ENCLOSURE 2

VOLUME 4

SEQUOYAH NUCLEAR PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 2.0
SAFETY LIMITS

Revision 0

LIST OF ATTACHMENTS

ATTACHMENT 1

ITS Chapter 2.0, SAFETY LIMITS (SLs)

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A01

ITS ITS Chapter 2.0

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (Tavg) shall not exceed the limits ~~shown in Figure 2.1-1~~ in the COLR and the following SLs shall not be exceeded:. LA01

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.132 for the BHTP correlation, ≥ 1.21 for the BWU-N correlation, and ≥ 1.21 for the BWCMV correlation.

2.1.1.2 The maximum local fuel pin centerline temperature shall be maintained ≤ 4901°F, decreasing by 13.7°F per 10,000 MWD/MTU of burnup for COPERNIC applications, and ≤ 4642°F, decreasing by 58°F per 10,000 MWD/MTU of burnup for TACO3 applications.

Applicability APPLICABILITY: MODES 1 and 2.

ACTION:

2.2.1 If SL 2.1.1 is violated, restore compliance and be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

Applicability APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

2.2.2.1 MODES 1 and 2

2.2.2.1 Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

2.2.2.2 MODES 3, 4 and 5

2.2.2.2 Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

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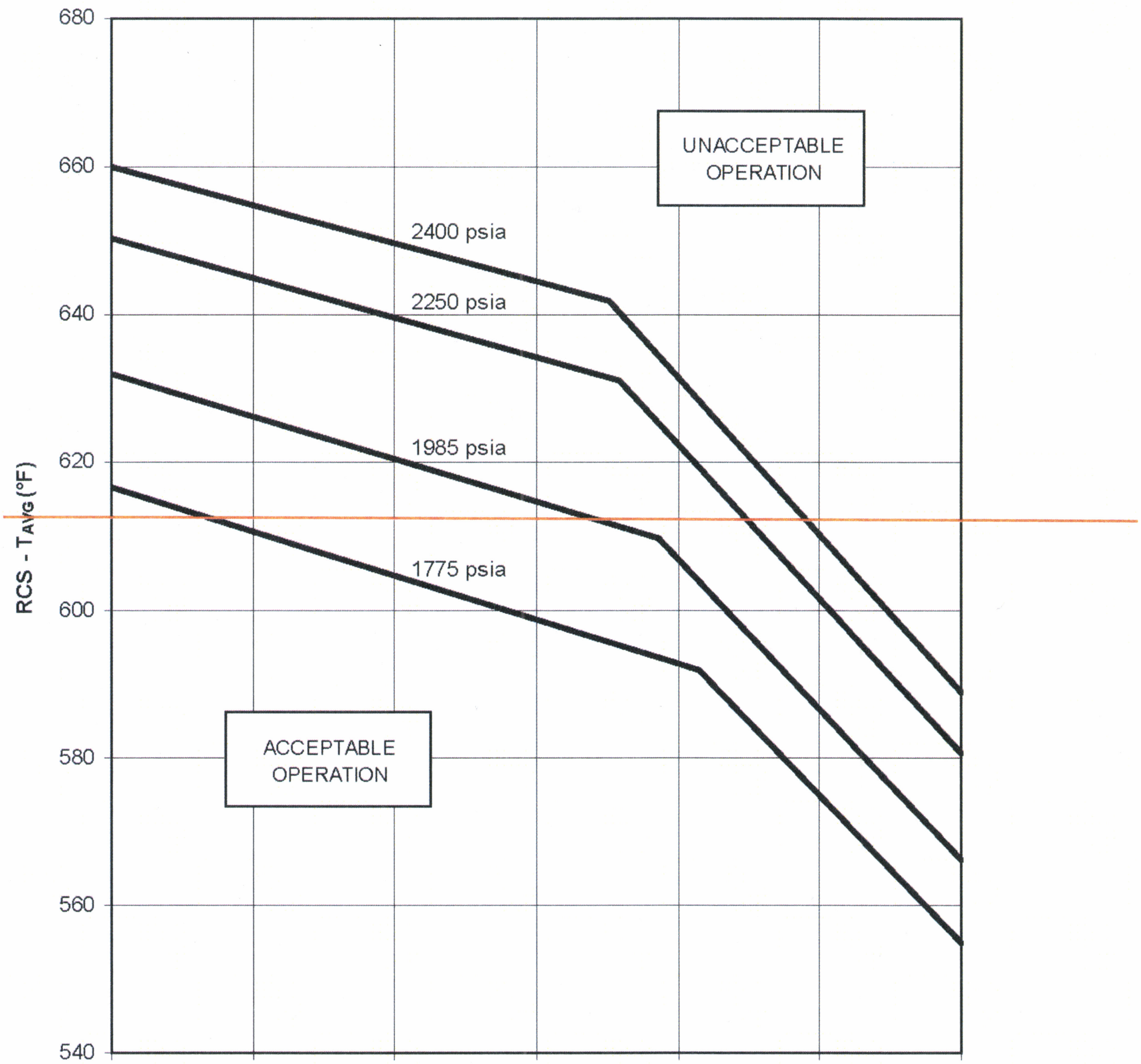
ITS A01 ITS Chapter 2.0

