NORTH ANNA POWER STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

CHAPTER 2.0 - SAFETY LIMITS

CHAPTER 2.0 - SAFETY LIMITS

IMPROVED TECHNICAL SPECIFICATIONS

2.0 SAFETY LIMITS (SLs)

2.1 SLs

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 The departure from nucleate boiling ratio (DNBR) shall be maintained greater than or equal to the 95/95 DNBR criterion for the DNB correlations and methodologies specified in Section 5.6.5.

2.1.1.2 The peak fuel centerline temperature shall be maintained < 4700°F.

2.1.2 RCS Pressure SL

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

2.2 SL Violations

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

2.2.2 If SL 2.1.2 is violated:

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

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

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CHAPTER 2.0 - SAFETY LIMITS

IMPROVED TECHNICAL SPECIFICATIONS BASES

B 2.1 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

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.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that 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.

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.

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.

(continued)

BASES

BACKGROUND (continued)

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

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 allowable values (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, and flow, AFD, 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.

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

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

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.

(continued)

BASES

SAFETY LIMITS (continued)

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.

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 and main steam safety valves ensures that for variations in the THERMAL POWER, RCS pressure, RCS average temperature, RCS flow rate, and AFD 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 main steam 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. Allowable values 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.

BASES

SAFETY LIMIT VIOLATIONS	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. UFSAR, Section 3.1.6. 2. UFSAR, Section 7.2.

B 2.1 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 during operating conditions, the continued integrity of the RCS is ensured. According to GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure 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. The design pressure of the RCS is 2500 psia. 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). 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 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 allowable values (Ref. 5), together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The reactor high pressure trip allowable value 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 the following:

a. Pressurizer power operated relief valves (PORVs);

b. Steam Generator PORVs;

c. Steam Dump System;

d. Reactor Control System;

e. Pressurizer Level Control System; or

f. Pressurizer spray valve.

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.

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. UFSAR, Sections 3.1.10, 3.1.11, and 3.1.24.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.
4. 10 CFR 100.

BASES

REFERENCES (continued)

5. UFSAR, Section 7.2.

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

CHAPTER 2.0 - SAFETY LIMITS

IMPROVED STANDARD TECHNICAL SPECIFICATIONS

MARKUP AND JUSTIFICATION FOR DEVIATIONS

CTS

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1

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 ~~SLs~~ limits specified in ~~Figure 2.1.1-1.~~ Insert

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2.1.2

2.1.2 RCS Pressure SL

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

2.1.1 Action

2.2 SL Violations

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:

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

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

~~2.2.3 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.~~

~~2.2.4 Within 24 hours, notify the [Plant Superintendent and Vice President—Nuclear Operations].~~

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~~2.2.5 Within 30 days a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the [offsite review function], and the [Plant Superintendent, and Vice President—Nuclear Operations].~~

~~2.2.6 Operation of the unit shall not be resumed until authorized by the NRC.~~

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INSERT

the COLR; and the following SLs shall not be exceeded.

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained greater than or equal to the 95/95 DNBR criterion for the DNB correlations and methodologies specified in Section 5.6.5.

2.1.1.2 The peak fuel centerline temperature shall be maintained < 4700 °F.

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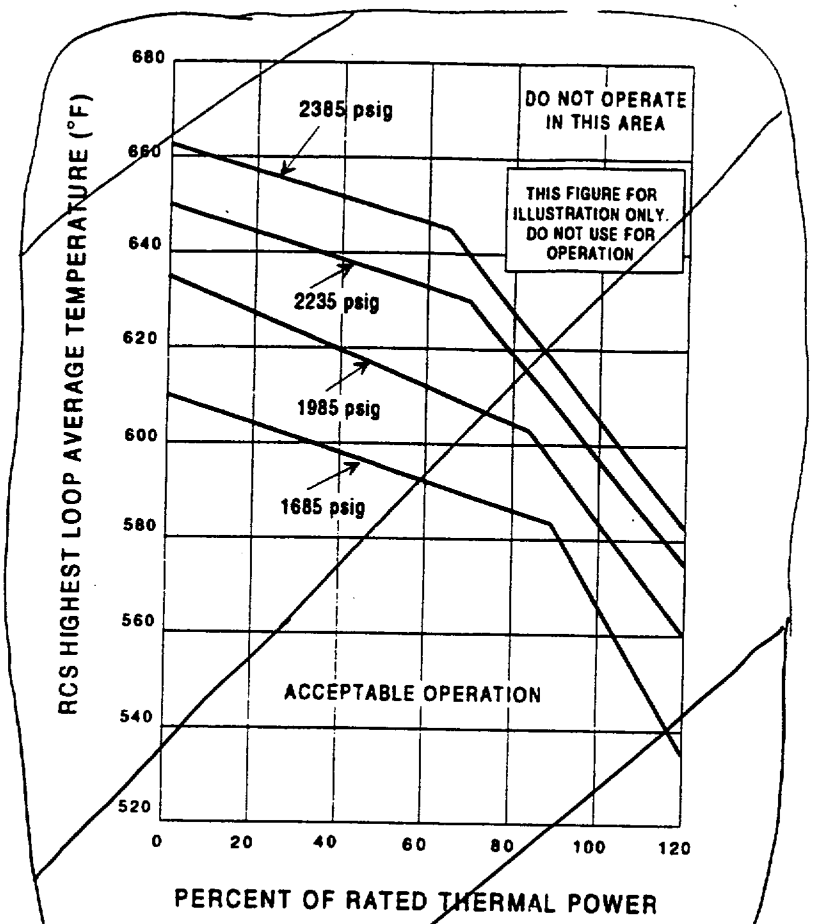

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits

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1. The brackets have been removed and the proper plant specific information/value has been provided.

