STS, and the current TS. In no case did the licensee propose a change in the TS that would result in the relaxation of the current design requirements as stated in the Updated Final Safety Analysis Reports (UFSAR) for Dresden or Quad Cities.

In response to the staff's recommendations, the licensee submitted identical TS for Quad Cities and Dresden except for plant-specific equipment and design differences. Technical differences between the units are identified as appropriate in the proposed amendment.

2.0 EVALUATION

Review Guidelines - The licensee's purpose for the TSUP was to reformat the existing Dresden and Quad Cities TS into the easier to use STS format. Plant specific data, values, parameters, and equipment specific operational requirements contained in the current TS for Dresden and Quad Cities were retained by the licensee in the TSUP.

The STS contained in NUREG-0123 were developed by the NRC and industry because of the shortcomings associated with the custom TS which were issued to plants licensed in early 1970's (i.e., Dresden (1971) and Quad Cities (1972)). The STS developed by the NRC and industry provided an adequate level of protection for plant operation by assuring required systems are operable and have been proven to be able to perform their intended functions. The limiting conditions for operation (LCO), the allowed out-of-service times, and the required surveillance frequencies were developed based on industry operating experience, equipment performance, and probabilistic risk assessment analysis during the 1970's. The STS were used as the licensing basis for plants licensed starting in the late 1970's.

For the most part, ComEd's adoption of the STS resulted in more restrictive LCOs and surveillance requirements (SR). In some cases, however, the STS provides relief from the Dresden and Quad Cities current TS requirements. In all these cases, the adoption of the STS requirements for LCOs or SR does not change the current design requirements of either plant as described in each

plant's UFSAR. In addition, the success criteria for the availability and operability of all required systems contained in the current TS are maintained by the adoption of the STS requirements in the proposed TSUP TS.

In addition to adopting the STS guidelines and requirements in the TSUP, ComEd has also evaluated Generic Letters (GLs) concerning line item improvements for TS. These GLs were factored into TSUP to make the proposed TS in the TSUP reflect industry lessons learned in the 1980's and early 1990's.

Deviations between the proposed specifications, the STS, and the current TS were reviewed by the staff to determine if they were due to plant specific features or if they posed a technical deviation from the STS guidelines. Plant specific data, values, parameters, and equipment specific operational requirements contained in the current TS for Dresden and Quad Cities were retained by the licensee in the upgraded TS.

Administrative Changes - Non-technical, administrative changes were intended to incorporate human factor principles into the form and structure of the STS so that they would be easier for plant operation's personnel to use. These changes are editorial in nature or involve the reorganization or reformatting of requirements without affecting technical content of the current TS or operational requirements. Every section of the proposed TS reflects this type of change.

More Restrictive Requirements - The proposed TSUP TS include certain more restrictive requirements than are contained in the existing TS. Examples of more restrictive requirements include the following: placing an LCO on plant equipment which is not required by the present TS to be operable; adding more restrictive requirements to restore inoperable equipment; and adding more restrictive SR.

Less Restrictive Requirements - The licensee provided a justification for less restrictive requirements on a case-by-case basis as discussed in this SE. When requirements have been shown to provide little or no safety benefit, their removal from the TS may be appropriate. In most cases, these relaxations had previously been granted to individual plants on a plant-specific basis as the result of (a) generic NRC actions, and (b) new NRC staff positions that have evolved from technological advancements and operating experience.

The Dresden and Quad Cities plant designs were reviewed to determine if the specific design basis was consistent with the STS contained in NUREG-0123. All changes to the current TS and deviations between the licensee's proposed TS and the STS were reviewed by the staff for acceptability to determine if adequate justification was provided (i.e., plant specific features, retention of existing operating values, etc.).

Deviations the staff finds acceptable include: (1) adding clarifying statements, (2) incorporating changes based on GLs, (3) reformatting multiple steps included under STS action statements into single steps with unique

identifiers, (4) retaining plant specific steps, parameters, or values, (5) moving ACTION statements within a TS, (6) moving ACTION statements from an existing TS to form a new TS section, and (7) omitting the inclusion of STS steps that are not in existing TS.

Relocation of Technical Specifications - The proposed TS include the relocation of some requirements from the TS to licensee-controlled documents. Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to state Technical Specifications to be included as part of the license. The Commission's regulatory requirements related to the content of TS are set forth in 10 CFR 50.36. That regulation requires that the TS include items in five specific categories, including (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. However, the regulation does not specify the particular requirements to be included in a plant's TS.

The Commission has provided guidance for the contents of TS in its "Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," 58 Fed. Reg. 39132 (July 22, 1993), in which the Commission indicated that compliance with the Final Policy Statement satisfies Section 182a of the Energy Reorganization Act. The Final Policy Statement identified four criteria to be used in determining whether a particular matter is required to be included in the TS, as follows: (1) Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary; (2) a process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (3) a structure, system, or component that is part of a primary success path and which functions or actuates to mitigate a Design Basis Accident of Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (4) a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety. As a result, existing TS requirements which fall within or satisfy any of the criteria in the Final Policy Statement must be retained in the TS, while those TS requirements which do not fall within or satisfy these criteria may be relocated to other, licensee-controlled documents.

The following sections provide the staff's evaluations of the specific proposed TS changes.

3.0 EVALUATION OF TSUP PROPOSED TS SECTION 2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

The following sections provide the staff's evaluation of the TS changes reflected in proposed TSUP TS Section 2.0. Proposed TSUP TS Section 2.0 results from combining current TS Sections 1/2.1 and 1/2.2 and reformatting to STS format. Section 2.0 establishes operating parameters to assure specific

acceptable fuel design limits are not exceeded during steady state operation, normal operational transients and anticipated operational occurrences.

3.1 Section 2.1.A: THERMAL POWER, Low Pressure or Low Flow

Proposed TS 2.1.A, "THERMAL POWER, Low Pressure or Low Flow," incorporates the requirements of the STS Section 2.1.1 and current TS requirements from Section 1.1.B of the current TS for both stations. The proposed section has been reformatted adopting the STS guidelines. Plant specific values for the listed parameters are included to be consistent to the UFSAR for both stations. The current TS do not contain specific applicability requirements or required actions. Specific applicability statements in accordance with STS guidelines have been adopted in the proposed TS. In addition, required actions have been adopted in the proposed TS in accordance with STS guidelines when the safety limit is violated. There are no changes and/or deviations from STS guidance in the proposed TS Section.

