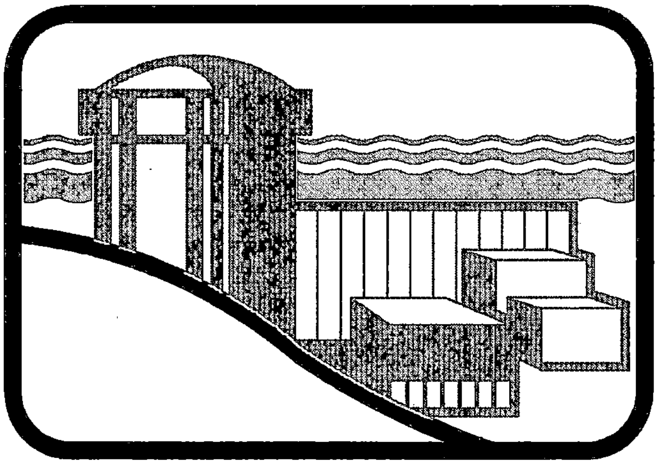
IMPROVED TECHNICAL SPECIFICATIONS



ALISADES

UCLEAR

LANT

Volume 4 CHAPTER 2.0

Consumers Energy

PALISADES NUCLEAR PLANT
CONSUMERS ENERGY
Docket 50-255
Conversion to Improved Technical Specifications
License DPR-20

INTRODUCTION: CHAPTER 2.0, SAFETY LIMITS (SLs)

A. ARRANGEMENT AND CONTENT OF THIS CHAPTER OF THE CHANGE REQUEST

This Chapter of the Technical Specification Change Request (TSCR) proposes changes to those Palisades Technical Specifications addressing SAFETY LIMITS (SLs). These changes are intended to result in requirements which are appropriate for the Palisades Nuclear Plant, but closely emulate those of the Standard Technical Specifications, Combustion Engineering Plants, NUREG 1432, Revision 1, Chapter 2.0.

This discussion and its supporting information frequently refer to three sets of Technical Specifications, and to two groups of discussions associated with the proposed changes; the following abbreviations are used for clarity and brevity:

CTS - The Palisades Current Technical Specifications,
ITS - The Palisades Improved Technical Specifications,
ISTS - NUREG 1432, Revision 1.
DOC - Discussions of Change; these discussions explain and justify the differences between the requirements of CTS and ITS.
JFD - Justifications for Deviation; these discussions explain the differences between the requirements of the ITS and the ISTS.

Six attachments are provided to assist the reviewer:

1. Proposed ITS Chapter 2.0 pages

2. Proposed ITS Chapter 2.0 Bases

3. A set of all those CTS pages which contain requirements associated with those in ITS Chapter 2.0, marked up to show the changes from CTS to ITS, and arranged by specification in the order in which the requirements occur in ITS. This attachment also includes a DOC for each change.

   Each change from CTS to ITS is classified in the following categories:

   ADMINISTRATIVE - A change which is editorial in nature, which only involves movement of requirements within the TS without affecting their technical content, or clarifies CTS requirements.

   MORE RESTRICTIVE - A change which only adds new requirements, or which revised an existing requirement resulting in additional operational restrictions.

   RELOCATED - A change which only moves requirements, not meeting the 10 CFR 50.36(c)(2)(ii) criteria, from the CTS to the Operating Requirements Manual (which has been included in the FSAR by reference).

A. ARRANGEMENT AND CONTENT OF THIS CHAPTER OF THE CHANGE REQUEST (continued)

LESS RESTRICTIVE - REMOVAL OF DETAIL (LA) - A change in which certain details from otherwise retained Specifications are removed from the ITS and placed in the Bases, FSAR, or other licensee controlled documents.

LESS RESTRICTIVE - A change which deletes any existing requirement, or which revises any existing requirement resulting in reduced operational restriction.

4. No Significant Hazards Analyses for the changes from CTS to ITS.

   An individual No Significant Hazards Analysis is provided for each Less Restrictive change; generic No Significant Hazards Analyses are provided for each of the other categories of change.

5. ISTS Chapter 2.0, Specifications and Bases, marked to show the differences between ISTS and ITS.

6. JFDs for the differences between ISTS and ITS.

B. REFERENCE DOCUMENTS

This Chapter of the TSCR is based on the following reference documents:

1. CTS as revised through Amendment 178.

2. The following TSCRs which are currently under review by the NRC:

   a. None.

3. ISTS, as revised by Industry Generic Changes (TSTF) approved as of October 15, 1997.

4. The following changes to ISTS which are currently under review by the NRC:

   a. None.

C. THE UNIQUE PALISADES NUCLEAR PLANT FEATURES AFFECTING THIS CHAPTER

Palisades has several physical, analytical, and administrative features which differ from those newer CE plants upon which the ISTS was based. Palisades was the first CE plant designed and built. Its design and licensing preceded the issuance of the General Design Criteria so that, in some aspects, its physical systems are not like those of newer plants; its Technical Specifications preceded the issuance of Standard Technical Specifications (STS) so that LCOs, Actions, and Surveillance Requirements are not coordinated as they would be for a STS plant. Palisades has purchased all its core reloads from Siemens Power Corporation (or its predecessors), therefore, reload analyses and the associated core physics parameters, as well as certain Safety Analyses are not like those plants using all CE fuel and analyses as were modeled in the ISTS.

D. THE DIFFERENCES BETWEEN CTS "OPERATING CONDITIONS" AND ITS "MODES"

The CTS definitions of plant operating conditions have been replaced with the operation Mode definitions used in ISTS. In several instances the name for a CTS defined "operating condition" is the same as that for an ISTS "Mode," but the definition differs.

CTS contain the following definitions for operating conditions:

1. The POWER OPERATION condition shall be when the reactor is critical and the neutron flux power range instrumentation indicates greater than 2% of RATED POWER.

