ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATIONS

Technical Specification 2.0

"SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS"

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

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

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

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

THERMAL POWER, High Pressure and High Flow

2.1.B The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.05 (Unit 2), or 1.08 (Unit 3), with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow. During single recirculation loop operation, this MCPR limit shall be increased by 0.01.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

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

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

Reactor Coolant System Pressure

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

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

ACTION:

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

Reactor Vessel Water Level

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

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

ACTION:

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

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.2 LIMITING SAFETY SYSTEM SETTINGS

Reactor Protection System (RPS) Instrumentation Setpoints

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

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

ACTION:

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

TABLE 2.2.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

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

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

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

TABLE 2.2.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

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

BASES

2.1 SAFETY LIMITS

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

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

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

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

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

BASES

2.1.A THERMAL POWER, Low Pressure or Low Flow

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

2.1.B THERMAL POWER, High Pressure and High Flow

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

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

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

BASES

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

2.1.C Reactor Coolant System Pressure

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

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

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

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

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2.1.D Reactor Vessel Water Level

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

2.2 LIMITING SAFETY SYSTEM SETTINGS

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

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

2.2.A Reactor Protection System Instrumentation Setpoints

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

1. Intermediate Range Monitor, Neutron Flux - High

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

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

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

2. Average Power Range Monitor

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

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

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

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

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

3. <u>Reactor Vessel Steam Dome Pressure - High</u>

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The scram will quickly reduce the neutron flux, counteracting the pressure increase. The scram setting is slightly higher than the operating pressure to permit normal operation without spurious scrams. The scram setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement (reactor vessel steam space) compared to the highest pressure that occurs in the system during a transient.

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

4. <u>Reactor Vessel Water Level - Low</u>

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

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

5. Main Steam Line Isolation Valve - Closure

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

6. Main Steam Line Radiation - High

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

7. Drywell Pressure - High

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

8. Scram Discharge Volume Water Level - High

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

9. Turbine Stop Valve - Closure

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

10. Turbine EHC Control Oil Pressure - Low

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

11. Turbine Control Valve Fast Closure

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

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

12. Turbine Condenser Vacuum - Low

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

13. <u>Reactor Mode Switch Shutdown Position</u>

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

14. <u>Manual Scram</u>

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

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

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

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

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

THERMAL POWER, High Pressure and High Flow

2.1.B The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.06 with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow. During single recirculation loop operation, this MCPR limit shall be increased by 0.01.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

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

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

Reactor Coolant System Pressure

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

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

ACTION:

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

Reactor Vessel Water Level

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

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

ACTION:

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

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.2 LIMITING SAFETY SYSTEM SETTINGS

Reactor Protection System (RPS) Instrumentation Setpoints

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

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

ACTION:

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

TABLE 2.2.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

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

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

TABLE 2.2.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

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

BASES

2.1 SAFETY LIMITS

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

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

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

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

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

BASES

2.1.A THERMAL POWER, Low Pressure or Low Flow

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

2.1.B THERMAL POWER, High Pressure and High Flow

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

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

However, if transition boiling were to occur, cladding perforation would not necessarily be expected. Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicates that BWR fuel can survive for an extended period in an environment of transition boiling.

2.1.C Reactor Coolant System Pressure

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

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

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

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

BASES

2.1.D Reactor Vessel Water Level

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

2.2 LIMITING SAFETY SYSTEM SETTINGS

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

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

2.2.A Reactor Protection System Instrumentation Setpoints

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

1. Intermediate Range Monitor, Neutron Flux - High

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

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

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

2. Average Power Range Monitor

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

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

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

BASES

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

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

3. Reactor Vessel Steam Dome Pressure - High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure crease while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The scram will quickly reduce the neutron flux, counteracting the pressure increase. The scram setting is slightly higher than the operating pressure to permit normal operation without spurious scrams. The scram setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement (reactor vessel steam space) compared to the highest pressure that occurs in the system during a transient.

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

4. Reactor Vessel Water Level - Low

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

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

5. Main Steam Line Isolation Valve - Closure

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

6. Main Steam Line Radiation - High

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

7. Drywell Pressure - High

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

8. Scram Discharge Volume Water Level - High

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

9. Turbine Stop Valve - Closure

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

10. Turbine EHC Control Oil Pressure - Low

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

11. <u>Turbine Control Valve Fast Closure</u>

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

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

12. <u>Turbine Condenser Vacuum - Low</u>

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

BASES

13. Reactor Mode Switch Shutdown Position

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

14. Manual Scram

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

ATTACHMENT 4

EXISTING TECHNICAL SPECIFICATIONS

Technical Specification 2.0

"SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS"

ATTACHMENT 4

DELETION OF CURRENT TECHNICAL SPECIFICATIONS

This technical specification amendment will replace the current sections 1.1/2.1 and 1.2/2.2, Fuel Cladding Integrity and Reactor Coolant System, for the Dresden Unit 2 and Unit 3 Technical Specifications. The specifications are replaced in its entirety with revised pages that combine the Unit 2 and Unit 3 specifications.

Delete the following pages:

DPR - 19	DPR - 25
1/2.1-1	1/2.1-1
1/2.1-2	1/2.1-2
1/2.1-3	1/2.1-3
1/2.1-4	1/2.1-4
1/2.1-5	1/2.1-5
B 1/2.1-6	B 1/2.1-6
B 1/2.1-7	B 1/2.1-7
B 1/2.1-8	B 1/2.1-8
B 1/2.1-9	B 1/2.1-9
B 1/2.1-10	B 1/2.1-10
B 1/2.1-11	B 1/2.1-11
B 1/2.1-12	B 1/2.1-12
B 1/2.1-13	B 1/2.1-13
B 1/2.1-14	B 1/2.1-14
B 1/2.1-15	B 1/2.1-15
B 1/2.1-16	B 1/2.1-16
B 1/2.1-17	B 1/2.1-17
1/2.2-1	1/2.2-1
B 1/2.2-2	B 1/2.2-2
B 1/2.2-3	B 1/2.2-3
B 1/2.2-4	B 1/2.2-4

ATTACHMENT 4

DELETION OF CURRENT TECHNICAL SPECIFICATIONS

This technical specification amendment will replace the current sections 1.1/2.1 and 1.2/2.2, Fuel Cladding Integrity and Reactor Coolant System, for the Quad Cities Unit 1 and Unit 2 Technical Specifications. The specifications are replaced in its entirety with revised pages that combine the Unit 1 and Unit 2 specifications.

Delete the following pages:

DPR - 29	DPR - 30
1.1/2.1-1	1.1/2.1-1
1.1/2.1-2	1.1/2.1-2
1.1/2.1-3	1.1/2.1-2a
1.1/2.1-4	1.1/2.1-3
1.1/2.1-5	1.1/2.1-4
1.1/2.1-6	1.1/2.1-5
1.1/2.1-7	1.1/2.1-6
1.1/2.1-8	1.1/2.1-7
1.1/2.1-9	1.1/2.1-7a
1.1/2.1-10	1.1/2.1-8
1.1/2.1-11	1.1/2.1-9
1.1/2.1-12	1.1/2.1-10
1.1/2.1-13	1.1/2.1-11
1.1/2.1-14	1.1/2.1-12
1.1/2.1-15	Figure 2.1-1
1.1/2.1-16	Figure 2.1-3
1.1/2.1-17	1.2/2.2-1
Figure 2.1-1	1.2/2.2-2
Figure 2.1-3	1.2/2.2-2a
1.2/2.2-1	1.2/2.2-3
1.2/2.2-2	--
1.2/2.2-3	--
1.2/2.2-4	--

ATTACHMENT 5

DRESDEN 2/3 DIFFERENCES

Technical Specification 2.0

"SAFETY LIMITS AND
LIMITING SAFETY SYSTEM SETTINGS"

ATTACHMENT 5

COMPARISON OF DRESDEN UNIT 2 AND UNIT 3
TECHNICAL SPECIFICATIONS FOR THE
IDENTIFICATION OF TECHNICAL DIFFERENCES

SECTION 1.1/2.1
"FUEL CLADDING INTEGRITY"

Commonwealth Edison has conducted a comparison review of the Dresden Unit 2 and Unit 3 Technical Specifications to identify any technical differences in support of combining the Technical Specifications into one document. The intent of the review was not to identify any differences in presentation style (e.g. table formats, use of capital letters, etc.), punctuation or spelling errors, but rather to identify areas which the Technical Specifications are technically or administratively different.