A change to current TS Section 1.1.B. for both Dresden and Quad Cities Stations has been proposed. The Reactor Vessel Steam Space Pressure which defines the transition between high and low pressure for this safety limit in the current TS is 800 psig, while in the proposed TS it is 785 psig. The 800 psig is an incorrect value for the safety limit. The current specifications for both Dresden and Quad Cities contain an error, using both 800 psig and 800 psia values for this set point. The current TS LCOs use 800 psig while the Bases use 800 psia. The 800 psia value is the correct value. To be consistent with STS guidelines, the proposed TS will express pressures in units of psig. The UFSAR and the Core Operating Limits Reports for Dresden and Quad Cities both use the value of 785 psig which is equivalent to 800 psia. The proposed TS is consistent with the UFSARs and design requirements for both stations and is also consistent with the guidance of the STS.

Based on the above evaluation, the staff finds that the proposed TS 2.1.A: THERMAL POWER, Low Pressure or Low Flow has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS and current TS requirements and finds that there are no deviations from STS guidelines and the proposed TS have incorporated the current TS requirements for this safety limit. In addition, the staff finds the change to the current pressure transition point is consistent with the current plants UFSAR and is, therefore, acceptable. Therefore, the staff finds that proposed TSUP TS 2.1.A: "THERMAL POWER, Low Pressure or Low Flow" is acceptable.

3.2 Section 2.1.B: THERMAL POWER, High Pressure and High Flow

Proposed TS 2.1.B, "THERMAL POWER, High Pressure and High Flow," incorporates the requirements of the STS Section 2.1.2 and current TS Section 1.1.A. for both stations. Plant specific values for the listed parameters are included to be consistent to the UFSAR for the plants. The current TS do not contain specific required action or applicability statements for this specific safety limit. The proposed TS incorporates actions and applicability statements in accordance with the STS guidelines and format.

A change to current TS Section 1.1.A for both Dresden and Quad Cities Stations has been proposed. The Reactor Vessel Steam Space Pressure which defines the transition between high and low pressure for this safety limit in the current TS is 800 psig, while in the proposed TS it is 785 psig. The 800 psig is an incorrect value for the safety limit. The current specifications for both Dresden and Quad Cities contain an error, using both 800 psig and 800 psia values for this set point. The current TS LCOs use 800 psig while the Bases use 800 psia. The 800 psia value is the correct value. To be consistent with STS guidelines, the proposed TS will express pressures in units of psig. The UFSAR and the Core Operating Limits Reports for Dresden and Quad Cities both use the value of 785 psig which is equivalent to 800 psia. The proposed TS is consistent with the UFSARs and design requirements for both stations and is also consistent with the guidance of the STS.

The proposed TS conservatively maintains the 0.01 adder for the Minimum Critical Power Ratio (MCPR) limit when in Single Loop Operation to account for increased uncertainties in the core flow and neutron flux monitoring measurements. This is a deviation from STS guidance, but is consistent with current TS requirements and is consistent with the guidance provided in GL 86-09, "Technical Resolution of Generic Issue No. B-59 - (N-1) Loop Operation in BWRs and PWRs."

Based on the above evaluation, the staff finds that the proposed TS for Section 2.1.B: "THERMAL POWER, High Pressure and High Flow" has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS and current TS requirements and finds that the deviation from STS is acceptable and that all existing TS requirements have been incorporated into the proposed TS for this safety limit. In addition, the staff finds that the change to the current pressure transition point is consistent with the current plant designs as described in the UFSARs and is, therefore, acceptable. The staff finds that the proposed TSUP TS Section 2.1.B: "THERMAL POWER, High Pressure and High Flow" is acceptable.

3.3 Section 2.1.C: Reactor Coolant System Pressure

Proposed TS Section 2.1.C, "Reactor Coolant System Pressure," incorporates the requirements of STS Section 2.1.3, and current TS Section 1.2 for Dresden and current TS Section 1.2.A. for Quad Cities. Plant specific values for the safety limit are included to be consistent with the UFSAR for the plants. The current TS do not have specific required action or applicability statements for this specific safety limit. Proposed actions and applicability statements have been incorporated into the proposed TS in accordance with STS guidelines and format. There are no changes and/or deviations from STS guidance in the proposed TS and the proposed TS have incorporated all the current TS requirements concerning this safety limit.

The current TS for Quad Cities states that compliance with the safety limit for the Reactor Coolant System Pressure is measured in the reactor vessel steam space. In the proposed TS, the licensee has adopted the STS nomenclature and indicates that compliance with the safety limit shall be

measured in the reactor vessel steam dome. The instrumentation used to verify compliance has not changed from the current TS to the proposed TS and, therefore, the staff finds that the proposed nomenclature change from steam space to steam dome to match STS guidelines is acceptable. The proposed TS changes are administrative in nature and, therefore, do not reduce the margin of safety for Dresden or Quad Cities Stations.

The staff finds that the proposed TS 2.1.C: "Reactor Coolant System Pressure" has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS and current TS requirements and finds that the proposed TS do not deviate from STS guidelines and do not relax any existing TS requirements. Therefore, the staff finds the proposed TSUP TS Section 2.1.C, "Reactor Coolant System Pressure" acceptable.

3.4 Section 2.1.D: Reactor Vessel Water Level

Proposed TS Section 2.1.D, "Reactor Vessel Water Level" incorporates the requirements of STS Section 2.1.4 and current TS Section 1.1.D for both stations. Plant specific values for the listed parameters are included in the proposed TS to be consistent to the UFSAR for the plants. The current TS do not contain specific required action or applicability requirements. Proposed actions and applicability statements have been incorporated into the proposed TS in accordance with the STS guidelines and format. The proposed TS have incorporated all the current TS requirements concerning this safety limit.

The proposed TS conservatively specifies the Reactor Vessel Water Level Safety Limit as greater than or equal to twelve inches above top of active fuel (TAF). STS Section 2.1.4 only specifies TAF. The proposed TS maintains the current TS value and is consistent with the UFSAR which provides a point which can be adequately monitored and provides adequate margin for effective action. The staff, therefore, finds the deviation from the STS acceptable.