2. The HOT STANDBY condition shall be when T_{ave} is greater than 525°F and any of the CONTROL RODS are withdrawn and the neutron flux power range instrumentation indicates less than 2% of RATED POWER.

3. The HOT SHUTDOWN condition shall be when the reactor is subcritical by an amount greater than or equal to the margin as specified in Technical Specification 3.10 and T_{ave} is greater than 525°F.

4. The COLD SHUTDOWN condition shall be when the primary coolant is at SHUTDOWN BORON CONCENTRATION and T_{ave} is less than 210°F.

5. The REFUELING SHUTDOWN condition shall be when the primary coolant is at REFUELING BORON CONCENTRATION and T_{ave} is less than 210°F.

ITS contain the following definition table for Modes:

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[(a)]	AVERAGE PRIMARY COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 300
4	Hot Shutdown[(b)]	< 0.99	NA	$300 > T_{ave} > 200$
5	Cold Shutdown[(b)]	< 0.99	NA	≤ 200
6	Refueling[(c)]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

E. MODE CHANGES USING CTS "OPERATING CONDITIONS" VERSUS ITS "MODES"

1. CTS "Power Operation" is essentially equivalent to ITS "MODE 1." Each represents a condition with the reactor critical and the turbine generator in operation. The only effective difference is the power level which separates that Condition or Mode from the next lower one. During plant startup, the plant must meet all CTS "Power Operation" or ITS "MODE 1" LCOs before the turbine generator is placed on the line; similarly, during plant shutdown, the plant exits CTS "Power Operation" or ITS "MODE 1" when the turbine generator is no longer in service. Therefore, this change in definition will have no operational effect.

2. CTS "Hot Standby" is similar to ITS "MODE 2." Each represents a condition with the reactor critical, or nearly so, and the turbine generator shut down. During plant startup, the plant must meet all CTS "Hot Standby" or ITS "MODE 2" LCOs before a reactor startup is started; during plant shutdown, the plant exits CTS "Hot Standby" or ITS "MODE 2" when the reactor is shutdown. CTS action statements requiring that the plant be placed in "Hot Standby" are effectively equivalent to ITS Actions requiring the plant be placed in "MODE 2." Therefore, this change in definition will have no operational effect.

3. CTS "Hot Shutdown" and ITS "MODE 3" are similar at their upper temperature boundary. During plant shutdown, the plant exits CTS "Hot Standby" or ITS "MODE 2" when the reactor is shutdown. CTS action statements requiring that the plant be placed in "Hot Shutdown" are effectively equivalent to ITS Actions requiring the plant be placed in "MODE 3." CTS "Hot Shutdown" and ITS "MODE 3" are quite different at their lower temperature boundary; CTS "Hot Shutdown" is exited when Tave drops below 525°F, ITS "MODE 3" is not exited until Tave drops below 300°F.

4. CTS does not provide a defined term for the condition when Tave is between 525°F and 210°F (the upper bound for CTS "Cold Shutdown").

5. CTS "Cold Shutdown" is essentially equivalent to ITS "MODE 5." Each represents a condition with Tave below boiling. There is no technical significance to the difference between the CTS 210°F and the ITS 200°F. CTS action statements requiring that the plant be placed in "Cold Shutdown" are effectively equivalent to ITS Actions requiring the plant be placed in "MODE 5." Therefore, this change in definition will have no operational effect.

6. CTS "Refueling Shutdown" is essentially equivalent to ITS "MODE 6." Each, when taken with other definitions and LCO requirements, represents a condition with the reactor at least 5% shutdown. Therefore, this change in definition will have no operational effect.

F. THE MAJOR CHANGES FROM CTS (as modified by pending TSCRs) TO ITS

1. An additional Safety Limit has been proposed. CTS do not contain a Safety Limit for Linear Heat Rate, yet such a limit is appropriate. A Linear Heat Rate Safety Limit of 21.0 kW per foot has been included.

G. THE MAJOR DIFFERENCES BETWEEN ITS AND ISTS

1. The use of three fuel design specific values for the DNBR Safety Limit has been retained from CTS. An explanation of the reason for having three values for this Safety Limit is provided in the Applicable Safety Analyses portion of the bases.

2. The NUREG provides "Analog" and "Digital" versions of the Safety Limits. Palisades instrumentation is of the analog type. Therefore, only the "Analog" versions are included.

ATTACHMENT 1

PALISADES NUCLEAR PLANT

CHAPTER 2.0, SAFETY LIMITS

PROPOSED TECHNICAL SPECIFICATIONS

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained at or above the following DNB correlation safety limits:

Correlation	Safety Limit
XNB	1.17
ANFP	1.154
HTP	1.141

2.1.1.2 In MODES 1 and 2, the peak Linear Heat Rate (LHR) (adjusted for fuel rod dynamics) shall be maintained at ≤ 21.0 kW/ft.

2.1.2 Primary Coolant System (PCS) Pressure SL

In MODES 1, 2, 3, 4, 5, and 6, the PCS pressure shall be maintained at ≤ 2750 psia.

2.2 SL Violations

2.2.1 If SL 2.1.1.1 or SL 2.1.1.2 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, 5, or 6, restore compliance within 5 minutes.

ATTACHMENT 2

PALISADES NUCLEAR PLANT

CHAPTER 2.0, SAFETY LIMITS

PROPOSED BASES

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

The Palisades Nuclear Plant design criteria (Ref. 1) requires, and these SLs ensure, 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 (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 and possible cladding perforation that would result in the release of fission products to the primary 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 primary coolant.

Operation above the boundary of the nucleate boiling regime beyond onset of DNB could result in excessive cladding temperature because of the resultant sharp reduction in the heat transfer coefficient in the transition and film boiling regimes. If a steam film is allowed to form, 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 primary coolant.

