May 23, 1997

Ms. Irene Johnson, Acting Manager
Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M95822 AND M95823)

Dear Ms. Johnson:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 177 to Facility Operating License No. DPR-29 and Amendment No. 175 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 June 10, 1996, as supplemented by a letter dated February 17, 1997.

The amendments revise the Technical Specifications (TS) to reflect the transition from General Electric Company (GE) to Siemens Power Corporation (SPC) as the fuel supplier for the Quad Cities Nuclear Power Station, Units 1 and 2.

As an administrative action by the Commission that only involves the format of the licenses and does not authorize any activities outside the scope of your application and supplement, the NRC has amended the licenses to include an Appendix C that lists additional license conditions. The additional license condition as a result of the review of this application reflects the relocation of the contents of TS 5.4 to the Updated Final Safety Analysis Report.

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

Sincerely,

Original signed by:

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

Docket Nos. 50-254, 50-265

OFO1

Enclosures: 1. Amendment No. 177 to DPR-29
2. Amendment No. 175 to DPR-30
3. Safety Evaluation

cc w/encl: see next page

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*concurred by memo dated 3/13/97; no major changes and on previous page.

DOCUMENT NAME: QUAD\QC95822.AMD

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I. Johnson
Commonwealth Edison Company

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Unit Nos. 1 and 2

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

COMMONWEALTH EDISON COMPANY

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 177
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 June 10, 1996, as supplemented on February 17, 1997, 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, paragraph 3.B. is hereby amended and paragraph 3.N is added to Facility Operating License No. DPR-29* to read as follows:

---

*Page 6 is attached, for convenience, for the composite license to reflect this change.

3.B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 177 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3.N. Additional Conditions

The additional conditions contained in Appendix C, as revised through Amendment No. 177 , are hereby incorporated into this license. Commonwealth Edison Company shall operate the facility in accordance with the Additional Conditions.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days.

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:
1. License page 6
2. Appendix C - Additional Conditions
3. Technical Specifications

Date of Issuance: May 23, 1997

The above Surveillance Requirements shall be successfully demonstrated prior to entering into MODE 2 on the first plant startup following the fifteenth refueling outage (Q1R15).

N. <u>Additional Conditions</u>

The additional conditions contained in Appendix C, as revised through Amendment No. 177, are hereby incorporated into this license. Commonwealth Edison Company shall operate the facility in accordance with the Additional Conditions.

4. This license is effective as of the date of issuance, and shall expire at midnight, December 14, 2012.

FOR THE ATOMIC ENERGY COMMISSION

Original signed by:

A. Giambusso, Deputy Director
for Reactor Projects
Division of Licensing

Enclosures:
Appendixes A and B --
Technical Specifications
Appendix C -- Additional Conditions

APPENDIX C

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-29

Commonwealth Edison Company shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Conditions	Implementation Date
177	This amendment authorizes the licensee to incorporate in the Updated Final Safety Analysis Report (UFSAR), the description of the Reactor Coolant System design pressure, temperature and volume that was removed from Technical Specification Section 5.4, and evaluated in staff safety evaluation dated May 23, 1997.	60 days from the date of issuance.

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

COMMONWEALTH EDISON COMPANY

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 175
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 June 10, 1996, as supplemented on February 17, 1997, 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, paragraph 3.B. is hereby amended and paragraph 3.M is added to Facility Operating License No. DPR-30* to read as follows:

---

*Page 6 is attached, for convenience, for the composite license to reflect this change.

3.B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 175 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3.M. Additional Conditions

The additional conditions contained in Appendix C, as revised through Amendment No. 175 , are hereby incorporated into this license. Commonwealth Edison Company shall operate the facility in accordance with the Additional Conditions.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days.

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:
1. License page 6
2. Appendix C - Additional Conditions
3. Technical Specifications

Date of Issuance: May 23, 1997

e. Surveillance Requirement 4.4.A.4.a - Standby Liquid Control Initiation.

f. Surveillance Requirement 4.9.A.8.h - Emergency Diesel Generator 24 hour test.

g. Surveillance Requirement 4.9.A.8.c - Emergency Diesel Generator full load reject test.

h. Surveillance Requirement 4.1.A.1 - Logic System Functional Test for Reactor Protection System Instrumentation, Table 4.1.A-1, Item 5, Main Steam Line Isolation Valve Closure RPS Calibration.

Each of the above Surveillance Requirements shall be successfully demonstrated prior to entering into MODE 2 on the first plant startup following the fourteenth refueling outage (Q2R14).

M. Additional Conditions

The additional conditions contained in Appendix C, as revised through Amendment No. 175 , are hereby incorporated into this license. Commonwealth Edison Company shall operate the facility in accordance with the Additional Conditions.

4. This license is effective as of the date of issuance, and shall expire at midnight, December 14, 2012.

FOR THE ATOMIC ENERGY COMMISSION

Original signed by:

A. Giambusso, Deputy Director
for Reactor Projects
Division of Licensing

Enclosures:
Appendixes A and B --
Technical Specifications
Appendix C -- Additional Conditions

Date of Issuance: December 14, 1972

APPENDIX C

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-30

Amendment Number	Additional Conditions	Implementation Date
175	This amendment authorizes the licensee to incorporate in the Updated Final Safety Analysis Report (UFSAR), the description of the Reactor Coolant System design pressure, temperature and volume that was removed from Technical Specification Section 5.4, and evaluated in staff safety evaluation dated May 23 1997.	60 days from the date of issuance.

ATTACHMENT TO LICENSE AMENDMENT NOS. 177 AND 175

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 and contain marginal lines indicating the area of change.

REMOVE	INSERT
I	I
II	II
XIII	XIII
XIV	XIV
XVII	XVII
XXVI	XXVI
1-3	1-3
1-4	1-4
1-5	1-5
1-6	1-6
1-7	1-7
B 2-1	B 2-1
B 2-2	B 2-2
B 2-5	B 2-5
B 2-6	B 2-6
B 2-8	B 2-8
B 3/4.2-1	B 3/4.2-1
B 3/4.2-2	B 3/4.2-2
3/4.3-2	3/4.3-2
B 3/4.3-2	B 3/4.3-2
B 3/4.3-3	B 3/4.3-3
B 3/4.3-4	B 3/4.3-4
B 3/4.3-6	B 3/4.3-6
B 3/4.3-7	B 3/4.3-7
B 3/4.6-3	B 3/4.6-3
3/4.11-2	3/4.11-2
B 3/4.11-1	B 3/4.11-1
B 3/4.11-2	B 3/4.11-2
B 3/4.11-3	B 3/4.11-3
---	B 3/4.11-4
5-5	5-5
5-5a	---
5-6	5-6
6-15	6-15
6-16	6-16
6-16a	6-16a

SECTION PAGE

Section 1 DEFINITIONS

DEFINITIONS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

Section 5 DESIGN FEATURES

ADMINISTRATIVE CONTROLS

SECTION	PAGE

3/4.11 POWER DISTRIBUTION LIMITS

3/4.12 SPECIAL TEST EXCEPTIONS

1.0 DEFINITIONS

FRACTION OF LIMITING POWER DENSITY (FLPD)

The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle (applicable to GE fuel).

FRACTION OF RATED THERMAL POWER (FRTP)

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

FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1-1.

FUEL DESIGN LIMITING RATIO (FDLRX)

The FUEL DESIGN LIMITING RATIO (FDLRX) shall be the limit used to assure that the fuel operates within the end-of-life steady-state design criteria by, among other items, limiting the release of fission gas to the cladding plenum (applicable to SPC fuel).