The staff finds that the proposed TS 2.1.D, "Reactor Vessel Water Level" has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS and current TS requirements and finds that the proposed deviation between the proposed TS and the STS is acceptable. In addition, the staff finds that the proposed TS does not relax any existing TS requirements. Therefore, the staff finds the proposed TSUP TS Section 2.1.D, "Reactor Vessel Water Level" acceptable.

3.5 Power Transient Safety Limit

Current TS 1.1.C contains a requirement for Power Transient Safety Limit. The proposed TS has eliminated this requirement. The STS guidelines have also eliminated this requirement. The current TS safety limit assumes the high neutron flux set point is exceeded for some period of time without a reactor scram. The Reactor Protection System (RPS) at Dresden and Quad Cities provides a highly reliable system which provides a reactor scram when neutron flux exceeds a given setpoint. This safety limit is assumed to be exceeded in the current analysis in the UFSAR. In addition, in accordance with 10 CFR

50.62, additional equipment has been added to Dresden and Quad Cities to mitigate the effects of an anticipated transient without scram (ATWS) independent of the RPS. The Power Transient Safety Limit was in the original TS prior to the implementation of the modifications to meet 10 CFR 50.62. The safety limits recommended by STS guidelines and adopted in Sections 2.0, 3.2, and 3.11 of the proposed TSUP TS are sufficient to provide a conservative margin of safety against conditions which could represent a challenge to the fuel cladding integrity and/or reactor coolant system integrity. The requirement for a Power Transient Safety Limit is outdated and other safety limits and safety systems provide equivalent protection. The staff has determined that the Power Transient Safety Limit is not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act. Further, it does not fall within any of the four criteria set forth in the Commission's Final Policy Statement discussed in Section 2.0 above. The staff, therefore, finds removal of the current TS Power Transient Safety Limit acceptable.

3.6 Section 2.2.A: Reactor Protection System (RPS) Instrumentation Setpoints

Proposed TS Section 2.2.A, "Reactor Protection System (RPS) Instrumentation Setpoints," has been reformatted and incorporates the requirements of STS Section 2.2.1. Proposed TS 2.2.A contains the trip setpoints for the reactor protection system instrumentation. The functional units specified in the proposed TS Section 2.2.A, Table 2.2.A-1 represent instrumentation which is used to protect the integrity of the reactor core and reactor coolant system. The trip setpoints specified in proposed TS Section 2.2.A, Table 2.2.A-1 have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist in mitigating the consequences of accidents. The proposed TS incorporates the existing safety limits from the current Dresden and Quad Cities TS, respectively. The proposed TS has been reformatted into the Table format of the STS guidelines. The current TS limiting safety system settings (LSSS) are currently divided into Section 2.1, "Fuel Cladding Integrity," and Section 2.2, "Reactor Coolant System." All of these safety limits have been incorporated into the proposed TS Table 2.2.A-1. To be consistent with the format guidelines of the STS, safety limits functional units contained in current Dresden TS Table 3.1.1 and Quad Cities TS Table 3.1-1, 3.1-2, and 3.1-3 have been relocated to proposed TS Table 2.2.A-1. Proposed TS Section 2.2.A and associated Table 2.2.A-1 have been formatted in accordance with the STS guidelines. The following provides a discussion of the deviations between the proposed TS and the STS guidelines and an evaluation of differences and/or relocations of the current TS requirements.

Proposed TS 2.2.A and Table 2.2.A-1, Action statements, do not adopt STS nomenclature for "Allowable Values." The proposed TS have replaced "Allowable Values" with the current TS nomenclature, "Trip Level Settings." The proposed TS "Trip Setpoint" is equivalent to STS "Allowable Values." The STS also contain values for Trip Setpoint which are lower than the STS Allowable Values. Operation with a trip set less conservative than its STS

Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analysis. The STS requirements for "Trip Setpoint" have not been adopted in the proposed TS nor are they included in the current TS. This deviation from STS guidelines maintains consistency with current TS requirements and nomenclature. These values/parameters (STS "Trip Setpoint") are more appropriately controlled administratively in other documents, i.e., UFSAR. The proposed TS has included the appropriate OPERABILITY limitation for the parameter ("Trip Setpoint") and this value is retained in the proposed TS. Changes to the acceptance criteria detailed in these documents will be controlled by the provisions of 10 CFR 50.59. In addition, including the STS allowable values may cause confusion with plant operators. The staff, therefore, finds this deviation from the STS guidelines acceptable.

FUNCTIONAL UNIT - IRM inoperative scram

The "IRM Inoperative Scram 'N/A'" has been included in the proposed TS based on the plant design and to ensure consistency in function and format to current industry practice. This additional requirement deviates from STS guidance, but is merely an administrative enhancement to the STS requirements and has been added to maintain consistency in format to the TSUP requirements for APRMs which include an inoperative statement (STS Table 2.2.1-1, Item 2.d). Because the proposed changes and deviations from STS do not reduce existing plant safety margins, the staff finds the proposed TS changes and deviations from STS guidelines acceptable.

FUNCTIONAL UNIT - Average Power Range Monitor

Proposed TS Table 2.2.A-1, Item 2.a for "Setdown Neutron Flux - High" is equivalent to the current TS Section 2.1.A.2. STS terminology ("Neutron Flux - Upscale, Setdown") has been reworded in the proposed TS to be consistent with existing plant terminology and the current design requirements. The Average Power Range Monitor (APRM) trip remains at 15 percent of rated thermal power. The proposed TS for "Flow Biased Neutron Flux - High" deviates from STS guidelines to be consistent to the current TS.

TSUP Table 2.2.A-1, Item 2.b is changed as follows:

1. STS terminology is modified to be consistent to the current TS. The setpoints are maintained and the proposed TS deviates from STS terminology in order to maintain the current TS requirements. STS terminology ("Flow Biased Simulated Thermal Power - Upscale") is equivalent to the proposed TS ("Flow Biased Neutron Flux - High") and provides protection for the same requirement - the fuel cladding integrity safety limit. STS terminology ("High Flow Clamped") is equivalent to the proposed TS for Dresden ("High Flow Maximum") and provides protection for the same requirement.

2. The proposed TS section is separated into two categories depending on the number of recirculation pumps running. The STS requirements for single recirculation loop operation (SLO) and dual recirculation loop operation (DLO) are not explicitly delineated. Splitting the requirements for SLO and DLO enhances STS requirements by providing greater clarity to plant operations personnel and maintains the requirements of the current TS and the design requirements for each station. The staff, therefore, finds the above deviations from STS guidelines acceptable.