CHAPTER 2.0 - SAFETY LIMITS

IMPROVED STANDARD TECHNICAL SPECIFICATIONS BASES

MARKUP AND JUSTIFICATION FOR DEVIATIONS

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

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.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that 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.

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.

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.

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BASES

Main steam

BACKGROUND (continued)

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

②

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.

allowable values

②

and flow, AFD,

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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, 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.

Automatic enforcement of these reactor core SLs is provided by the ~~following functions:~~

appropriate operation of the RPS and the main steam safety valves.

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②

~~a. High pressurizer pressure trip;~~

~~b. Low pressurizer pressure trip;~~

~~c. Overtemperature ΔT trip;~~

~~d. Overpower ΔT trip;~~

~~e. Power Range Neutron Flux trip; and~~

~~f. Steam generator safety valves.~~

~~The limitation that the average enthalpy in the hot leg be less than or equal to the enthalpy of saturated liquid also ensures that the ΔT measured by instrumentation, used in the RPS design as a measure of core power, is proportional to core power.~~

(continued)

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BASES

allowable values

APPLICABLE SAFETY ANALYSES (continued)

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.

(2) (4)

(2)

figure

the COLR show

SAFETY LIMITS

The ~~curves~~ provided in ~~Figure B 2.1.1-1 show~~ 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.

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~~The curves are based on enthalpy hot channel factor limits provided in the COLR. The dashed line of Figure B 2.1.1-1 shows an example of a limit curve at 2235 psig. In addition, it illustrates the various RPS functions that are designed to prevent the unit from reaching the limit.~~

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~~The SL is higher than the limit calculated when the AFD is within the limits of the $F_1(\Delta I)$ function of the overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature ΔT reactor trips will reduce the setpoints to provide protection consistent with the reactor core SLs (Refs. 3 and 4).~~

Insert

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,

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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.

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 and main steam safety valves ensures that for variations in the THERMAL POWER, RCS pressure, RCS average temperature, RCS flow rate, and AFD that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

BASES

APPLICABILITY (continued)	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.~~ (4) TSTF-5

~~2.2.1~~

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.

~~2.2.3~~

~~If SL 2.1.1 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 5).~~

~~2.2.4~~

~~If SL 2.1.1 is violated, the Plant Superintendent and the Vice President—Nuclear Operations shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.~~ TSTF-5

~~2.2.5~~

~~If SL 2.1.1 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 6). A copy of the report shall also be provided to the Plant Superintendent and the Vice President—Nuclear Operations.~~

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BASES

SAFETY LIMIT VIOLATIONS (continued)

~~2.2.6~~

~~If SL 2.1.1 is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.~~

TSTF-5

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 10.~~ UFSAR Section 3.1.6. ①

2. UFSAR, Section [7.2]. ②③

3. ~~WCAP-8746-A, March 1977.~~

4. ~~WCAP-9273-NP-A, July 1985.~~ TSTF-339

5. ~~10 CFR 50.72.~~

6. ~~10 CFR 50.73.~~ TSTF-5

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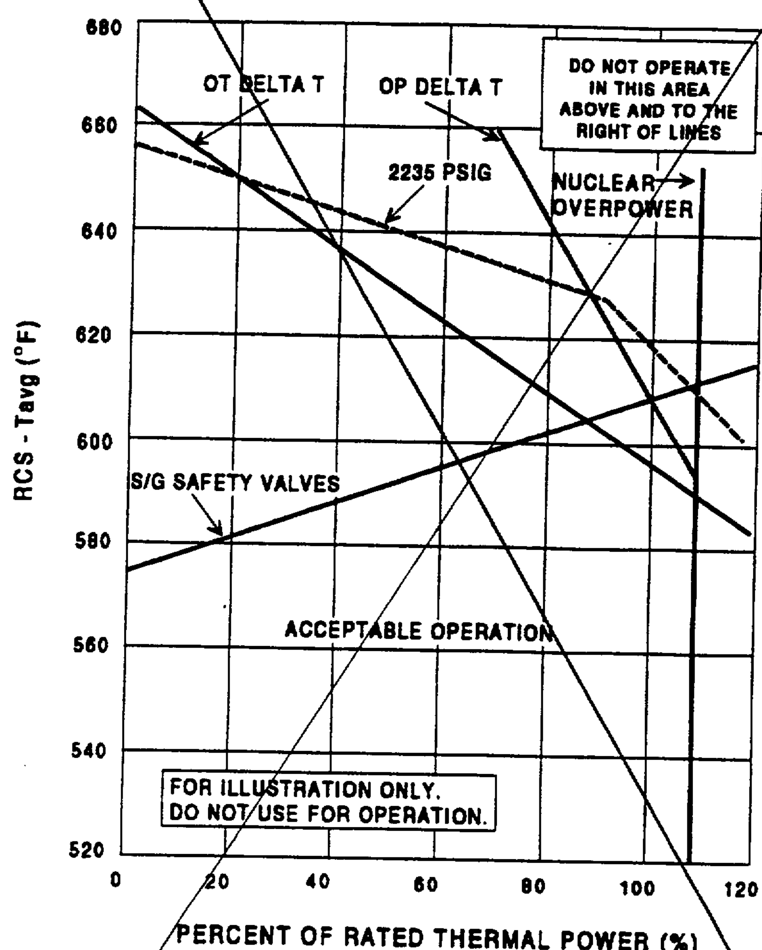

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Figure B 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits vs. Boundary of Protection

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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 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.

during operating conditions (5)

(1)

(4)

The design pressure of the RCS is 2500 psia. 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).