~~Figure 2.1-1 Reactor Core Safety Limit - Four Loops in Operation~~



LA01

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 2.1 SAFETY LIMITS

REACTOR CORE

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Applicability APPLICABILITY: MODES 1 and 2.

ACTION:

2.2.1 If SL 2.1.1 is violated, restore compliance and be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

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Applicability APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

2.2.2.1 MODES 1 and 2

2.2.2.1 Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

2.2.2.2 MODES 3, 4 and 5

2.2.2.2 Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

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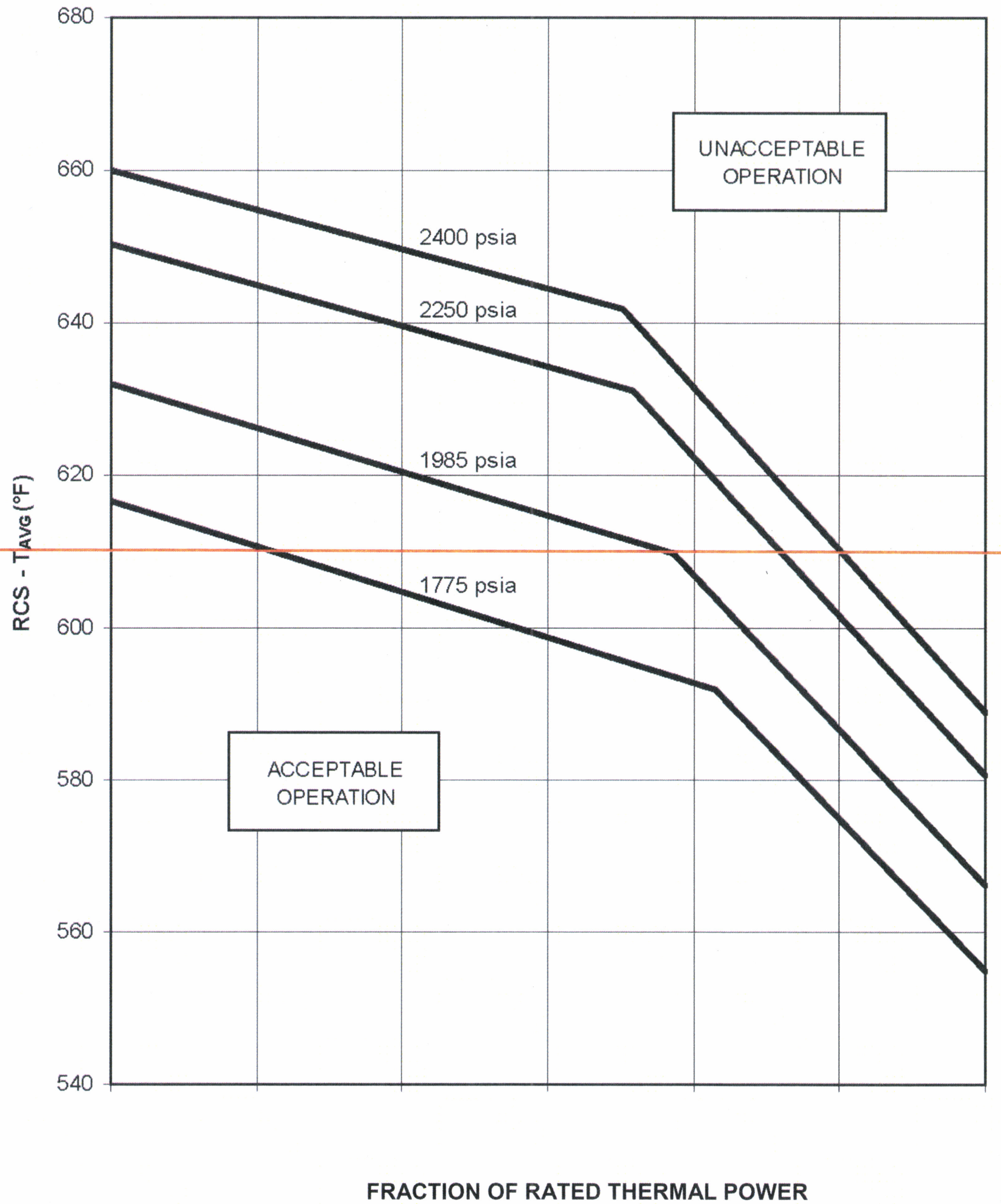
ITS