FUEL DESIGN LIMITING RATIO FOR CENTERLINE MELT (FDLRC)

The FUEL DESIGN LIMITING RATIO FOR CENTERLINE MELT (FDLRC) shall be 1.2 times the LHGR at a given location divided by the product of the TRANSIENT LINEAR HEAT GENERATION RATE limit and the FRACTION OF RATED THERMAL POWER (applicable to SPC fuel).

IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE shall be: a) leakage into primary containment collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or b) leakage into the primary containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

LIMITING CONTROL ROD PATTERN (LCRP)

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

LINEAR HEAT GENERATION RATE (LHGR)

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

1.0 DEFINITIONS

LOGIC SYSTEM FUNCTIONAL TEST (LSFT)

A LOGIC SYSTEM FUNCTIONAL TEST (LSFT) shall be a test of all required logic components, i.e., all required relays and contacts, trip units, solid state logic elements, etc, of a logic circuit, from as close to the sensor as practicable up to, but not including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping or total system steps so that the entire logic system is tested.

MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)

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

MINIMUM CRITICAL POWER RATIO (MCPR)

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

OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Specification 6.8 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specification 6.9.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE

An OPERATIONAL MODE, i.e., MODE, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1-2.

PHYSICS TESTS

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

1.0 DEFINITIONS

PRESSURE BOUNDARY LEAKAGE

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

PRIMARY CONTAINMENT INTEGRITY (PCI)

PRIMARY CONTAINMENT INTEGRITY (PCI) shall exist when:

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

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

   2) Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are open under administrative control as permitted by Specification 3.7.D.

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

c. Each primary containment air lock is in compliance with the requirements of Specification 3.7.C.

d. The primary containment leakage rates are maintained within the limits of Specification 3.7.A.

e. The suppression chamber is in compliance with the requirements of Specification 3.7.K.

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

PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analysis, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

RATED THERMAL POWER (RTP)

RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 2511 MWT.

1.0 DEFINITIONS

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME shall be the time interval for each trip function from the opening of the sensor contact up to and including the opening of the trip actuator.

REPORTABLE EVENT

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

SECONDARY CONTAINMENT INTEGRITY (SCI)

SECONDARY CONTAINMENT INTEGRITY (SCI) shall exist when:

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

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

   2) Closed by at least one manual valve, blind flange, or deactivated automatic damper secured in its closed position, except as permitted by Specification 3.7.O.

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

c. The standby gas treatment system is in compliance with the requirements of Specification 3.7.P.

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

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

f. The pressure within the secondary containment is less than or equal to the value required by Specification 4.7.N.1.

SHUTDOWN MARGIN (SDM)

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

SOURCE CHECK

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

1.0 DEFINITIONS

THERMAL POWER
THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR)
The TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR) limit protects against fuel centerline melting and 1% plastic cladding strain during transient conditions throughout the life of the fuel (applicable to SPC fuel).

TRIP SYSTEM
A TRIP SYSTEM shall be an arrangement of instrument CHANNEL trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A TRIP SYSTEM may require one or more instrument CHANNEL trip signals related to one or more plant parameters in order to initiate TRIP SYSTEM action. Initiation of protective action may require the tripping of a single TRIP SYSTEM or the coincident tripping of two TRIP SYSTEMs.

UNIDENTIFIED LEAKAGE
UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

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 fuel damage is calculated to 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 x 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 x 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 such that, with the limiting fuel assembly operating at the MCPR Safety Limit, more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. This includes consideration of 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

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 local power is limited to 7.7% of rated bundle 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.

BASES

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. The top of active fuel is defined to be 360 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.

3/4.2 INSTRUMENTATION

In addition to reactor protection instrumentation which initiates a reactor scram (Sections 2.2 and 3/4.1), protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or which terminates operator errors before they result in serious consequences. The objectives of these specifications are to assure the effectiveness of the protective instrumentation when required and to prescribe the trip settings required to assure adequate performance. As indicated, one CHANNEL may be required to be made inoperable for brief intervals to conduct required surveillance. Some of the settings have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. It should be noted that the setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations. Surveillance requirements for the instrumentation are selected in order to demonstrate proper function and OPERABILITY. Additional instrumentation for REFUELING operations is identified in Sections 3/4.10.B.

Current fuel designs incorporate slight variations in the length of the active fuel and, thus, the actual top of active fuel, when compared with the original fuel designs. Safety Limits, instrument water level setpoints, and associated LCOs refer to the top of active fuel. In these cases, the top of active fuel is defined as 360 inches above vessel zero. Licensing analyses, both accident and transient, utilize this definition for the automatic initiation and manual intervention associated with these events.

3/4.2.A Isolation Actuation Instrumentation

The isolation actuation instrumentation automatically initiates closure of appropriate isolation valves and/or dampers, which are necessary to prevent or limit the release of fission products from the reactor coolant system, the primary containment and the secondary containment in the event of a loss-of-coolant accident or other reactor coolant pressure boundary (RCPB) leak. The parameters which result in isolation of the secondary containment also actuate the standby gas treatment system. The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of primary and secondary containment and RCPB system isolation. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. Redundant sensor input signals for each parameter are provided for initiation of isolation (one exception is standby liquid control system initiation).

The reactor low level instrumentation is set to trip at greater than or equal to 144 inches above the top of active fuel (which is defined to be 360 inches above vessel zero). This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps. For this trip setting and a 60-second valve closure time, the valves will be closed before perforation of the cladding occurs, even for the maximum break.

3/4.2.B Emergency Core Cooling System Actuation Instrumentation

The emergency core cooling system (ECCS) instrumentation generates signals to automatically actuate those safety systems which provide adequate core cooling in the event of a design basis transient or accident. The instrumentation which actuates the ECCS is generally arranged in a one-out-of-two taken twice logic circuit. The logic circuit is composed of four CHANNEL(s) and each CHANNEL contains the logic from the functional unit sensor up to and including all relays which actuate upon a signal from that sensor. For core spray and low pressure coolant injection, the divisionally powered actuation logic is duplicated and the redundant components are powered from the other division's power supply. The single-failure criterion is met through provisions for redundant core cooling functions, e.g., sprays and automatic blowdown and high pressure coolant injection. Although the instruments are listed by system, in some cases the same instrument is used to send the actuation signal to more than one system at the same time.

For effective emergency core cooling during small pipe breaks, the high pressure coolant injection (HPCI) system must function since reactor pressure does not decrease rapidly enough to allow either core spray or the low pressure coolant injection (LPCI) system to operate in time. The automatic pressure relief function is provided as a backup to HPCI, in the event HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing or calibration and also minimizes the risk of inadvertent operation, i.e., only one instrument CHANNEL out-of-service.

3/4.2.C ATWS - RPT Instrumentation

The anticipated transient without scram (ATWS) recirculation pump trip (RPT) provides a means of limiting the consequences of the unlikely occurrence of a failure to scram concurrent with the associated anticipated transient. The response of this plant to this postulated event falls within the bounds of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO24222, dated December 1979. Tripping the recirculation pumps adds negative reactivity by increasing steam voiding in the core area as core flow decreases.

3/4.2.D Reactor Core Isolation Cooling Actuation Instrumentation

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

3.3 - LIMITING CONDITIONS FOR OPERATION

B. Reactivity Anomalies

The reactivity equivalence of the difference between the actual critical control rod configuration and the predicted critical control rod configuration shall not exceed 1% Δk/k.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the reactivity equivalence difference exceeding 1% Δk/k, within 12 hours perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.