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.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

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 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.

allowable values (2)

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 allowable value ~~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. (2)

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

a. Pressurizer power operated relief valves (PORVs);

b. ~~Steam line relief valve~~; Steam Generator PORVs (2)

c. Steam Dump System;

d. Reactor Control System;

e. Pressurizer Level Control System; or

f. Pressurizer spray valve.

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. (3)

(continued)

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BASES (continued)

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

~~The following SL violations are applicable to the RCS pressure SL.~~

~~2.2.2.1~~

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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.

~~2.2.2.2~~

TSTF-5

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.

(continued)

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BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.3

If the RCS pressure SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 7).

2.2.4

If the RCS pressure SL is violated, the Plant Superintendent and the Vice President—Nuclear Operations shall be notified within 24 hours. The 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.

2.2.5

If the RCS pressure SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 8). A copy of the report shall also be provided to the Plant Superintendent and the Vice President—Nuclear Operations.

2.2.6

If the RCS pressure SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

TSTF-5

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.~~ UFSAR Sections 3.1.10, 3.1.11, and 3.1.2.4. ①
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.
4. 10 CFR 100.

(continued)

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BASES

REFERENCES (continued)	5. (a) FSAR, Section [7.2]. (2) (3)
	6. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.
	7. ~~10 CFR 50.72.~~ TSTF-5
	~~8. 10 CFR 50.73.~~

Rev. 0

1. North Anna Units 1 and 2 were designed and constructed on the basis of the proposed General Design Criteria, published in 1966. Since February 20, 1971, when the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50, were published, the Company attempted to comply with the intent of the newer criteria to the extent practical, recognizing previous design commitments. The NRC's Safety Evaluation Report for North Anna Units 1 and 2 reviewed the plant against 10 CFR Part 50, Appendix A and concluded that the facility design conforms to the intent of the newer criteria. The North Anna UFSAR contains discussions comparing the design of the plant to the 10 CFR 50, Appendix A, General Design Criteria. Bases references to the 10 CFR 50, Appendix A criteria have been replaced with references to the appropriate section of the UFSAR.

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

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. Editorial correction made to the Bases.

5. Clarifying information is added to the Bases. The maximum RCS pressure SL is only applicable at operating temperatures. At lower temperatures, a lower maximum RCS pressure is required.

CHAPTER 2.0 - SAFETY LIMITS

CURRENT TECHNICAL SPECIFICATIONS

MARKUP AND DISCUSSION OF CHANGES

UNIT 1

(A.1)

Chapter 2.0

ITS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

Insert proposed 2.1.1

LA.1

REACTOR CORE

2.1.1 2.1.1 ~~The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1* for 3 loop operation and 2.1-2 and 2.1-3 for 2 Loop operation.~~ A.2

APPLICABILITY: MODES 1 and 2.

ACTION:

2.2.1 Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, 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: MODES 1, 2, 3, 4 and 5.

ACTION:

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.

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.

~~* For the period of operation until steam generator replacement, the combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1a.~~ A.3

