Consumers Energy

A CMS Energy Company

Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043

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U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

DOCKET 50-255 - LICENSE DPR-20 - PALISADES PLANT - CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS - RESPONSE TO DECEMBER 4, 1998 REQUEST FOR ADDITIONAL INFORMATION - ITS SECTIONS 2.0, 3.1, AND 3.2

On January 26, 1998, Consumers Energy Company submitted a Technical Specifications Change Request (TSCR) to revise the Palisades Technical Specifications to closely emulate the Standard Technical Specifications for Combustion Engineering Plants, NUREG-1432. On December 4, 1998, the NRC requested additional information regarding Sections 2.0, Safety Limits; 3.1, Reactivity Control Systems; and 3.2, Power Distribution Limits, of that TSCR. This letter provides both responses to the NRC questions and associated editorial revisions to the pages of our January 26, 1998 submittal. It also includes one technical change, which was made as a result of comments from the Palisades staff.

The technical change identified by the Palisades staff revises the frequency of the SR which verifies operability of the control rod position deviation alarm from 92 days to 18 months. Verification of that alarm's operability involves misaligning each control rod group until the alarm actuates. This involves both exceeding the LCO 3.1.4 group alignment limits and moving part length rods. Neither of these actions is desired during power operation. The CTS neither requires this alarm to be operable nor includes any associated surveillance requirement. Since Palisades rods are manually controlled, and rod group alignments are verified after moving rods, the alarm is not as significant as in a plant with automatic rod control. The revised pages are included in Enclosure 4.

The NRC RAI of December 4, 1998, requested that Consumers Energy provide a response within 60 days of our receipt of that RAI. Subsequently, in a telephone conversations with the NRR Project Manager for Palisades, Consumers Energy received permission to delay the response to allow additional time for preparation and internal review.

A001/1

The following Enclosures to this letter have been provided:

Enclosure 1 contains: a) answers to the Request for Additional Information (RAI) and, b) markups of the previously submitted pages to show where revisions have been made. The corrections made in response to one of the Section 3.1 questions also affected Section 5.0, Administrative Controls.

Enclosure 2 contains marked-up ITS submittal pages incorporating editorial corrections.

Enclosures 3, 4, 5, and 6 contain revised pages for Sections 2.0, 3.1, 3.2, and 5.0 respectively, along with lists of revised pages and instructions for page replacement. These revised pages reflect changes resulting from our response to the RAI questions and the editorial changes itentified in Enclosure 2. Each revised page is dated for identification.

The changes being submitted herein do not alter the conclusions of the No Significant Hazards Considerations contained in our January 29, 1998 submittal.

SUMMARY OF COMMITMENTS

This submittal contains no new commitments and no revisions to existing commitments.

Kurt M. Haas
Director, Engineering

CC Administrator, Region III, USNRC
Project Manager, NRR, USNRC
NRC Resident Inspector - Palisades

Enclosures

CONSUMERS ENERGY COMPANY

RESPONSE TO DECEMBER 4, 1998 RAI

To the best of my knowledge, the content of this response to the NRC Request for Additional Information dated December 4, 1998 concerning Sections 2.0, 3.1, 3.2, and 5.0 of our January 26, 1998 License Amendment request for conversion to Improved Technical Specifications, is truthful and complete.

Kurt M. Haas
Director, Engineering

Sworn and subscribed to before me this 1st day of March ~~1998~~ 1999.

Mary Ann Engle

Mary Ann Engle, Notary Public
Berrien County, Michigan
(Acting in Van Buren County, Michigan)
My commission expires February 16, 2000

50-255 CE PALASADES

PROPOSED CHANGE TO TECH SPECS RE
IMPROVED TECH SPECS - RESPONSE
TO RAI -ITS SECTS 2,& 3

REC'D W/LTR DTD 03/01/99....9903080053

ENCLOSURE 1

CONSUMERS ENERGY COMPANY
PALISADES PLANT
DOCKET 50-255

CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS
RESPONSE TO DECEMBER 4, 1998
REQUEST FOR ADDITIONAL INFORMATION

RESPONSE TO NRC QUESTIONS
CHAPTER 2.0, SAFETY LIMITS
SECTION 3.1, REACTIVITY CONTROL SYSTEMS
SECTION 3.2, POWER DISTRIBUTION LIMITS

CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS
RESPONSE TO DECEMBER 04, 1998 REQUEST FOR ADDITIONAL INFORMATION
CHAPTER 2.0, SAFETY LIMITS APPLICABILITY

NRC REQUEST:

2.0-01 Safety Limits Applicability
B SL 2.1.2 Applicability, Bases page B 2.1.2-3
JFD-7

The STS SL 2.1.2 applicability has been expanded in ITS 2.1.2 to include Mode 6, in accordance with the CTS 2.2 applicability that includes "when there is fuel in the reactor."

Comment: The ITS Bases, B SL 2.1.2, Applicability addressing Mode 6 is not stated in a logical way; "The SL is applicable in MODE 6 because the ... closure bolts are less than fully tensioned, making it possible that the PCS can be pressurized." Suggest that the bases be more clearly written; i.e., "When the closure bolts are less than fully tensioned the SL is applicable because it is possible ...."

Consumers Energy Response:

The Applicability discussion in the Bases for ITS 2.1.2 has been revised to better clarify the requirement for the PCS Pressure Safety Limit in Mode 6.