Quad Cities current TS have the "APRM Downscale" trip in TS Table 3.1-3, "Reactor Protection System (Scram) Instrumentation Requirements Run Mode." This requirement (APRM Downscale Trip) was not retained in the proposed TS. The current action associated with this requirement is that "the APRM downscale trip function is automatically bypassed when the Intermediate Range Monitor (IRM) instrumentation is operable and not high." Removal of the APRM/IRM companion scram eliminates the APRM downscale scram which occurs in the Run mode with the simultaneous IRM scram that occurs with IRMs "high" or inoperable. This requirement provides no enhancement to safety and its elimination has been previously approved by the NRC staff (Dresden Units 2 and 3, Amendment Nos. 100/96 by letter dated August 24, 1988) from the Dresden TS. The proposed changes clarify the intent of the current Quad Cities specification by clearly defining the scram functions needed to be operable in each mode of operation and does not involve any modification of the reactor protection system wiring or circuitry. Proper overlap between the IRMs and APRMs is not affected. The staff has determined that the requirement for the APRM Downscale trip is not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act. Further, it does not fall within any of the four criteria set forth in the Commission's Final Policy Statement discussed in Section 2.0 above. Based on the above evaluation the staff finds that the deletion of the APRM downscale trip in the proposed TS is acceptable.

The requirements contained in the current Dresden and Quad Cities TS Section 2.1 for the operability and surveillance testing for both the APRM scram and the APRM Rod Block Trip Setting have been relocated to TSUP TS Section 3.2.E. The acceptability of Section 3.2.E of the proposed TS will be evaluated along with the other proposed TS for Section 3/4.2. This change is acceptable.

<u>FUNCTIONAL UNIT - Reactor Vessel Water Level - Low</u>

The proposed TS Table 2.2.A-1, "Reactor Vessel Water Level - Low" deviates from STS terminology (STS specifies "Reactor Vessel Water Level - Low, Level 3"). This deviation from STS guidance is based on current plant design and terminology at Dresden and Quad Cities Stations, is administrative in nature and does not compromise plant safety in any way. The proposed specification provides protection for the same requirements as STS - the fuel cladding integrity and reactor coolant system pressure safety limits. Therefore, the staff finds that maintaining the current TS terminology concerning the functional unit is an acceptable deviation from the STS guidelines.

The "Reactor Vessel Water Level - Low Low ECCS Initiation Trip Point," currently TS item 2.1.D, has been relocated to proposed TS 3.2.B. The staff finds this change acceptable.

FUNCTIONAL UNIT - Main Steam Line Radiation - High

The "Main Steam Line Radiation" trip function is included in proposed TS Table 2.2.A-1 in accordance with STS guidelines. The trip setpoint of 15 times normal full power background remains unchanged from existing TS requirements for Quad Cities Station. The trip setpoint for Dresden of three times normal full power background is also unchanged from the current TS. The Dresden "Main Steam Line Radiation" trip setpoint for Unit 2 uses a dual setting dependent on the status of the Hydrogen Addition system. This deviates from the STS guidelines which do not account for hydrogen addition. Both setpoints remain unchanged from the current Dresden TS. Since the existing TS requirements are maintained and ensure existing plant safety margins are maintained, there is no reduction in plant safety proposed by these changes. The staff, therefore, finds the deviation from STS acceptable.

FUNCTIONAL UNIT - Scram Discharge Volume Water Level - High

The "Scram Discharge Volume (SDV) Water Level - High" setpoints are included in proposed TS Table 2.2.A-1. The specification is differentiated by the type of instrumentation for the high level trip. The differential pressure switches and the thermal switches or float switches are identified with the corresponding trip levels. The proposed setpoints are given in gallons of water in the instrument volume. In the current Dresden TS, the scram discharge volume trip setpoints are given in inches. For Dresden, the trip setpoints are different for Unit 2 and Unit 3 due to design differences. The current TS values in inches are equivalent to the proposed TS values expressed in gallons. The STS specifies the setpoint by "percent of full scale." The change in terms in the proposed TS does not adversely affect the current safety margins and provides protection for the same requirements - ensuring sufficient volume exists in the SDV system to accommodate a scram. Since existing requirements are maintained and ensure existing plant safety margins are maintained, there is no reduction in plant safety proposed by these changes. The staff, therefore, finds the proposed changes acceptable.

FUNCTIONAL UNIT - Turbine EHC Control Oil Pressure Low

Current TS Section 2.1.I, "Turbine Low Electro Hydraulic Control (EHC) Pressure" scram function is included in proposed Section 2.2.A and Table 2.2.A-1. The proposed specification is renamed "Turbine EHC Control Oil Pressure - Low." The STS requirements for this setpoint do not exist. The setpoint of 900 psig remains unchanged from the current TS. As such, the proposed TS provides equivalent protection to existing requirements. Since existing TS requirements and existing plant safety margins are maintained, there is no reduction in plant safety. The staff, therefore, finds this deviation from STS guidelines acceptable.

FUNCTIONAL UNIT - Turbine Condenser Vacuum Low

Current TS 2.1.J, "Condenser Low Vacuum Scram" (Quad Cities only) is included in proposed Table 2.2.A-1. For Dresden Station, the "Condenser Low Vacuum Scram" is currently in Table 3.1.1, "Reactor Protection Instrumentation," and is relocated to proposed Table 2.2.A-1. Although this requirement is not included within the STS, ComEd has conservatively retained the "Condenser Low Vacuum Scram" function in the proposed TS.