page 1 of 6

Rev. 0

Chapter 2.0

8-25-86

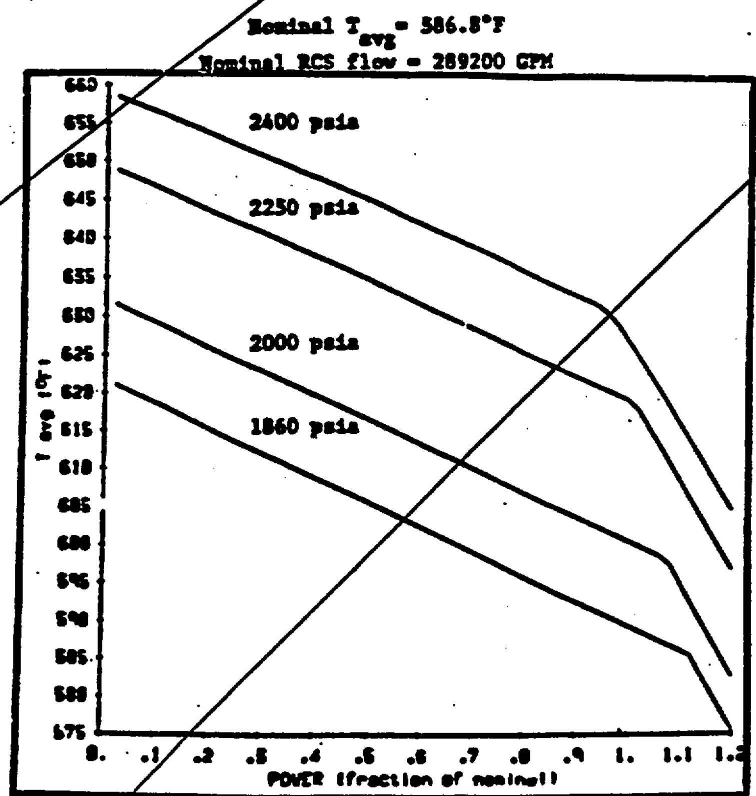

Figure 2.1-1 REACTOR CORE SAFETY LIMITS FOR THREE LOOP OPERATION

LA.1

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3-3-92

A.3

Nominal T_{avg} = 586.8°F
Nominal RCS flow = 268,500 GPM

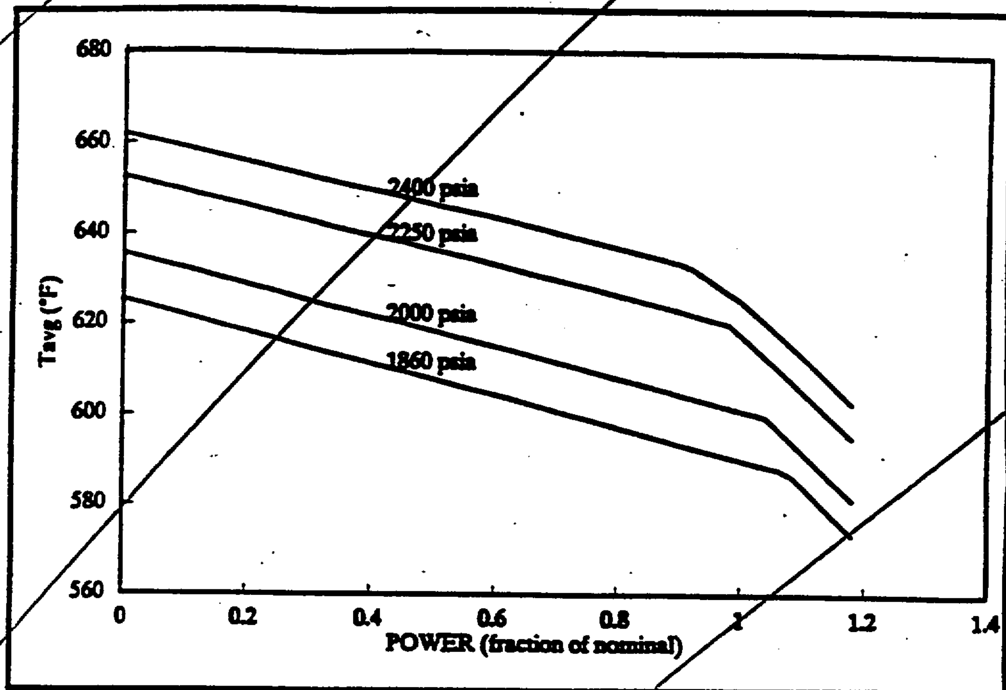

Figure 2.1-1a REACTOR CORE SAFETY LIMITS FOR THREE LOOP OPERATION FOR THE PERIOD OF OPERATION UNTIL STEAM GENERATOR REPLACEMENT

11-26-77

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A.2

REACTOR CORE SAFETY LIMIT - TWO LOOPS IN OPERATION
(ONE LOOP ISOLATED)

FIGURE 2.1-2

11-26-77

This page left blank pending NRC approval of ECCS evaluation of two loops in operation with the third loop not isolated.

REACTOR CORE SAFETY LIMIT - TWO LOOP OPERATION

(LOOP STOP VALVES OPEN)

FIGURE 2.1-3

A.2

A.1

Chapter 2.0

8-7-90

ITS

6.6 REPORTABLE EVENT ACTION

See ITS Chapter 5.0

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the SNSOC and the results of this review shall be submitted to the Vice President-Nuclear Operations and the MSRC.

6.7 SAFETY LIMIT VIOLATION

2.1.1
2.2.1

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

~~b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Vice President-Nuclear Operations and MSRC shall be notified within 24 hours.~~

~~c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SNSOC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.~~

~~d. The Safety Limit Violation Report shall be submitted to the Commission, the Vice President-Nuclear Operations and the MSRC within 14 days of the violation.~~

L.1

6.8 PROCEDURES AND PROGRAMS

See ITS chapter 5.0

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

page 6 of 6

Rev. A

UNIT 2

(A.1)

Chapter 2.0

8-21-80

ITS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

Insert proposed 2.1.1

(LA.1)

2.1.1 2.1.1 ~~The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1 for 3 loop operation and 2.1-2 and 2.1-3 for 2 loop operation.~~ (A.2)

APPLICABILITY: MODES 1 and 2.

ACTION:

2.2.1 Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, 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: MODES 1, 2, 3, 4 and 5.

ACTION:

2.2.2.1 MODES 1 and 2

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.

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.

Chapter 2.0

8-25-86

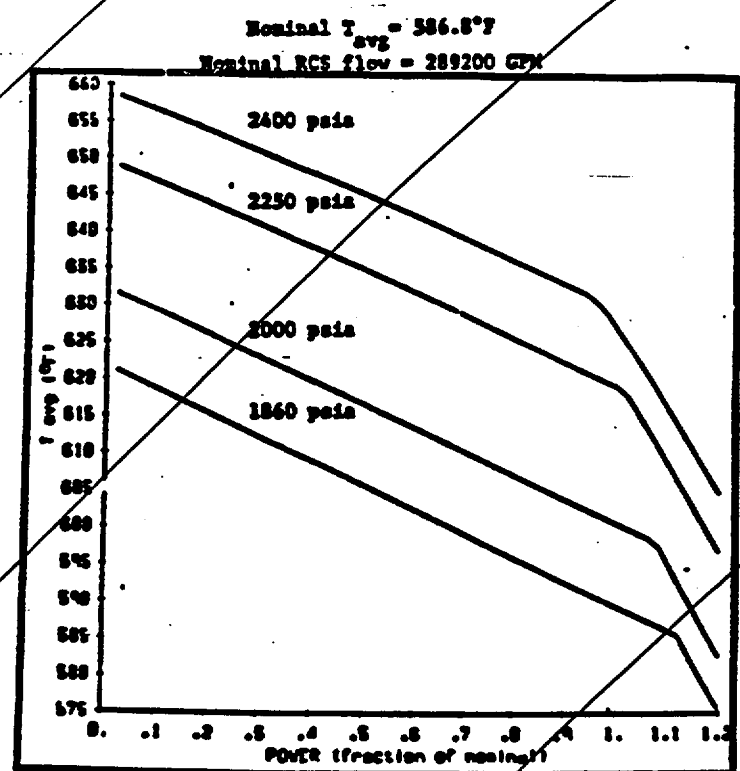

Figure 2.1-1 REACTOR CORE SAFETY LIMITS FOR THREE LOOP OPERATION

LA.1

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Rev. 0

Chapter 2.0

8-21-80

This page left blank pending NRC approval of ECCS evaluation of two loops in operation with the third loop isolated.