Affected Submittal Pages:

Att 2 ITS, page B 2.1.2-3
Att 5 NUREG, page B 2.0-8

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.

RAI 2.0-01

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 with the reactor vessel head installed and ~~because~~ the reactor vessel head closure bolts ~~only must be~~ less than fully tensioned~~, making it possible that the PCS could be pressurized.~~ the Potential for an overpressurization event still exists.

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.

Although overpressurization of the PCS is impossible once the reactor Vessel head is removed, the requirements of this SL apply as long as fuel is in the reactor. Once all the fuel has been removed from the reactor, the requirements of SL 2.1.2 no longer apply.

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 fully tightened, making it unlikely that the RCS can be pressurized.~~ 7

In Mode 6 with the reactor head installed and the reactor vessel closure bolts less than fully tensioned, the potential for an overpressurization event still exists. Although overpressurization of the RCS is impossible once the reactor vessel head is removed, the requirements of this SL apply as long as fuel is in the reactor. Once all the fuel has been removed from the reactor, the requirements of SL 2.1.2 no longer apply.

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 7

(continued)

NRC REQUEST:

3.1-01 ITS 3.1.1 Shutdown Margin (SDM)
LCO 3.1.1, SR 3.1.1.1 and SR 3.1.1.2
DOC A.6 and DOC M.2
JFD 7 and JFD 9

ITS LCO 3.1.1 states that "SDM shall be within limits," without referring to a COLR or explicitly stating the SDM limits. The ITS 3.1.1 limits and their applicability are defined in SR 3.1.1.1 and SR 3.1.1.2. TSTF-9 revised the STS from having the limits explicitly stated in the LCO to referencing the COLR in the LCO.

Comment: The ITS uses an unacceptable and cumbersome method to define LCO limits. Recommend either including the limits in the LCO, thereby enabling the use of only one SR, or utilizing the COLR as is done with other specifications.

Consumers Energy Response:

The specific values for SDM have been removed from the CTS and relocated to the COLR consistent with NUREG-1432 as modified by TSTF-9. By adopting TSTF-9, the cumbersome method of stipulating the limits for Shutdown Margin in multiple LCOs, or multiple Surveillance Requirements, has been eliminated. Justification for this change, as well as the related conforming changes, are provided in the "Affected Submittal Pages" listed below. Reference to SR 3.1.1.2 in Section 3.3, "Instrumentation" will be deleted as part of Consumers Energy response to NRC's Request for Additional Information related to Section 3.3.

As a result of relocating the SDM limits to the COLR, a revision has been made to Discussion of Change (DOC) 3.1.1, A.8.

This revision supersedes the response previously submitted by Consumers Energy to NRC RAI 5.6-02.

Affected Submittal Pages:

Att 1 ITS 3.1.1, page 3.1.1-1
Att 1 ITS 3.1.1, page 3.1.1-2
Att 2 ITS 3.1.1, page B 3.1.1-3
Att 2 ITS 3.1.1, page B 3.1.1-5
Att 2 ITS 3.1.1, page B 3.1.1-6
Att 3 CTS, page 3-50 (ITS 3.1.1 page 1 of 2)
Att 3 DOC 3.1.1, page 2 of 6
Att 3 DOC 3.1.1, page 3 of 6
Att 3 DOC 3.1.1, page 4 of 6
Att 3 DOC 3.1.1, page 5 of 6
Att 3 DOC 3.1.1, page 6 of 6
Att 4 NSHC 3.1.1, page 1 of 3
Att 4 NSHC 3.1.1, page 2 of 3
Att 4 NSHC 3.1.1, page 3 of 3
Att 5 NUREG, page 3.1-1
Att 5 NUREG, page 3.1-1 Insert
Att 5 NUREG, page B 3.1-4
Att 5 NUREG, page B 3.1-5
Att 5 NUREG, page B 3.1-6 Insert
Att 6 JFD 3.1.1, page 2 of 3

Att 1 ITS, page 5.0-25
Att 3 CTS, page 6-20
Att 3 DOC 5.0, page 2 of 7
Att 5 NUREG, page 5.0-21
Att 5 NUREG, page 5.0-21 Insert

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

RAI 3.1-01

LCO 3.1.1 SDM shall be within the limits ~~.~~ specified in the COLR. Y

APPLICABILITY: MODE 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	~~NOTE: Only required to be met in MODE 3 when T_{ave} is ≥ 525°F and four primary coolant pumps are operating.~~	X RAI 3.1-01
	Verify SDM ~~is ≥ 2.0% Δρ.~~ to be within limits.	24 hours X

SURVEILLANCE REQUIREMENTS

SR 3.1.1.2	------------------NOTE------------------ Only required to be met in MODE 3 when T_{ave} is ≥ 525°F with less than four primary coolant pumps operating, in MODE 3 when T_{ave} is < 525°F, and in MODES 4 and 5. -- Verify SDM is ≥ 3.5% $\Delta\rho$.	24 hours

RAI 3.1-01

BASES

APPLICABLE SAFETY ANALYSES (continued)

In addition to the limiting MSLB transient, the SDM requirement for MODES 3 and 4 must also protect against an inadvertent boron dilution; (Ref. 3) and an uncontrolled control rod bank withdrawal from subcritical conditions (Ref. 5).

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the PCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life when critical boron concentrations are highest.