A01

ITS Chapter 2.0

~~Figure 2.1-1 Reactor Core Safety Limit-Four Loops in Operation~~



LA01

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DISCUSSION OF CHANGES
ITS CHAPTER 2.0, SAFETY LIMITS (SLs)

ADMINISTRATIVE CHANGES

A01 In the conversion of the Sequoyah Nuclear Plant (SQN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 4.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 6 – Removal of Cycle – Specific Limits from the Technical Specifications to the Core Operating Limits Report)* CTS 2.1.1 requires the combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) not to exceed the limits shown in Figure 2.1-1. ITS 2.1.1 states the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR. This changes the CTS by moving limits that must be confirmed on a cycle specific bases to the COLR. The Reactor Core safety limits are retained in Technical Specification Chapter 2.0.

The removal of these cycle specific parameter limits from the Technical Specifications to the COLR and the retention of the limiting Safety Limits in the Technical Specifications is acceptable because the cycle specific limits are developed or utilized under NRC-approved methodologies that ensure the Safety Limits are met. The NRC documented in Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," that this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the Safety Limits. NRC-approved Topical Report WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report," determined that the specific values for these parameters may be relocated to the COLR provided the SLs continue to appear in the Technical Specifications. The methodologies used to develop the parameters in the COLR were approved by the NRC in accordance with Generic Letter 88-16. Additionally, this change is acceptable because the removed information will be adequately controlled in the COLR under the requirements provided in ITS 5.6.3, "Core Operating Limits Report." ITS 5.6.3 ensures that the applicable limits of the safety analysis are met (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such as SDM, transient analysis limits, and accident analysis limits). This change is designated as a less restrictive removal of detail change because information relating to cycle specific parameter limits is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

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Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

SLs
2.0

2.0 SAFETY LIMITS (SLs)

2.1

2.1 SLs

2.1.1, 2.1.1 Applicability

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

2.1.1.1

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ ~~[1.17 for the WRB-1/WRB-2 DNB correlations]~~. INSERT 1 1

maximum local fuel pin

2.1.1.2

2.1.1.2 The ~~peak fuel~~ centerline temperature shall be maintained ~~< [5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup]~~. INSERT 2 2 1

2.1.2, 2.1.2 Applicability

2.1.2 Reactor Coolant System Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ ~~[~~2735~~]~~ psig. 1

2.2 SAFETY LIMIT VIOLATIONS

2.1.1 ACTION

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.1.2 ACTION

2.2.2 If SL 2.1.2 is violated:

2.1.2 ACTION

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.1.2 ACTION

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

CTS 2.1.1

1 **INSERT 1**

1.132 for the BHTP correlation, ≥ 1.21 for the BWU-N correlation, and ≥ 1.21 for the BWCMV correlation.