In addition, the current setpoint for Dresden "Condenser Low Vacuum Scram" is 23 inches mercury (Hg) while Quad Cities' "Condenser Low Vacuum Scram" is 21 inches Hg. To maintain consistency between Dresden and Quad Cities where no design difference exists, ComEd proposes to reduce Dresden's setpoint to 21 inches Hg. The loss of the main condenser vacuum results in the closure of the turbine stop valves and disables the turbine bypass function should the loss of vacuum continue to drop significantly. Closure of the stop valves without bypass, in turn, causes a pressure transient, neutron flux rise and increase in fuel cladding surface heat flux. However, closure of the turbine stop valves results in a reactor scram signal as soon as the valve position reaches 90 percent open. Thus, for a loss of condenser vacuum, the low condenser vacuum scram anticipates the subsequent stop valve closure and thereby results in a less severe reactor pressurization and fuel thermal heat flux rise. An instantaneous and total loss of vacuum represents the most severe loss of condenser vacuum which can be postulated. Such an event would simultaneously cause fast closure of the turbine stop valves and prevent the bypass function while minimizing the beneficial effects of the anticipatory low condenser vacuum scram function. The consequences of this postulated event are bounded by the turbine trip without bypass transient, which assumes no credit for the anticipatory scram. Since the most severe licensing basis turbine trip without bypass event does not take credit for the low condenser vacuum scram, a change of the scram setpoint value will not affect the consequences for this limiting event. The staff, therefore, finds this reduction of the existing TS setpoint acceptable.

Conclusion

The staff finds that the proposed TS for Section 2.2.A, "RPS Instrumentation Setpoints" has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS and current TS requirement and finds that all deviations from STS are acceptable and that the relaxation of the current TS maintains the design requirements for each station. Therefore, the staff finds the proposed TSUP TS Section 2.2.A, "RPS Instrumentation Setpoints" is acceptable.

3.7 Technical Specification Bases

The staff has reviewed the proposed TS Bases for proposed TS 2.0, "Safety Limits and Limiting Safety System Settings." The Bases have been developed in accordance with the guidelines of the STS. Therefore, the staff finds the proposed TS Bases acceptable.

3.8 Open Items

All items in the current Dresden and Quad Cities TS Section 2.1 were not incorporated into Section 2.0 of the proposed TS. The remaining items from Section 2.1 will be relocated to other sections of the proposed TS. These should be left as open items, contingent upon their implementation in future TSUP amendments.

1. Current TS 2.1.B, "APRM Rod Block Trip Setting" will be relocated to proposed TSUP TS 3.2.E.

2. Current TS 2.2.B, "Primary System Safety Valve Nominal Settings" will be relocated to proposed TSUP TS 3.6.F.

3. Current TS 2.1.D, "Reactor Water Level - Low Low ECCS Initiation Trip Point" will be relocated to proposed TSUP TS 3.2.B.

4. Current TS 2.1.H, "Main Steamline Pressure Initiation of Main Steamline Isolation Valve Closure" will be relocated to proposed TSUP TS section 3.2.A.

3.9 SUMMARY FOR SECTION 2.0

The proposed TSUP Section 2.0, "Safety Limits and Limiting Safety System Settings," clarifies the requirements of the current TS through the adoption of STS format, adds more restrictive requirements, and incorporates changes to correct inconsistencies with the STS. In addition, all deviations between the STS guidelines and proposed TS and the relaxations of current TS have been determined by the staff to be consistent with the plant design requirements and adequate justification has been provided to support these deviations and relaxations. Therefore, the staff finds the proposed amendment for TSUP TS Section 2.0 acceptable.

4.0 EVALUATION OF PROPOSED TS SECTION 3/4.11 "POWER DISTRIBUTION LIMITS"

The following sections provide the staff's evaluation of the TS changes reflected in proposed TSUP TS Section 3/4.11. The proposed TS 3/4.11 is a new specification that results from the consolidation and rearrangement of the existing TS requirements for power distribution limits. The majority of the proposed specifications are currently contained in TS Section 3/4.5, "ECCS Systems." The new specifications have been developed using the STS guidelines. The proposed TS are evaluated below.

4.1 TS 3/4.11.A: AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

Proposed TS 3/4.11.A, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," incorporates the requirements of STS 3/4.2.1 and requirements from current TS Section 3/4.5.I. Plant specific values for the listed parameters are included which are consistent with the UFSAR for the plants. Both Dresden and Quad Cities had placed the cycle specific parameters in a Core Operating Limits

Report (COLR) through a previous amendment request. The COLR is controlled in Section 6.0 of the current and proposed TSUP TS. The proposed changes are consistent with the STS guidelines and Generic Letter 88-16, "Removal of Cycle Specific Parameters From Technical Specifications."

4.1.1 Applicability

The applicability of the current and proposed TS is equivalent based upon comparison of the SR of current TS Section 4.5.I and the applicability statements of proposed TS Section 3.11.A. The applicability of the proposed TS enforces APLHGR determination when the reactor is in OPERATIONAL MODE 1 and greater than or equal to 25 percent rated thermal power. Similarly, the SR in the current TS only require the APLHGR values to be determined when the reactor is at greater than or equal to 25 percent rated thermal power.

4.1.2 LCO

The proposed LCO for 3.11.A deviates in terminology from STS for Dresden Station only. The proposed Dresden TS uses the terminology "bundle average exposure" as opposed to STS terminology "AVERAGE PLANAR EXPOSURE." The STS terminology is not applicable to the fuel design used at Dresden Station. The proposed deviation is administrative in nature, is consistent to STS requirements, and does not affect the safe operation of Dresden Station. The staff, therefore, finds the deviation acceptable.

4.1.3 Actions

The proposed actions for TS Section 3/4.11.A are adopted from STS, but are separated for clarification purposes. The proposed wording is consistent with STS guidelines.

An alteration of the current TS action statement was identified in the proposed TS. Current TS Section 3.5.I requires corrective action to be taken within 15 minutes if it is determined by normal surveillance that the limiting value for APLGHR is being exceeded. This is consistent with proposed TS Section 3.11.A Action 1. The current TS then allows a 2-hour period in which to return APLHGR within prescribed limits, and, if this is not achieved, the reactor shall be brought to cold shutdown within 36 hours. The proposed TS alters the action requirement by stating that if the limits are not restored within 2 hours, then the reactor is to be brought to less than 25 percent rated thermal power within 4 hours. The action of the proposed TS is consistent with STS guidance.

The APLHGR limits are primarily derived from fuel design evaluations and loss-of-coolant accident (LOCA) and transient analyses that are assumed to occur at high power levels. Design calculations and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5 percent to 15 percent rated thermal power when entry into Mode 2 occurs. When in Mode 2, the intermediate range monitor scram function provides prompt scram initiation during any

significant transient, thereby effectively removing any APLHGR limit compliance concern. Therefore, at thermal power levels less than 25 percent rated thermal power, the reactor is operating with substantial margin to the APLHGR limits and the level of safety provided by the APLHGR limit is minimal. In addition, the current TS SR only requires APLHGR determination above 25 percent rated thermal power. Therefore, the applicability of the thermal limits is only above 25 percent rated thermal power. Reducing power beyond the proposed TS action requirement (i.e., to cold shutdown within 36 hours) results in unnecessary thermal cycles of the reactor vessel and unnecessary challenges to safety systems. Therefore, the staff finds that the deletion of the requirement to bring the reactor to cold shutdown is acceptable.

