ATTACHMENT 1

VOLUME 4

KEWAUNEE POWER STATION IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 2.0 SAFETY LIMITS (SLs)

Revision 0

LIST OF ATTACHMENTS

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

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS - REACTOR CORE

APPLICABILITY

2.1.1 Applies to the limiting combination of thermal power, Reactor Coolant System pressure and coolant temperature during the OPERATING and HOT STANDBY MODES.

~~**OBJECTIVE**~~

~~To maintain the integrity of the fuel cladding.~~

SPECIFICATION

A01 pressurizer pressure

Reactor Coolant System (RCS) highest loop average temperature M01

2.1.1 a. The combination of RATED POWER level, ~~coolant pressure,~~ ~~and coolant temperature~~ shall not exceed the limits specified in the COLR. ~~The SAFETY LIMIT is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.~~ LA01

2.1.1.1 b. The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.14 for the HTP DNB correlation and 1.17 for the WRB-1 DNB correlation.

2.1.1.2 c. The peak fuel centerline temperature shall be maintained < 5080°F decreasing by 58°F per 10,000 MWD/MTU of burnup.

2.2.1 d. If SAFETY LIMIT 2.1.a, 2.1.b, or 2.1.c is violated, restore compliance and be in HOT SHUTDOWN within 1 hour.

2.2 SAFETY LIMIT - REACTOR COOLANT SYSTEM PRESSURE

~~**APPLICABILITY**~~

~~Applies to the maximum limit on Reactor Coolant System pressure.~~

~~**OBJECTIVE**~~

~~To maintain the integrity of the Reactor Coolant System.~~

SPECIFICATION

2.1.2 a. The Reactor Coolant System pressure shall not exceed 2735 psig with fuel assemblies installed in the reactor vessel.

in MODES 1, 2, 3, 4 and 5

L01

b. IF SAFETY LIMIT 2.2.a is violated then:

2.2.2.1 1. If in the OPERATING or HOT STANDBY modes, restore compliance and be in HOT SHUTDOWN within 1 hour.

2.2.2.2 2. If in the HOT SHUTDOWN, INTERMEDIATE SHUTDOWN, COLD SHUTDOWN, ~~or~~ ~~REFUELING~~ modes, restore compliance within 5 minutes.

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DISCUSSION OF CHANGES
ITS 2.0, SAFETY LIMITS

ADMINISTRATIVE CHANGES

A01 In the conversion of the Kewaunee Power Station (KPS) 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. 3.0, "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.

MORE RESTRICTIVE CHANGES

M01 CTS 2.1.a states that the combination of RATED POWER level, coolant pressure, and coolant temperature shall not exceed the limits specified in the COLR. ITS 2.1.1, in part, states that the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR. This changes the CTS by requiring the highest loop average temperature to be used in lieu of the average reactor coolant temperature.

The purpose of CTS 2.1.a is to ensure that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation. The KPS CTS 2.1 Bases states that this is the "reactor coolant system average temperature." The ITS 2.1.1 requirement that the temperature value used to determine the Safety Limit is met is the maximum of the coolant average temperatures is conservative in lieu of using the average of the two coolant loops. Therefore, this change is acceptable and is designated as more restrictive because the proposed temperature limit is now the highest loop average temperature in lieu of the average of the two loops.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 2.1.a states, in part, that the SAFETY LIMIT is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line. This statement is in reference to the Reactor Core Safety Limits Curve (Figure 1) in the Core Operating Limits Report (COLR). ITS 2.1.1 does not contain this statement but does state, in part, that the combination of THERMAL POWER, RCS highest loop average temperature, and pressurizer pressure shall not exceed the limits in the COLR. This changes the CTS by

DISCUSSION OF CHANGES
ITS 2.0, SAFETY LIMITS

moving the details of when the SAFETY LIMIT is exceeded from the Technical Specifications to the COLR.