The withdrawal of a control rod bank from subcritical conditions adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The withdrawal of control rod banks also produce a time dependent redistribution of core power.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled control rod banks withdrawal transient is terminated by either a high power trip or a high pressurizer pressure trip. In all cases, power level, PCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2).

RAI 3.1-01

LCO

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value for ~~of the LCO~~. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, then the minimum required time assumed for operator action to terminate dilution may no longer be applicable. X

RAI 3.1-04

SDM is a core physics design condition that can be ensured through full-length control rod positioning (regulating and shutdown rods) and through the soluble boron concentration. X

BASES

RAI 3.1-01

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 ~~and SR 3.1.1.2~~

SDM is verified by ~~performing~~ a reactivity balance calculation, considering the listed reactivity effects:

a. PCS boron concentration;

b. Control rod positions;

c. PCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration; and

f. Isothermal Temperature Coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical and the fuel temperature will be changing at the same rate as the PCS.

Samarium is not considered in the reactivity analysis since the analysis assumes that the negative reactivity due to Samarium is offset by the positive reactivity of Plutonium build in.

~~To maintain consistency with the assumptions used in the MSLB analysis, two values of SDM are specified in the surveillance requirements.~~

RAI 3.1-01

SR 3.1.1.1 requires SDM to be ~~≥ 2% Δρ.~~ within the limits specified in the COLR. ~~This SDM value ensures the consequences of an MSLB, as well as the other events described in the Applicable Safety Analysis, will be acceptable whenever the plant is in MODE 3 with T_{ave} ≥ 525°F and four Primary Coolant Pumps (PCPs) are in operation. Therefore, SR 3.1.1.1 is modified by a Note which only requires this SR to be met in MODE 3 when T_{ave} ≥ 525°F and four PCPs are in operation.~~

BASES

RAI 3.1-d

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 and SR 3.1.1.2 (continued)

SR 3.1.1.2 requires SDM to be ≥ 3.5% Δρ. This SDM value ensures the consequences of an MSLB, as well as the other events described in the Applicable Safety Analysis, will be acceptable as a result of a cooldown of the PCS which adds positive reactivity in the presence of a negative moderator temperature coefficient. As such, the requirements of this SR 3.1.1.2 must be met whenever the plant is in MODE 3 with T_{ave} ≥ 525°F with less than four PCPs operating, MODE 3 with T_{ave} < 525°F, and MODES 4 and 5. Therefore, SR 3.1.1.2 is modified by a Note which only requires this SR to be met in MODE 3 with T_{ave} ≥ 525°F with less than four PCPs operating, MODE 3 with T_{ave} < 525°F, and MODES 3, 4 and 5.

The Frequency of 24 hours for the verification of SDM is based on the generally slow change in required boron concentration, and also allows sufficient time for the operator to collect the required data, which includes performing a boron concentration analysis, and may completing the calculation.

REFERENCES

1. FSAR, Section 5.1
2. FSAR, Section 14.14
3. FSAR, Section 14.3
4. 10 CFR 100
5. FSAR, Section 14.2

Reactivity Control Systems (A.1)

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

~~Applicability~~

~~Applies to operation of CONTROL RODS and hot channel factors during operation~~

~~Objective~~

~~To specify limits of CONTROL ROD movement to assure an acceptable power distribution during power operation, limit worth of individual rods to values analyzed for accident conditions, maintain adequate shutdown margin after a reactor trip and to specify acceptable power limits for power tilt conditions.~~

(A.1) RAI 3.1-01

(A.6)

Specifications

3.10.1 Shutdown Margin Requirements

LA.1 (A.2)

~~SR 3.1.1.1 Note~~

a. With four primary coolant pumps in operation at hot shutdown and above the shutdown margin shall be 2%.

LA.1

~~SR 3.1.1.1~~ SR 3.1.1.2 Note b. With less than four primary coolant pumps in operation at hot shutdown and above, boration shall be immediately initiated to increase and maintain the shutdown margin at ≥3% — LA.1 (A.5) RA A.1 (L.1)

SR 3.1.1.2

(A.3) (A.4) c. At less than the hot shutdown condition, with at least one primary coolant pump in operation or at least one shutdown cooling pump in operation, with a flow rate ≥2810 gpm, the boron concentration shall be greater than the cold shutdown boron concentration for normal cooldowns and heatups, ie, non-emergency conditions.

~~SR 3.1.1.2 Note~~

~~During non-emergency conditions, at less than the hot shutdown condition with no operating primary coolant pumps and a primary system recirculating flow rate < 2810 gpm but ≥ 650 gpm, then within one hour either:~~

~~1. (a) Establish a shutdown margin of ≥ 3.5% and~~

~~(b) Assure two of the three charging pumps are electrically disabled.~~

~~OR~~

~~2. At least every 15 minutes verify that no charging pumps are operating. If one or more charging pumps are determined to be operating in any 15 minute surveillance period, terminate charging pump operation and insure that the shutdown margin requirements are met and maintained.~~

(See 3.4)

(A.6) — <ADD LCO & Applicability>

(M.1) — <ADD RA A.1>

(M.2) — <ADD SR 3.1.1.1 and FREQ>

Amendment No. ~~31, 43, 57, 68, 70, 118~~, 162
October 26, 1994

A.3 CTS 3.10.1c specifies SHUTDOWN MARGIN requirements at "less than the hot shutdown condition" (below 525°F). In the proposed ITS this corresponds to MODE 3 <525°F, MODE 4, and MODE 5. The requirements for the refueling condition (MODE 6) are addressed in proposed ITS 3.9.1. This is an administrative change to reflect the NUREG-1432 defined MODES. This change is consistent with the intent of NUREG-1432. RAI 3.1-01