BASES

BACKGROUND (continued)	The Reactor Protective System (RPS), in combination with the LCOs, is designed to prevent any anticipated combination of transient conditions for Primary Coolant System (PCS) temperature, pressure, and THERMAL POWER level that would result in a 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 a 95% probability at a 95% confidence level (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.

Palisades uses three DNB correlations; the XNB, ANFP, and HTP detailed in References 3 through 8. The DNB correlations are used solely as analytical tools to ensure that plant conditions will not degrade to the point where DNB could be challenged. The XNB correlation is used for non-High Thermal Performance (HTP) assemblies (assemblies loaded prior to cycle 9), when the non-HTP assemblies could have been limiting. The non-HTP fuel assemblies are used for vessel fluence reduction and reside on the core periphery. The core periphery locations operate at relatively low relative power fractions; therefore, they are not DNB limiting assemblies. The XNB correlation provides administrative justification for using non-HTP assemblies in Palisades low leakage core design. The ANFP and HTP correlations are used for Palisades High Thermal Performance (HTP) fuel assemblies (assemblies loaded in cycle 9 and later).

The HTP correlation can be used when the calculated reactor coolant conditions fall within the correlation's applicable coolant condition ranges. Outside of the applicable range of the HTP correlation, the ANFP correlation can be used. The ANFP correlation may be used over a broader range of coolant conditions than the HTP correlation. The HTP correlation is an extension of the ANFP correlation and incorporates the results of test sections designed to represent HTP fuel design for CE plants.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

The prediction of DNB is a function of several measured parameters. The following trip functions and LCOs, limit these measured parameters to protect the Palisades reactor from approaching conditions that could lead to DNB:

Parameter	Protection
Core Flow Rate	Low PCS Flow Trip
Core Power	Variable High Power Trip
PCS Pressure/Core Power	TM/LP Trip
Core Inlet Temperature	T_{inlet} LCO
Axial Shape Index (ASI)	ASI LCO
Assembly Power	Incore Power Monitoring (LHR and Radial Peaking Factor LCOs)

The RPS setpoints, LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation," in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for PCS 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.

The SL represents a design requirement for establishing the protection system trip setpoints identified previously. LCO 3.2.1, "Linear Heat Rate (LHR)," and LCO 3.2.2, "Radial Peaking Factors," or the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

SL 2.1.1.1 and SL 2.1.1.2 ensure that the minimum DNBR is not less than the safety analyses limit and that fuel centerline temperature remains below melting.

The minimum value of the DNBR during normal operation and design basis AOOs is limited to the following DNB correlation safety limit:

Correlation	Safety Limit
XNB	1.17
ANFP	1.154
HTP	1.141

BASES

SAFETY LIMITS (continued)

The fuel centerline melt LHR value assumed in the safety analysis is 21 kw/ft. Operation ≤ 21 kw/ft maintains the dynamically adjusted peak LHR and ensures that fuel centerline melt will not occur during normal operating conditions or design AOOs.

APPLICABILITY

SL 2.1.1.1 and SL 2.1.1.2 only apply 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 are available to prevent PCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the plant into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1.

In MODES 3, 4, 5, and 6, a reactor core SL is not required, since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1.1 or SL 2.1.1.2 is violated, the requirement to go to MODE 3 places the plant in a MODE in which this SL is not applicable.

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

BASES

REFERENCES

1. FSAR, Section 5.1
2. FSAR, Chapter 14
3. XN-NF-621(A), Rev 1
4. XN-NF-709
5. ANF-1224(A), May 1989
6. ANF-89-192, January 1990
7. XN-NF-82-21, Rev 1
8. EMF-92-153(A) and Supplement 1, March 1994

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Primary Coolant System (PCS) Pressure SL

BASES

BACKGROUND

The SL on PCS pressure protects the integrity of the PCS against overpressurization. In the event of fuel cladding failure, fission products are released into the primary coolant. The PCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on PCS pressure, continued PCS integrity is ensured. According to Palisades Nuclear Plant design criteria (Ref. 1), the Primary Coolant Pressure Boundary (PCPB) design conditions are not to be exceeded during normal operation and Anticipated Operational Occurrences (AOOs). Also, according to Palisades Nuclear Plant design criteria (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the PCPB greater than limited local yielding.

The design pressure of the PCS is 2500 psia. During normal operation and AOOs, the PCS pressure is kept from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2) and by the piping, valve, and fitting limit of 120% of design pressure (Ref. 6). The initial hydrostatic test was conducted at 125% of design pressure (3125 psia) to verify the integrity of the primary coolant system (Ref. 2). Following inception of plant operation PCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the PCS could result in a breach of the PCPB. If this 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).

BASES

APPLICABLE SAFETY ANALYSES

The PCS primary safety valves, the Main Steam Safety Valves (MSSVs), and the High Pressurizer Pressure trip have settings established to ensure that the PCS pressure SL will not be exceeded.

The PCS primary safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence the 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 Protective System (RPS) trip setpoints (LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation"), together with the settings of the MSSVs (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") and the primary safety valves, provide pressure protection for normal operation and AOOs. In particular, the High Pressurizer Pressure Trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). Conservative values for all system parameters, delay times and core moderator coefficient are assumed.

More specifically, for the limiting case, no credit is taken for operation of any other pressure relieving system including the following:

a. Pressurizer Power Operated Relief Valves (PORVs);

b. Turbine Bypass Control System;

c. Atmospheric Steam Dump Valves;

d. Pressurizer Level Control System; or

e. Pressurizer Pressure Control System.

BASES

SAFETY LIMITS

The maximum transient pressure allowable in the PCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the PCS piping, valves, and fittings under 120% of design pressure (Ref. 6). The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable PCS pressure is established at 2750 psia.

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, 5, and 6 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is applicable in MODE 6 because the reactor vessel head closure bolts only must be less than fully tensioned, making it possible that the PCS could be pressurized.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the PCS pressure SLs.

2.2.2.1

If the PCS 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.