With the provisions of the ACTION above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

B. Reactivity Anomalies

The reactivity equivalence of the difference between the actual critical control rod configuration and the predicted critical control rod configuration shall be verified to be less than or equal to 1% Δk/k:

1. During the first startup following CORE ALTERATION(s), and

2. At least once per 31 effective full power days.

BASES

During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

3/4.3.B Reactivity Anomalies

During each fuel cycle, excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern selected base states to the predicted rod inventory at that state. Alternatively, monitored K_{eff} can be compared with the predicted K_{eff} as calculated by an approved 3-D core simulator code. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons. Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% $\Delta k/k$. Deviations in core reactivity greater than 1% $\Delta k/k$ are not expected and require thorough evaluation. A 1% $\Delta k/k$ reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

3/4.3.C Control Rod OPERABILITY

Control rods are the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the control rods provide the means for reliable control of reactivity changes to ensure the specified acceptable fuel design limits are not exceeded. This specification, along with others, assures that the performance of the control rods in the event of an accident or transient, meets the assumptions used in the safety analysis. Of primary concern is the trippability of the control rods. Other causes for inoperability are addressed in other Specifications following this one. However, the inability to move a control rod which remains trippable does not prevent the performance of the control rod's safety function.

The specification requires that a rod be taken out-of-service if it cannot be moved with drive pressure. Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

BASES

Control rods that are inoperable due to exceeding allowed scram times, but are movable by control rod drive pressure, need not be disarmed electrically if the shutdown margin provisions are met for each position of the affected rod(s).

If the rod is fully inserted and then disarmed electrically or hydraulically, it is in a safe position of maximum contribution to shutdown reactivity. (Note: To disarm the drive electrically, four amphenol-type plug connectors are removed from the drive insert and withdrawal solenoids, rendering the drive immovable. This procedure is equivalent to valving out the drive and is preferred, as drive water cools and minimizes crud accumulation in the drive.). If it is disarmed electrically in a non-fully inserted position, that position shall be consistent with the SHUTDOWN MARGIN limitation stated in Specification 3.3.A. This assures that the core can be shut down at all times with the remaining control rods, assuming the strongest OPERABLE control rod does not insert. The occurrence of more than eight inoperable control rods could be indicative of a generic control rod drive problem which requires prompt investigation and resolution.

In order to reduce the potential for Control Rod Drive (CRD) damage and more specifically, collet housing failure, a program of disassembly and inspection of CRDs is conducted during or after each refueling outage. This program follows the recommendations of General Electric SIL-139 with nondestructive examination results compiled and reported to General Electric on collet housing cracking problems.

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

3/4.3.D Control Rod Maximum Scram Insertion Times;

3/4.3.E Control Rod Average Scram Insertion Times; and

3/4.3.F Four Control Rod Group Scram Insertion Times

These specifications ensure that the control rod insertion times are consistent with those used in the safety analyses. The control rod system is analyzed to bring the reactor subcritical at a rate fast enough to prevent fuel damage, i.e., to prevent the MCPR from becoming less than the fuel cladding integrity Safety Limit. The analyses demonstrate that if the reactor is operated within the limitation set in Specification 3.11.C, the negative reactivity insertion rates associated with the scram performance result in protection of the MCPR Safety Limit.

Analysis of the limiting power transient shows that the negative reactivity rates, resulting from the scram with the average response of all the drives, as given in the above specification, provide the required protection, and MCPR remains greater than the fuel cladding integrity SAFETY LIMIT. In the analytical treatment of most transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 210 milliseconds. Approximately 90 milliseconds after neutron flux reaches the trip point, the pilot scram valve

BASES

solenoid de-energizes and 120 milliseconds later the control rod motion is estimated to actually begin. However, 200 milliseconds rather than 120 milliseconds is conservatively assumed for this time interval in the transient analyses and is also included in the allowable scram insertion times specified in Specifications 3.3.D, 3.3.E, and 3.3.F.

The performance of the individual control rod drives is monitored to assure that scram performance is not degraded. Transient analyses are performed for both Technical Specification Scram Speed (TSSS) and nominal scram speed (NSS) insertion times. These analyses result in the establishment of the cycle dependent TSSS MCPR limits and NSS MCPR limits presented in the COLR. Results of the control rod scram tests performed during the current cycle are used to determine the operating limit for MCPR. Following completion of each set of scram testing, the results will be compared with the assumptions used in the transient analysis to verify the applicability of the MCPR operating limits. Prior to the initial scram time testing for an operating cycle, the MCPR operating limits will be based on the TSSS insertion times.

Individual control rod drives with excessive scram times can be fully inserted into the core and de-energized in the manner of an inoperable rod drive provided the allowable number of inoperable control rod drives is not exceeded. In this case, the scram speed of the drive shall not be used as a basis in the re-determination of thermal margin requirements. For excessive average scram insertion times, only the individual control rods in the two-by-two array which exceed the allowed average scram insertion time are considered inoperable.

The scram times for all control rods are measured at the time of each refueling outage. Experience with the plant has shown that control drive insertion times vary little through the operating cycle; hence no re-assessment of thermal margin requirements is expected under normal conditions. The history of drive performance accumulated to date indicates that the 90% insertion times of new and overhauled drives approximate a normal distribution about the mean which tends to become skewed toward longer scram times as operating time is accumulated. The probability of a drive not exceeding the mean 90% insertion time by 0.75 seconds is greater than 0.999 for a normal distribution. The measurement of the scram performance of the drives surrounding a drive, which exceeds the expected range of scram performance, will detect local variations and also provide assurance that local scram time limits are not exceeded. Continued monitoring of other drives exceeding the expected range of scram times provides surveillance of possible anomalous performance.

The test schedule provides reasonable assurance of detection of slow drives before system deterioration beyond the limits of Specification 3.3.C. The program was developed on the basis of the statistical approach outlined above and judgement. The occurrence of scram times within the limits, but significantly longer than average, should be viewed as an indication of a systematic problem with control rod drives, especially if the number of drives exhibiting such scram times exceeds eight, which is the allowable number of inoperable rods.

BASES

3/4.3.J Control Rod Drive Housing Support

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Section 4.6.3.5 of the UFSAR. This support is not required if the reactor coolant system is at atmospheric pressure, since there would then be no driving force to rapidly eject a drive housing.

3/4.3.K Scram Discharge Volume Vent and Drain Valves

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required. The operability of the scram discharge volume vent and drain valves assures the proper venting and draining of the volume, so that water accumulation in the volume does not occur. These specifications designate the minimum acceptable level of scram discharge volume vent and drain valve OPERABILITY, provide for the periodic verification that the valves are open, and for the testing of these valves under reactor scram conditions during each refueling outage.

3/4.3.L Rod Worth Minimizer

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not have sufficient reactivity worth to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. These low power (up to the LPSP) sequences are verified during the cycle reload analysis to ensure that the 280 cal/gm limit is not exceeded. The requirement that an operator follow these sequences is supervised by the RWM or a second technically qualified individual. These sequences are developed to limit reactivity worth of control rods and, together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data. Therefore, the energy deposited during a postulated rod drop accident is significantly less than that required for rapid fuel dispersal.