A.4 CTS 3.10.1c includes the statement "...with at least one primary coolant pump in operation or at least one shutdown cooling pump in operation, with a flow rate ≥ 2810 gpm, the boron concentration shall be greater than the cold shutdown boron concentration." In the proposed ITS for operation with Tave < 525°F, ~~the~~ SHUTDOWN MARGIN (SDM) ~~requirement~~ will be ~~3.5% Δρ~~ within the limits specified in the COLR regardless of the primary system flow rate and ~~The SDM requirement of 3.5% Δρ will exist~~ throughout the temperature range as a cooldown occurs. ~~The ITS definition of SDM also allows credit to be taken for the most reactive rod, which was assumed to be fully withdrawn, if all control rods can be verified to be inserted by independent means. This would allow from 1 to 1.5 % Δρ credit for inserted control rod worth to be added, depending on the assumed reactivity from the most reactive rod which is a function of core burnup.~~

~~Therefore, adding this value to the 2% Δρ SDM required at power, will approximate the 3.5% Δρ required SDM. Boron will be added as required during the cooldown to account for the temperature defect.~~ Overall, this is considered to be an administrative change since the "cold shutdown boron concentration" requirement is replaced by the requirement to have ~~a~~ SDM ~~of 3.5% Δρ~~ throughout the temperature range. This change could be more or less restrictive depending on a particular primary coolant temperature evaluated, however, overall the requirement is considered an administrative "substitution" of one requirement for another while still preserving the SDM requirements.

within the limits specified in the COLR

A.5 CTS 3.10.1b states in part that "...boration shall be immediately initiated to increase and maintain the shutdown margin at...." In the proposed ITS this statement becomes Action A and the term "immediately" is changed to 15 minutes. In the proposed NUREG-1432, the time frame of 15 minutes is used in lieu of "immediately" to specify a specific time in which an action must be started. The terminology conveys the same meaning in the CTS in that quick action must be taken. In NUREG-1432, a Completion Time of "immediately" is defined in Section 1.3 as "pursue continuously in a controlled manner without delay." Therefore, while a Completion Time of "15 minutes" is used in the proposed ITS as compared to the CTS "Immediately" the effective meaning is the same. Therefore, this is considered to be an Administrative Change. This change is consistent with NUREG-1432.

RAI 3.1-01

A.6 CTS 3.10.1a, CTS 3.1.10.1b and CTS 3.1.10c contain the requirements for SHUTDOWN MARGIN. The amount of required SHUTDOWN MARGIN is dependent on the plant operating conditions (e.g., above or below hot shutdown) and the number of primary coolant pumps in operation. To establish consistency with the format and style of the ITS, the values of the required SHUTDOWN MARGIN have been moved to ~~surveillance requirements (SR 3.1.1.1 and SR 3.1.1.2) and~~ the COLR including the plant specific operating conditions and pump configurations ~~have been placed in the surveillance requirement Notes.~~ (See DOC LA.1) A new LCO statement has been added which states that the SHUTDOWN MARGIN must be within the limits specified in the COLR and an Applicability of MODES 3, 4, and 5 stipulated. These changes do not alter the actual CTS requirement for SHUTDOWN MARGIN, nor do they impose any additional requirements. These changes merely present the same information in a different format necessary to convert to the ITS. As such, these changes are considered administrative in nature.

X

X

MORE RESTRICTIVE CHANGES (M)

M.1 CTS 3.10.1a specifies "With four primary coolant pumps in operation at hot shutdown and above, the shutdown margin shall be 2%." However there is no action specified in the CTS if the shutdown margin is found to be less than 2% and so the plant would have to enter LCO 3.0.3. In the proposed ITS, if the SHUTDOWN MARGIN is found to be below the limit, boration must be initiated within 15 minutes. This is similar to the restoration action specified in CTS 3.10.1b which specifies if shutdown margin is below the required amount that "boration shall be immediately initiated to increase and maintain the shutdown margin." Since in the CTS, LCO 3.0.3 would be have to be entered if the SHUTDOWN MARGIN was found to be below the 2% limit, the 15 minutes to initiate boration is considered to be a more restrictive change. Initiating boration to restore the required amount of SHUTDOWN MARGIN is the appropriate action to take in this situation to return the plant to a safe condition. Furthermore, CTS 3.10.1c does not specify actions to take if flow is ≥ 2810 and the shutdown margin requirements (boron concentration greater than the cold shutdown boron concentration) have not been met. Therefore, if the SHUTDOWN MARGIN was not met, and the plant was above the CTS Cold Shutdown (210°F) then the plant would have to be shutdown in accordance with LCO 3.0.3. In the proposed ITS, ACTION A requires that if the SHUTDOWN MARGIN (SDM) requirement is not within limit, then boration must be initiated within 15 minutes to restore SDM to within limit. Therefore, since the proposed ITS requires that action be taken with 15 minutes, it is considered to be a more restrictive action. This change is consistent with NUREG-1432.