A.2

REACTOR CORE SAFETY LIMIT - TWO LOOP OPERATION

(ONE LOOP ISOLATED)

FIGURE 2.1-2

8-21-80

This page left blank pending NRC approval of ECCS evaluation of two loops in operation with the third loop not isolated.

A.2

REACTOR CORE SAFETY LIMIT - TWO LOOP OPERATION
(LOOP STOP VALVES OPEN)
FIGURE 2.1-3

(A.1)

Chapter 2.0

ITS

6.6 REPORTABLE EVENT ACTION

See ITS Chapter 5.0

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the SNSOC and the results of this review shall be submitted to the Vice President- Nuclear Operations and the MSRC.

6.7 SAFETY LIMIT VIOLATION

2.1.1
2.2.1

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. ~~The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Vice President- Nuclear Operations and MSRC shall be notified within 24 hours.~~

L.1

c. ~~A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SNSOC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.~~

d. ~~The Safety Limit Violation Report shall be submitted to the Commission, the Vice President-Nuclear Operations and the MSRC within 14 days of the violation.~~

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

(See ITS Chapter 5.0)

page 5 of 5 Rev. 0

ADMINISTRATIVE CHANGES

A.1 In the conversion of the North Anna 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. 1, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

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

A.2 CTS 2.1.1 references three curves providing limits on THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (Tavg). One curve applies to three loop operation (Figure 2.1-1) and two apply to two-loop operation (Figures 2.1-2 and 2.1-3). In the CTS, Figures 2.1-2 and 2.1-3 are replaced with a note stating, "This page left blank pending NRC approval of ECCS evaluation of two loops in operation with the third loop isolated" and "This page left blank pending NRC approval of ECCS evaluation of two loops in operation with the third loop not isolated," respectively. ITS 2.1.1 does not contain an allowance to operate with less than three reactor coolant loops in operation. This changes the CTS by eliminating references and place holders for curves applying to two-loop operation.

This change is acceptable because the requirements have not changed. Both the ITS and the CTS require all three loops in operation in the applicable MODES (MODES 1 and 2). This change is designated as administrative because it eliminates an option in the CTS which cannot be used.

A.3 Unit 1 CTS 2.1.1 contains a Note and an additional Figure, Figure 2.1-1a, which is to be used for the period of operation until steam generator replacement. ITS 2.1.1 does not contain a similar Note or additional Figure.

This change is acceptable because the North Anna Unit 1 steam generators have been replaced and the Note and the Figure are no longer applicable. This change is designated as administrative because it eliminates information from the CTS that is no longer applicable.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 5 – Removal of Cycle-Specific Parameter Limits from the Technical Specifications to the Core Operating Limits Report)* CTS 2.1.1 requires that the combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature not exceed the limits in CTS Figure 2.1-1. ITS 2.1.1 states that the combination of THERMAL POWER, RCS highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR and provides specific limits on DNBR and peak fuel centerline temperature. This changes the CTS by relocating the reactor core SLs to the COLR with limiting parameters retained in the SL.

The removal of these cycle-specific parameter limits from the Technical Specifications and their relocation into the COLR is acceptable because these limits are developed or utilized under NRC-approved methodologies. The NRC documented in Generic Letter 88-16, Removal of Cycle-Specific Parameter Limits From the 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 requirements and Surveillances that verify that the cycle-specific parameter limits are being met. NRC-approved Topical Report WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report" determined that the specific values for the reactor core SLs may be relocated to the COLR. The reactor SLs continue to require that the core be operated within the SLs, and limiting values for the SLs continue to appear in the Technical Specifications. The methodologies used to develop the SLs in the COLR have obtained prior approval by the NRC in accordance with Generic Letter 88-16. Also, this change is acceptable because the removed information will be adequately controlled in the COLR under the requirements provided in ITS 5.6.5, Core Operating Limits Report. ITS 5.6.5 ensures that the applicable limits (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) of the safety analysis are met. 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

L.1 *(Category 8 – Deletion of Reporting Requirements)* CTS 6.7.1 states that when a Safety Limit is violated, the NRC Operations Center must be notified within one hour, the Vice President - Nuclear Operations and the MSRC shall be notified within 24 hours, and a Safety Limit Violation Report must be prepared and submitted to the NRC, the Vice

President - Nuclear Operations, and the MSRC within 14 days. The ITS does not contain these reporting requirements. This changes the CTS by eliminating the explicit reporting requirements and relying on the reporting required by regulations.

The purpose of CTS 6.7.1 is to ensure that Company management, oversight organizations, and the NRC are notified with a safety limit is violated. This change is acceptable because the regulations provide adequate reporting requirements, or the reports do not affect continued plant operation. If a Safety Limit is violated, 10 CFR 50.72 and 10 CFR 50.73 describe the required notification and reporting to the NRC. Internal reporting to management and internal oversight organizations is a Company-internal procedural issue not appropriate for the Technical Specifications. This change is designated as less restrictive because reports that would be submitted under the CTS will not be required under the ITS.