The analysis of the control rod drop accident was originally presented in Sections 7.9.3, 14.2.1.2, and 14.2.1.4 of the original SAR. Improvements in analytical capability have allowed a more refined analysis of the control rod drop accident which is discussed below.

Every operating cycle the peak fuel rod enthalpy rise is determined by comparing cycle specific parameters with the results of parametric analyses. This peak fuel rod enthalpy is then compared to the analysis limit of 280 cal/gm to demonstrate compliance for that operating cycle. If the cycle

BASES

specific parameters are outside the range used in the parametric study, an extension of the enthalpy may be required. Some of the cycle specific parameters used in the analysis are: maximum control rod worth, Doppler coefficient, effective delayed neutron fraction and maximum four bundle local peaking factor. The methodology used for the control rod drop accident analysis is NRC-approved and is part of the license bases referenced in Specification 6.9.A.6.

The rod worth minimizer provides automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted, i.e., it limits operator deviations from planned withdrawal sequences (reference UFSAR Section 7.7.2). It serves as a backup to procedural control of control rod worth. In the event that the rod worth minimizer is out-of-service when required, a second licensed operator or other technically qualified individual who is present at the reactor console can manually fulfill the control rod pattern conformance function of the rod worth minimizer. In this case, the normal procedural controls are backed up by independent procedural controls to assure conformance.

3/4.3.M Rod Block Monitor

The rod block monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator, who withdraws control rods according to a written sequence. The specified restrictions with one channel out-of-service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

3/4.3.N Economic Generation Control System

Operation of the facility with the economic generation control system (EGC) (automatic flow control) is limited to the range of 65% to 100% of rated core flow. In this flow range and above 20% of RATED THERMAL POWER, the reactor could safely tolerate a rate of change of load of 8 MWe/sec (reference UFSAR Section 7.7.3.2). Limits within the EGC and the flow control system prevent rates of change greater than approximately 4 MWe/sec. When EGC is in operation, this fact will be indicated on the main control room console.

BASES

3/4.6.E Safety Valves

3/4.6.F Relief Valves

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of safety valves are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary. The overpressure protection system must accommodate the most severe pressurization transient. SPC methodology determines the most limiting pressurization transient each cycle. Evaluations have determined that the most severe transient is the closure of all the main steam line isolation valves followed by a reactor scram on high neutron flux. The analysis results demonstrate that the design safety valve capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of the reactor pressure vessel design pressure.

The relief valve function is not assumed to operate in response to any accident, but are provided to remove the generated steam flow upon turbine stop valve closure coincident with failure of the turbine bypass system. The relief valve opening pressure settings are sufficiently low to prevent the need for safety valve actuation following such a transient.

Each of the five relief valves discharge to the suppression chamber via a dedicated relief valve discharge line. Steam remaining in the relief valve discharge line following closure can condense, creating a vacuum which may draw suppression pool water up into the discharge line. This condition is normally alleviated by the vacuum breakers; however, subsequent actuation in the presence of an elevated water leg can result in unacceptably high thrust loads on the discharge piping. To prevent this, the relief valves have been designed to ensure that each valve which closes will remain closed until the normal water level in the relief valve discharge line is restored. The opening and closing setpoints are set such that all pressure induced subsequent actuation are limited to the two lowest set valves. These two valves are equipped with additional logic which functions in conjunction with the setpoints to inhibit valve reopening during the elevated water leg duration time following each closure.

Each safety/relief valve is equipped with diverse position indicators which monitor the tailpipe acoustic vibration and temperature. Either of these provide sufficient indication of safety/relief valve position for normal operation.

3/4.6.G Leakage Detection Systems

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. Limits on leakage from the reactor coolant pressure boundary are required so that appropriate action can be taken before the integrity of the reactor coolant pressure boundary is impaired. Leakage detection systems for the reactor coolant system are provided to alert the operators when leakage rates above the normal background levels are detected and also to supply quantitative measurement of leakage rates.

3.11 - LIMITING CONDITIONS FOR OPERATION

B. TRANSIENT LINEAR HEAT GENERATION RATE

The TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR) shall be maintained such that the FUEL DESIGN LIMITING RATIO for CENTERLINE MELT (FDLRC)[a] is less than or equal to 1.0. Where FDLRC is equal to:

$$\frac{(LHGR)(1.2)}{(TLHGR)(FRTP)}$$

APPLICABILITY:

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

ACTION:

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

1. Restore FDLRC to less then or equal to 1.0, or
2. Adjust the flow biased APRM setpoints specified in Specifications 2.2.A and 3.2.E by 1/FDLRC, or
3. Adjust[b] each APRM gain such that the APRM readings are ≥100 times 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. TRANSIENT LINEAR HEAT GENERATION RATE

The value of FDLRC[a] 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.

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a For GE fuel, MFLPD/FRTP is substituted for FDLRC. Adjustments are based on the lowest APRM setpoint or highest APRM reading resulting from the two limits.

b 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.

BASES

3/4.11.A AVERAGE PLANAR LINEAR HEAT GENERATION RATE

GE Fuel

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 6.9.

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.

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.

SPC Fuel

This specification assures that the peak cladding temperature of SPC fuel following a postulated design basis loss-of-coolant accident will not exceed the Peak Cladding Temperature (PCT) and maximum oxidation limits specified in 10CFR50.46. The calculational procedure used to establish the Average Planar Linear Heat Generation Rate (APLHGR) limits is based on a loss-of-coolant accident analysis.

The 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 not strongly influenced by the rod-to-rod power distribution within the assembly.

BASES

The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for two-loop and single-loop operation are specified in the Core Operating Limits Report (COLR).

3/4.11.B TRANSIENT LINEAR HEAT GENERATION RATE

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 $\geq 1\%$ 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 or FDLRC indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

SPC Fuel

The Fuel Design Limiting Ratio for Centerline Melt (FDLRC) is incorporated to protect the above criteria at all power levels considering events which cause the reactor power to increase to 120% of rated thermal power.

The scram settings must be adjusted to ensure that the TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR) is not violated for any power distribution. This is accomplished using FDLRC. The scram setting is decreased in accordance with the formula in Specification 3.11.B, when FDLRC is greater than 1.0.

The adjustment may also be accomplished by increasing the gain of the APRM by FDLRC. This provides the same degree of protection as reducing the trip setting by 1/FDLRC by raising the initial APRM reading closer to the trip setting such that a scram would be received at the same point in a transient as if the trip setting had been reduced.

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 are analyzed to determine which result in the largest reduction in the CRITICAL POWER RATIO (CPR). The type of transients evaluated are 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 6.9.

MCPR Operating Limits are presented in the CORE OPERATING LIMITS REPORT (COLR) for both Nominal Scram Speed (NSS) and Technical Specification Scram Speed (TSSS) insertion times. The negative reactivity insertion rate resulting from the scram plays a major role in providing the required protection against violating the Safety Limit MCPR during transient events. Faster scram insertion times provide greater protection and allow for improved MCPR performance. The application of NSS MCPR limits utilizes measured data that is faster than the times required by the Technical Specifications, while the TSSS MCPR limits provide the necessary protection for the slowest allowable average scram insertion times identified in Specification 3.3.E. The measured scram times are compared with the nominal scram insertion times and the Technical Specification Scram Speeds. The appropriate operating limit is applied, as specified in the COLR.