RAI 3.1-01

M.2 The Palisades Nuclear Plant CTS does not contain an explicit surveillance requirement for SHUTDOWN MARGIN even though there was a requirement that the limits be met as specified in 3.10.1. Proposed ITS 3.1.1 adds SR 3.1.1.1 ~~and SR 3.1.1.2~~ to verify SHUTDOWN MARGIN "every 24 hours." Since the requirement to verify SHUTDOWN MARGIN was not explicitly required in the CTS, the addition of the proposed Frequency is considered a "more restrictive" change. This change is consistent with NUREG-1432.

X

M.3 CTS 3.10.7 includes an exception which allows a deviation from the requirement for shutdown margin during performance of CRDM exercises. Proposed ITS 3.1.1 does not contain this same exception since violation of the LCO is not expected during the performance of the control rod drive exercise surveillance (SR 3.1.4.4). During the performance of SR 3.1.4.4, control rods will be exercised between 6 inches and 8 inches. The change in reactivity as a result of this movement is small due to the relative worth of the control rods which is largely determined by their position in the core at the time this SR is performed. This small change in reactivity is not enough to cause a violation of the Shutdown Margin requirements of ITS 3.1.1. Thus, reliance on the exception contained in CTS 3.10.7 is not needed. This change is consistent with NUREG-1432.

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

There were ~~no~~ "Removal of ~~Details~~" changes in ~~this~~ ~~specification~~ RAI 3.1-01

LA.1 New

CTS 3.10.1 contains the requirements for Shutdown Margin including specific values based on plant conditions and configuration. This proposed change relocates the values for Shutdown Margin to the COLR in order to provide core design and operational flexibility that can be used for improved fuel management and to solve plant specific issues. Placing the Shutdown Margin values in the COLR allows the core design to be finalized after shutdown when the actual end of cycle burnup is known. This would save redesign efforts if the actual burnup differs from the projected value. Current reload design efforts and the resolution of plant specific issues are restricted by the guidelines to not change the Shutdown Margin since it would result in a License Amendment Request. Although the actual value of Shutdown Margin is not derived through calculations, it is assumed to be an initial input in the plant safety analyses. As such, a change in Shutdown Margin must be evaluated for its impact on the safety analyses to determine if the revised value results in an unreviewed safety question. Placing the Shutdown Margin limits in the COLR does not result in a significant impact on plant safety since changes to the safety analyses (including a change in Shutdown Margin limits) are done in accordance with NRC approved methodologies.

LESS RESTRICTIVE CHANGES (L)

There were no "Less Restrictive" changes in this specification.

X

L.1 ~~CTS 3.10.1b requires SHUTDOWN MARGIN to be maintained ≥ 3.75% whenever less than four Primary Coolant Pumps (PCPs) are in operations and the plant is in hot shutdown or above. The basis for this requirement is to ensure an adequate amount of SHUTDOWN MARGIN is available to prevent a return to power following a Main Steam Line Break (MSLB). Inclusion of this requirement in the CTS was approved in Amendment 31 to the Provisional Operating License for the Palisades Plant (November 1, 1977) which authorized power levels up to 2530 Mwt. In support of Amendment 31, an analysis of the MSLB with two operating PCPs was performed to address operation of the plant with less than four operating PCPs since this configuration was permitted by the technical specifications during plant heatups and cooldowns, and for a restricted period of time at reduced power level. As part of the conversion to the ISTS and to establish a single value for SHUTDOWN MARGIN in Mode 3 with less than four PCPs in operation, a re-evaluation of the MSLB with two operating PCPs was performed assuming a minimum SHUTDOWN MARGIN of 3.5%. The evaluation shows that this event does not present as great a challenge to DNB and fuel centerline melt as the steamline break analysis of record. As such, ITS 3.1.1, "SDM" is proposed with a minimum SHUTDOWN MARGIN limit of 3.5%. Relaxing the requirement of CTS 3.10.1b to maintain a minimum SHUTDOWN MARGIN of 3.75% whenever less than four PCPs are operating in hot standby (ITS Mode 3) is acceptable since the consequences of an MSLB with less than four PCPs operating with a SHUTDOWN MARGIN of 3.5% are bounded by the MSLB analysis of record.~~

X

RAI 3.1-6

RAI 3.1-01

LESS RESTRICTIVE CHANGES L.1

There were no "Less Restrictive' Changes associated with this Specification.

~~CTS 3.10.1b requires SHUTDOWN MARGIN to be maintained ≥ 3.75% whenever less~~ than four Primary Coolant Pumps (PCPs) are in operations and the plant is in hot shutdown or above. The basis for this requirement is to ensure an adequate amount of SHUTDOWN MARGIN is available to prevent a return to power following a Main Steam Line Break (MSLB). Inclusion of this requirement in the CTS was approved in Amendment 31 to the Provisional Operating License for the Palisades Plant (November 1, 1977) which authorized power levels up to 2530 Mwt. In support of Amendment 31, an analysis of the MSLB with two operating PCPs was performed to address operation of the plant with less than four operating PCPs since this configuration was permitted by the technical specifications during plant heatups and cooldowns, and for a restricted period of time at reduced power level. As part of the conversion to the ISTS and to establish a single value for SHUTDOWN MARGIN in MODE 3 with less than four PCPs in operation, a re-evaluation of the MSLB with two operating PCPs was performed assuming a minimum SHUTDOWN MARGIN of 3.5%. The evaluation shows that this event does not present as great a challenge to DNB and fuel centerline melt as the steamline break analysis of record. As such, ITS 3.1.1, "SHUTDOWN MARGIN (SDM)" is proposed with a minimum SHUTDOWN MARGIN limit of 3.5%. Relaxing the requirement of CTS 3.10.1b to maintain a minimum SHUTDOWN MARGIN of 3.75% whenever less than four PCPs are operating in hot standby (ITS MODE 3) is acceptable since the consequences of an MSLB with less than four PCPs operating with a SHUTDOWN MARGIN of 3.5% are bounded by the MSLB analysis of record.

