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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 160
License No. DPR-19

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated June 20, 1996, as supplemented December 30, 1996, and March 5, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954 (the Act), as amended, and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraphs 2.C.(2) and 2.C.(7) of Facility Operating License No. DPR-19 are hereby amended to read as follows:

2.C.(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 160, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

2.C.(7) Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 160 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stang

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

Attachments:
1. Changes to the Appendix B
2. Changes to the Technical Specifications

Date of Issuance: June 12, 1997

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-19

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

Amendment Number	Additional Condition	Implementation Date
157	The license is amended to authorize changing the UFSAR to allow credit for containment overpressure as detailed below, to assure adequate Net Positive Suction Head is available for low pressure Emergency Core Cooling System pumps following a design basis accident.	Effective as of the issuance of Amendment No. 157 and shall be implemented within 30 days.

Time (seconds)	Containment Pressure (PSIG)
0-240	9.5
240-480	2.9
480-6000	1.9
6000-accident end	2.5

Amendment Number	Additional Condition	Implementation Date
157	The EOPs shall be changed to alert operators to NPSH concerns and to make containment spray operation consistent with the overpressure requirements for NPSH.	Shall be implemented within 30 days after issuance of Amendment No. 157.
160	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 a safety evaluation dated June 12, 1997.	30 days from the date of issuance of Amendment No. 160

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 155
License No. DPR-25

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated June 20, 1996, as supplemented December 30, 1996, and March 5, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954 (the Act), as amended, and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraphs 3.B. and 3.0 of Facility Operating License No. DPR-25 are hereby amended to read as follows:

3.B Technical Specifications

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

3.O Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 155 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachments:
1. Changes to Appendix B
2. Changes to the Technical Specifications

Date of Issuance: June 12, 1997

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-25

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

Amendment Number	Additional Condition	Implementation Date
152	The license is amended to authorize changing the UFSAR to allow credit for containment overpressure as detailed below, to assure adequate Net Positive Suction Head is available for low pressure Emergency Core Cooling System pumps following a design basis accident.	Prior to Unit 3 returning to Mode 3 from refueling outage D3R14.

Time (seconds)	Containment Pressure (PSIG)
0-240	9.5
240-480	2.9
480-6000	1.9
6000-accident end	2.5

Amendment Number	Additional Condition	Implementation Date
152	The licensee shall complete the evaluation of the torus attached piping.	Prior to Unit 3 returning to Mode 3 from refueling outage D3R14.
152	The EOPs shall be changed to alert operators to NPSH concerns and to make containment spray operation consistent with the overpressure requirements for NPSH.	Shall be implemented within 30 days after issuance of Amendment No. 152.
155	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 a safety evaluation dated June 12, 1997.	30 days from the date of issuance of Amendment No. 155

ATTACHMENT TO LICENSE AMENDMENT NOS. 160 AND 155

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

DOCKET NOS. 50-237 AND 50-249

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment numbers and contain marginal lines indicating the area of change.

REMOVE	INSERT
XIV	XIV
2-2	2-2
2-4	2-4
B 2-1	B 2-1
B 2-2	B 2-2
B 2-4	B 2-4
B 2-5	B 2-5
B 2-8	B 2-8
B 3/4.2-1	B 3/4.2-1
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 3/4.6-3	B 3/4.6-3
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
5-6	5-6
6-14	6-14
6-15	6-15

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

Reactor Coolant System Pressure

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

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

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1345 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1345 psig within 2 hours and comply with the requirements of Specification 6.7.

Reactor Vessel Water Level

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

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

ACTION:

With the reactor vessel water level at or below twelve inches above the top of the active irradiated fuel, manually initiate the ECCS to restore the water level, after depressurizing the reactor vessel, if required, and comply with the requirements of Specification 6.7.

a The top of active irradiated fuel is defined to be 360 inches above vessel zero.

TABLE 2.2.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

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

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

b The top of active fuel is defined to be 360 inches above vessel zero.

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

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 28×10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. At 25% of RATED THERMAL POWER, the peak powered bundle would have to be operating at 3.86 times the average powered bundle in order to achieve this bundle power. Thus, a core thermal power limit of 25% for reactor pressures below 785 psig is conservative.