Based on the above evaluation, the staff finds that the proposed TS for Section 3/4.11.A, APLHGR, has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS and current TS requirements and finds that all deviations from the STS are acceptable and the proposed TS do not relax any existing TS requirements. Therefore, the staff finds the proposed TS 3/4.11.A acceptable.

4.2 TS 3/4.11.B, "APRM SETPOINTS"

Proposed TS 3/4.11.B, "APRM SETPOINTS," incorporates the requirements of STS 3/4.2.2. Plant specific values for the listed parameters are included to be consistent with the UFSAR for the plants. The proposed TS incorporates the existing TS requirements concerning APRM setpoints from current TS Sections 2.1.A.1, 2.1.A.2, and 2.1.B. The proposed changes are consistent with the STS and Generic Letter 88-16, "Removal of Cycle Specific Parameters from Technical Specifications." Both Dresden and Quad Cities had placed the cycle specific parameters in a COLR through a previous amendment request. The COLR is controlled in Section 6.0 of the current and proposed TS. There are no significant changes and/or deviations from STS guidance or relaxation of current TS requirements. The proposed TS change is administrative in nature and, therefore, does not reduce the margin of safety for Dresden or Quad Cities Stations.

4.2.1 LCO

Proposed TS LCO 3.11.B adopts the STS format, but deviates due to plant specific/fuel vendor specific terminology/methodology. Enhancements to STS guidelines are made to avoid duplicating setpoints within the proposed TS and to more clearly delineate when the actions for the specification are to be implemented. The proposed LCO requirements are consistent with the current TS requirements in place at Dresden and Quad Cities Stations, and includes the intent of the STS guidelines.

Proposed TS 3.11.B is provided to require the APRM gain or APRM flow biased scram and rod block trip setpoints to be adjusted when operating under conditions of abnormal power peaking so that acceptable margin to the fuel cladding integrity limits are maintained. For Quad Cities, abnormal power peaking is represented by the Maximum Fraction of Limiting Power Density

(MFLPD) being greater than the Fraction of Rated Thermal Power (FRTP). For Dresden, abnormal power peaking is represented by the Fuel Design Limiting Ratio For Centerline Melt (FDLRC) being greater than 1.0. These thermal limits are different for Dresden and Quad Cities due to the different fuel vendor suppliers for the plants (General Electric Company (GE) currently supplies fuel to Quad Cities and Siemens Nuclear Power (SNP) currently supplies fuel to Dresden). The different fuel vendors utilize different requirements and terminology to describe the thermal limitations placed upon their respective fuel designs. However, the basis of these thermal limits remains equivalent - to maintain the integrity of the fuel cladding.

To maintain the appropriate margin under conditions of abnormal power peaking, either the APRM gain must be adjusted upward or the flow biased neutron flux upscale scram trip and rod block setpoints be reduced. This is accomplished by multiplying the APRM gain or setpoints by a factor that is representative of the reduction in margin to the fuel cladding integrity limits. Adjustment to the scram and rod block setpoints are made by multiplying the setpoint by the inverse of the factor for the APRM gains. This factor will be less than one and, therefore, will cause the setpoints to be lowered to maintain the margin. When the reactor is operating with normal peaking (i.e., FDLRC < 1.0 or MFLPD < FRTP) it is not necessary to modify the APRM flow biased scram or rod block setpoints. These requirements are consistent with current TS requirements and do not reduce existing plants safety margins.

4.2.2 Actions

The proposed Actions in TSUP 3.11.B are adopted from STS, but are separated for clarification purposes. The current TS do not contain an action statement associated with this TS. The licensee has adopted required actions in accordance with the STS guidelines. The proposed action requires that if FDLRC is greater than 1.0 for Dresden or MFLPD is greater than FRTP for Quad Cities, that within 6 hours the limit is restored, or the APRM setpoints in the proposed TS Section 2.2.A and 3.2.E are adjusted, or the APRM gains are adjusted. These thermal limits are different for Dresden and Quad Cities due to the different fuel vendor suppliers. The different fuel vendors utilize different requirements and terminology to describe the thermal limitations placed upon their respective fuel designs. However, the basis of these thermal limits remains equivalent to maintain the integrity of the fuel cladding. If the action provisions are not met, thermal power is required to be reduced to below 25 percent of rated thermal power. The proposed required actions in the TS are consistent with STS guidelines. Based on the above evaluation, the staff finds the proposed actions acceptable.

4.2.3 Surveillance Requirements

The proposed TS SRs are adopted from the STS guidelines. Proposed TS 4.11.B requires that the value of FDLRC (Dresden) or MFLPD and FRTP (Quad Cities) be checked at least once per 24 hours; initially within 12 hours after a thermal power increase of 15 percent or more, and once per 12 hours when the reactor is operating on a limiting control rod pattern. Proposed TSUP TS 4.11.B.4 is

added to stipulate that the provisions of proposed TS 4.0.D are not applicable per the requirement of GL 87-09. The proposed changes are consistent and/or more conservative than existing TS requirements; therefore, the proposed TS does not reduce the margin of safety for the plants. The staff, therefore, finds the proposed TS SR acceptable.

4.2.4 Conclusion

Based on the above evaluation, the staff finds the proposed TS 3/4.11.B, "APRM SETPOINTS" has adopted the guidelines of STS and has not relaxed any current TS requirements. Therefore, the staff finds the proposed TS acceptable.

4.3 TS 3/4.11.C: MINIMUM CRITICAL POWER RATIO

Proposed TS 3/4.11.C, "MINIMUM CRITICAL POWER RATIO," incorporates the requirements of STS 3/4.2.3 and current TS Section 3/4.5.L for Dresden and 3.5/4.5.K for Quad Cities. Plant specific values for the listed parameters are included which are consistent with the UFSAR for both Dresden and Quad Cities. The proposed changes are consistent with the STS and GL 88-16. Both Dresden and Quad Cities have cycle specific parameters placed in a COLR. The proposed amendment request is administrative in nature and, therefore, does not reduce the margin of safety for Dresden or Quad Cities Stations.