CHAPTER 2.0 - SAFETY LIMITS

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS

GENERIC NSHCs

10 CFR 50.92 EVALUATION
FOR
ADMINISTRATIVE CHANGES

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve reformatting, renumbering, and rewording of Technical Specifications with no change in intent. These changes, since they do not involve technical changes to the Technical Specifications, are administrative.

This type of change is connected with the movement of requirements within the current requirements, or with the modification of wording that does not affect the technical content of the current Technical Specifications. These changes will also include nontechnical modifications of requirements to conform to the Writer's Guide or provide consistency with the Improved Standard Technical Specifications in NUREG-1431. Administrative changes are not intended to add, delete, or relocate any technical requirements of the current Technical Specifications.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change involves reformatting, renumbering, and rewording the existing Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. As such, this change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no effect on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION
FOR
MORE RESTRICTIVE CHANGES

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve adding more restrictive requirements to the existing Technical Specifications by either making current requirements more stringent or by adding new requirements that currently do not exist.

These changes include additional commitments that decrease allowed outage times, increase the frequency of surveillances, impose additional surveillances, increase the scope of specifications to include additional plant equipment, increase the applicability of specifications, or provide additional actions. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change provides more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more restrictive requirements either has no effect on or increases the margin of plant safety. As provided in the discussion of change, each change in this category is, by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION FOR RELOCATED SPECIFICATIONS

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve relocating existing Technical Specification LCOs to licensee controlled documents.

The the Company has evaluated the current Technical Specifications using the criteria set forth in 10 CFR 50.36. Specifications identified by this evaluation that did not meet the retention requirements specified in the regulation are not included in the Improved Technical Specifications (ITS) submittal. These specifications have been relocated from the current Technical Specifications to the Technical Requirements Manual.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relocates requirements and surveillances for structures, systems, components or variables that do not meet the criteria of 10 CFR 50.36 (c)(2)(ii) for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the North Anna Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to the Technical Requirements Manual, which will be maintained pursuant to 10 CFR 50.59. In addition, the affected structures, systems, components or variables are addressed in existing surveillance procedures which are also controlled by 10 CFR.50.59 and subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements and adequate control of existing requirements will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The proposed change will not reduce a margin of safety because it has no significant effect on any safety analyses assumptions, as indicated by the fact that the requirements do not meet the 10 CFR 50.36 criteria for retention. In addition, the relocated requirements are moved without change and any future changes to these requirements will be evaluated per 10 CFR 50.59.

NRC prior review and approval of changes to these relocated requirements, in accordance with 10 CFR 50.92, will no longer be required. This review and approval does not provide a specific margin of safety which can be evaluated. However, since the proposed change is consistent with the Westinghouse Standard Technical Specifications, NUREG-1431 issued by the NRC, revising the Technical Specifications to reflect the approved level of detail gives assurance that this relocation does not result in a significant reduction in the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES - REMOVED DETAIL

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve moving details out of the Technical Specifications and into the Technical Specifications Bases, the UFSAR, the TRM or other documents under regulatory control such as the Quality Assurance Program Topical Report. The removal of this information is considered to be less restrictive because it is no longer controlled by the Technical Specification change process. Typically, the information moved is descriptive in nature and its removal conforms with NUREG-1431 for format and content.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relocates certain details from the Technical Specifications to other documents under regulatory control. The Bases, UFSAR, and Technical Requirement Manual will be maintained in accordance with 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specification Bases are subject to the change control provisions in the Administrative Controls Chapter of the Technical Specifications. The UFSAR is subject to the change control provisions of 10 CFR 50.71(e). Other documents are subject to controls imposed by Technical Specifications or regulations. Since any changes to these documents will be evaluated, no significant increase in the probability or consequences of an accident previously evaluated will be allowed. Therefore this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operations. The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The proposed change will not reduce a margin of safety because it has no effect on any safety analysis assumptions. In addition, the details to be moved from the Technical Specifications to other documents are not being changed. Since any future changes to these details will be evaluated under the applicable regulatory change control mechanism, no significant reduction in a margin of safety will be allowed. A significant reduction in the margin of safety is not associated with the elimination of the 10 CFR 50.92 requirement for NRC review and approval of future changes to the relocated details. The proposed change is consistent with the Westinghouse Standard Technical Specifications, NUREG-1431, issued by the NRC Staff, revising the Technical Specifications to reflect the approved level of detail, which indicates that there is no significant reduction in the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 1
RELAXATION OF LCO REQUIREMENTS

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve relaxation of the current Technical Specification (CTS) Limiting Conditions for Operation (LCOs) by the elimination of specific items from the LCO or Tables referenced in the LCO, or the addition of exceptions to the LCO.

These changes reflect the ISTS approach to provide LCO requirements that specify the protective conditions that are required to meet safety analysis assumptions for required features. These conditions replace the lists of specific devices used in the CTS to describe the requirements needed to meet the safety analysis assumptions. The ITS also includes LCO Notes which allow exceptions to the LCO for the performance of testing or other operational needs. The ITS provides the protection required by the safety analysis and provides flexibility for meeting the conditions without adversely affecting operations since equivalent features are required to be OPERABLE. The ITS is also consistent with the plant current licensing basis, as may be modified in the discussion of individual changes. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change provides less restrictive LCO requirements for operation of the facility. These less restrictive LCO requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event in that the requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the current safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change does impose different requirements. However, the change is consistent with the assumptions in the current safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The imposition of less restrictive LCO requirements does not involve a significant reduction in the margin of safety. As provided in the discussion of change, this change has been evaluated to ensure that the current safety analyses and licensing basis requirements are maintained. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION FOR LESS RESTRICTIVE CHANGES – CATEGORY 2 RELAXATION OF APPLICABILITY

The North Anna Nuclear Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve relaxation of the applicability of current Technical Specification (CTS) Limiting Conditions for Operation (LCOs) by reducing the conditions under which the LCO requirements must be met.