The removal of this detail from the Technical Specifications is acceptable because this type of information is not necessary to provide adequate protection of public health and safety. The ITS still retains the requirement that the combination of THERMAL POWER, RCS highest loop average temperature, and pressurizer pressure shall not exceed the limits in the COLR. This change is acceptable because the removed details will be adequately controlled in the COLR under the requirements provided in ITS 5.6.3, "CORE OPERATING LIMITS REPORT (COLR)." ITS 5.6.3 ensures that the applicable limits of safety analyses are met. This change is designated as a less restrictive removal of detail change because the manner in which the TS requirement is met is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 2 – Relaxation of Applicability)* CTS 2.2.a requires the Reactor Coolant System (RCS) pressure not exceed 2735 psig with fuel assemblies installed in the reactor vessel. CTS 2.2.b.2 requires that if RCS pressure of 2735 psig is exceeded during HOT SHUTDOWN, INTERMEDIATE SHUTDOWN, COLD SHUTDOWN, or REFUELING, compliance be restored within 5 minutes. ITS 2.1.2 requires the RCS pressure be maintained ≤ 2735 psig in MODES 1, 2, 3, 4, and 5. ITS 2.2.2.2 requires compliance be restored within 5 minutes if in MODE 3, 4, or 5. This changes the CTS by only requiring the RCS pressure limit be met in MODES 1, 2, 3, 4, and 5, in lieu of all times when fuel assemblies are installed in the reactor vessel (i.e., deletes the requirement for compliance in REFUELING (equivalent to ITS MODE 6)).

The purpose of the RCS pressure limit is to ensure the continued integrity of the RCS. In the event of a 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 which would raise concerns to limits on radioactive releases specified in 10 CFR 50.67. This change is acceptable because the safety limit could be approached or exceeded in MODES 1, 2, 3, 4, and 5 due to overpressurization events. In addition, the safety limit is not applicable in MODE 6 (Refueling) because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized. This change is designated as less restrictive because the Safety Limit is applicable in less conditions in the ITS than in the CTS.

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

CTS

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1, 2.1.a 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.b 2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ [1.17 for the WRB-1~~/WRB-2~~ DNB correlation and ≥ 1.14 for the HTP DNB correlations]. 1

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

2.1.2 Reactor Coolant System Pressure SL

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

2.2 SAFETY LIMIT VIOLATIONS

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

2.2.b.2 2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

JUSTIFICATION FOR DEVIATIONS
ITS 2.0, SAFETY LIMITS

1. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is provided. This is acceptable since the generic specific information/value is revised to reflect the current plant design.

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

All changes are 1 unless otherwise noted

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core

BASES

INSERT 1

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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. (Ref. 2)

INSERT 2

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.

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

2

INSERT 1

USAR GDC 6, "Reactor Core Design,"

2

INSERT 2

the reactor core with its related controls and protection systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated

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All changes are 1 unless otherwise noted

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation ~~and AOOs~~. INSERT 3 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 3

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

The Reactor ~~Trip~~ Protection System setpoints (Ref. 2 3), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, ~~RCS~~ and ~~F~~flow, ΔI, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities. 4

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

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the ~~F~~SAR, Ref. 2 3) provide more restrictive limits to ensure that the SLs are not exceeded. U 5

SAFETY LIMITS

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

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

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB ; and 3

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.

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1 **INSERT 3**

with appropriate margins for uncertainties and specific transient situations which can be anticipated

All changes are 1 unless otherwise noted

BASES

SAFETY LIMITS (continued)

normal

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)~~. INSERT 4 To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during and normal ~~steady state~~ operation, ~~normal~~ operational transients, and ~~AOOs~~. INSERT 4

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."~~ 7 In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

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

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

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 10.~~ USAR, Section 3.1.2.1. 2

2. USAR, Section 3.1.3.3.

3 ~~2~~. U ~~F~~SAR, Section ~~[~~7.2~~]~~. 6

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1

INSERT 4

with appropriate margins for uncertainties and specific transient situations which can be anticipated

JUSTIFICATION FOR DEVIATIONS
ITS 2.1.1 BASES, REACTOR CORE SL

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

2. The ISTS lists GDC 10 of Appendix A to 10 CFR 50 as the reference document for the requirement that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences. Per the information contained in USAR Section 1.8, Kewaunee Power Station (KPS) was designed, constructed, and is being operated to comply with the Atomic Energy Commission (AEC) General Design Criteria (GDC) for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967. Since the plant was approximately 50% complete prior to the February 20, 1971 issuance of 10 CFR 50 Appendix A General Design Criteria, KPS was not required to be reanalyzed and the Final Safety Analysis Report (FSAR) was not required to be revised to reflect these later criteria. However, the AEC Safety Evaluation Report (SER), issued July 24, 1972, acknowledged that the AEC staff assessed the plant, as described in the FSAR (Amendment No. 7), against the Appendix A design criteria and determined that the plant design generally conforms to the intent of the Appendix A criteria. As a result, KPS utilizes AEC GDC 6, Reactor Core Design, as the licensing reference document for the requirement that specified acceptable fuel design limits are not exceeded.

3. The punctuation corrections have been made consistent with the Writer's Guide from the Improved Technical Specifications, TSTF-GG-05-01, Section 5.1.3.