With PCS pressure greater than the value specified in SL 2.1.2 in MODE 1 or 2, the pressure must be reduced to below this value. A pressure greater than the value specified in SL 2.1.2 exceeds 110% of the PCS design pressure and may challenge system integrity.

The allowed Completion Time of 1 hour provides the operator time to complete the necessary actions to reduce PCS pressure by terminating the cause of the pressure increase, removing mass or energy from the PCS, or a combination of these actions, and to establish MODE 3 conditions.

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.2.2

If the PCS pressure SL is exceeded in MODE 3, 4, 5 or 6, PCS pressure must be restored to within the SL value within 5 minutes.

Exceeding the PCS pressure SL in MODE 3, 4, 5 or 6 is potentially 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. This 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. FSAR, Section 5.1
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
5. FSAR, Section 4.3
6. ASA B31.1-1955, Code for Pressure Piping, 1967

ATTACHMENT 3

PALISADES NUCLEAR PLANT

CHAPTER 2.0, SAFETY LIMITS

CTS MARKUP

AND

DISCUSSION OF CHANGES

Chapter 2.0

A.1

2.0 SAFETY LIMITS ~~AND LIMITING SAFETY SYSTEM SETTINGS~~ (SLs)

2.1 SLs

2.1.1 Safety Limit - Reactor Core

2.1.1.1 The Minimum DNBR of the reactor core shall be maintained greater than or equal to the DNB correlation safety limit.

Correlation	Safety Limit
XNB	1.17
ANFP	1.154
HTP	1.141

M.1

2.1.1.2 <INSERT FROM BELOW>

~~Applicability~~

A.4

Safety Limit 2.1 is applicable in ~~HOT STANDBY and POWER OPERATION.~~ MODES 1 and 2

2.2 ~~Action~~ SL Violations

M.2

2.2.1 If a Safety Limit is exceeded, ~~the reactor shall be shut down immediately~~ restore compliance and be in MODE 3 within 1 hour ~~and not restarted until the Commission authorizes resumption of operation in accordance with 10 CFR 50.36(c)(1)(i)(A).~~ A.3

A.1

2.1.2 Safety Limit - Primary Coolant System Pressure (PCS)

The PCS Pressure shall not exceed 2750 psia.

~~Applicability~~ A.1

Safety Limit 2.1.2 is applicable ~~when there is fuel in the reactor.~~ in MODES 1-6 A.2

A.1

~~Action~~

M.3

A.1

2.2.2 If a Safety Limit is exceeded, ~~the reactor shall be shut down immediately and not restarted until the Commission authorizes resumption of operation in accordance with 10 CFR 50.36(c)(1)(i)(A).~~ A.3

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, 5, or 6 restore compliance within 5 minutes.

~~2.3 Limiting Safety System Settings - Reactor Protective System (RPS)~~

~~The RPS trip setting limits shall be as stated in Table 2.3.1.~~

~~Applicability~~

~~Limiting Safety System Settings of Table 2.3.1 are applicable when the associated RPS channels are required to be OPERABLE by Specification 3.17.1.~~

~~Action~~

~~2.3.1 If an RPS instrument setting is not within the allowable settings of Table 2.3.1, immediately declare the instrument inoperable and complete corrective action as directed by Specification 3.17.1.~~

<See 3.3>

Amendment No. ~~31~~, ~~25~~, ~~43~~, ~~118~~, ~~137~~, ~~150~~, ~~168~~, 174
October 31, 1996

<INSERT ABOVE>

M.1

2.1.1.2 In MODES 1 and 2, the peak linear heat rate (LHR) (adjusted for fuel rod dynamics) shall be maintained at ≤ 21.0 kW/ft.

TABLE 2.3.1

REACTOR PROTECTIVE SYSTEM TRIP SETTING LIMITS

	RPS Trip Unit	Four Primary Coolant Pumps Operating	Three Primary Coolant Pumps Operating
1.	Variable High Power	≤15% above core power, with a minimum of ≤30% RATED POWER and a maximum of ≤106.5% RATED POWER.	≤15% above core power with a minimum of ≤15% RATED POWER and a maximum of ≤49% RATED POWER.
2.	PCS Flow	≥95% Full PCS Flow.	≥60% Full PCS Flow.
3.	High Pressure Pressurizer	≤2255 psia.	≤2255 psia.
4.	Thermal Margin/ Low Pressure	(a)	(a)
5.	Steam Generator Low Water Level	≥25.9% Narrow Range	≥25.9% Narrow Range
6.	Steam Generator Low Pressure	≥500 psia.	≥500 psia.
7.	Containment High Pressure	≤3.70 psig.	≤3.70 psig.

<See 3.3>

(a) The pressure setpoint for the Thermal Margin/Low Pressure Trip, P_{trip}, is the higher of two values, P_{min} and P_{var}, both in psia:

P_{min} = 1750
P_{var} = 2012(QA)(QR_1) + 17.0(T_{in}) - 9493

where:

QA = -0.720(ASI) + 1.028;		when -0.628 ≤ ASI < -0.100
QA = -0.333(ASI) + 1.067;		when -0.100 ≤ ASI < +0.200
QA = +0.375(ASI) + 0.925;		when +0.200 ≤ ASI ≤ +0.565
ASI = Measured ASI		when Q ≥ 0.0625
ASI = 0.0		when Q < 0.0625
QR_1 = 0.412(Q) + 0.588;		when Q ≤ 1.0
QR_1 = Q;		when Q > 1.0

Q = Core Power/RATED POWER
T_{in} = Maximum primary coolant inlet temperature, in °F.

ASI, T_{in}, and Q are the existing values as measured by the associated instrument channel.

Amendment No. ~~31~~, ~~80~~, ~~118~~, ~~138~~, ~~150~~, 162
October 26, 1994

ADMINISTRATIVE CHANGES (A)

A.1 All reformatting and renumbering are in accordance with NUREG-1432. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1432. During Improved Technical Specification (ITS) development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or implied) to the TS. Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1432. Since the design is already approved by the NRC, adding more details does not result in a technical change.