The review of Section 1.1/2.1 "Fuel Cladding Integrity" revealed one technical difference:

> The MCPR Safety Limit for Unit 2 is different than the MCPR Safety Limit for Unit 3. This due to the different resident fuel types in each of the Dresden Units and the difference is retained in the proposed specifications.

ATTACHMENT 5

COMPARISON OF DRESDEN UNIT 2 AND UNIT 3
TECHNICAL SPECIFICATIONS FOR THE
IDENTIFICATION OF TECHNICAL DIFFERENCES

SECTION 1.2/2.2
"REACTOR COOLANT SYSTEM"

Commonwealth Edison has conducted a comparison review of Dresden Unit 2 and Unit 3 Technical Specifications to identify any technical differences in support of combining the Technical Specifications into one document. The intent of the review was not to identify any differences in presentation style (e.g. table formats, use of capital letters, etc.), punctuation, or spelling errors but rather identify areas which the Technical Specifications are technically or administratively different.

The review of Section 1.2/2.2 "Reactor Coolant System" did not reveal any technical differences.

ATTACHMENT 5

QUAD CITIES 1/2 DIFFERENCES

Technical Specification 2.0

"SAFETY LIMITS AND
LIMITING SAFETY SYSTEM SETTINGS"

ATTACHMENT 5

COMPARISON OF QUAD CITIES UNIT 1 AND UNIT 2 TECHNICAL SPECIFICATIONS FOR THE IDENTIFICATION OF TECHNICAL DIFFERENCES

SECTION 1.1/2.1 "FUEL CLADDING INTEGRITY"

Commonwealth Edison has conducted a comparison review of the Quad Cities Unit 1 and Unit 2 Technical Specifications to identify any technical differences in support of combining the Technical Specifications into one document. The intent of the review was not to identify any differences in presentation style (e.g. table formats, use of capital letters, etc.), punctuation or spelling errors, but rather to identify areas which the Technical Specifications are technically or administratively different.

The review of Section 1.1/2.1 "Fuel Cladding Integrity" revealed the following technical differences:

> The third paragraph, last sentence on page 1.1/2.1-6 (DPR-29) states "Basis of the values derived for this safety limit for each fuel type is documented in Reference 1." The Unit 2 Technical Specifications states, "...is documented in References 1 and 2." NEDO-24259-A (Reference 2 in the current Quad TS) contained information concerning the use of barrier fuel. The latest revision of NEDE-24011-P-A contains the information regarding barrier fuel which was previously only contained in NEDO-24259-A. As a result, the reference can be deleted.
>
> The last sentence of paragraph B on page 1.1/2.1-9 (DPR30) states, "As with the scram setting, this may be accomplished by adjusting the APRM gains." This information is not contained in the Unit 1 Technical Specifications. The Unit 2 Technical Specification information will be retained in the combination since the information is consistent with the requirements of Limiting Safety System Setting 2.1.B.

ATTACHMENT 5

COMPARISON OF QUAD CITIES UNIT 1 AND UNIT 2
TECHNICAL SPECIFICATIONS FOR THE
IDENTIFICATION OF TECHNICAL DIFFERENCES

SECTION 1.2/2.2
"REACTOR COOLANT SYSTEM"

Commonwealth Edison has conducted a comparison review of the Quad Cities Unit 1 and Unit 2 Technical Specifications to identify any technical differences in support of combining the Technical Specifications into one document. The intent of the review was not to identify any differences in presentation style (e.g. table formats, use of capital letters, etc.), punctuation, or spelling errors but rather identify areas which the Technical Specifications are technically or administratively different.

The review of Section 1.2/2.2 "Reactor Coolant System" did not reveal any technical differences.

ATTACHMENT 6

SIGNIFICANT HAZARDS CONSIDERATIONS AND ENVIRONMENTAL ASSESSMENT EVALUATION

Technical Specification 2.0

"SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS"

ATTACHMENT 6

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Commonwealth Edison has evaluated this proposed amendment and determined that it involves no significant hazards consideration. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility, in accordance with the proposed amendment, would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3) Involve a significant reduction in a margin of safety.

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

The proposed changes to Specifications 1/2.1 and 1/2.2 to delete the present Applicability and Objective sections represent administrative changes to format and presentation of material. The proposed changes provide the user with a format that will allow better access to needed information and provides concise Safety Limit, Limiting Safety System Settings, Applicability and Action requirements. The additions of Applicability and Action requirements represent clarification of intended requirements that do not presently state all required conditions of operability or provide clearly stated Action statements if the requirements are not met. The combining of the two sections and the added requirements follow STS guidelines that are in use at many operating BWRs with similar design and operating configurations as Dresden and Quad Cities Stations. Operability requirements for Safety Limits have been chosen to reflect only those Operational Modes where the Safety Limits apply. Operability requirements for Limiting Safety System Settings are already stated in other sections of the Technical Specifications, thus reference to the appropriate operability requirement is made rather than repeating the requirement in the Limiting Safety System Setting Specification.

Deletion of the Power Transient Safety Limit does not impact any safety analyses. The safety analyses assume the Reactor Protection System (RPS) operates as designed and the reactor scrams when the neutron flux exceeds the limiting safety system setting. The proposed Technical Specifications will continue to provide a highly reliable system to operate as assumed in the safety analyses. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The reactor water level low scram setpoint is changed (for Quad Cities) to be consistent with other reactor water level setpoints in the Technical Specifications and the STS. The setpoint is equivalent to the current requirement but is expressed as the reactor water level above the top of active fuel.

ATTACHMENT 6

The scram discharge volume scram level is converted for Dresden Unit 2 and Unit 3 to gallons to be consistent with the Quad Cities Units. The proposed setpoints are consistent with the current specifications. The change in the units does not represent a change in the physical setpoint.

The proposed change to delete the APRM Downscale Scram trip function for Quad Cities has been evaluated by Commonwealth Edison and General Electric and previously approved for Dresden Station. The events of concern with respect to the APRM/IRM companion trip are the Control Rod Drop Accident and the low power Rod Withdrawal Error. The FSAR and reload safety analyses do not credit this scram function in the termination of either of these events. Since this scram function is not credited in the termination of these events, the elimination of this scram function has no adverse effect of previously evaluated accidents.

The change to the low condenser vacuum scram setpoint from 23 inches of Hg to 21 inches of Hg is consistent with an identical change made to Quad Cities Units 1 and 2. The low condenser vacuum scram is an anticipatory scram and is not credited in any transient analysis. Thus the reduction in the setpoint will not affect any transient analysis.

The proposed changes do not alter the intent of existing setpoints or accident assumptions and follow existing requirements at other operating BWRs for operability and Action statements. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated because:

The proposed administrative changes to the format and arrangement of material do not affect technical requirements or assumptions of any potential accident and; therefore, cannot create the possibility of a new or different kind of accident from any previously evaluated.

The proposed addition of Applicability and Action requirements enhance the understanding and usability of the Technical Specifications and thus represent an improvement over present specifications. New requirements are modeled after those in use at operating BWRs and do not represent requirements that will adversely affect potential accident analyses or assumptions. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

Deletion of the Power Transient Safety Limit does not involve a change in the design or operation of any systems assumed to operate in the safety analyses. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

ATTACHMENT 6

The change in the units for the Reactor Water Level scram function do not change any physical plant setpoints. The setpoint will remain the same but will be expressed as the level above the top of active fuel. The change does not create the possibility of a new or different kind of accident.

The conversion of the Scram Discharge Volume scram setpoint from inches to gallons does not alter any physical plant setpoints. The setpoint will remain the same but will be expressed in gallons rather than inches. The change will provide consistency between Dresden and Quad Cities.

The deletion of the APRM Downscale Scram Trip Function does not introduce any new accident. The limiting accidents, Control Rod Drop, Rod Withdrawal Error, in the operating region of transition between the Startup and Run Operational Modes are well understood and are evaluated in FSAR and reload analyses. Other control rod initiated events which are less limiting in this region are subsets of the low power Rod Withdrawal Error event and are bounded by it and the design basis Control Rod Drop Accident. General Electric has indicated that, for reactivity insertion mechanisms at very low power, the only effect of the deletion of the APRM downscale scram would be that the initial power level could be a few percent lower which would not have a significant effect on the severity of the event. In addition, proper overlap between the IRMs and APRMs is not affected since the calibration requirements are not being changed.