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

   Analyzed events are assumed to be initiated by the failure of plant structures, systems or components. The proposed change relaxes the required SHUTDOWN MARGIN from 3.75% to 3.5% when less than four PCPs are in operation in MODE 3. SHUTDOWN MARGIN is neither a accident initiator, nor accident precursor and therefore can not affect the probability of an accident. Therefore, the proposed change does not result in a significant increase in the probability of an accident previously evaluated.

1. (continued)

The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, and the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. The proposed change relaxes the required SHUTDOWN MARGIN from 3.75% to 3.5% when less than four PCPs are in operation in MODE 3. The minimum required SHUTDOWN MARGIN is an initial assumption used in the MSLB accident which ensures specified acceptable fuel design limits are not exceeded. A minimum SHUTDOWN MARGIN value of 3.75% prevents a return to power in the event of the worst steam line break assuming less than four operating PCPs. The maximum return to power with a 3.5% SHUTDOWN MARGIN is approximately 150 MWt. Although a reduction in available SHUTDOWN MARGIN from 3.75% to 3.5% results in a higher return to power following a MSLB, the consequences of a MSLB with less than four PCPs operating is bounded by the analysis of record for a MSLB. As such, the acceptable fuel design limits and radiological consequences resulting from a change in SHUTDOWN MARGIN are with the limits derived from Standard Review Plan section 15.1.5 appendix A, and 10 CFR 100. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant. No new equipment is being introduced, and no installed equipment is being operated in a new or different manner. The proposed change relaxes the required SHUTDOWN MARGIN from 3.75% to 3.5% when less than four PCPs are in operation in MODE 3. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

RAI 3 1-01

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

The margin of safety is determined by the design and qualification of the plant equipment, the operation of the plant within analyzed limits, and the point at which protective or mitigative actions are initiated. The proposed change relaxes the required SHUTDOWN MARGIN from 3.75% to 3.5% when less than four PCPs are in operation in MODE 3. The proposed change does not effect established safety limits, operating restrictions, or design assumptions. The margin of safety for an MSLB is established by the event described in the FSAR which considers the most limiting case initiated from hot full power. This case bounds the consequences (radiological and fuel cladding failure) from other initial operating states including operation with less than four PCPs and an initial SHUTDOWN MARGIN of 3.5%. Thus, the margin of safety previously established for the MSLB accident of record has remained unchanged. Therefore, this change does not involve a significant reduction in a margin of safety.

RAI 3.1-01

RAI 3.1-01

6

3.1 REACTIVITY CONTROL SYSTEMS

TSTF-136

3.1.1 SHUTDOWN MARGIN (SDM) ~~—T_{avg} > 200°F (Analog)~~

7

LCO 3.1.1 SDM shall be ~~≥ [4.5]% Δk/k.~~ within the limits provided in the COLR,

TSTF-9

APPLICABILITY: MODES 3, ~~and~~ 4, and 5

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

TSTF-9

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify SDM ~~is ≥ [4.5]% Δk/k.~~ 2.0 / 6 to be within limits.	24 hours

~~INSERT 1~~

7

~~INSERT 2~~

~~7~~

~~TSTF-9 NOT USED~~

~~9~~

RAI 3.1-01

SECTION 3.1

INSERT 1

-----NOTE-----

Only required to be met in MODE 3 when T_{ave} is ≥ 525°F and four primary coolant pumps are operating.

INSERT 2

-----NOTE-----

Only required to be met in MODE 3 when T_{ave} is ≥ 525°F with less than four primary coolant pumps operating, in Mode 3 when T_{ave} is < 525°F, and in MODES 4 and 5.

SR 3.1.1.2	Verify SDM is ≥ 3.5% Δρ.	24 hours

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~SDM. An idle RCP cannot, therefore, produce a return to power from the hot standby condition.~~ RAI 3.1-01

SDM satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2). 2

LCO

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value for ~~of the LCO~~. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, then the minimum required time assumed for operator action to terminate dilution may no longer be applicable. 9

SDM is a ~~core~~ control rod physics design condition that can be ensured through ~~CEA~~ positioning (regulating and shutdown ~~CEA~~ rods) and through the soluble boron concentration. 4

APPLICABILITY

In MODES 3 ~~and~~ 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.~~5~~5, ~~"Shutdown Control Element Assembly (CEA) Insertion Limits,"~~ and LCO 3.1.~~6~~6. ~~If the insertion limits of LCO 3.1.6 or LCO 3.1.7 are not being complied with, SDM is not automatically violated. The SDM must be calculated by performing a reactivity balance calculation (considering the listed reactivity effects in Bases Section SR 3.1.1.1). In MODE 5, SDM is addressed by LCO 3.1.2, "SHUTDOWN MARGIN (SDM)—T_{avg} ≤ 200°F."~~ In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." 4

TSTF-67 14

TSTF-136 7

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

(continued)