4.3.1 Action

The proposed action statement implements the requirements of the current TS with the exception that the proposed TS do not require the reactor to be taken to cold shutdown if the MCPR limit is not restored within the 2-hour time frame. The proposed TS alters the action requirement by stating that if the limits are not restored within 2 hours, then the reactor is to be brought to less than 25 percent rated thermal power within 4 hours. The action of the proposed TS is consistent with STS guidance.

The MCPR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Design calculations and operating experience have shown that as power is reduced, the margin to the required MCPR limits increases. This trend continues down to the power range of 5 percent to 15 percent rated thermal power when entry into Mode 2 occurs. When in Mode 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any MCPR limit compliance concern. Therefore, at thermal power levels less than 25 percent rated thermal power, the reactor is operating with substantial margin to the MCPR limits and the level of safety provided by the MCPR limit is minimal. In addition, the current TS SR only requires MCPR determination above 25 percent rated thermal power. Therefore, the applicability of the thermal limits is only above 25 percent rated thermal power. Reducing power beyond the proposed TS action requirement (i.e., to cold shutdown within 36 hours) results in unnecessary thermal cycles of the reactor vessel, and unnecessary challenges to safety systems. Therefore, the

deletion of the requirement to bring the reactor to cold shutdown is acceptable.

4.3.2 Surveillance Requirements

In proposed TS 4.11.C.1, the licensee proposed differing values for t_{ave} for Dresden and Quad Cities. The basis for the differing values is differing fuel vendors. The proposed values are consistent with current TS requirements and do not reduce the margin of safety for either Dresden or Quad Cities. The staff, therefore, finds this change acceptable.

4.3.3 Conclusion

Based on the above evaluation, the staff finds that proposed TS 3/4.11.C, "MINIMUM CRITICAL POWER RATIO" has adopted the guidelines of STS and the deviation from the current TS requirements does not reduce the margin of safety. Therefore, the staff finds the proposed TS acceptable.

4.4. TS 3/4.11.D: STEADY STATE LINEAR HEAT GENERATION RATE (Dresden)
TS 3/4.11.D: LINEAR HEAT GENERATION RATE (Quad Cities)

Proposed TS 3/4.11.D, "STEADY STATE LINEAR HEAT GENERATION RATE" (Dresden) and "LINEAR HEAT GENERATION RATE" (Quad Cities), incorporate the requirements of STS 3/4.2.4 and current TS 3/4.5.J for both stations. The difference in title for these proposed TS is due to the difference in fuel vendors at each site (Quad Cities - GE, Dresden - SNP). Plant specific values for the listed parameters are included which are consistent with the UFSAR for the plants. The proposed changes are consistent with the STS and Generic Letter 88-16. Both Dresden and Quad Cities have the cycle specific parameters placed in a COLR. The proposed TS change is administrative in nature and, therefore, does not reduce the margin of safety for Dresden or Quad Cities Stations.

The proposed action statement implements the requirements of the current TS with the exception that the proposed TS do not require the reactor to be taken to cold shutdown if the LHGR limit is not restored within the 2-hour time frame. The proposed TS alters the action requirement by stating that if the limits are not restored within 2 hours, then the reactor is to be brought to less than 25 percent rated thermal power within 4 hours. The action of the proposed TS is consistent with STS guidance.

The LHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Design calculations and operating experience have shown that as power is reduced, the margin to the required LHGR limits increases. This trend continues down to the power range of 5 percent to 15 percent rated thermal power when entry into Mode 2 occurs. When in Mode 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any LHGR limit compliance concern. Therefore, at thermal power levels less than 25 percent rated thermal power, the reactor is operating with substantial margin to the LHGR limits and the level of safety

provided by the LHGR limit is minimal. In addition, the current TS SR only requires LHGR determination above 25 percent rated thermal power. Therefore, the applicability of the thermal limits is only above 25 percent rated thermal power. Reducing power beyond the proposed TS action requirement (i.e., to cold shutdown within 36 hours) results in unnecessary thermal cycles of the reactor vessel, and unnecessary challenges to safety systems.

Based on its evaluation, the staff finds that the proposed TS 3/4.11.D, "STEADY STATE LINEAR HEAT GENERATION RATE" (Dresden) and 3/4.11.D, "LINEAR HEAT GENERATION RATE" (Quad Cities) have been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS and current TS requirement and finds that there are no significant deviations from STS and the deviation from current TS requirements does not reduce the margin of safety. Therefore, the staff finds the proposed TS 3/4.11.D, "STEADY STATE LINEAR HEAT GENERATION RATE" (Dresden) and 3/4.11.D, "LINEAR HEAT GENERATION RATE" (Quad Cities) are acceptable.

4.5 TS 3/4.11.E: TRANSIENT LINEAR HEAT GENERATION RATE (Dresden Only)

Proposed TS 3/4.11.E, "TRANSIENT LINEAR HEAT GENERATION RATE" (Dresden only), incorporates the requirements of STS 3/4.2.4 and current TS 3/4.5.K. Plant specific values for the listed parameters are included which are consistent with the UFSAR for the plant. This additional power distribution limit for Dresden is due to fuel vendor specific design differences. The proposed changes are consistent with the STS and Generic Letter 88-16. There are no significant changes and/or deviations from STS guidance. TS 3/4.11.E is unchanged from the current TS with the exception of an additional SR (TS 4.11.D.1) which is based on STS. There is no relaxation of current TS requirements. The proposed TS change is administrative in nature and, therefore, does not reduce the margin of safety for Dresden.

Based on the above evaluation, the staff finds that the proposed TS for Section 3/4.11.E, "TRANSIENT LINEAR HEAT GENERATION RATE" (Dresden only) has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS and current TS requirement and finds that there are no significant deviations from STS and the proposed TS do not relax any existing TS requirements. Therefore, the staff finds the proposed TS 3/4.11.E, "TRANSIENT LINEAR HEAT GENERATION RATE" (Dresden only) acceptable.

4.6 Technical Specification Bases

The staff has reviewed the proposed TS Bases for proposed TS 3/4.11, "POWER DISTRIBUTION LIMITS." The Bases have been developed in accordance with the guidelines of the STS. Therefore, the staff finds the proposed TS Bases acceptable.