The change in the low condenser vacuum scram function will not create the possibility of a new or different kind of accident because the function is not recognized in any of the transient analysis. The low condenser vacuum scram function is an anticipatory scram.

The proposed changes do not involve a significant reduction in the margin of safety because:

The proposed administrative changes to format, arrangement of material, clarification of requirements and other non- technical changes do not affect any safety aspects of the plant and as such can not involve a significant reduction in the margin of safety.

The proposed Applicability statements require availability of Safety Limits and Limiting Safety System Settings when required to perform their respective functions. Proposed Actions for Safety Limits allow only 2 hours to be in Hot Shutdown and then reference Specification 6.4 to ensure that proper reports are made and restart is prohibited until approved by the NRC. These provisions help ensure that present margins are not significantly reduced.

Deletion of the Power Transient Safety Limit does not impact the margin assumed in the safety analyses. The safety analyses assume the RPS operates as designed and the reactor scrams when the neutron flux exceeds the limiting safety system setting. The margins assumed in the design of the RPS and in the safety and transient analyses calculations have not been revised. Therefore, this change does not involve a significant reduction in the margin of safety.

The change in units to the Reactor Water Level scram setpoint and the Scram Discharge Volume scram setpoint do not involve a significant reduction in the margin of safety because the changes do not represent a change in the physical setpoints.

The reduction in the Low Condenser Vacuum scram setpoint does not represent a reduction in the margin of safety because the scram is not credited in any transient analysis.

The APRM Downscale Scram Trip Function is not credited in the termination of any FSAR or reload safety analysis event. As such, the elimination of this scram function has no effect on any margin of safety.

ATTACHMENT 6

ENVIRONMENTAL ASSESSMENT STATEMENT APPLICABILITY REVIEW

Commonwealth Edison has evaluated the proposed amendment against the criteria for the identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.20. It has been determined that the proposed changes meet the criteria for a categorical exclusion as provided under 10 CFR 51.22 (c)(9). This conclusion has been determined because the changes requested do not pose significant hazards consideration or do not involve a significant increase in the amounts, and no significant changes in the types, of any effluent that may be released offsite. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure. Therefore, the Environmental Assessment Statement is not applicable for these changes.

ATTACHMENT 1

EXECUTIVE SUMMARY

Technical Specification 3/4.11

"POWER DISTRIBUTION LIMITS"

EXECUTIVE SUMMARY

The Dresden Technical Specification Upgrade Program (TSUP) was conceptualized in response to lessons learned from the Diagnostic Evaluation Team inspection and the frequent need for Technical Specification interpretations. A comparison study of the Standard Technical Specification (STS), later operating plant's Technical Specifications provisions and Quad Cities Technical Specifications was performed prior to the Dresden TSUP effort. The study identified potential improvements in clarifying requirements and requirements which are no longer consistent with current industry practices. The Dresden TSUP will enhance the Quad Cities TSUP currently under review by the NRC. As a result of the inconsistencies in the Quad Cities submittal compared to the Standard Technical Specifications (STS), Dresden's submittal will more closely follow the provisions of STS and in conjunction, Quad Cities will amend their submittal so that Quad Cities and Dresden are identical within equipment and plant design. The format for the Dresden TSUP will remain as a two column layout for human factors considerations. Additionally, chapter organizations will remain essentially unchanged.

The TSUP is not intended to be a complete adoption for the STS. Overall, the Dresden custom Technical Specifications provide for the safe operation of the plant and therefore, only an upgrade is deemed necessary.

In response to an NRC recommendation, Quad Cities combined the Unit 1 and Unit 2 Technical Specifications into one document. The Dresden Unit 2 and Unit 3 Technical Specifications will also be combined into one document. To accomplish the combination of the Technical Specifications, a comparison of the individual Technical Specifications was performed to identify any technical differences. The technical differences are identified in the proposed amendment package for each section.

The TSUP was identified as a station top priority and is currently contained in the Dresden Management Action Plan (DMAP). The TSUP goal is to provide a better tool to station personnel to implement their responsibilities and to ensure Dresden Station is operated in accordance with current industry practices. The improved Technical Specifications provide for enhanced operation of the plant. The program improves the operator's ability to use the Technical Specifications by more clearly defining the Limiting Conditions for Operation and required actions. The most significant improvement to the specifications is the addition of equipment operability requirements during shutdown conditions.

EXECUTIVE SUMMARY
(continued)
PROPOSED CHANGES TO TECHNICAL SPECIFICATION SECTION 3/4.11, "POWER DISTRIBUTION LIMITS"

The current Dresden and Quad Cities Technical Specifications contain Applicability and Objective statements at the beginning of most sections. The proposed amendment will delete the "Objective" statement and integrates appropriate applicability statements within the specifications. This provides a clarification of the intended requirements and actions which are required when the specification cannot be met.

The proposed Section 3/4.11 is a new section that results in the consolidation and rearrangement of the power distribution limits. The majority of the proposed specifications are currently contained in section 3/4.5, ECCS Systems. The new specifications are adopted from the STS.

ATTACHMENT 2

DESCRIPTION OF CHANGES

Technical Specification 3/4.11

"POWER DISTRIBUTION LIMITS"

ATTACHMENT 2

DESCRIPTION OF AMENDMENT REQUEST

The changes proposed in this amendment request are made to 1) improve the understanding and usability of the present technical specifications, 2) incorporate technical improvements, and 3) include some provisions from later operating BWR plants.

GENERIC CHANGES

The present Dresden and Quad Cities technical specifications contain applicability and objective statements at the beginning of most sections. These statements are generic in nature and do not provide any useful information to the user of the technical specifications. The proposed change will delete the objective statement and provide applicability statements within each specification similar to the Standard Technical Specifications (STS). The proposed applicability statement to be included in each specification will include the reactor operational modes or other conditions for which the Limiting Condition for Operation (LCO) must be satisfied.

The proposed rearrangement of the power distribution limits from current section 3/4.5 to a new section will provide consistency in presentation of material and present the material in a fashion consistent with the STS. The addition of the applicable operational modes will provide readily accessible information concerning when the system is required to be operable and when surveillance requirements must be performed.

The proposed section contains several differences between Dresden and Quad Cities as a result of the different fuel vendors at the two sites. Dresden uses Siemens Nuclear Power fuel and thus has fuel limits defined by Siemens. Quad Cities uses General Electric fuel and uses thermal limits identical to those presented in the STS.

The proposed changes are consistent with the STS and Generic Letter 88-16, Removal of Cycle Specific Parameters From Technical Specifications. Both Dresden and Quad Cities have the cycle specific parameters in a Core Operating Limits Report (COLR).

SPECIFIC CHANGES

Proposed specification 3/4.11.A, Average Planar Linear Heat Generation Rate (APLHGR) is a complete adoption of the STS requirements. Dresden APLHGR limits are a function of bundle average exposure versus average planar exposure for General Electric fuel. Specification 3.11.A requires that all the APLHGR limits specified in the COLR be met in operational mode 1 when thermal power is greater than 25% of rated thermal power. When the condition is not satisfied the action requires that corrective action be initiated within 15 minutes, the APLHGR values restored within 2 hours or thermal power reduced below 25% within the next 4 hours. The proposed actions are adopted from STS but are separated for clarification purposes. Surveillance Requirement (SR) 4.11.A.1 requires the values of APLHGR be checked at least once per 24 hours, within 12 hours after a thermal power increase of at least 15% of rated thermal power, and initially and once per 12 hours when the reactor is operating on a limiting control rod pattern. The proposed specifications implement the current specifications for APLHGR in the Dresden and Quad Cities Technical Specifications but do not require the reactor to be placed in cold shutdown in 36 hours if the APLHGR limit is not restored within the 2 hours. The requirement to monitor APLHGR within 12 hours of a power increase is a new requirement based on STS. The requirement to monitor APLHGR while operating on a limiting control rod pattern is a new requirement and is consistent with the STS definition for a limiting control rod pattern. Proposed SR 4.11.A.4 states that the provisions of specification 4.0.D are not applicable. The provision allows power to be increased above 25% of rated thermal power applicability limit prior to performing the SRs as long as the SR time limits are met. Without the provision, a hold in the power ascension would have to occur prior to exceeding 25% of rated thermal power to perform the required surveillance.