2.1.B THERMAL POWER, High Pressure and High Flow

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

2.1.D Reactor Vessel Water Level

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

2.2 LIMITING SAFETY SYSTEM SETTINGS

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

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

2.2.A Reactor Protection System Instrumentation Setpoints

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

1. Intermediate Range Monitor, Neutron Flux - High

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

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

Additional conservatism was taken in this analysis by assuming that the IRM CHANNEL closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power is limited to 7.7% of rated power, thus maintaining MCPR above the fuel cladding integrity Safety Limit. Based on the above analysis, the IRM provides protection against local 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 increased based on the new background levels to allow for the injection of hydrogen. This trip function provides an anticipatory scram to limit offsite dose consequences, but is not assumed to occur in the analysis of any design basis event.

BASES

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 to the original fuel designs. Safety Limits, water level instrument 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 initiations 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.

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 the 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. 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.

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

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

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

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 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 fuel cycle dependent TSSS MCPR operating limits and NSS MCPR operating limits which are presented in the COLR. Results of the control rod scram timing tests performed during the current fuel cycle are used to determine the operating limit for MCPR. Following the completion of each set of scram time 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.

reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if operation can continue. The evaluation must verify the reactor coolant system integrity remains acceptable and must be completed if continued operation is desired. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

The 72 hour completion time is reasonable to accomplish the evaluation of a mild violation. More severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed if continued operation is desired.

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.

BASES

3/4.11.A AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature 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) operating limits is based on a loss-of-coolant accident analysis. The analysis is performed using calculational models which are consistent with the requirements of Appendix K of 10 CFR Part 50.

The PCT following a postulated loss-of-coolant accident is primarily a function of the initial condition's 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 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 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.

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 ≥ 1% plastic strain does not occur; and, the fuel does not experience centerline melt during anticipated operational occurrences beginning at any power level and terminating at 120% of RATED THERMAL POWER.

The APRM scram settings must be adjusted to ensure that the LHGR transient limit (TLHGR) is not violated for any power distribution. This is accomplished by using FDLRC. The APRM scram setting is decreased in accordance with the formula in Specification 3.11.B, when FDLRC is greater than 1.0.

BASES

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 daily requirement for calculating FDLRC 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 FDLRC 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 FDLRC after initially determining FDLRC is greater than 1.0 exists to ensure that FDLRC will be known following a change in THERMAL POWER or power shape that could place operation above a thermal limit.

The FUEL DESIGN LIMIT RATIO FOR CENTERLINE MELT (FDLRC) is defined as:

$$\text{FDLRC} = \frac{(\text{LHGR})(1.2)}{(\text{TLHGR})(\text{FRTP})}\ ;$$

where LHGR is the LINEAR HEAT GENERATION RATE, and TLHGR is the TRANSIENT LINEAR HEAT GENERATION RATE. The TLHGR is specified in the CORE OPERATING LIMITS REPORT.

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 takes advantage of improved scram insertion rates, 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 insertion 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 greatest 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.

3/4.11.D STEADY STATE LINEAR HEAT GENERATION RATE

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

5.0 DESIGN FEATURES

5.3 REACTOR CORE

Fuel Assemblies

5.3.A The rsactor core shall contain 724 fuel assemblies[1,2]. 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 a water box. Limited substitutions of Zircaloy or ZIRLO or stainless steel 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[3]. 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.

1 ATRIUM-9B fuel with exception of lead test assemblies is only allowed in the reactor core in Operational Modes 3, 4 and 5, and with no more than one control rod withdrawn, for Unit 2 only.

2 Operation in all modes with ATRIUM-9B fuel is allowed for Dresden, Unit 3, Cycle 15, only.

3 The design bases applicable to ATRIUM-9B fuel are those which are applicable to Operational Modes 3, 4, and 5, for Unit 2 only.

5.4 [INTENTIONALLY LEFT BLANK]

[INTENTIONALLY LEFT BLANK]

3. Annual Radiological Environmental Operating Report

   The Annual Radiological Environmental Operating Report covering the operation of the Unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

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 Steady State Linear Heat Generation Rate (SLHGR) for Specification 3.11.D.

      (4) The Minimum Critical Power Operating Limit (including scram insertion times) 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) ANF-1125(P)(A), "Critical Power Correlation - ANFB."

(2) ANF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors."

(3) XN-NF-79-71(P)(A), "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors."

(4) XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors."

(5) XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump Boiling Water Reactors Reload Fuel."

(6) ANF-913(P)(A), "CONTRANSA2: A Computer Program for Boiling Water Reactor Transient Analysis."

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

(8) ANF-89-14(P)(A), Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advance Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel, Revision 1 and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, October 1991.

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

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

(11) Commonwealth Edison Company Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods", and associated Supplements on Neutronics Licensing Analyses (Supplement 1) and La Salle County Unit 2 Benchmarking (Supplement 2).