BASES

ACTIONS A.1 (continued)

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid storage tank ~~or the borated water storage tank.~~ (3) The operator should borate with the best source available for the plant conditions. (4)

injection flow (P)

(4) concentrated

In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle, when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of (P) 1% Δk/k must be recovered and a boration flow rate of {35} gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately (2)25 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% Δk/k. These boration parameters of {35} gpm and {100} ppm represent typical values and are provided for the purpose of offering a specific example. (2) (4) (1) (4) (4) (4) (1)

10E-4 Δρ/ppm

SURVEILLANCE REQUIREMENTS SR 3.1.1.1

SDM is verified by ~~performing~~ a reactivity balance calculation, considering the listed reactivity effects:

a. (P) RCS boron concentration; (4)

(5) Control rod

b. CEA positions; (4)

c. (P) RCS average temperature; (4)

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

~~f. Samarium concentration;~~ and (10)

(f) ~~g.~~ Isothermal temperature coefficient (ITC). (2)

(continued)

3-r

SECTION 3.1

INSERT 1

Samarium is not considered in the reactivity analysis since the analysis assumes that the negative reactivity due to samarium is offset by the positive reactivity of plutonium build in.

RAI 3.1-01

INSERT 2

To maintain consistency with the assumptions used in the MSLB analysis, two values of SDM are specified in the surveillance requirements.

SR 3.1.1.1 requires SDM to be ≥ 2% Δρ. This SDM value ensures the consequences of an MSLB, as well as the other events described in the Applicable Safety Analysis, will be acceptable whenever the plant is in MODE 3 with T_{ave} ≥ 525°F and four Primary Coolant Pumps (PCPs) are in operation. Therefore, SR 3.1.1.1 is modified by a Note which only requires this SR to be met in MODE 3 when T_{ave} ≥ 525°F and four PCPs are in operation. SR 3.1.1.2 requires SDM to be ≥ 3.5% Δρ. This SDM value ensures the consequences of an MSLB, as well as the other events described in the Applicable Safety Analysis, will be acceptable as a result of a cooldown of the PCS which adds positive reactivity in the presence of a negative moderator temperature coefficient. As such, the requirements of SR 3.1.1.2 must be met whenever the plant is in MODE 3 with T_{ave} ≥ 525°F with less than four PCPs operating, MODE 3 with T_{ave} < 525°F and MODES 4 and 5. Therefore, SR 3.1.1.2 is modified by a Note which only requires this SR to be met in MODE 3 with T_{ave} ≥ 525°F with less than four PCPs operating, MODE 3 with T_{ave} < 525°F, and MODES 4 and 5.

Change	**Discussion**

RAI 3.1-01

7. ISTS Change Traveler TSTF-136 combines ISTS 3.1.1 and ISTS 3.1.2 into a single specification in order to eliminate unnecessary and confusing duplication, and renumbers the remaining specifications in Section 3.1. The impetus for this change was the approval of TSTF-9 which allowed the values for shutdown margin to be moved to the COLR. As a result of TSTF-9, the LCO, Actions, and Surveillance Requirements of ISTS 3.1.1 and ISTS 3.1.2 were the same. ~~Although the~~ Palisades ~~plant~~ has ~~not~~ relocated the shutdown margin values to the COLR, it has adopted the consolidation of ISTS 3.1.1 and ISTS 3.1.2 into a single specification. Proposed ITS 3.1.1 address the plant conditions encompassed in MODEs 3, 4, and 5. As a result of this consolidation, a new Surveillance Requirement has been added (ITS SR 3.1.1.2), and Note has been included to SR 3.1.1.1 to modify the performance of the surveillances. The format of the Surveillance Requirements are consistent with the style and format presented in the Writers Guide (NUMARC 93-03).

in accordance with TSTF-9 and has consolidated

8. The Palisades plant was designed prior to issuance of the General Design Criteria (GDC) in 10 CFR 50. Therefore, reference to the GDCs is omitted and appropriately replaced by reference to "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 7, 1971. It was this updated discussion, including the identified exemptions, which formed the original plant Licensing Basis for future compliance with the GDCs.

RAI 3.1-02

9. TSTF-9 permits relocation of the shutdown margin values specified in ISTS 3.1.1 and ISTS 3.1.2 to the COLR. At this time, the Palisades plant has elected not to exercise this option and has maintained the required shutdown values in the ITS. The appropriate justification for this change is provided in DOC LA.1 for ITS 3.1.1.

10. Samarium is not considered in the Palisades Nuclear Plant reactivity balance due to the fact the that Palisades Nuclear Plant fuel vendor does not account for Samarium in fuel design calculations. The vendor assumes that the negative reactivity defect due to Samarium is offset by the positive reactivity of Plutonium build in. Plutonium build in and Samarium are equally competing reactivity effects that are accounted for in fuel design calculations. Therefore, including Samarium into the SDM calculation would not be correct for the Palisades Nuclear Plant.