4.7 CONCLUSION - PROPOSED TSUP SECTION 3/4.11 POWER DISTRIBUTION LIMITS

The proposed TS changes for TS 3/4.11, "Power Distribution Limits," are administrative in nature and do not adversely affect the safe operation of

Dresden or Quad Cities Stations. The proposed TS clarifies the requirements of the present TS through the adoption of STS format, adds more restrictive requirements, and incorporates changes to correct inconsistencies with the STS. As indicated above, the staff has evaluated all deviations between the proposed TS and the STS and found them acceptable. In addition, the staff compared the proposed TS with the current TS requirements and found no relaxation of current requirements. Therefore, the staff finds the proposed TS 3/4.11 is acceptable.

5.0 EVALUATION OF PROPOSED TS 3/4.12, "SPECIAL TEST EXCEPTIONS"

5.1 TS 3/4.12.A: PRIMARY CONTAINMENT INTEGRITY

Proposed TS 3/4.12.A, "PRIMARY CONTAINMENT INTEGRITY," incorporates the requirements of STS 3/4.10.1 and requirements from Section 3.7.A.2 of the current TS for both stations. Plant specific values for the listed parameters are included which are consistent with the UFSAR for the plants. The proposed TS change request does not reduce the margin of safety for Dresden or Quad Cities Stations.

The staff identified one deviation from STS in the proposed TS. The proposed specification allows the primary containment integrity requirements to be suspended for the purpose of performing low power physics tests with thermal power less than 1 percent of rated thermal power and the reactor coolant temperature less than 212 °F. The STS specifies a temperature requirement of 200 °F. A temperature of 212 °F is consistent with the current TS and is retained in the proposed TS for consistency with the proposed operational modes defined in proposed TSUP TS Section 1.0. The difference between 212 °F and 200 °F results in an insignificant change in water density so the coolant provides the same moderator effect and essentially an equivalent level of safety protection. The evaporation rate may be higher at 212 °F, but the airborne limits are controlled by 10 CFR 20 limits and are encompassed with the safety analysis. The staff, therefore, finds this deviation from STS guidelines acceptable.

A relaxation of current TS was also identified. Current TS Section 3.7.A.2 requires the reactor power to be less than five MWt to perform low power physics tests. Five MWt equals approximately 0.2 percent rated thermal power. The proposed TS uses 1 percent rated thermal power for consistency to STS. The intent of the current TS is to complete the low power physics testing at the low end of the APRM scale. To meet this intent, 1 percent rated thermal power and 0.2 percent rated thermal power fulfill the same purpose. Five MWt is very difficult to detect on the APRM scale and, therefore, was never measured accurately. Performing these tests at 1 percent versus 0.2 percent power will have no effect on the safety limits. The staff has determined that the change from the current TS does not reduce existing plant safety margins, and, therefore, the change is acceptable.

Based on its evaluation, the staff finds that the proposed TS 3/4.12.A, "PRIMARY CONTAINMENT INTEGRITY" has been reformatted adopting the STS

guidelines. The staff has reviewed the proposed TS against the STS and current TS and finds that the deviation from STS is acceptable and the relaxation of existing TS requirements is acceptable. Therefore, the staff finds the proposed TSUP TS Section 3/4.12.A, "PRIMARY CONTAINMENT INTEGRITY" acceptable.

5.2 Section 3/4.12.B: SHUTDOWN MARGIN DEMONSTRATIONS

Proposed TS 3/4.12.B, "SHUTDOWN MARGIN DEMONSTRATIONS," incorporates the requirements of STS 3/4.10.3 and current TS 3.3.B.3.b. Plant specific values for the listed parameters are included which are consistent with the UFSAR. There is no relaxation of current TS requirements. The proposed amendment request is administrative in nature and, therefore, does not reduce the margin of safety for Dresden or Quad Cities Stations. The staff identified the following deviations from STS:

Proposed TS 4.12.B.1 deviates from STS guidelines by stipulating that the shorting links also be removed to demonstrate operability. The proposed SR is consistent with current requirements and removes an inconsistency between LCO 3.12.B.1 and the SR. The staff, therefore, finds this deviation from STS guidelines acceptable.

Proposed TS 4.12.B.2 deviates from STS guidelines by allowing "other technically qualified individual" to be used instead of STS terminology which specifies "other technically qualified member of the unit technical staff." This proposed terminology clarifies ambiguities in the STS that are unnecessary and does not constitute a reduction in current requirements. The proposed change is administrative in nature and, therefore, does not reduce the safety margin of the plant. The staff, therefore, finds this deviation from the STS guidelines acceptable.

Based on its evaluation, the staff finds that the proposed TS 3/4.12.B, "SHUTDOWN MARGIN DEMONSTRATIONS" has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS and current TS requirement and finds that all deviations from STS are acceptable and the proposed TS do not relax any existing TS requirements. Therefore, the staff finds the proposed TS 3/4.12.B, "SHUTDOWN MARGIN DEMONSTRATIONS" acceptable.

5.3 Technical Specification Bases

The staff has reviewed the proposed Bases for TS 3/4.12. The proposed Bases have been prepared using the guidelines of the STS. The staff finds these proposed Bases acceptable.

5.4 CONCLUSION

The changes in proposed TS 3/4.12, "Special Test Exceptions," are administrative in nature and do not adversely affect the safe operation of Dresden and Quad Cities Stations. The proposed TS clarifies the requirements

of the present TS through the adoption of STS format, adds more restrictive requirements, and incorporates changes to correct inconsistencies with the STS. Based on the above evaluation, the staff finds the proposed TS 3/4.12 acceptable.

6.0 SUMMARY

The proposed TS for Sections 2.0, 3/4.11 and 3/4.12 will be clearer and easier to use as a result of the adaptation of the STS format. The changes result in additional limitations, restrictions, or changes based on generic guidance. It is the staff's assessment that the changes proposed in this amendment do not pose any decrease in safety, or an increase in the probability of an analyzed or unanalyzed accident. The revised TS changes do not reduce the existing margin of safety set forth by the current TS. Therefore, the staff finds the proposed TS changes acceptable.

7.0 STATE CONSULTATION

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

8.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (60 FR 24906). Accordingly, the amendments meet 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 amendments.

9.0 CONCLUSION

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

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