Proposed specification 3/4.11.B, Average Power Range Monitor (APRM) Setpoints, is adopted from STS with several enhancements. The enhancements are made to avoid duplicating setpoints within the proposed technical specifications and to more clearly delineate when the actions for the specification are to be implemented. This specification is provided to require the APRM gain or APRM flow biased scram and rod block trip setpoints to be adjusted when operating under conditions of abnormal power peaking so that acceptable margin to the fuel cladding integrity limits are maintained. Abnormal power peaking is represented when the Maximum Fraction of Limiting Power Density (MFLPD) is greater than the Fraction of Rated Thermal Power (FRTP) for Quad Cities and for Dresden when the Fuel Design Limiting Ratio For Centerline Melt (FDLRC) is greater than 1.0. To maintain the appropriate margin under conditions of abnormal power peaking, either the APRM gain must be adjusted upward or the flow biased neutron flux upscale scram trip and rod block setpoints be reduced. This is accomplished by multiplying the APRM gain or setpoints by a factor that is representative of the reduction in margin to the fuel cladding integrity limits. Adjustment to the scram and rod block setpoint are made by multiplying the setpoint by the inverse of the factor for the APRM gains. This factor will be less than one and thus cause the setpoints to be lowered to maintain the margin. When the reactor is operating with normal peaking (i.e. FDLRC < 1.0 or MFLPD < FRTP) it is not necessary to modify the APRM flow biased scram or rod block setpoints. The proposed actions are adopted from STS but are separated for clarification purposes. The action requires that if FDLRC is greater than 1.0 for Dresden or MFLPD is greater than FRTP for Quad Cities, that within 6 hours the limit is restored, or the APRM setpoints in 2.2.A and 3.2.E are adjusted by the factors described above, or the APRM gains are adjusted by the factor described above. If the action provisions are not met, thermal power is required to be reduced to below 25% of rated thermal power. The proposed SR requires that the value of FDLRC (Dresden) or MFLPD and FRTP (Quad Cities) be checked at least once per 24 hours, within 12 hours after a thermal power increase of 15% of more, and initially and once per 12 hours when the reactor is operating on a limiting control rod pattern. The SR are modified from STS in accordance with the proposed LCO. Proposed SR 4.11.B.4 is added to stipulate that the provisions of specification 4.0.D are not applicable. Footnote (a) is adopted from the STS but does not restrict the adjustment of the APRM gains to less than 90% of rated thermal power. The footnote implements current requirements for adjusting APRM gains during operation with abnormal power peaking.

Proposed specification 3/4.11.C, Minimum Critical Power Ratio (MCPR) is adopted from STS. The MCPR is required to be equal to or greater than the limit specified in the COLR in operational mode 1 when thermal power is greater than 25% of rated thermal power.

ATTACHMENT 2

The actions for MCPR are adopted from the applicable STS actions and are separated for clarification. Proposed action 3.11.C.1 requires that when the MCPR is less than the applicable limit specified in the COLR, that corrective action be initiated within 15 minutes, the MCPR restored within 2 hours or thermal power be reduced to less than 25% of rated within the next 4 hours. The proposed specification implements the requirements of the current specifications but do not require the reactor to be taken to cold shutdown if the MCPR limit is not restored within the 2 hour time frame. The surveillance requirements are adopted from the STS and are based on a scram insertion time value, t_{ave}. The definition of t_{ave} is contained in the COLR. Dresden and Quad Cities use different scram insertion values for t_{ave} and they are identified in the COLR. Dresden uses a 90% mean insertion value whereas Quad Cities uses a 20% mean insertion value. The difference is a result of the different fuel vendors at Dresden and Quad Cities. The value of MCPR is required to be determined to be greater than the MCPR limit at least every 24 hours, within 12 hours after completion of a power increase of at least 15% of rated thermal power and initially and once per 12 hours when the reactor is operating on a limiting control rod pattern. The value used in the development of the MCPR limit is required to be verified within 72 hours of the completion of specification 4.3.D, Scram insertion times. Proposed SR 4.11.C.4 is added to stipulate that the provisions of specification 4.0.D are not applicable.

The limits on Linear Heat Generation Rate are different for Dresden and Quad Cities and are discussed individually due to the differences.

Proposed specification 3/4.11.D, Linear Heat Generation Rate (LHGR) for Quad Cities is adopted from the STS. The LHGR is required to be less than the value specified in the COLR when in operational mode 1 and thermal power is greater than 25% of rated thermal power. When the condition is not satisfied the action requires that corrective action be initiated within 15 minutes, the LHGR values restored within 2 hours or thermal power reduced below 25% within the next 4 hours. The proposed actions are adopted from STS but are separated for clarification purposes. SR 4.11.D.1 requires the values of LHGR be checked at least once per 24 hours, within 12 hours after a thermal power increase of at least 15% of rated thermal power, and initially and once per 12 hours when the reactor is operating on a limiting control rod pattern. The proposed specifications implement the current specifications for LHGR in the Quad Cities Technical Specifications. The requirement to monitor LHGR within 12 hours of a power increase is a new requirement based on STS. The requirement to monitor LHGR while operating on a limiting control rod pattern is a new requirement and is a result of the adoption of the STS definition for a limiting control rod pattern. Proposed SR 4.11.D.4 is added to stipulate that the provisions of specification 4.0.D are not applicable.

Proposed specification 3/4.11.D, Steady State Linear Heat Generation Rate (SLHGR) for Dresden is retained from the current specifications but reformatted in accordance with the proposed specifications. The specification requires that the SLHGR values be less than the limits specified in the COLR in operational mode 1 with thermal power greater than 25% of rated thermal power. When the condition is not satisfied the action requires that corrective action be initiated within 15 minutes, the SLHGR values restored within 2 hours or thermal power reduced below 25% within the next 4 hours. The proposed actions are adopted from STS but are separated for clarification purposes. SR 4.11.D.1 requires the values of SLHGR be checked at least once per 24 hours, within 12 hours after a thermal power increase of at least 15% of rated thermal power, and initially and once per 12 hours when the reactor is operating on a limiting control rod pattern. The proposed specifications implement the current specifications for SLHGR in the Dresden Technical Specifications. The requirement to monitor SLHGR within 12 hours of a power increase is a new requirement based on STS. The requirement to monitor SLHGR while operating on a limiting control rod pattern is a new requirement and is a result of the adoption of the STS definition for a limiting control rod pattern. Proposed SR 4.11.D.4 is added to stipulate that the provisions of specification 4.0.D are not applicable.

Proposed specification 3/4.11.E, Transient Linear Heat Generation Rate (TLHGR) for Dresden is retained from the current specifications. The specification requires that the TLHGR values be less than the limits specified in the COLR in operational mode 1 with thermal power greater than 25% of rated thermal power. When the condition is not satisfied the action requires that corrective action be initiated within 15 minutes, the TLHGR values restored within 2 hours or thermal power reduced below 25% within the next 4 hours. The proposed actions are adopted from STS but are separated for clarification purposes. SR 4.11.D.1 requires the values of TLHGR be checked at least once per 24 hours, within 12 hours after a thermal power increase of at least 15% of rated thermal power, and initially and once per 12 hours when the reactor is operating on a limiting control rod pattern. The proposed specifications implement the current specifications for TLHGR in the Dresden Technical Specifications. The requirement to monitor TLHGR within 12 hours of a power increase is a new requirement based on STS. The requirement to monitor TLHGR while operating on a limiting control rod pattern is a new requirement and is a result of the adoption of the STS definition for a limiting control rod pattern. Proposed SR 4.11.E.4 is added to stipulate that the provisions of specification 4.0.D are not applicable.

The changes proposed to the Bases for proposed Section 3/4.11 are administrative in nature and include the capitalization of terms defined in proposed Section 1.0, Definitions.