For core flows less than rated, the MCPR Operating Limit established in the specification is adjusted to provide protection of the Safety Limit MCPR in the event of an uncontrolled recirculation flow increase to the physical limit of the pump. Protection is provided for manual and automatic flow control by applying the appropriate flow dependent MCPR limits presented in the COLR. The MCPR Operating Limit for a given power/flow state is the greater value of MCPR as given by the rated conditions MCPR limit or the flow dependent MCPR limit. For automatic flow control, in addition to protecting the Safety Limit MCPR during the flow run-up event, protection is provided to prevent exceeding the rated flow MCPR Operating Limit during an automatic flow increase to rated core flow.

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.

BASES

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 fo 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.

5.0 DESIGN FEATURES

5.3 REACTOR CORE

Fuel Assemblies

5.3.A The reactor core shall contain 724 fuel assemblies. Each assembly consists of a matrix of Zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. The assemblies may contain water rods or water boxes. Limited substitutions of Zircaloy or ZIRLO filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

Control Rod Assemblies

5.3.B The reactor core shall contain 177 cruciform shaped control rod assemblies. The control material shall be boron carbide powder (B_4C) and/or hafnium metal. The control rod assembly shall have a nominal axial absorber length of 143 inches.

5.0 DESIGN FEATURES

5.4 [INTENTIONALLY BLANK]

4. Radioactive Effluent Release Report

   The Radioactive Effluent Release Report covering the operation of the facility during the previous calendar year shall be submitted prior to April 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the facility. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

5. Monthly Operating Report

   Routine reports of operating statistics and shutdown experience, including documentation of all challenges to safety valves or safety/relief valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

6. CORE OPERATING LIMITS REPORT

   a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

      (1) The Control Rod Withdrawal Block Instrumentation for Table 3.2.E-1 of Specification 3.2.E.

      (2) The Average Planar Linear Heat Generation Rate (APLHGR) Limit for Specification 3.11.A.

      (3) The Linear Heat Generation Rate (LHGR) for Specification 3.11.D.

      (4) The Minimum Critical Power Operating Limit (including scram insertion time) for Specification 3.11.C. This includes rated and off-rated flow conditions.

   b. The analytical methods used to determine the operating limits shall be those previously reviewed and approved by the NRC in the latest approved revision or supplement of topical reports:

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

      (2) Commonwealth Edison Topical Report NFSR-0085, "Benchmark of BWR Nuclear Design Methods," (latest approved revision).

(3) Commonwealth Edison Topical Report NFSR-0085, Supplement 1, "Benchmark of BWR Nuclear Design Methods - Quad Cities Gamma Scan Comparisons," (latest approved revision).

(4) Commonwealth Edison Topical Report NFSR-0085, Supplement 2, "Benchmark of BWR Nuclear Design Methods - Neutronic Licensing Analyses," (latest approved revision).

(5) Advanced Nuclear Fuels Methodology for Boiling Water Reactors, XN-NF-80-19(P)(A), Volume 1, Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.

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

(7) Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description, XN-NF-80-19(P)(A), Volume 3, Revision 2, Exxon Nuclear Company, January 1987.

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

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

(10) Qualification of Exxon Nuclear Fuel for Extended Burnup Supplement 1: Extended Burnup Qualification of ENC 9x9 BWR Fuel, XN-NF-82-06(P)(A) Supplement 1, Revision 2, Advanced Nuclear Fuels Corporation, May 1988.

(11) Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels 9x9-IX and 9x9-9X BWR Reload Fuel, ANF-89-014(P)(A), Revision 1 and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, October 1991.

(12) Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A) Revision 1, and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.

(13) Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A), Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.

(14) ANFB Critical Power Correlation, ANF-1125(P)(A) and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.

(15) Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, ANF-524(P)(A), Revision 2, Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.

(16) COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analyses, ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.

(17) Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993.

(18) Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods," Revision 0, Supplements 1 and 2, December 1991, March 1992, and May 1992, respectively; SER letter dated March 22, 1993.

(19)*ComEd letter, "ComEd Response to NRC Staff Request for Additional Information (RAI) Regarding the Application of Siemens Power Corporation ANFB Critical Power Correlation to Coresident General Electric Fuel for LaSalle Unit 2 Cycle 8 and Quad Cities Unit 2 Cycle 15, NRC Docket No.'s 50-373/374 and 50-254/265", J.B. Hosmer to U.S. NRC, July 2, 1996, transmitting the topical report, Application of the ANFB Critical Power Correlation to Coresident GE Fuel for Quad Cities Unit 2 Cycle 15, EMF-96-051(P), Siemens Power Corporation - Nuclear Division, May 1996, and related information.

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.9.B Special Reports

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

*Applicable to Unit 2 for cycle 15 only.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 177 TO FACILITY OPERATING LICENSE NO. DPR-29

AND AMENDMENT NO. 175 TO FACILITY OPERATING LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY

AND

MIDAMERICAN ENERGY COMPANY

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-254 AND 50-265

1.0 INTRODUCTION

By letter dated June 10, 1996, as supplemented by letter dated February 17, 1997, Commonwealth Edison Company (ComEd, the licensee) requested changes to the Quad Cities Nuclear Power Station, Units 1 and 2, Technical Specifications (TS). Quad Cities, Units 1 and 2, currently use General Electric (GE) fuel and licensing methodologies. Siemens Power Corporation (SPC) fuel and licensing methodologies are planned for use at Quad Cities beginning with Unit 2 Cycle 15 and Unit 1 Cycle 16. The Siemens' loss-of-coolant accident (LOCA) methodology and fuel assembly designs are approved for use at other licensed boiling water reactor (BWR) facilities. Thus, the proposed changes to the Quad Cities, Units 1 and 2, TS represent the transition from one NRC-approved methodology to another NRC-approved methodology. Other minor editorial changes are also proposed.

By letter dated February 17, 1997, ComEd submitted revisions that were also required for the approval of TS changes for SPC fuel transition for LaSalle County Station, Units 1 and 2. The revision lists the specific NRC approval date and the revision/supplement for each of the new topical reports, and revises the Section 5.3.A description of fuel assemblies. This letter provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

2.1 Mechanical Design

The ATRIUM-9B fuel design is a 9x9 lattice design with an internal water box to enhance neutron moderation. The ATRIUM-9B fuel design was analyzed and assessed by Siemens according to the approved methodology, entitled "Generic Mechanical Design for Advanced Nuclear Fuels 9x9-IX and 9x9-9X BWR Reload Fuel," ANF-89-014(P)(A), Revision 1 Supplement 1. The ATRIUM-9B fuel mechanical design followed the approved methodology and, therefore, is acceptable for Quad Cities Nuclear Power Station, Unit 2 Cycle 15.

2.2 Definitions

Linear heat generation rate (LHGR) limits are monitored for GE fuel by the parameters fraction of limiting power density (FLPD) and maximum fraction of limiting power density (MFLPD). The licensee proposed to add "(applicable to GE fuel)" to the end of each of these definitions to distinguish GE parameters from SPC parameters. SPC uses Fuel Design Limiting Ratio For Centerline Melt (FDLRC) and Fuel Design Limiting Ratio (FDLRX) to monitor LHGR. The licensee has proposed to add the following definitions of FDLRC and FDLRX, which are applicable to SPC fuel:

> FUEL DESIGN LIMITING RATIO FOR CENTERLINE MELT (FDLRC)
> The FUEL DESIGN LIMITING RATIO FOR CENTERLINE MELT (FDLRC) shall be 1.2 times the LHGR at a given location divided by the product of the TRANSIENT LINEAR HEAT GENERATION RATE limit and the FRACTION OF RATED THERMAL POWER (applicable to SPC fuel).
>
> FUEL DESIGN LIMITING RATIO (FDLRX)
> The FUEL DESIGN LIMITING RATIO (FDLRX) shall be the limit used to assure that the fuel operates within the end-of-life steady-state design criteria by, among other items, limiting the release of fission gas to the cladding plenum (applicable to SPC fuel).