1 **INSERT 2**

≤ 4901°F, decreasing by 13.7°F per 10,000 MWD/MTU of burnup for COPERNIC applications, and ≤ 4642°F, decreasing by 58°F per 10,000 MWD/MTU of burnup for TACO3 applications

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2.0 SAFETY LIMITS (SLs)

2.1 **2.1 SLs**

2.1.1, 2.1.1 Applicability

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

2.1.1.1 — 2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ ~~[1.17 for the WRB-1/WRB-2 DNB correlations]~~ INSERT 1. 1

2.1.1.2 — 2.1.1.2 The ~~peak fuel~~ maximum local fuel pin centerline temperature shall be maintained ~~< [5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup]~~ INSERT 2. 2 1

2.1.2, 2.1.2 Applicability

2.1.2 Reactor Coolant System Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ ~~[~~2735~~]~~ psig. 1

2.2 SAFETY LIMIT VIOLATIONS

2.1.1 ACTION — 2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.1.2 ACTION — 2.2.2 If SL 2.1.2 is violated:

2.1.2 ACTION — 2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.1.2 ACTION — 2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

1 **INSERT 1**

1.132 for the BHTP correlation, ≥ 1.21 for the BWU-N correlation, and ≥ 1.21 for the BWCMV correlation.

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≤ 4901°F, decreasing by 13.7°F per 10,000 MWD/MTU of burnup for COPERNIC applications, and ≤ 4642°F, decreasing by 58°F per 10,000 MWD/MTU of burnup for TACO3 applications

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JUSTIFICATION FOR DEVIATIONS
ITS CHAPTER 2.0, SAFETY LIMITS (SLs)

1. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

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Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

INSERT 1

The restrictions of this SL prevent overheating of the fuel ~~and~~ cladding~~, as well as possible~~ cladding perforation, which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature. 1

~~Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.~~

corresponding significant

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the ~~resultant sharp~~ reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant. INSERT 2 1

INSERT 3 1

The proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents violation of the reactor core SLs.

1 **INSERT 1**

(due to departure from nucleate boiling) and overheating of the fuel pellet (centerline fuel melt(CFM)), either of which could result in

1 **INSERT 2**

from the outer surface of the cladding to the reactor coolant water

1 **INSERT 3**

DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. The DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

To meet the DNB Design Basis, a statistical core design (SCD) process has been used to develop an appropriate statistical DNBR design limit. Uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 percent probability at a 95 percent confidence level that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. This DNBR uncertainty derived from the SCD analysis, combined with the applicable DNB critical heat flux correlation limit, establishes the statistical DNBR design limit which must be met in plant safety analysis using values of input parameters without adjustment for uncertainty.

Operation above the maximum local linear heat generation rate for fuel melting could result in excessive fuel pellet temperature and cause melting of the fuel at its centerline. Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant. The melting point of uranium dioxide varies slightly with burnup. As uranium is depleted and fission products produced, the net effect is a decrease in the melting point. Fuel centerline temperature is not a directly measurable parameter during operation. The maximum local fuel pin centerline temperature is maintained by limiting the local linear heat generation rate in the fuel. The local linear heat generation rate in the fuel is limited so that the maximum fuel centerline temperature will not exceed the value acceptance criteria in the safety analysis.