A.2 CTS 2.1.1 states in the Applicability that "Safety Limit 2.1.2 is applicable when there is fuel in the reactor." In the proposed ITS 2.1.2, it states, "In MODES 1, 2, 3, 4, 5, and 6, the PCS pressure shall be maintained at ≤ 2750 psia." These statements are equivalent since in the proposed ITS a "MODE" is defined in part by "...with fuel in the reactor vessel.

A.3 CTS Safety Limit-Reactor Core Action 2.1.1 and Safety Limit-Primary Coolant System (PCS) specify in part that if a Safety Limit is exceeded the reactor shall be "...and not restarted until the Commission authorizes resumption of operation in accordance with 10 CFR 50.36(c)(1)(I)(A)." The requirement that the reactor not be restarted until authorized by the Commission is contained in the referenced section of 10 CFR 50 as stated. To place this requirement in the Palisades Technical Specifications duplicates a current regulation which must be adhered to. Therefore, the wording in CTS SL 2.1.1 and SL 2.2.1 Action requiring Commission authorization for resumption of operation is not included in the proposed ITS. This is considered to be an administrative change since the requirement to receive Commission authorization to restart exists in regulation. This change is consistent with NUREG-1432 as modified by TSTF-5.

A.4 CTS 2.1 states that the Safety Limit 2.1 is applicable in "HOT STANDBY and POWER OPERATION." In the proposed ITS, the Applicability is MODES 1 and 2. The operating range covered by the CTS "HOT STANDBY and POWER OPERATION" is from <2% power with any control rods withdrawn to 100% power. The proposed ITS MODES 1 and 2 covers the operating range from $K_{eff} \geq 0.99$ to 100% RTP. This range of operating conditions are essentially identical. Therefore, this is considered to be an administrative change. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - MORE RESTRICTIVE (M)

M.1 The CTS does not include a "linear heat rate" limit for a reactor core safety limit. Proposed ITS 2.1.1.2 includes the following requirement "In MODES 1 and 2, the peak Linear Heat Rate (LHR) (adjusted for fuel rod dynamics) shall be maintained at ≤ 21.0 kw/ft. This is included in the proposed ITS because the peak linear heat rate is an assumption of the safety analysis. Since this limit is now included in the TS, it is considered to be a more restrictive change. This change is consistent with NUREG-1432.

M.2 CTS 2.1, Safety Limit-Reactor Core, and CTS 2.2, Safety Limit-Primary Coolant System Pressure (MODE 1 and 2) have an Action that if the Safety Limit is exceeded "the reactor shall be shut down immediately." This is modified in the proposed ITS 2.2, Safety Limit Violations, section 2.2.1 to "restore compliance and be in MODE 3 within 1 hour." The intent of the both the CTS and the ITS actions is to place the reactor in a safe condition such that the SL is no longer exceeded. For the CTS there is no explicit requirement to restore compliance. The CTS has an Action time of "immediately" to accomplish the reactor shutdown as compared to the ITS which specifies one hour to restore compliance and be in MODE 3. The CTS does not define "immediately" in the TS but several places in the Bases (primarily in 3.17, Instrumentation Systems), have the following discussion in reference to the use of "immediately": "The completion time of "immediately" does not mean "instantaneously," rather it implies "start as quickly as plant conditions permit and continue until completed." The CTS usage is similar to the ITS usage of "Immediately" which is "...the Required Action should be pursued without delay and in a controlled manner." Therefore, it is hard to make an absolute comparison between "immediately" and "MODE 3 within one hour" for the time frame in which the reactor must be shutdown. Furthermore, the use of "shut down" in the CTS is not defined but taken to be subcritical. In the ITS, MODE 3 requires that K_{eff} be less than .99 which could theoretically be "shutdown" to a greater extent than that required by the CTS. From a practical standpoint, in most cases there is little, if any, difference between the CTS and the ITS requirements as the reactor will most likely be "tripped" in either situation resulting in the reactor being shutdown by the same amount. However, since the ITS specifically requires that the Safety Limits be restored and ITS versus "shutdown" in the CTS, the ITS action is considered to be overall "more restrictive." This change is consistent with NUREG-1432.

M.3 CTS 2.2, Safety Limit - Primary Coolant System Pressure has an Applicability of when there is fuel in the reactor. However the Action specified in CTS 2.2.1 only has an action to shut down the reactor if a Safety Limit is exceeded, and does not specify any operational actions to take if the reactor is already shutdown. Proposed ITS 2.2.2 specifies that if SL 2.1.2 is violated, "In MODE 3, 4, 5, or 6 restore compliance within 5 minutes." Therefore, since the CTS does not specify that compliance be restored as required by the ITS, the ITS is considered to be more restrictive. This change is consistent with NUREG-1432.

LESS RESTRICTIVE CHANGES - REMOVAL OF DETAILS TO LICENSEE CONTROLLED DOCUMENTS (LA)

There were no "Removal of Details" changes associated with Chapter 2.0.

LESS RESTRICTIVE CHANGES (L)

There were no "Less Restrictive" changes associated with Chapter 2.0.

RELOCATED (R)

There were no "Relocated" changes associated with Chapter 2.0.