4. Typographical/grammatical error corrected.

5. Editorial correction made to the Bases. The ISTS contains the phrase "(as indicated in the FSAR, (Ref. 2)." The intent of this phrase is to reference the location of the RPS trip setpoints identified in a previous paragraph of the Applicable Safety Analysis. The ITS markup moves the aforementioned phrase to the proper place in the paragraph.

6. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is provided. This is acceptable since the generic specific information/value is revised to reflect the current plant design.

7. The Allowable Values for the Reactor Protection System are not included in LCO 3.3.1. Therefore, this sentence has been deleted.

All changes are 1 unless otherwise noted

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"~~ INSERT 1 (Ref. 1), the reactor coolant pressure ~~coolant~~ boundary (RCPB) ~~design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs)~~ INSERT 2. 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~~ INSERT 3. 2 3 2 2

The design pressure of the RCS is ~~2500 psia~~ 2485 psig. During normal operation and ~~AOOs~~ anticipated operational transients, 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~~ 50.67, "~~Reactor Site Criteria~~ Accident Source Term" (Ref. 4). 4

APPLICABLE SAFETY ANALYSES

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

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

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INSERT 1

USAR GDC 9, "Reactor Coolant Pressure Boundary"

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INSERT 2

shall be designed, fabricated, and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime.

2

INSERT 3

USAR GDC 33, "Reactor Coolant Pressure Boundary" (Ref. 1), the reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

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All changes are 1 unless otherwise noted

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

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

a. Pressurizer power operated relief valves (PORVs)~~,~~ ; 5

b. ~~Steam line relief valve~~ Steam generator PORVs~~,~~ ;

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

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. 6
The ~~most~~ more limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig. 7

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BASES

APPLICABILITY

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

SAFETY LIMIT VIOLATIONS

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

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR ~~100~~, 50.67 "~~Reactor Site Criteria~~," Accident Source Term limits (Ref. 4). 4

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

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

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28~~ USAR, Sections 4.1.3.1 and 4.1.3.3. 2

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~~ 50.67. 4

5. ~~F~~ U SAR, Section ~~[~~7.2~~]~~. 1 6

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

JUSTIFICATION FOR DEVIATIONS
ITS 2.1.2 BASES, REACTOR COOLANT SYSTEM (RCS) PRESSURE SL

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

2. The ISTS lists GDC 14 and GDC 15 of Appendix A to 10 CFR 50 as the reference documents for the Reactor Coolant Pressure Boundary (RCPB) design requirements. The ISTS also lists GDC 28 of Appendix A to 10 CFR 50 as the reference document for the Reactivity Limit design requirements. Per the information contained in USAR Section 1.8, Kewaunee Power Station (KPS) was designed, constructed, and is being operated to comply with the Atomic Energy Commission (AEC) General Design Criteria (GDC) for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967. Since the plant was approximately 50% complete prior to the February 20, 1971 issuance of 10 CFR 50 Appendix A General Design Criteria, KPS was not required to be reanalyzed and the Final Safety Analysis Report (FSAR) was not required to be revised to reflect these later criteria. However, the AEC Safety Evaluation Report (SER), issued July 24, 1972, acknowledged that the AEC staff assessed the plant, as described in the FSAR (Amendment No. 7), against the Appendix A design criteria and determined that the plant design generally conforms to the intent of the Appendix A criteria. As a result, KPS utilizes AEC GDC 9 and AEC GDC 33 as the licensing reference documents for the RCPB requirements.

3. Editorial correction made to the Bases. The ISTS Background portion of the Bases contains the phrase and associated acronym "reactor pressure coolant boundary (RCPB)"; however, the words "pressure" and "coolant" are in the incorrect order. The words "pressure" and "coolant" have been reversed to accurately reflect the acronym "RCPB".

4. Changes are made to the ISTS Bases which reflect the Kewaunee Power Station (KPS) design. License Amendment 166, issued March 17, 2003 (ADAMS accession No. ML030210062), revised the radiological consequence analyses for the KPS design basis accidents to implement the alternate source term (AST) as described in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors" and pursuant to 10 CFR 50.67, "Accident Source Term."

5. The punctuation corrections have been made consistent with the Writer's Guide from the Improved Technical Specifications, TSTF-GG-05-01, Section 5.1.3.

6. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is provided. This is acceptable since the generic specific information/value is revised to reflect the current plant design.

7. Typographical/grammatical error corrected.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 2.0, SAFETY LIMITS

There are no specific NSHC discussions for this Specification.