The licensee has proposed to delete the definition of Rod Density. Rod density will be replaced by critical control rod configuration in order to make use of the capability to monitor actual K_{eff} versus predicted K_{eff}.

The licensee proposed to add the definition of the SPC transient LHGR limit. The proposed definition is as follows:

> TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR)
> The TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR) limit protects against fuel centerline melting and 1 percent plastic cladding strain during transient conditions throughout the life of the fuel (applicable to SPC fuel).

The staff notes that the GE LHGR limits will be applied to the co-resident GE fuel in the core and the SPC LHGR limits will be applied to the SPC fuel in the core. The existing LHGR TS Bases will be modified to show applicability to both GE and SPC fuel. The staff finds these definition changes acceptable.

2.3 Safety Limits Bases

The licensee proposed an editorial change to Section 2.1, third paragraph, of the Bases. The current wording states "the fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an AOO." The proposed change would consist of the following:

> The fuel cladding integrity limit is set such that no fuel damage is calculated to occur as a result of an AOO.

The staff concludes that the change clarifies the meaning of the sentence and is acceptable.

In Section 2.1.B, Thermal Power, High Pressure and High Flow, the licensee proposed editorial changes to paragraph one. The current wording of the last sentence states that "the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9 percent of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties." The editorial change would have the last two sentences of paragraph one consist of the following:

> Therefore, the fuel cladding integrity Safety Limit is defined such that, with the limiting fuel assembly operating at the MCPR Safety Limit, more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. This includes consideration of the power distribution within the core and all uncertainties.

The staff notes that this wording is consistent with the bases in Improved Standard Technical Specifications, NUREG-1433, Revision 1 and, therefore, it is acceptable.

2.4 Limiting Safety System Settings Bases

In Section 2.2.A.1, Reactor Protection System Instrumentation Setpoints - Intermediate Range Monitor, Neutron Flux - High, the licensee proposed an editorial change to the third paragraph. The sentence with the proposed change states that "the results of this analysis show that the reactor is scrammed and peak power is limited to 1 percent of rated power, thus maintaining minimum critical power ratio (MCPR) above the fuel cladding integrity Safety Limit." The licensee proposed to change 1 percent to 7.7 percent to reflect the correct value in the Updated Final Safety Analysis Report (UFSAR) and Safety Analysis Report (SAR). Section 7.6.1 of the UFSAR cites, in graphical form, 7.7 percent as the power level at which the IRMs terminate the low power RWE transient. With two other editorial changes, the proposed statement will read as follows:

> The results of this analysis show that the reactor is scrammed and peak local power is limited to 7.7% of rated bundle power, thus maintaining MCPR above the fuel cladding integrity Safety Limit.

Based on this information, the staff finds this editorial change acceptable.

In Section 2.2.A.4, Reactor Protection System Instrumentation Setpoints - Reactor Vessel Water Level - Low, the licensee proposed to add a clarification of the top of active fuel at the end of the last paragraph. The proposed last sentence of the paragraph would read "The top of active fuel is defined to be

360 inches above vessel zero." This statement is consistent with footnotes and other sections of the bases and, therefore, is acceptable.

2.5 Instrumentation Bases

The licensee proposed a clarification to Section 3/4.2, Instrumentation. The licensee proposed to add the following paragraph to Section 3/4.2.

> Current fuel designs incorporate slight variations in the length of the active fuel and, thus, the actual top of active fuel, when compared with the original fuel designs. Safety Limits, instrument water level setpoints, and associated LCOs refer to the top of active fuel. In these cases, the top of active fuel is defined as 360 inches above vessel zero. Licensing analyses, both accident and transient, utilize this definition for the automatic initiation and manual intervention associated with these events.

The proposed additions provide a clear definition and use of the top of active fuel reference point. The staff finds this addition to the bases acceptable.

2.6 Reactivity Control Limiting Conditions For Operation And Bases

TS 3.3.B, Reactivity Anomalies, currently requires that the reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall not exceed 1 percent $\Delta k/k$. In addition, Surveillance Requirement 4.3.B requires that the reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall be verified to be less than or equal to 1 percent $\Delta k/k$. This limit ensures that plant operation is maintained within the assumptions of the safety analyses.

The licensee has proposed to replace "ROD DENSITY" in both the TS and the surveillance requirement with "critical control rod configuration." This proposed change is necessitated by the additional proposal to include an alternative to monitoring reactivity anomalies in the TS bases. Both the SPC core monitoring code, Powerplex, and the GE Core Monitoring Code (CMC) provides the capability to monitor actual K_{eff} versus predicted K_{eff}. The licensee stated that the change from Rod Density to critical control rod configuration was necessary in order to use this capability. The staff notes that this method is currently used at Dresden to monitor reactivity anomalies. Thus, the following will be added to Section 3/4.3.B, Reactivity Anomalies Bases:

> Alternatively, monitored K_{eff} can be compared with the predicted K_{eff} as calculated by an approved 3-D core simulator code.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The staff notes that this proposed change only revises the current method of measuring the difference between predicted and monitored core reactivity and does not change the required limit; therefore, the change to TS 3/4.3.B and its Bases is acceptable.

In Sections 3/4.3.D, 3/4.3.E, and 3/4.3.F, Control Rod Maximum Scram Insertion Times, Control Rod Average Scram Insertion Times, and Four Control Rod Group Scram Insertion Times, of the TS Bases, the licensee proposed to remove the following comments:

first paragraph: "(as adjusted for statistical variation in the observed data);"

second paragraph: "In the statistical treatment of the limiting transients, a statistical distribution of total scram delay is used rather than the bounding value described above;"

third paragraph: "Observed plant data or Technical Specification limits were used to determine the average scram performance used in the transient analyses, and the results of each set of control rod scram tests performed during the current cycle are compared against earlier results to verify that the performance of the control rod insertion system has not changed significantly;" and

fourth paragraph: "If test results should be determined to fall outside of the statistical population defining the scram performance characteristics used in the transient analyses, a re-determination of thermal margin requirements is undertaken as required by Specification 3.11.C. A smaller test sample than that required by these specifications is not statistically significant and should not be used in the re-determination of thermal margins."

The licensee stated that the above information is based on past data, which is a GE methodology. Current SPC methods used to evaluate the 5 percent, 20 percent, 50 percent and 90 percent control rod scram insertion times, collected during the performance of the scram timing Surveillance Requirement 4.3.D, will replace the above information as follows:

> Transient analyses are performed for both Technical Specification Scram Speed (TSSS) and nominal scram speed (NSS) insertion times. These analyses result in the establishment of the cycle dependent TSSS MCPR limits and NSS MCPR limits presented in the COLR. Results of the control rod scram tests performed during the current cycle are used to determine the operating limit for MCPR. Following completion of each set of scram testing, the results will be compared with the assumptions used in the transient analysis to verify the applicability of the MCPR operating limits. Prior to the initial scram time testing for an operating cycle, the MCPR operating limits will be based on the TSSS insertion times.