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1

INSERT 3 (cont)

The curves provided in the COLR show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the safety analysis DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These lines are bounding for all fuel types. The curves provided in the COLR are based upon enthalpy rise hot channel factors that result in acceptable DNBR performance of each fuel type. Acceptable DNBR performance is assured by operation within the DNB-based Limiting Safety Limit System Settings (Reactor Trip System trip limits). The plant trip set points are verified to be less than the limits defined by the safety limit lines provided in the COLR converted from power to delta-temperature and adjusted for uncertainty.

The limiting heat flux conditions for DNB are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance or Delta-I (ΔI) is within the limits of the f_1 (Delta I) function of the Overtemperature Delta Temperature trip. When the axial power imbalance exceeds the tolerance (or deadband) of the $f_1(\Delta I)$ trip reset function, the Overtemperature Delta Temperature trip set point is reduced by the values in the COLR to provide protection required by the core safety limits.

Similarly, the limiting linear heat generation rate conditions for CFM are higher than those calculated for the range of all control rods from fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance or Delta-I (ΔI) is within the limits of the $f_2(\Delta I)$ function of the Overpower-Delta Temperature trip. When the axial power imbalance exceeds the tolerance (or deadband) of the $f_2(\Delta I)$ trip reset function, the Overpower-Delta Temperature trip set point is reduced by the values specified in the COLR to provide protection required by the core safety limits.

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Trip System setpoints (Ref. 2), in combination with ~~all~~ the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, RCS Flow, ΔI, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities. 1

Automatic enforcement of these reactor core SLs is provided by the appropriate operation of the RPS and the steam generator safety valves.

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the U FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded. 1

SAFETY LIMITS

The figure provided in the COLR shows the loci of points of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation.

The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB and

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

BASES

SAFETY LIMITS (continued)

The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs. If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. U FSAR, Section ~~[~~7.2~~]~~. 1 2

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure ~~coolant~~ boundary (RCPB) design conditions are not to be exceeded 3
during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is ~~2500 psia~~ 2485 psig. During normal operation 1
and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of

BASES

APPLICABLE SAFETY ANALYSES (continued)

external load without a direct reactor trip. During the transient, no control actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.

The Reactor Trip System setpoints (Ref. 5), together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of any of the following:

a. Pressurizer power operated relief valves (PORVs)~~,~~ ;

~~b. Steam line relief valve,~~

b ~~c~~. Steam Dump System~~,~~ ;

c ~~d~~. Reactor Control System~~,~~ ;

d ~~e~~. Pressurizer Level Control System~~,~~ ; or

e ~~f~~. Pressurizer spray valve.

4 4 5 4 4 4

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under ~~[~~USAS, Section B31.1 (Ref. 6)~~]~~ is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig. 2

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BASES

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

SAFETY LIMIT VIOLATIONS

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4).

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000, 1971 1

3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.

4. 10 CFR 100.

5. U FSAR, Section ~~[~~7.2~~]~~. 1 2

6. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

INSERT 1

The restrictions of this SL prevent overheating of the fuel ~~and~~ cladding~~, as well as possible~~ cladding perforation, which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature. 1

~~Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.~~

corresponding significant

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the ~~resultant sharp~~ reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

INSERT 2 1

INSERT 3 1

The proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents violation of the reactor core SLs.

1 **INSERT 1**

(due to departure from nucleate boiling) and overheating of the fuel pellet (centerline fuel melt(CFM)), either of which could result in

1 **INSERT 2**

from the outer surface of the cladding to the reactor coolant water

1 **INSERT 3**

DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. The DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

To meet the DNB Design Basis, a statistical core design (SCD) process has been used to develop an appropriate statistical DNBR design limit. Uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 percent probability at a 95 percent confidence level that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. This DNBR uncertainty derived from the SCD analysis, combined with the applicable DNB critical heat flux correlation limit, establishes the statistical DNBR design limit which must be met in plant safety analysis using values of input parameters without adjustment for uncertainty.