ATTACHMENT 4

PALISADES NUCLEAR PLANT

CHAPTER 2.0, SAFETY LIMITS

NO SIGNIFICANT HAZARDS CONSIDERATION

ADMINISTRATIVE CHANGES (A)

The Palisades Nuclear Plant is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants." Some of the proposed changes involve reformatting, renumbering, and rewording of Technical Specifications. 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 which 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-1432. 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, Palisades Nuclear Plant staff 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 changes involve reformatting, renumbering, and rewording of the existing Technical Specification. These modifications involve no technical changes to the existing Technical Specifications. The majority of changes were done in order to be consistent with NUREG-1432. During the development of NUREG-1432, certain wording preferences or English language conventions were adopted. The changes are administrative in nature and do not impact initiators of analyzed events. They also do not impact the assumed mitigation of accidents or transient events. Therefore, the changes do 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 changes involve reformatting, renumbering, and rewording of the existing Technical Specifications. The changes do 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 changes will not impose any new or different requirements or eliminate any existing requirements. Therefore, the changes do 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 margin of safety?**

   The proposed changes involve reformatting, renumbering, and rewording of the existing Technical Specifications. The changes are administrative in nature and will not involve any technical changes. The changes will not reduce a margin of safety because it has no impact on any safety analysis assumptions. Also, since these changes are administrative in nature, no question of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

MORE RESTRICTIVE CHANGES (M)

The Palisades Nuclear Plant is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, "Standard Technical Specifications, Combustion Engineering 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 which currently do not exist.

These changes may include additional requirements that decrease allowed outage time, increase frequency of surveillance, impose additional surveillance, increase the scope of a specification to include additional plant equipment, increase the applicability of a specification, or provide additional actions. These changes are generally made to conform with the NUREG-1432.

In accordance with the criteria set forth in 10 CFR 50.92, the Palisades Nuclear Plant 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 changes provide more stringent requirements than previously existed in the Technical Specifications. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event. If anything, the new requirements may decrease the probability or consequences of an analyzed event by incorporating the more restrictive changes. The changes 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, the changes do 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 changes provide more stringent requirements than previously existed in the Technical Specifications. The changes do not alter the plant configuration (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The changes do impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Therefore, the changes do 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 margin of safety?

The proposed changes provide more stringent requirements than previously existed in the Technical Specifications. Adding more restrictive requirements either increases or has no impact on the margin of safety. The changes, by definition, provide additional restrictions to enhance plant safety. The changes maintain requirements within the safety analyses and licensing basis. As such, no question of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGES - REMOVAL OF DETAILS TO LICENSEE CONTROLLED DOCUMENTS (LA)

There were no "Removal of Details" changes made to Chapter 2.

LESS RESTRICTIVE CHANGES (L)

There were no "Less Restrictive Changes" made in Chapter 2.

ATTACHMENT 5

PALISADES NUCLEAR PLANT

CHAPTER 2.0, SAFETY LIMITS

MARKUP OF NUREG-1432

TECHNICAL SPECIFICATIONS

2.0 SAFETY LIMITS (SLs) [(Digital)] 6

2.1 SLs

Correlation	Safety Limit
XNB	1.17
ANFP	1.154
HTP	1.141

1

2.1.1 Reactor Core SLs

4 | 2.1.1.1 In MODES 1 and 2, the ~~d~~eparture from ~~n~~ucleate ~~b~~oiling ~~r~~atio (DNBR) shall be maintained at ~~≥ [1.19]~~ or above the following DNB correlation safety limits:

2.1.1.2 In MODES 1 and 2, the peak ~~l~~inear ~~h~~eat ~~r~~ate (LHR) (adjusted for fuel rod dynamics) shall be maintained at ≤ ~~[~~21.0~~]~~ kW/ft.

2.1.2 Primary ~~Reactor~~ Coolant System (~~R~~CS) Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained at ≤ ~~[~~2750~~]~~ psia. | 7

1 6 P

2.2 SL Violations

2.2.1 If SL 2.1.1.1 or SL 2.1.1.2 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. | 7

6

8

~~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].~~

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

(continued)

2.0 SLs ~~(Digital)~~ 6

8

2.2 SL Violations (continued)
2.2.6 Operation of the unit shall not be resumed until authorized by the NRC.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs ~~(Digital)~~ 6

BASES The Palisades Nuclear Plant design criteria 4

BACKGROUND

~~GDC 10~~ (Ref. 1) requires, and these SLs ensure 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 (95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature. 2

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the ~~reactor~~ primary 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. 2

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~~ primary coolant. 2

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of ~~the onset of DNB and~~ beyond the onset of DNB the resultant sharp reduction in the heat transfer coefficient. If a steam film is allowed to form ~~inside the steam film~~ in the transition and film boiling regimes 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~~ primary coolant. 2

(continued)

BASES

BACKGROUND (continued)

The Reactor Protective System (RPS), in combination with the LCOs, is 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 violation of the reactor core SLs. primary P

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 a 95% probability at a 95% confidence level (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.

4 <INSERT>

P

The RPS setpoints, LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation," in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for 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:~~

~~a. Pressurizer Pressure—High trip;~~

~~b. Pressurizer Pressure—Low trip;~~

~~c. Linear Power Level—High trip;~~

~~d. Steam Generator Pressure—Low trip;~~

~~e. Local Power Density—High trip;~~

~~f. DNBR—Low trip;~~

~~g. Steam Generator Level—Low trip;~~

4

(continued)

4

CHAPTER 2.0

INSERT

Palisades uses three DNB correlations: the XNB, ANFP, and HTP detailed in References 3 through 8. The DNB correlations are used solely as analytical tools to ensure that plant conditions will not degrade to the point where DNB could be challenged. The XNB correlation is used for non-HTP fuel assemblies (assemblies loaded prior to cycle 9), when the non-HTP assemblies could have been limiting. The non-HTP fuel assemblies are used for vessel fluence reduction and reside on the core periphery. The core periphery locations operate at relatively low relative power fractions; therefore, they are not DNB limiting assemblies. The XNB correlation provides administrative justification for using the non-HTP assemblies in Palisades low leakage core design. The ANFP and HTP correlations are used for Palisades High Thermal Performance (HTP) fuel assemblies (assemblies loaded in cycle 9 and later).

The HTP correlation can be used when the calculated reactor coolant conditions fall within the correlation's applicable coolant condition ranges. Outside of the applicable range of the HTP correlation, the ANFP correlation can be used. The ANFP correlation may be used over a broader range of coolant conditions than the HTP correlation. The HTP correlation is an extension of the ANFP correlation and incorporates the results of test sections designed to represent HTP fuel designs for CE plants.