The NSS insertion times are typically faster than the TSSS insertion times; thus, the NSS insertion times are used to calculate the NSS MCPR operating limit. If any of the average scram insertion times do not meet the NSS times, the TSSS MCPR operating limit is used. TS 3.11.C, Minimum Critical Power Ratio, requires that the MCPR shall be equal to or greater than the MCPR operating limit specified in the COLR. These changes to the bases clarify the SPC methodology that will be used at Quad Cities and how it will be used to meet TS 3.11.C. Based on this information, the changes to Section 3/4.3.D, 3.E, and 3.F bases are acceptable.

In Section 3/4.3.L, Rod Worth Minimizer, the licensee proposed editorial changes to the first paragraph of the Bases. Currently, the first two sentences state that "control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow these sequences is supervised by the RWM or a second technically qualified individual." The editorial changes would result in a passage that reads as follows:

> Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not have sufficient reactivity worth to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. These low power (up to the LPSP) sequences are verified during the cycle reload analysis to ensure that the 280 cal/gm limit is not exceeded. The requirement that an operator follow these sequences is supervised by the RWM or a second technically qualified individual.

The licensee also proposed editorial changes to the last sentence of the third paragraph. The last sentence will be replaced by the following:

> The methodology used for the control rod drop accident analysis is NRC-approved and is part of the license bases referenced in Specification 6.9.A.6.

The staff notes that these editorial changes clarify that the control rod sequences used during the cycle are not all written prior to cycle startup, but are verified to meet the 280 cal/gm limit up to the Low Power Set Point (LPSP). This verification is completed using NRC-approved methodologies which are referenced in TS 6.9.A.6. Based on the above, the staff finds the editorial changes acceptable.

2.7 Primary System Boundary Bases

In Sections 3/4.6.E and 3/4.6.F, Safety Valves and Relief Valves, the licensee proposed to add the following sentence to the middle of the first paragraph:

> SPC methodology determines the most limiting pressurization transient each cycle.

The addition of this statement clarifies the SPC methodology for analyzing the overpressurization event and, therefore, is acceptable.

2.8 Power Distribution Limits - Limiting Conditions For Operation And Bases

As stated above, FLPD and MFLPD are LHGR terms that are specific to GE fuel. The co-resident GE fuel in the core will be monitored by the GE fuel dependent LHGR limits, FLPD and MFLPD, and the SPC fuel will be monitored by the SPC LHGR limits, FDLRC and FDLRX. The staff notes that the SPC fuel is protected from off rated transients by the application of FDLRC to the Average Power Range Monitor (APRM) setpoints. Based on this, the licensee proposed to revise TS 3/4.11.B, Average Power Range Monitor Setpoints, to reflect the SPC FDLRC limit and the requirement to modify the APRM setpoints if FDLRC is greater than 1.0 for SPC fuel. The proposed change to TS 3/4.11.B is identical to Dresden, Units 2 and 3, TS 3/4.11.B except for the following: 1) footnote (a) is added to the appropriate FDLRC statements, and 2) the current footnote (a) will become footnote (b).

The proposed footnote (a) will state the following:

> For GE fuel, MFLPD/FRTP is substituted for FDLRC. Adjustments are based on the lowest APRM setpoint or highest APRM reading resulting from the two limits.

The staff notes that TS 3/4.11.B will be titled Transient Linear Heat Generation Rate instead of the current title, Average Power Range Monitor Setpoints. The staff has reviewed the Dresden TS 3/4.11.B and compared it to the proposed changes above. Since the TLHGR and FDLRC limits for SPC fuels are applied to the APRM setpoints, the staff finds the propose changes to TS 3/4.11.B acceptable.

The licensee also proposed changes to the TS Bases Sections 3/4.11.A, 3/4.11.B and 3/4.11.C, Average Planar Linear Heat Generation Rate (APLHGR), APRM Setpoints and Minimum Critical Power Ratio, in order to provide clarification of the SPC methodology for the application of thermal limits.

TS 3.11.A requires that all APLHGR for each type of fuel as a function of bundle average exposure shall not exceed the limits specified in the COLR. For Section 3/4.11.A, the licensee proposed the following changes: 1) relocate the last two paragraphs of Section 3/4.11.A on Bases page B3/4.11-1 to the beginning of Section 3/4.11.A, 2) insert "GE Fuel" in front of the current first paragraph, and 3) add the following paragraphs to describe the SPC methodology:

SPC Fuel

This specification assures that the peak cladding temperature of SPC fuel following a postulated design basis loss-of-coolant accident will not exceed the Peak Cladding Temperature (PCT) and maximum oxidation limits specified in 10 CFR 50.46. The calculational procedure used to establish the Average Planar Linear Heat Generation Rate (APLHGR) limits is based on a loss-of-coolant accident analysis.

The 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 not strongly influenced by the rod-to-rod power distribution within the assembly.

The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for two-loop and single-loop operation are specified in the Core Operating Limits Report (COLR).

The staff finds the Section 3/4.11.A Bases changes described above acceptable.

TS 3.11.B requires, based on the proposed change discussed above, that the TLHGR shall be maintained such that the FDLRC is less than or equal to 1.0. The licensee proposed to delete the first sentence of the paragraph since it is no longer applicable. Furthermore, the licensee proposed to add "or FDLRC" following "MFLPD" in the last sentence of the original paragraph and add the following paragraphs to expand on the SPC methodology:

SPC Fuel

The Fuel Design Limiting Ratio for Centerline Melt (FDLRC) is incorporated to protect the above criteria at all power levels considering events which cause the reactor power to increase to 120% of rated thermal power.

The scram settings must be adjusted to ensure that the TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR) is not violated for any power distribution. This is accomplished using FDLRC. The scram setting is decreased in accordance with the formula in Specification 3.11.B, when FDLRC is greater than 1.0.

The adjustment may also be accomplished by increasing the gain of the APRM by FDLRC. This provides the same degree of protection as reducing the trip setting by 1/FDLRC by raising the initial APRM reading closer to the trip setting such that a scram would be received at the same point in a transient as if the trip setting had been reduced.

The added paragraphs provide clarification of LCO Action Statements 3.11.B.2 and 3.11.B.3. Therefore, the addition of the above paragraphs clarifies the SPC methodology and is acceptable.

In Section 3/4.11.C, the licensee proposed minor editorial changes to the second paragraph. These changes affect the first two sentences and are as follows:

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

The licensee proposed to replace the fourth paragraph to again clarify the SPC methodology, which uses four scram insertion points to calculate MCPR Operating Limit and MCPR Safety Limit:

> MCPR Operating Limits are presented in the CORE OPERATING LIMITS REPORT (COLR) for both Nominal Scram Speed (NSS) and Technical Specification Scram Speed (TSSS) insertion times. The negative reactivity insertion rate resulting from the scram plays a major role in providing the required protection against violating the Safety Limit MCPR during transient events. Faster scram insertion times provide greater protection and allow for improved MCPR performance. The application of NSS MCPR limits utilizes measured data that is faster than the times required by the Technical Specifications, while the TSSS MCPR limits provide the necessary protection for the slowest allowable average scram insertion times identified in Specification 3.3.E. The measured scram times are compared with the nominal scram insertion times and the Technical Specification Scram Speeds. The appropriate operating limit is applied, as specified in the COLR.
>
> For core flows less than rated, the MCPR Operating Limit established in the specification is adjusted to provide protection of the Safety Limit MCPR in the event of an uncontrolled recirculation flow increase to the physical limit of the pump. Protection is provided for manual and automatic flow control by applying the appropriate flow dependent MCPR limits presented in the COLR. The MCPR Operating Limit for a given power/flow state is the greater value of MCPR as given by the rated conditions MCPR limit or the flow dependent MCPR limit. For automatic flow control, in addition to protecting the Safety Limit MCPR during the flow run-up event, protection is provided to prevent exceeding the rated flow MCPR Operating Limit during an automatic flow increase to rated core flow.