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATIONS

Technical Specification 3/4.11

"POWER DISTRIBUTION LIMITS"

3.11 - LIMITING CONDITIONS FOR OPERATION

A. AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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

APPLICABILITY:

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

ACTION:

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

1. Initiate corrective action within 15 minutes, and
2. Restore APLHGR to within the required limit within 2 hours.

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

4.11 - SURVEILLANCE REQUIREMENTS

A. AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.
4. The provisions of Specification 4.0.D are not applicable.

3.11 - LIMITING CONDITIONS FOR OPERATION

B. Average Power Range Monitor Setpoints

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

APPLICABILITY:

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

ACTION:

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

1. Restore FDLRC to within its limit, or
2. Adjust the flow biased APRM setpoints specified in Specifications 2.2.A and 3.2.E by 1/FDLRC, or
3. Adjust[a] the APRM gain such that the APRM readings are ≥ 100% of the FRACTION OF RATED THERMAL POWER (FRTP) times FDLRC.

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

4.11 - SURVEILLANCE REQUIREMENTS

B. Average Power Range Monitor Setpoints

The value of FDLRC shall be verified:

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with FDLRC greater than or equal to 1.0.
4. The provisions of Specification 4.0.D are not applicable.

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

3.11 - LIMITING CONDITIONS FOR OPERATION

C. MINIMUM CRITICAL POWER RATIO

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

APPLICABILITY:

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

ACTION:

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

1. Initiate corrective ACTION within 15 minutes, and
2. Restore MCPR to within the required limit within 2 hours.

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

4.11 - SURVEILLANCE REQUIREMENTS

C. MINIMUM CRITICAL POWER RATIO

MCPR, with:

1. t_{ave} = 3.50 prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.3.D, or
2. t_{ave} determined within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.3.D,

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

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.
4. The provisions of Specification 4.0.D are not applicable.

3.11 - LIMITING CONDITIONS FOR OPERATION

D. STEADY STATE LINEAR HEAT GENERATION RATE

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

APPLICABILITY:

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

ACTION:

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

1. Initiate corrective ACTION within 15 minutes, and
2. Restore the SLHGR to within the required limit within 2 hours.

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

4.11 - SURVEILLANCE REQUIREMENTS

D. STEADY STATE LINEAR HEAT GENERATION RATE

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

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for SLHGR.
4. The provisions of Specification 4.0.D are not applicable.

3.11 - LIMITING CONDITIONS FOR OPERATION

E. TRANSIENT LINEAR HEAT GENERATION RATE

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

APPLICABILITY:

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

ACTION:

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

1. Initiate corrective ACTION within 15 minutes, and
2. Restore the TLHGR to within the required limit within 2 hours.

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

4.11 - SURVEILLANCE REQUIREMENTS

E. TRANSIENT LINEAR HEAT GENERATION RATE

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

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for TLHGR.
4. The provisions of Specification 4.0.D are not applicable.

BASES

3/4.11.A AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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

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

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

The daily requirement for calculating APLHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate APLHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating APLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that APLHGR will be known following a change in THERMAL POWER or power shape, that could place operation above a thermal limit.

3/4.11.B APRM SETPOINTS

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

BASES

3/4.11.C MINIMUM CRITICAL POWER RATIO

The required operating limit MCPR at steady state operating conditions as specified in Specification 3.11.C are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in the CRITICAL POWER RATIO (CPR). The type of transients evaluated were change of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.11.C is obtained and presented in the CORE OPERATING LIMITS REPORT.

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

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

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

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value has considerable margin. Thus, the demonstration of MCPR below this power level is unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR after initially determining that a LIMITING CONTROL ROD PATTERN exists ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation above a thermal limit.

BASES

3/4.11.D STEADY STATE LINEAR HEAT GENERATION RATE

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

3/4.11.E TRANSIENT LINEAR HEAT GENERATION RATE

This specification provides assurance that the fuel will neither experience centerline melt nor exceed 1% plastic strain for transient overpower events beginning at any power and terminating at 120% of RATED THERMAL POWER. The daily requirement for calculating TLHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate TLHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating TLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that TLHGR will be known following a change in THERMAL POWER or power shape that could place operation above a thermal limit.

3.11 - LIMITING CONDITIONS FOR OPERATION

A. AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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

APPLICABILITY:

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

ACTION:

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

1. Initiate corrective ACTION within 15 minutes, and
2. Restore APLHGR to within the required limit within 2 hours.

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

4.11 - SURVEILLANCE REQUIREMENTS

A. AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.
4. The provisions of Specification 4.0.D are not applicable.

3.11 - LIMITING CONDITIONS FOR OPERATION

B. Average Power Range Monitor Setpoints

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

APPLICABILITY:

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

ACTION:

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

1. Restore MFLPD to within its limit, or
2. Adjust the flow biased APRM setpoints specified in Specifications 2.2.A and 3.2.E by FRTP/MFLPD, or
3. Adjust[a] the APRM gain such that the APRM readings are $\geq$100% of the MFLPD.

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

4.11 - SURVEILLANCE REQUIREMENTS

B. Average Power Range Monitor Setpoints

The value of MFLPD shall be verified:

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with MFLPD greater than or equal to FRTP.
4. The provisions of Specification 4.0.D are not applicable.

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

3.11 - LIMITING CONDITIONS FOR OPERATION

C. MINIMUM CRITICAL POWER RATIO

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

APPLICABILITY:

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

ACTION:

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

1. Initiate corrective ACTION within 15 minutes, and
2. Restore MCPR to within the required limit within 2 hours.

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

4.11 - SURVEILLANCE REQUIREMENTS

C. MINIMUM CRITICAL POWER RATIO

MCPR, with:

1. $t_{ave} = 0.86$ prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.3.D, or
2. t_{ave} determined within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.3.D,

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

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.
4. The provisions of Specification 4.0.D are not applicable.

3.11 - LIMITING CONDITIONS FOR OPERATION

D. LINEAR HEAT GENERATION RATE

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

APPLICABILITY:

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

ACTION:

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

1. Initiate corrective ACTION within 15 minutes, and
2. Restore the LHGR to within the required limit within 2 hours.

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

4.11 - SURVEILLANCE REQUIREMENTS

D. LINEAR HEAT GENERATION RATE

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

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for LHGR.
4. The provisions of Specification 4.0.D are not applicable.

BASES

3/4.11.A AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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

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

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

The daily requirement for calculating APLHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate APLHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating APLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that APLHGR will be known following a change in THERMAL POWER or power shape, that could place operation above a thermal limit.

3/4.11.B APRM SETPOINTS

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

3/4.11.C MINIMUM CRITICAL POWER RATIO

The required operating limit MCPR at steady state operating conditions as specified in Specification 3.11.C are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in the CRITICAL POWER RATIO (CPR). The type of transients evaluated were change of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.11.C is obtained and presented in the CORE OPERATING LIMITS REPORT.

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

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

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

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value has considerable margin. Thus, the demonstration of MCPR below this power level is unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR after initially determining that a LIMITING CONTROL ROD PATTERN exists ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation above a thermal limit.

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 to calculate LHGR within 12 hours after the completion [illegible] POWER increase of at least 15% of RATED THERMAL POWER ensures the [illegible] after power distribution shifts while still allotting time for the power distribution to [illegible] 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.

ATTACHMENT 4

EXISTING TECHNICAL SPECIFICATIONS

Technical Specification 3/4.11

"POWER DISTRIBUTION LIMITS"

ATTACHMENT 4

DELETION OF CURRENT TECHNICAL SPECIFICATIONS

This technical specification amendment will create a new section 3/4.11 that will replace several specifications in the current section 3.5/4.5, ECCS Systems, for the Dresden Unit 2 and Unit 3 Technical Specifications. Sections 3.5/4.5 will be replaced in its entirety with revised pages that combine the Unit 2 and Unit 3 specifications when the upgraded section 3/4.5 is developed and therefore, no pages are being deleted with this amendment.

ATTACHMENT 4

DELETION OF CURRENT TECHNICAL SPECIFICATIONS

This technical specification amendment will create a new section 3/4.11 that will replace several specifications in the current section 3.5/4.5, ECCS Systems, for the Quad Cities Unit 1 and Unit 2 Technical Specifications. Sections 3.5/4.5 will be replaced in its entirety with revised pages that combine the Unit 1 and Unit 2 specifications when the upgraded section 3/4.5 is developed and therefore, no pages are being deleted with this amendment.