Reactor operating conditions are used in CTS to define when the LCO features are required to be OPERABLE. CTS Applicabilities can be specific defined terms of reactor conditions or more general such as, "all MODES" or "any operating MODE." Generalized applicability conditions are not contained in ITS, therefore the ITS eliminates CTS requirements such as "all MODES" or "any operating MODE," replacing them with ITS defined MODES or applicable conditions that are consistent with the application of the plant safety analysis assumptions for operability of the required features.

CTS requirements may also be eliminated during conditions for which the safety function of the specified safety system is met because the feature is performing its intended safety function. Deleting applicability requirements that are indeterminate or which are inconsistent with application of accident analyses assumptions is acceptable because when LCOs cannot be met, the TS may be satisfied by exiting the applicability which takes the plant out of the conditions that require the safety system to be OPERABLE.

This change provides the protection required by the safety analysis and provides flexibility for meeting limits by restricting the application of the limits to the conditions assumed in the safety analyses. The ITS is also consistent with the plant current licensing basis, as may be modified in the discussion of individual changes. The change is generally made to conform with NUREG-1431 and has been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relaxes the conditions under which the LCO requirements for operation of the facility must be met. These less restrictive applicability requirements for the LCOs do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event in that the requirements continue to ensure that process variables, structures, systems, and components are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change does impose different requirements. However, the requirements are consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The relaxed applicability of LCO requirements does not involve a significant reduction in the margin of safety. As provided in the discussion of change, this change has been evaluated to ensure that the LCO requirements are applied in the MODES and specified conditions assumed in the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 3
RELAXATION OF COMPLETION TIME

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve relaxation of the Completion Times for Required Actions in the current Technical Specifications (CTS).

Upon discovery of a failure to meet an LCO, the ITS specifies times for completing Required Actions of the associated TS Conditions. Required Actions of the associated Conditions are used to establish remedial measures that must be taken within specified Completion Times (referred to as Allowed Outage Times (AOTs) in the CTS). These times define limits during which operation in a degraded condition is permitted. Adopting Completion Times from the ITS is acceptable because the Completion Times take into account the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. In addition, the ITS provides consistent Completion Times for similar conditions. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relaxes the Completion Time for a Required Action. Required Actions and their associated Completion Times are not initiating conditions for any accident previously evaluated and the accident analyses do not assume that required equipment is out of service prior to the analyzed event. Consequently, the relaxed Completion Time does not significantly increase the probability of any accident previously evaluated. The consequences of an analyzed accident during the relaxed Completion Time are the same as the consequences during the existing AOT. As a result, the consequences of any accident previously evaluated are not significantly increased. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the method governing normal plant operation. The Required Actions and associated Completion Times in the ITS have been evaluated to ensure that no new accident initiators are introduced. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The relaxed Completion Time for a Required Action does not involve a significant reduction in the margin of safety. As provided in the discussion of change, the change has been evaluated to ensure that the allowed Completion Time is consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION FOR LESS RESTRICTIVE CHANGES – CATEGORY 4 RELAXATION OF REQUIRED ACTION

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve relaxation of the Required Actions in the current Technical Specifications (CTS).

Upon discovery of a failure to meet an LCO, the ITS specifies Required Actions to complete for the associated Conditions. Required Actions of the associated Conditions are used to establish remedial measures that must be taken in response to the degraded conditions. These actions minimize the risk associated with continued operation while providing time to repair inoperable features. Some of the Required Actions are modified to place the plant in a MODE in which the LCO does not apply. Adopting Required Actions from the ISTS is acceptable because the Required Actions take into account the operability status of redundant systems of required features, the capacity and capability of the remaining features, and the compensatory attributes of the Required Actions as compared to the LCO requirements. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relaxes Required Actions. Required Actions and their associated Completion Times are not initiating conditions for any accident previously evaluated and the accident analyses do not assume that required equipment is out of service prior to the analyzed event. Consequently, the relaxed Required Actions do not significantly increase the probability of any accident previously evaluated. The Required Actions in the ITS have been developed to provide appropriate remedial actions to be taken in response to the degraded condition considering the operability status of the redundant systems of required features, and the capacity and capability of remaining features while minimizing the risk associated with continued operation. As a result, the consequences of any accident previously evaluated are not significantly increased. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The Required Actions and associated Completion Times in the ITS have been evaluated to ensure that no new accident initiators are introduced. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The relaxed Required Actions do not involve a significant reduction in the margin of safety. As provided in the discussion of change, this change has been evaluated to minimize the risk of continued operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 5
DELETION OF SURVEILLANCE REQUIREMENT

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve deletion of Surveillance Requirements in the current Technical Specifications (CTS).

The CTS require safety systems to be tested and verified Operable prior to entering applicable operating conditions. The ITS eliminates unnecessary CTS Surveillance Requirements that do not contribute to verification that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change deletes Surveillance Requirements. Surveillances are not initiators to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be Operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The remaining Surveillance Requirements are consistent with industry practice and are considered to be sufficient to prevent the removal of the subject Surveillances from creating a new or different type of accident. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The deleted Surveillance Requirements do not result in a significant reduction in the margin of safety. As provided in the discussion of change, the change has been evaluated to ensure that the deleted Surveillance Requirements are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 6
RELAXATION OF SURVEILLANCE REQUIREMENT ACCEPTANCE CRITERIA

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve the relaxation of Surveillance Requirements acceptance criteria in the current Technical Specifications (CTS).