The prediction of DNB is a function of several measured parameters. The following trip functions and LCOs, limit these measured parameters to protect the Palisades reactor from approaching conditions that could lead to DNB:

Parameter	Protection
Core Flow Rate	Low PCS Flow Trip
Core Power	Variable High Power Trip
PCS Pressure/Core Power	TM/LP Trip
Core Inlet Temperature	T_{inlet} LCO
Axial Shape Index (ASI)	ASI LCO
Assembly Power	Incore Power Monitoring (LHR and Radial Peaking Factor LCOs)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~h. Steam Generator Level—High trip;~~

~~i. Reactor Coolant Flow—Low trip; and~~

~~j. 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 protection system design as a measure of the core power is proportional to core power.~~

4

The SL represents a design requirement for establishing the protection system trip setpoints identified previously. LCO 3.2.1, "Linear Heat Rate (LHR)," and LCO 3.2.~~4~~ 2, ~~"Departure From Nucleate Boiling Ratio (DNBR),"~~ Radial Peaking Factors or the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded. 4

SAFETY LIMITS

1

SL 2.1.1.1 and SL 2.1.1.2 ensure that the minimum DNBR is not less than the safety analyses limit and that fuel centerline temperature remains below melting.

The minimum value of the 2 DNBR during normal operation and design basis AOOs is ~~limited to [1.19],~~ ~~based on a statistical combination of CE-1 CHF correlation and engineering factor uncertainties, and is established as an SL. Additional factors such as rod bow and spacer grid size and placement will determine the limiting safety system settings required to ensure that the SL is maintained.~~

the following DNB correlation safety limit:

Correlation	Safety Limit
XNB	1.17
ANFP	1.154
HTP	1.141

4

~~Maintaining~~ Operation ≤ 21 kW/ft the dynamically adjusted peak LHR ~~to ≤ 21 kW/ft~~ and ensures that fuel centerline melt will not occur during normal operating conditions or design AOOs.

The fuel centerline melt LHR value assumed in the safety analysis is 21 kW/ft.

APPLICABILITY

SL 2.1.1.1 and SL 2.1.1.2 only apply 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~~ are available to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip 4

P

(continued)

BASES

APPLICABILITY (continued)

function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1.

In MODES 3, 4, 5, and 6, ~~Applicability~~ a reactor core SL is not required, since the reactor is not generating significant THERMAL POWER. 2

SAFETY LIMIT VIOLATIONS

The following violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1.1 or SL 2.1.1.2 is violated, the requirement to go to MODE 3 places the ~~unit~~ plant in a MODE in which this SL is not applicable. 2

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

~~2.2.3~~

~~If SL 2.1.1.1 or SL 2.1.1.2 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 3).~~

~~2.2.4~~

~~If SL 2.1.1.1 or SL 2.1.1.2 is violated, the appropriate senior management of the nuclear plant and the utility 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 the senior management.~~ 8

~~2.2.5~~

~~If SL 2.1.1.1 or SL 2.1.1.2 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. 4). A copy of the~~

(continued)

BASES

SAFETY LIMIT VIOLATIONS

~~2.2.5 (continued)~~

~~report shall also be provided to the senior management of the nuclear plant, and the utility Vice President—Nuclear Operations.~~

~~2.2.6~~

~~If SL 2.1.1.1 or SL 2.1.1.2 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.~~

8

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 10, 1988.~~ FSAR, Section 5.1
2. FSAR, ~~Section 1.1.~~ Chapter 14 1
3. ~~10 CFR 50.72.~~
4. ~~10 CFR 50.73.~~ 8

4
3. XN-NF-621 (A), Rev. 1
4. XN-NF-709
5. ANF-1224 (A), May 1989
6. ANF-89-192, January 1990
7. XN-NF-82-21(A), Rev. 1
8. EMF-92-153 (A) and Supplement 1, March 1994

B 2.0 SAFETY LIMITS (SLs)

4 | B 2.1.2 ~~Reactor~~ Coolant System (RCS) Pressure SL (~~Digital~~)

Primary — 4 — P

BASES

4 | 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~~ primary 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, continued RCS integrity 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~~ primary coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, according to ~~GDC 28~~ (Ref. 1), ~~"Reactivity Limits,"~~ reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding. | 4 | 9 | 9

Palisades Nuclear Plant design criteria

FSAR 5.1.3.5

FSAR 5.1.3.6

FSAR 5.1.4.9

FSAR Chapter 4

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, the RCS pressure is kept from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2) and by the piping, valve and fitting limit of 120% of design pressure. (Ref. 6). ~~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.~~ The initial hydrostatic test was conducted at 125% of design pressure (3125 psia) to verify the integrity of the primary coolant system (Ref. 2). Following inception of ~~unit~~ plant operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3). | 4

Overpressurization of the RCS could result in a breach of the RCPB. If this 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~~ primary safety valves, the main steam safety valves (MSSVs), and the ~~Reactor Pressure—High~~ High Pressurizer Pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded. | 4

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The ~~R~~ P CS ~~pressurizer~~ primary safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence the 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. (4)

FSAR 4.3.9.4

The Reactor Protective System (RPS) trip setpoints (LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation"), together with the settings of the MSSVs (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") and the ~~pressurizer~~ primary safety valves, provide pressure protection for normal operation and AOOs. In particular, the Pressurizer Pressure ~~—High~~ Trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). ~~Safety analyses for both the Pressure—High Trip and the RCS pressurizer safety valves are performed, using conservative assumptions relative to pressure control devices.~~ (4)

FSAR 4.3.9.4

Conservative values for all system parameters, delay times and core moderator coefficient are assumed.