ATTACHMENT 5

DRESDEN 2/3 DIFFERENCES

Technical Specification 3/4.11

"POWER DISTRIBUTION LIMITS"

ATTACHMENT 5

COMPARISON OF DRESDEN UNIT 2 AND UNIT 3 TECHNICAL SPECIFICATIONS FOR THE IDENTIFICATION OF TECHNICAL DIFFERENCES

SECTION 3/4.5 "ECCS SYSTEMS"

Commonwealth Edison has conducted a comparison review of the Dresden Unit 2 and Unit 3 Technical Specifications to identify any technical differences in support of combining the Technical Specifications into one document. The intent of the review was not to identify any differences in presentation style (e.g. table formats, use of capital letters, etc.) or punctuation but rather to identify areas in which the Technical Specifications are technically different.

The review of Section 3.5/4.5 "ECCS System" Sections I, J, K, and L did not reveal any technical differences.

ATTACHMENT 5

QUAD CITIES 1/2 DIFFERENCES

Technical Specification 3/4.11

"POWER DISTRIBUTION LIMITS"

ATTACHMENT 5

COMPARISON OF QUAD CITIES UNIT 1 AND UNIT 2
TECHNICAL SPECIFICATIONS FOR THE
IDENTIFICATION OF TECHNICAL DIFFERENCES

SECTION 3.5/4.5
"ECCS SYSTEM"

Commonwealth Edison has conducted a comparison review of the Quad Cities Unit 1 and Unit 2 Technical Specifications to identify any technical differences in support of combining the Technical Specifications into one document. The intent of the review was not to identify any differences in presentation style (e.g. table formats, use of capital letters, etc.), punctuation or spelling errors, but rather to identify areas which the Technical Specifications are technically or administratively different.

The review of Section 3.5/4.5 "ECCS System" Sections I, J, and K revealed the following technical differences:

1. Unit 1 Page 3.5/4.5-20: The Average Planar Linear Heat Generation Rate (APLHGR) also serves a secondary function which is to assure fuel rod mechanical integrity.

   Unit 2 Page 3.5/4.5-13 and 14: Power operation with LHGRs at or below those specified in the CORE OPERATING LIMITS REPORT assures that the peak cladding temperature following a postulated loss-of-coolant accident will not exceed the 2200 °F limit. These values represent limits for operation to ensure conformance with 10 CFR 50 Appendix K only if they are more limiting than other design parameter. The maximum average planar LHGRs specified in the CORE OPERATING LIMITS REPORT at higher exposures result in a peak cladding temperature of less than 2200 °F. However, the maximum average planar LHGRs are specified in the CORE OPERATING LIMITS REPORT as limits because conformance calculations have not been performed to justify operation at LHGRs in excess of those shown.

The Unit 2 Bases material is adopted in theory by adopting the Bases material presented in the Standard Technical Specifications.

2. Unit 1 Page 3.5/4.5-21: The MCPR Operating Limit reflects an increase of 0.03 over the most limiting transient to allow continued operation with one feedwater heater out of service.

   Unit 2: Unit 2 does not have this paragraph.

ATTACHMENT 5

The material presented in the Unit 1 Bases represents a cycle specific added that is currently contained in the core operating limits report. Therefore, the material is not adopted in accordance with the STS.

3. Unit 2 Page 3.5/4.5-14a: This page contains the information related to the ODYN option B determination of the scram time input for the MCPR limit.

The information is not adopted in the proposed specification in accordance with the Standard Technical Specifications. The information is contained partially in the core operating limits report and within the technical manuals.

ATTACHMENT 6

SIGNIFICANT HAZARDS CONSIDERATIONS AND ENVIRONMENTAL ASSESSMENT EVALUATION

Technical Specification 3/4.11

"<u>POWER DISTRIBUTION LIMITS</u>"

ATTACHMENT 6

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Commonwealth Edison has evaluated this proposed amendment and determined that it involves no significant hazards consideration. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility, in accordance with the proposed amendment, would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3) Involve a significant reduction in a margin of safety.

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

In general, the proposed changes represent the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Implementation of these changes will provide increased reliability of equipment assumed to operate in the current safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits, and as such, will not significantly increase the probability or consequences of a previously evaluated accident.

Some of the proposed changes represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. These proposed changes are consistent with the current safety analyses and have been previously determined to represent sufficient requirements for the assurance of reliability of equipment assumed to operate in the safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits. As such, these changes will not significantly increase the probability or consequences of a previously evaluated accident.

The Generic Changes to the technical specifications involve administrative changes to format and arrangement of the material. As such, these changes cannot involve a significant increase in the probability or consequences of an accident previously evaluated.

The current specifications require the reactor to be placed in cold shutdown when a thermal limit was exceeded and not restored within the allotted 2 hours, but the proposed specifications require the reactor to be less than 25% of rated thermal power if this condition occurred. The change eliminates a shutdown and requires the power level to be reduced to the point that the limits are no longer applicable.

Therefore, the change will not increase the probability or consequences of an accident.

Create the possibility of a new or different kind of accident from any previously evaluated because:

In general, the proposed changes represent the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Others represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. These changes do not involve revisions to the design of the station. Some of the changes may involve revision in the operation of the station; however, these changes provide additional restrictions which are in accordance with the current safety analyses, or are to provide for additional testing or surveillance which will not introduce new failure mechanisms beyond those already considered in the current safety analyses. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Since the Generic Changes proposed to the technical specifications are administrative in nature, they cannot create the possibility of a new or different kind of accident from any previously evaluated.

The requirement to reduce thermal power to less than 25% of rated thermal power rather than place the reactor in cold shutdown will not create a new or different kind of accident because the thermal limits are not required in operational mode 1 when thermal power is less than 25% of rated thermal power.

Involve a significant reduction in the margin of safety because:

In general, the proposed changes represent the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Others represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. Some of the latter individual items may introduce minor reductions in the margin of safety when compared to the current requirements. However, other individual changes are the adoption of new requirements which will provide significant enhancement of the reliability of the equipment assumed to operate in the safety analysis, or provide enhanced assurance that specified parameters remain within their acceptance limits. These enhancements compensate for the individual minor reductions, such that taken together, the proposed changes will not significantly reduce the margin of safety.

The Generic Changes proposed in this amendment request are administrative in nature and, as such, do not involve a reduction in the margin of safety.

ATTACHMENT 6

ENVIRONMENTAL ASSESSMENT STATEMENT APPLICABILITY REVIEW

Commonwealth Edison has evaluated the proposed amendment against the criteria for the identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.20. It has been determined that the proposed changes meet the criteria for a categorical exclusion as provided under 10 CFR 51.22 (c)(9). This conclusion has been determined because the changes requested do not pose significant hazards consideration or do not involve a significant increase in the amounts, and no significant changes in the types, of any effluent that may be released offsite. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure. Therefore, the Environmental Assessment Statement is not applicable for these changes.

ATTACHMENT 7

GENERIC LETTER 87-09 IMPLEMENTATION

Technical Specification 3/4.11

"POWER DISTRIBUTION LIMITS"

ATTACHMENT 7

APPLICATION OF GENERIC LETTER 87-09
REVISION TO SPECIFICATION 3.0.D

The Dresden/Quad Cities Technical Specification Upgrade Program has implemented the recommendations of Generic Letter 87-09. Included in these recommendations was a revision to Standard Technical Specification 3.0.4 for which these stations had no corresponding restriction. Under the proposed Specification, entry into an operational mode or other specified condition is permitted under compliance with the Action requirements. Indicated below is the method of implementation for this recommendation for each Action requirement in this package.