The CTS require safety systems to be tested and verified Operable prior to entering applicable operating conditions. The ITS eliminates or relaxes the Surveillance Requirement acceptance criteria that do not contribute to verification that the equipment used to meet the LCO can perform its required functions. For example, the ITS allows some Surveillance Requirements to verify Operability under actual or test conditions. Adopting the ITS allowance for "actual" conditions is acceptable because required features cannot distinguish between an "actual" signal or a "test" signal. Also included are changes to CTS requirements that are replaced in the ITS with separate and distinct testing requirements which, when combined, include Operability verification of all TS required components for the features specified in the CTS. Adopting this format preference in the ISTS is acceptable because Surveillance Requirements that remain include testing of all previous features required to be verified OPERABLE. Changes which provide exceptions to Surveillance Requirements to provide for variations which do not affect the results of the test are also included in this category. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relaxes the acceptance criteria of Surveillance Requirements. Surveillances are not initiators to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be Operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The relaxed acceptance criteria for Surveillance Requirements do not result in a significant reduction in the margin of safety. As provided in the discussion of change, the relaxed Surveillance Requirement acceptance criteria have been evaluated to ensure that they are sufficient to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner that gives confidence that the equipment can perform its assumed safety function. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 7
RELAXATION OF SURVEILLANCE FREQUENCY

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve the relaxation of Surveillance Frequencies in the current Technical Specifications (CTS).

CTS and ITS Surveillance Frequencies specify time interval requirements for performing surveillance testing. Increasing the time interval between Surveillance tests in the ITS results in decreased equipment unavailability due to testing which also increases equipment availability. In general, the ITS contain test frequencies that are consistent with industry practice or industry standards for achieving acceptable levels of equipment reliability. Adopting testing practices specified in the ITS is acceptable based on similar design, like-component testing for the system application and the availability of other Technical Specification requirements which provide regular checks to ensure limits are met. Relaxation of Surveillance Frequency can also include the addition of Surveillance Notes which allow testing to be delayed until appropriate unit conditions for the test are established, or exempt testing in certain MODES or specified conditions in which the testing can not be performed.

Reduced testing can result in a safety enhancement because the unavailability due to testing is reduced and; in turn, reliability of the affected structure, system or component should remain constant or increase. Reduced testing is acceptable where operating experience, industry practice or the industry standards such as manufacturers' recommendations have shown that these components usually pass the Surveillance when performed at the specified interval, thus the frequency is acceptable from a reliability standpoint. Surveillance Frequency changes to incorporate alternate train testing have been shown to be acceptable where other qualitative or quantitative test requirements are required which are established predictors of system performance. Surveillance Frequency extensions can be based on NRC-approved topical reports. The NRC staff has accepted topical report analyses that bound the plant-specific design and component reliability assumptions. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relaxes Surveillance Frequencies. The relaxed Surveillance Frequencies have been established based on achieving acceptable levels of equipment reliability. Consequently, equipment which could initiate an accident previously evaluated will continue to operate as expected and the probability of the initiation of any accident previously evaluated will not be significantly increased. The equipment being tested is still required to be Operable and capable of performing any accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The relaxed Surveillance Frequencies do not result in a significant reduction in the margin of safety. As provided in the discussion of change, the relaxation in the Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Thus, appropriate equipment continues to be tested at a Frequency that gives confidence that the equipment can perform its assumed safety function when required. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 8
DELETION OF REPORTING REQUIREMENTS

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve the deletion of requirements in the current Technical Specifications (CTS) to send reports to the NRC.

The CTS includes requirements to submit reports to the NRC under certain circumstances. However, the ITS eliminates these requirements for many such reports and, in many cases, relies on the reporting requirements of 10 CFR 50.73 or other regulatory requirements. The ITS changes to reporting requirements are acceptable because the regulations provide adequate reporting requirements, or the reports do not affect continued plant operation. Therefore, this change has no effect on the safe operation of the plant. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change deletes reporting requirements. Sending reports to the NRC is not an initiator to any accident previously evaluated. Consequently, the probability of any accident previously evaluated is not significantly increased. Sending reports to the NRC has no effect on the ability of equipment to mitigate an accident previously evaluated. As a result, the consequences of any accident previously evaluated is not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The deletion of reporting requirements does not result in a significant reduction in the margin of safety. The ITS eliminates the requirements for many such reports and, in many cases, relies on the reporting requirements of 10 CFR 50.73 or other regulatory requirements. The change to reporting requirements does not affect the margin of safety because the regulations provide adequate reporting requirements, or the reports do not affect continued plant operation. Therefore, this change does not involve a significant reduction in a margin of safety.

This proposed Technical Specification change has been evaluated against the criteria for and identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed change meets the criteria for categorical exclusion as provided for under 10 CFR 51.22(c)(9). The following is a discussion of how the proposed Technical Specification change meets the criteria for categorical exclusion.

10 CFR 51.22(c)(9): Although the proposed change involves changes to requirements with respect to inspection or surveillance requirements,

(i) proposed change involves No Significant Hazards Considerations (refer to the Determination of No Significant Hazards Considerations section of this Technical Specification Change Request);

(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite since the proposed changes do not affect the generation of any radioactive effluents nor do they affect any of the permitted release paths; and

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Based on the aforementioned and pursuant to 10 CFR 51.22 (b), no environmental assessment or environmental affect statement need be prepared in connection with issuance of an amendment to the Technical Specifications incorporating the proposed change of this request.

CHAPTER 2.0 - SAFETY LIMITS

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS

SPECIFIC NSHCs

There are no specific NSHC discussions for this Section.