The proposed change appropriately reflects the NRC-approved SPC methodology and does not change the current requirement that MCPR meet the limits specified in the COLR. Therefore, the proposed change is acceptable.

2.9 Reactor Core

TS 5.3.A, Fuel Assemblies, provides a description of the fuel assemblies. The licensee proposed to expand this description to be consistent with Improved Standard Technical Specifications, NUREG-1433, Revision 1, and to better reflect the ATRIUM-9B design. The revised description includes a discussion of the use of water rods or water boxes which is consistent with the SPC fuel design, and replaces "zirconium alloy" with "Zircaloy or Zirlo." Upon further review and discussions with ComEd on May 22, 1997, it was determined that to be consistent with Improved Standard Technical Specifications, NUREG-1433, Revision 1, and provide a clarification of what the substitutions will be for, additional words needed to be added. The words "filler rods for fuel rods" had been omitted from the amendment request. The sentence was changed, with approval from ComEd, to read, "Limited substitutions of Zircaloy or ZIRLO filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used." This change was approved in the NRC safety evaluation (SE) related to Amendment No. 173 issued for Unit 2 on May 2, 1997, and in the NRC SE related to Amendment No. 174 issued for Unit 2 on May 22, 1997. The proposed change accurately describes the SPC fuel design, is consistent with NUREG-1433, Revision 1, includes the cladding material cited in 10 CFR 50.44 and 50.46, and does not affect any current TS requirements. This change will make TS Section 5.3.A appropriate for Unit 1 and 2 and deletes Unit 2 specific page 5-5a; therefore, the proposed change is acceptable.

2.10 Reactor Coolant System

Section 182a of the Atomic Energy Act of 1954, as amended, requires applications for nuclear power plant operating licenses to include TSs 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 FR 39132 (July 22, 1993), in which the Commission indicated that compliance with the Final Policy Statement satisfies Section 182a of the Act. In particular, the Commission indicated that certain items could be relocated from the TS to licensee-controlled documents, consistent with the standard enunciated in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). In that case, the Atomic Safety and Licensing Appeal Board indicated that "technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety."

The Final Policy Statement identified four criteria to be used in determining whether a particular matter is required to be included in the TS limiting conditions for operation, 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 or 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 Commission recently amended 10 CFR 50.36 to codify and incorporate these four criteria (60 FR 36953).

TS 5.4 describes the design pressure, temperature, and volume of the reactor coolant system. The licensee proposed to relocate the contents of Specification 5.4 to the UFSAR. Page 5-6 and Table of Contents page XIV are modified to read, "[INTENTIONALLY BLANK]."

Design temperatures, pressures, and volumes of the Reactor Coolant System in existing TS Section 5.4 will be detailed in the Updated Final Safety Analysis Report (UFSAR). Changes to these facility design parameters are controlled by the requirements of 10 CFR 50.59. Furthermore, these design parameters are encompassed by existing TS Limiting Condition for Operation (LCOs) that establish acceptable requirements for ensuring that performance of the reactor coolant system is maintained. Any changes to the LCOs would receive prior NRC review and approval. Since the features with a potential to impact safety are sufficiently addressed by LCOs, and since design features, if altered in accordance with 10 CFR 50.59, would not result in a significant impact on safety, the criteria of 10 CFR 50.36(c)(4) for including the above design features in the TS are not met.

The above relocated requirements relating to design features are not required to be in the TS under 10 CFR 50.36, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to public health and safety. In addition, the staff finds that sufficient regulatory controls exist under 10 CFR 50.59 to assure continued protection of public health and safety. This proposed change is consistent with Improved Standard Technical Specifications, NUREG-1433, Revision 1, and is acceptable. The Additional Condition in Appendix C of the license will valuate the acceptability of removal of the contents of Specification 5.4. Accordingly, the staff has concluded that these requirements may be relocated from the TS to the licensee's UFSAR.

2.11 Reporting Requirements

TS 6.9 requires that, in addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the identified reports shall be submitted to the Regional Administrator of the appropriate Regional Office of the NRC unless otherwise noted. TS 6.9.A.6.a(4) describes the MCPR limit in the COLR. The licensee proposed to delete the 20 percent in the statement "including 20 percent scram insertion time" to reflect the SPC methodology. The proposed change will state "including scram insertion time." This reflects the current SPC methodology and is acceptable.

TS 6.9.A.6.b lists the analytical methods used to determine the operating limits that are previously reviewed and approved by the NRC in the latest approved revision or supplement of topical reports. The licensee proposed to include references to the list of topical reports which are used to determine the core operating limits by adding the following:

(5) Advanced Nuclear Fuels Methodology for Boiling Water Reactors, XN-NF-80-19(P)(A), Volume 1, Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.

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

(7) Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description, XN-NF-80-19(P)(A), Volume 3, Revision 2, Exxon Nuclear Company, January 1987.

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

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

(10) Qualification of Exxon Nuclear Fuel for Extended Burnup Supplement 1: Extended Burnup Qualification of ENC 9x9 BWR Fuel, XN-NF-82-06(P)(A) Supplement 1, Revision 2, Advanced Nuclear Fuels Corporation, May 1988.

(11) Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels 9x9-1X and 9x9-9X BWR Reload Fuel, ANF-89-014(P)(A), Revision 1 and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, October 1991.

(12) Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A), Revision 1 and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.

(13) Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A), Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.

(14) ANFB Critical Power Correlation, ANF-1125(P)(A) and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.

(15) Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, ANF-524(P)(A), Revision 2, Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.

(16) COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analyses, ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.

(17) Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993.

(18) Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods," Revision 0, Supplements 1 and 2, December 1991, March 1992, and May 1992, respectively; SER letter dated March 22, 1993.

The additional topical reports are those used in SPC methodology and have been approved by the NRC and are appropriate for the Quad Cities plant design and are acceptable for use. References (5), (6), (7), and (8) in the current TS for Unit 2 on page 6-16a were added in Amendment No. 173. These references are now items (5), (11), and (12) on new page 6-16 and (18) on new page 6-16a, for Units 1 and Unit 2. The staff finds this change acceptable because the use of identified NRC-approved methodologies will ensure that the values for cycle-specific parameters are determined consistent with applicable design bases and safety limits (e.g., fuel thermal and mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin and transient and accident analysis limits) and assist safe operation of the facility.

2.12 Conclusion

ComEd requested changes to the Quad Cities Nuclear Power Station, Units 1 and 2, TS which would incorporate NRC-approved thermal limit licensing methodology in the list of approved methodologies used in establishing the cycle specific thermal limits. Other minor editorial changes were also proposed. The staff concluded that these TS revisions are compatible with the STS, and SPC methodology. Based on the above, the staff concluded that operation in the proposed manner will not endanger the health and safety of proposed. The staff concluded that these TS revisions are compatible with the STS, and SPC methodology. Based on the above, the staff concluded that operation in the proposed manner will not endanger the health and safety of the public and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

3.0 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.

4.0 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 effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (61 FR 44355). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). The amendment also relates to changes in record keeping, reporting or administrative procedures or requirements. Accordingly, with respect to these items, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

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

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