PROPOSED TECH SPEC	ACTION	APPL. MODEs	CONT. OPS IN APP. COND?	CAT	CLARIFICATION
3.11.A		1, >25%	2 hrs	NO	Must reduce to < 25%
3.11.B	1	1, >25%	6 hrs	NO	Must reduce to < 25%
	2	1, >25%	6 hrs	NO	Must reduce to < 25%
	3	1, >25%	6 hrs	NO	Must reduce to < 25%
3.11.C	1	1, >25%	2 hrs	NO	Must reduce to < 25%
3.11.D		1, >25%	2 hrs	NO	Must reduce to < 25%
3.11.E (Dresden)		1, >25%	2 hrs	NO	Must reduce to < 25%

ATTACHMENT 1

EXECUTIVE SUMMARY

Technical Specification 3/4.12

"SPECIAL TEST EXCEPTIONS"

EXECUTIVE SUMMARY

The Dresden Technical Specification Upgrade Program (TSUP) was conceptualized in response to lessons learned from the Diagnostic Evaluation Team inspection and the frequent need for Technical Specification interpretations. A comparison study of the Standard Technical Specification (STS), later operating plant's Technical Specifications provisions and Quad Cities Technical Specifications was performed prior to the Dresden TSUP effort. The study identified potential improvements in clarifying requirements and requirements which are no longer consistent with current industry practices. The Dresden TSUP will enhance the Quad Cities TSUP currently under review by the NRC. As a result of the inconsistencies in the Quad Cities submittal compared to the Standard Technical Specifications (STS), Dresden's submittal will more closely follow the provisions of STS and in conjunction, Quad Cities will amend their submittal so that Quad Cities and Dresden are identical within equipment and plant design. The format for the Dresden TSUP will remain as a two column layout for human factors considerations. Additionally, chapter organizations will remain essentially unchanged.

The TSUP is not intended to be a complete adoption for the STS. Overall, the Dresden custom Technical Specifications provide for the safe operation of the plant and therefore, only an upgrade is deemed necessary.

In response to an NRC recommendation, Quad Cities combined the Unit 1 and Unit 2 Technical Specifications into one document. The Dresden Unit 2 and Unit 3 Technical Specifications will also be combined into one document. To accomplish the combination of the Units' Technical Specification, a comparison of the Unit 2 and Unit 3 Technical Specification was performed to identify any technical differences. The technical differences are identified in the proposed amendment package for each section.

The TSUP was identified as a station top priority and is currently contained in the Dresden Management Action Plan (DMAP). The TSUP goal is to provide a better tool to station personnel to implement their responsibilities and to ensure Dresden Station is operated in accordance with current industry practices. The improved Technical Specifications provide for enhanced operation of the plant. The program improves the operator's ability to use the Technical Specifications by more clearly defining the Limiting Conditions for Operation and required actions. The most significant improvement to the specifications is the addition of equipment operability requirements during shutdown conditions.

EXECUTIVE SUMMARY
(continued)
PROPOSED CHANGES TO TECHNICAL SPECIFICATION SECTION 3/4.12, "SPECIAL TEST EXCEPTIONS"

The current Dresden and Quad Cities Technical Specifications contain Applicability and Objective statements at the beginning of most sections. The proposed amendment will delete the "Objective" statement and integrates appropriate applicability statements within the specifications. This provides a clarification of the intended requirements and actions which are required when the specification cannot be met.

Proposed Section 3/4.12.A allows the primary containment integrity specifications to be suspended for low power physics testing.

Proposed Section 3/4.12.B delineates the requirements needed during shutdown margin demonstrations.

ATTACHMENT 2

DESCRIPTION OF CHANGES

Technical Specification 3/4.12

"SPECIAL TEST EXCEPTIONS"

ATTACHMENT 2

DESCRIPTION OF PROPOSED AMENDMENT

The changes proposed in this amendment request are made to improve the understanding and usability of the present technical specifications, and to incorporate the technical improvements from the Standard Technical Specifications (STS).

The present Dresden and Quad Cities Technical Specifications contain Applicability and Objective statements at the beginning of most sections. These statements are generic in nature and do not provide any useful information to the user of the technical specifications. The proposed change will delete the Objective statement and provide Applicability statements within each specification based on the STS. The proposed Applicability statement to be included in each specification will include the applicable operational modes or other conditions for which the Limiting Condition for Operation (LCO) must be satisfied.

The STS action provisions which delineate a specification 3.0.4 exception are not incorporated into the proposed specifications. The incorporation of the Generic Letter 87-09 change to STS specification 3.0.4 (Dresden and Quad Cities proposed 3.0.D specification) requires that each action be independently evaluated for applicability of the new specification. These evaluations are provided in Attachment 7.

Proposed section 3/4.12 is a new section but contains provisions that are currently allowed by the Technical Specifications.

SPECIFIC CHANGES

Section 3/4.12, Primary Containment Integrity

The proposed specification implements the current provisions contained in specification 3.7.A.2 of the Dresden and Quad Cities Technical Specifications in accordance with STS guidelines. The specification allows the primary containment integrity requirements to be suspended for the purpose of performing low power physics tests with thermal power less than 1% of rated thermal power and the reactor coolant temperature is less than 212°F. The temperature requirement is 212°F versus the STS requirement of 200°F for consistency with the proposed operational modes defined in Section 1.0 and the current specifications. The current technical specifications require the reactor power to be less than 5 MWt (~0.2%). The proposed specification uses 1% for consistency with other plants and STS. The proposed applicability is operational mode 2 during low power physics test. The proposed action requires the reactor mode switch to be placed in the shutdown position if thermal power is raised above the 1% level or the reactor coolant temperature becomes greater than 212°F. The proposed Surveillance Requirement (SR) is to verify thermal power and reactor coolant temperature are within the limits at least once per hour during low power physics tests.

ATTACHMENT 2

Section 3/4.12.B, Shutdown Margin Demonstrations

The proposed specification implements the current provisions contained in specification 3.3.B.3.b of the Dresden and Quad Cities Technical Specifications. The proposed specification allows the mode switch interlocks and control rod position provisions contained in specifications 3.10.A and 3.10.C to be suspended for the purpose of performing shutdown margin demonstrations provided the source range monitors are operable, the control rod sequence is verified by either the rod worth minimizer or a second qualified individual, the rod-out-notch-override function is not used, and no other core alterations are in progress. The proposed applicability is operational mode 5 during shutdown margin demonstrations. The proposed action requires the reactor mode switch to be placed in the shutdown or refuel position if any of the requirements are not met. The proposed SRs require that within 30 minutes prior to and at least once per 12 hours during the performance of the shutdown margin demonstration that the source range monitors are operable, the control rod sequence is being enforced, and that no other core alterations are in progress.

The remainder of the STS section 12 specifications are not adopted because the specifications are only applicable during the Startup Test Program.

The Bases for Section 3/4.12 are implemented in accordance with the proposed specifications.

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATIONS

Technical Specification 3/4.12

"SPECIAL TEST EXCEPTIONS"

3.12 - LIMITING CONDITIONS FOR OPERATION

A. PRIMARY CONTAINMENT INTEGRITY

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

APPLICABILITY:

OPERATIONAL MODE 2, during low power PHYSICS TESTS.

ACTION:

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

4.12 - SURVEILLANCE REQUIREMENTS

A. PRIMARY CONTAINMENT INTEGRITY

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

3.12 - LIMITING CONDITIONS FOR OPERATION

B. SHUTDOWN MARGIN Demonstrations

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

1. The source range monitors are OPERABLE with the RPS circuitry "shorting links" removed per Specification 3.10.B.
2. The rod worth minimizer is OPERABLE per Specification 3.3.L and is programmed for the SHUTDOWN MARGIN demonstration, or conformance with the SHUTDOWN MARGIN demonstration procedure is verified by a second licensed operator or other technically qualified individual.
3. The "rod-out-notch-override" control shall not be used during out-of-sequence movement of the control rods.
4. No other CORE ALTERATION(s) are in progress.

APPLICABILITY:

OPERATIONAL MODE 5, during SHUTDOWN MARGIN demonstrations.

ACTION:

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

4.12 - SURVEILLANCE REQUIREMENTS

B. SHUTDOWN MARGIN Demonstrations

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

1. The source range monitors are OPERABLE with the RPS circuitry "shorting links" removed per Specification 3.10.B,
2. The rod worth minimizer is OPERABLE with the required program per Specification 3.3.L or a second licensed operator or other technically qualified individual is present and verifies compliance with the SHUTDOWN MARGIN demonstration procedures, and
3. No other CORE ALTERATION(s) are in progress.

BASES

3/4.12.A PRIMARY CONTAINMENT INTEGRITY

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

3/4.12.B SHUTDOWN MARGIN Demonstrations

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

3.12 - LIMITING CONDITIONS FOR OPERATION

A. PRIMARY CONTAINMENT INTEGRITY

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

APPLICABILITY:

OPERATIONAL MODE 2, during low power PHYSICS TESTS.