More specifically, for the limiting case, no credit is taken for operation of any other pressure relieving system including the following: (2)

FSAR 14.12

FSAR 4.3.9.4

a. Pressurizer power operated relief valves (PORVs);

b. ~~Steam~~ Turbine Bypass Control System;

c. Steam Dump System

(d) ~~c~~. Pressurizer Level Control System; or

(e) ~~d~~. Pressurizer Pressure Control System. (4)

SAFETY LIMITS

The maximum transient pressure allowable in the ~~R~~ P CS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the ~~R~~ P CS 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 ~~R~~ P CS pressure is established at 2750 psia. (4) (1)

FSAR 14.12.2.2

(continued)

BASES (continued)

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, ~~and~~ 5 6 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~~ less than fully ~~tightened~~ tensioned, making it ~~unlikely~~ possible that the RCS can be pressurized. 7

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the RCS pressure SLs. 4

2.2.2.1

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. 4

With RCS pressure greater than the value specified in SL 2.1.2 in MODE 1 or 2, the pressure must be reduced to below this value. A pressure greater ~~than~~ the value than specified in SL 2.1.2 exceeds 110% of the RCS design pressure and may challenge system integrity. 4 2 4

The allowed Completion Time of 1 hour provides the operator time to complete the necessary actions to reduce RCS pressure by terminating the cause of the pressure increase, removing mass or energy from the RCS, or a combination of these actions, and to establish MODE 3 conditions. 4 4

2.2.2.2 4

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, 6 RCS pressure must be restored to within the SL value within 5 minutes. 7

Exceeding the RCS pressure SL in MODE 3, 4, or 5 6 is potentially 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. This action does not require reducing MODES, since this would require reducing temperature, which would 4 7

(continued)

BASES

SAFETY LIMIT VIOLATIONS

2.2.2.2 (continued)

compound the problem by adding thermal gradient stresses to the existing pressure stress.

~~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 appropriate senior management of the nuclear plant and the utility 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 to assess the condition of the unit before reporting to the 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 senior management of the nuclear plant, and the utility 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.~~

8

(continued)

BASES (continued) FSAR, Section 5.1

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28. 9
2. ASME, Boiler and Pressure Vessel Code, Section III, ~~Article NB-7000~~. 1965 edition
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.
4. 10 CFR 100.
5. FSAR, Section 4.3 ~~9.1.7~~.
6. [~~ASME, USAS B31.1~~, ~~Standard~~ Code for Pressure Piping, ~~1967~~.] ASA B31.1-1955 1
7. ~~10 CFR 50.72.~~ 8
8. ~~10 CFR 50.73.~~

ATTACHMENT 6

PALISADES NUCLEAR PLANT

CHAPTER 2.0, SAFETY LIMITS

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1432

TECHNICAL SPECIFICATIONS AND BASES

NOTE: The first five justifications for these changes from NUREG-1432 were generically used throughout the individual LCO section markups. Not all generic justifications are used in each section.

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

2. Editorial change for clarity or for consistency with the Improved Technical Specifications (ITS) Writer's Guide.

3. The requirement/statement has been deleted since it is not applicable to this facility. The following requirements have been renumbered, where applicable, to reflect this deletion.

4. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the facility specific nomenclature, number, reference, system description, or analysis description.

5. This change reflects the current licensing basis/technical specifications.

6. Palisades utilizes an analog protection system, but a safety limit for "Linear Heat Rate" was inadvertently not included in the Analog portion of NUREG-1432 for Section 2.0. This was recognized by the industry owner's groups and resulted in Technical Specification Task Force (TSTF)-144 which added a section for "Linear Heat Rate" in the Analog portion of NUREG-1432. However, since TSTF-144 was not approved by the NRC at the time of this markup, the NUREG-1432 "digital" portion of section 2.0 which does contain requirements for Linear Heat Rate was marked up to reflect the Palisades specific analysis in order to minimize the amount of markups required to show the plant specific differences. The ITS Bases also utilized the "digital" NUREG-1432 Safety Limits, Section 2.0, as a starting point since the Palisades analysis more closely resembles the information contained in the "digital" section than that contained in the "analog" section. Therefore, the Palisades Safety Limits, Section 2.0, will be modeled after the "digital" Section 2.0 of NUREG-1432.

7. CTS 2.2, Safety Limit-PCS has an applicability of "when there is fuel in the reactor." This is carried over in the proposed ITS in SL 2.2.2.2 by adding "MODE 6" to "In MODE 3, 4, or 5, ..." to read in MODE 3, 4, 5, or 6." In the ITS a "MODE" is defined as "...with fuel in the vessel." MODE 6 is defined as "One or more reactor vessel head closure bolts less than fully tensioned." Before all the reactor vessel head closure bolts are removed there is a possibility, albeit remote, that an overpressure condition could occur. Therefore, the CTS applicability is carried over to the proposed ITS.

8. NUREG-1432 sections 2.2.3 (NRC notification), 2.2.4 (utility management notification), 2.2.5 (Licensee Event Report requirements), and 2.2.6 (NRC restart authorization) specify administrative actions to be performed if a Safety Limit is violated. The industry owner's groups proposed Technical Specification Task Force (TSTF) Traveler number 5 which removed the notification requirements from the Improved Standard Technical Specifications (NUREG-1432 for Combustion Engineering Plants). Revision 1 of TSTF-5 was subsequently approved by the NRC. Therefore, these requirements and their accompanying references have been deleted in the proposed ITS.

9. The Palisades Nuclear Plant was designed prior to issuance of the General Design Criteria (GDC) in 10 CFR 50. Therefore, reference to the GDC is omitted and appropriately replaced by reference to the "Palisades Nuclear Plant design criteria." The Palisades Nuclear Plant design was compared to the GDCs as they appeared in 10 CFR 50, Appendix A on July 07, 1971. It was this updated discussion, including the identified exemptions, which formed the original plant Licensing Bases for future compliance with the GDCs.