Operation above the maximum local linear heat generation rate for fuel melting could result in excessive fuel pellet temperature and cause melting of the fuel at its centerline. Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant. The melting point of uranium dioxide varies slightly with burnup. As uranium is depleted and fission products produced, the net effect is a decrease in the melting point. Fuel centerline temperature is not a directly measurable parameter during operation. The maximum local fuel pin centerline temperature is maintained by limiting the local linear heat generation rate in the fuel. The local linear heat generation rate in the fuel is limited so that the maximum fuel centerline temperature will not exceed the value acceptance criteria in the safety analysis.

1

INSERT 3 (cont)

The curves provided in the COLR show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the safety analysis DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These lines are bounding for all fuel types. The curves provided in the COLR are based upon enthalpy rise hot channel factors that result in acceptable DNBR performance of each fuel type. Acceptable DNBR performance is assured by operation within the DNB-based Limiting Safety Limit System Settings (Reactor Trip System trip limits). The plant trip set points are verified to be less than the limits defined by the safety limit lines provided in the COLR converted from power to delta-temperature and adjusted for uncertainty.

The limiting heat flux conditions for DNB are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance or Delta-I (ΔI) is within the limits of the f_1 (Delta I) function of the Overtemperature Delta Temperature trip. When the axial power imbalance exceeds the tolerance (or deadband) of the $f_1(\Delta I)$ trip reset function, the Overtemperature Delta Temperature trip set point is reduced by the values in the COLR to provide protection required by the core safety limits.

Similarly, the limiting linear heat generation rate conditions for CFM are higher than those calculated for the range of all control rods from fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance or Delta-I (ΔI) is within the limits of the $f_2(\Delta I)$ function of the Overpower-Delta Temperature trip. When the axial power imbalance exceeds the tolerance (or deadband) of the $f_2(\Delta I)$ trip reset function, the Overpower-Delta Temperature trip set point is reduced by the values specified in the COLR to provide protection required by the core safety limits.

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BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Trip System setpoints (Ref. 2), in combination with ~~all~~ the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, RCS Flow, ΔI, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities. 1

Automatic enforcement of these reactor core SLs is provided by the appropriate operation of the RPS and the steam generator safety valves.

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the U FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded. 1

SAFETY LIMITS

The figure provided in the COLR shows the loci of points of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation.

The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB and

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

BASES

SAFETY LIMITS (continued)

The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs. If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. U FSAR, Section ~~[~~7.2~~]~~. 1 2

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B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure ~~coolant~~ boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding. 3

The design pressure of the RCS is ~~2500 psia~~ 2485 psig. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3). 1

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of

BASES

APPLICABLE SAFETY ANALYSES (continued)

external load without a direct reactor trip. During the transient, no control actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.

The Reactor Trip System setpoints (Ref. 5), together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of any of the following:

a. Pressurizer power operated relief valves (PORVs)~~,~~ ; 4

~~b. Steam line relief valve,~~ 4

b ~~c~~. Steam Dump System~~,~~ ; 5

c ~~d~~. Reactor Control System~~,~~ ; 4

d ~~e~~. Pressurizer Level Control System~~,~~ ; or 4

e ~~f~~. Pressurizer spray valve. 4

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under ~~[~~USAS, Section B31.1 (Ref. 6)~~]~~ is 120% of design pressure. 2
The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig.

BASES

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

SAFETY LIMIT VIOLATIONS

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4).

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000, 1971. 1

3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.

4. 10 CFR 100.

5. UFSAR, Section ~~[~~7.2~~]~~. 1 2

6. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.

JUSTIFICATION FOR DEVIATIONS
ITS CHAPTER 2.0 BASES, SAFETY LIMITS (SLs)

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Typographical/grammatical error corrected.

4. The steam line relief valves are removed from the list of items that have no credit taken for operation. The steam line safety valves are credited with protecting the Reactor Coolant System and the steam generators against overpressure for all load losses. Additionally, the subsequent items have been renumbered.

5. The punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS CHAPTER 2.0, SAFETY LIMITS (SLs)

There are no specific No Significant Hazards Considerations for this Specification.

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