ACTION:

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

4.12 - SURVEILLANCE REQUIREMENTS

A. PRIMARY CONTAINMENT INTEGRITY

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

3.12 - LIMITING CONDITIONS FOR OPERATION

B. SHUTDOWN MARGIN Demonstrations

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

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

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

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

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

APPLICABILITY:

OPERATIONAL MODE 5, during SHUTDCWN MARGIN demonstrations.

ACTION:

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

4.12 - SURVEILLANCE REQUIREMENTS

B. SHUTDOWN MARGIN Demonstrations

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

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

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

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

BASES

3/4.12.A PRIMARY CONTAINMENT INTEGRITY

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

3/4.12.B SHUTDOWN MARGIN Demonstrations

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

ATTACHMENT 4

EXISTING TECHNICAL SPECIFICATIONS

Technical Specification 3/4.12

"SPECIAL TEST EXCEPTIONS"

ATTACHMENT 4

DELETION OF CURRENT TECHNICAL SPECIFICATIONS

This technical specification amendment is a new section for the Dresden Unit 2 and Unit 3 Technical Specifications. Therefore, no technical specifications are being deleted with this amendment.

ATTACHMENT 4

DELETION OF CURRENT TECHNICAL SPECIFICATIONS

This technical specification amendment is a new section for the Quad Cities Unit 1 and Unit 2 Technical Specifications. Therefore, no technical specifications are being deleted with this amendment.

ATTACHMENT 5

DRESDEN 2/3 DIFFERENCES

Technical Specification 3/4.12

"SPECIAL TEST EXCEPTIONS"

ATTACHMENT 5

COMPARISON OF DRESDEN UNIT 2 AND UNIT 3
TECHNICAL SPECIFICATIONS FOR THE
IDENTIFICATION OF TECHNICAL DIFFERENCES

SECTION 3/4.12
"SPECIAL TEST EXCEPTIONS"

Commonwealth Edison has conducted a comparison review of the Dresden Unit 2 and Unit 3 Technical Specifications to identify any technical differences in support of combining the Technical Specifications into one document. The intent of the review was not to identify any differences in presentation style (e.g. table formats, use of capital letters, etc.), punctuation or spelling errors, but rather to identify areas which the Technical Specifications are technically or administratively different.

Proposed section 3/4.12, Special Test Exceptions is a new section and therefore, does not contain any technical differences.

ATTACHMENT 5

QUAD CITIES 1/2 DIFFERENCES

Technical Specification 3/4.12

"SPECIAL TEST EXCEPTIONS"

ATTACHMENT 5

COMPARISON OF QUAD CITIES UNIT 1 AND UNIT 2
TECHNICAL SPECIFICATIONS FOR THE
IDENTIFICATION OF TECHNICAL DIFFERENCES

SECTION 3/4.12
"SPECIAL TEST EXCEPTIONS"

Commonwealth Edison has conducted a comparison review of the Quad Cities Unit 1 and Unit 2 Technical Specifications to identify any technical differences in support of combining the Technical Specifications into one document. The intent of the review was not to identify any differences in presentation style (e.g. table formats, use of capital letters, etc.), punctuation or spelling errors, but rathe to identify areas which the Technical Specifications are technically or administratively different.

Proposed section 3/4.12, Special Test Exceptions is a new section and therefore, does not contain any technical differences.

ATTACHMENT 6

SIGNIFICANT HAZARDS CONSIDERATIONS AND ENVIRONMENTAL ASSESSMENT EVALUATION

Technical Specification 3/4.12

"<u>SPECIAL TEST EXCEPTIONS</u>"

ATTACHMENT 6

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Commonwealth Edison has evaluated this proposed amendment and determined that it involves no significant hazards consideration. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility, in accordance with the proposed amendment, would not:

1) Involve a signif.cant increase in the probability or consequences of an accident previously evaluated; or

2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3) Involve a significant reduction in a margin of safety.

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

The proposed Specification 3/4.12 is a new section which will provide the user with a format that will allow better access to needed information and provide concise Applicability and Action requirements. The additions of Applicability and Action requirements represent clarification of intended requirements that do not presently state all required conditions of operability or provide clearly stated Action statements if the requirements are not met. The combining of the two sections and the added requirements follow Standard Technical Specifications (STS) guidelines that are in use at many operating BWRs with similar design and operating configurations as Dresden and Quad Cities Stations.

The proposed Section 3/4.12 involves the relocation of present requirements into one section identical to STS provisions. The changes also implement the Applicability and Action provisions of the STS and later operating BWR plants that have been evaluated and found acceptable for use at Dresden and Quad Cities. Present Surveillance Requirements are replaced, where applicable, with proven STS guidelines that are being used at plants with a system similar to that at Dresden and Quad Cities. The changes in the present Surveillance Requirements add testing requirements that are not presently in the Dresden and Quad Cities technical specifications. The proposed changes do not affect accident assumptions other than a minor increase in the initial power level (~0.2% to 1%) and as such, do not involve a significant increase in the probability of an accident previously evaluated. The proposed specifications add additional requirements to specifications currently contained in the Technical Specifications. Since the proposed changes to the Technical Specifications implement requirements that have been demonstrated to provide acceptable operability provisions at other facilities with a design similar to that at Dresden and Quad Cities, the proposed changes do not significantly increase the consequences of an accident previously evaluated.

ATTACHMENT 6

The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated because:

> The proposed administrative changes to the format and arrangement of material do not affect technical requirements or assumptions of any potential accident and; therefore, cannot create the possibility of a new or different kind of accident from any previously evaluated.
>
> The proposed addition of Applicability and Action requirements enhance the understanding and usability of the Technical Specifications and thus represent an improvement over present specifications. New requirements are modeled after those in use at operating BWRs and do not represent requirements that will adversely affect potential accident analyses or assumptions. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed changes do not involve a significant reduction in the margin of safety because:

> The proposed administrative changes to format, arrangement of material, clarification of requirements and other non technical changes do not affect any safety aspects of the plant and as such can not involve a significant reduction in the margin of safety.

In addition, the commission has provided guidance concerning the application of standards for determining whether significant hazards consideration exists by providing certain examples (51 FR 7751) of amendments that are considered not likely to involve significant hazards considerations. Commonwealth Edison has reviewed the proposed changes against these examples and believes that the proposed changes fall within the scope of example (ii) "a change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications".

The proposed amendment does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings or a significant relaxation of the bases for the limiting conditions for operations. Therefore, based on the guidance provided in the Federal Register and the criteria established in 10 CFR 50.92(c), the proposed change does not constitute a significant hazards consideration.

ATTACHMENT 6

ENVIRONMENTAL ASSESSMENT STATEMENT APPLICABILITY REVIEW

Commonweal'h Edison has evaluated the proposed amendment against the criteria for the identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.20. It has been determined that the proposed changes meet the criteria for a categorical exclusion as provided under 10 CFR 51.22 (c)(9). This conclusion has been determined because the changes requested do not pose significant hazards consideration or do not involve a significant increase in the amounts, and no significant changes in the types, of any effluent that may be released offsite. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure. Therefore, the Environmental Assessment Statement is not applicable for these changes.

ATTACHMENT 7

GENERIC LETTER 87-09 IMPLEMENTATION

Technical Specification 3/4.12

"SPECIAL TEST EXCEPTIONS"

ATTACHMENT 7

APPLICATION OF GENERIC LETTER 87-09
REVISION TO SPECIFICATION 3.0.D

The Dresden/Quad Cities Technical Specification Upgrade Program has implemented the recommendations of Generic Letter 87-09. Included in these recommendations was a revision to Standard Technical Specification 3.0.4 (Dresden and Quad Cities proposed 3.0.D specification) for which these stations had no corresponding restriction. Under the proposed Specification, entry into an operational mode or other specified condition is permitted under compliance with the Action requirements. Indicated below is the method of implementation for this recommendation for each Action requirement in this package.

PROPOSED TECH SPEC	ACTION	APPL. MODEs	CONT. OPS IN APP. COND?	CAT	CLARIFICATION
3.12.A		2, during low power Physics Tests	No	No	Immediately place the reactor mode switch in the Shutdown position
3.12.B		5, during Shutdown Margin	No	No	Immediately place the reactor mode switch in the Shutdown or Refuel position