5.6 Reporting Requirements

5.6.4 Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the NRC no later than the fifteenth of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following: RAI 3.1-01

3.1.1 Shutdown Margin X
3.1.~~x~~6 Regulating Rod Group Position Limits X
3.2.1 Linear Heat Rate Limits Ed
3.2.2 Radial Peaking Factor Limits
3.2.4 ASI Limits

b. The analytical methods used to determine the core operating limits shall be those approved by the NRC, specifically those described in the latest approved revision of the following documents:

1. XN-75-27(A), "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," and Supplements 1(A), 2(A), 3(P)(A), 4(P)(A), and 5(P)(A); Exxon Nuclear Company. (LCOs 3.1.1 3.1.6, 3.2.1, 3.2.2, & 3.2.4) RAI 3.1-01 X

2. ANF-84-73(P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," and Appendix B(P)(A) and Supplements 1(P)(A), 2(P)(A); Advanced Nuclear Fuels Corporation. (LCOs 3.1.1 3.1.6, 3.2.1, 3.2.2, & 3.2.4) X

3. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company. (LCOs 3.2.1, 3.2.2, & 3.2.4)

4. ANF-84-093(P)(A), "Steamline Break Methodology for PWRs," and Supplement 1(P)(A); Advanced Nuclear Fuels Corporation. (LCOs 3.1.1 3.1.6, 3.2.1, 3.2.2, & 3.2.4) X

5. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," and Supplements 1(P)(A), 2(P)(A), 3(P)(A), and 4(P)(A); Exxon Nuclear Company. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

6.0 ADMINISTRATIVE CONTROLS

5.6.5 | ~~6.6.5~~ Core Operating Limits Report (COLR)

RAI 3.1-01

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following: A.8 X

3.1.1 Shutdown Margin
3.2.4 ~~3.1.1~~ ASI Limits. Rod Position
3.1.6 ~~3.10.5~~ Regulating Group ~~Insertion~~ Limits A.1
3.2.1 ~~3.23.1~~ Linear Heat Rate (LHR) Limits
3.2.2 ~~3.23.2~~ Radial Peaking Factor Limits

b. The analytical methods used to determine the core operating limits shall be those approved by the NRC, specifically those described in the latest approved revision of the following documents:

1. XN-75-27(A), "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," and Supplements 1(A), 2(A), 3(P)(A), 4(P)(A), and 5(P)(A); Exxon Nuclear Company. (LCOs ~~3.1.1~~ 3.2.4, ~~3.10.1~~ 3.1.1, ~~3.10.5~~ 3.1.6, ~~3.23.1~~ 3.2.1, & ~~3.23.2~~ 3.2.2) X A.1 A.8

2. ANF-84-73(P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," and Appendix B(P)(A) and Supplements 1(P)(A), 2(P)(A); Advanced Nuclear Fuels Corporation. (LCOs 3.1.1 ~~3.1.1~~ 3.2.4, ~~3.10.5~~ 3.1.6, ~~3.23.1~~ 3.2.1, & ~~3.23.2~~ 3.2.2) A.1 A.8

3. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company. (LCOs ~~3.1.1~~ 3.2.4, ~~3.23.1~~ 3.2.1, & ~~3.23.2~~ 3.2.2) A.1

4. ANF-84-093(P)(A), "Steamline Break Methodology for PWRs," and Supplement 1(P)(A); Advanced Nuclear Fuels Corporation. (LCOs 3.1.1 ~~3.10.1~~ 3.2.4, ~~3.10.5~~ 3.1.6, ~~3.23.1~~ 3.2.1, & ~~3.23.2~~ 3.2.2) A.1 A.8

5. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," and Supplements 1(P)(A), 2(P)(A), 3(P)(A), and 4(P)(A); Exxon Nuclear Company. (LCOs ~~3.1.1~~ 3.2.4, ~~3.10.5~~ 3.1.6, ~~3.23.1~~ 3.2.1, & ~~3.23.2~~ 3.2.2) A.1

6. EXEM PWR Large Break LOCA Model as defined by: (LCOs ~~3.10.5~~ 3.1.6, ~~3.23.1~~ 3.2.1, & ~~3.23.2~~ 3.2.2) A.1

   a) XN-NF-82-20(A), "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," and Supplements 1(P)(A), 2(P)(A), 3(P)(A), and 4(P)(A); Exxon Nuclear Company.

   b) XN-NF-82-07(P)(A), "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company.

   c) XN-NF-81-58(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," and Supplements 1(P)(A), 2(P)(A), 3(P)(A), and 4(P)(A); Exxon Nuclear Company.

A.5 CTS 6.4.1 requires that written procedures shall be established, implemented, and maintained for the activities listed. In this list, the CTS contains item b., "Refueling operations, and item c., "Surveillance and test activities of safety-related activities." These items are included in the procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978 which is referenced in CTS 6.4.1a and included in the proposed ITS 5.4.1a. Therefore, since these procedures are already required by the reference to Regulatory Guide 1.33, Revision 2, February 1978, they are not included in the proposed ITS. This change is an administrative change since no requirements have changed. This change maintains consistency with NUREG-1432.

A.6 CTS 6.4.1 requires that written procedures shall be established, implemented, and maintained for the activities listed. In this list, the CTS contains item f., "Site Security Plan implementation" and item g.,"Site Emergency Plan implementation.." These items were recommended to be removed from the Technical Specifications in NRC Generic Letter 93-07 since they are duplicative of regulations contained in the Code of Federal Regulations part 50 and 73. This change is considered to be an administrative change since these requirements must still be met as required by the Code of Federal Regulations. This change maintains consistency with NUREG-1432.

A.7 CTS 6.5.7 is entitled "Inservice Inspection and Testing Program." In the proposed ITS 5.5.7, the title is changed to the "Inservice Testing Program." This change is considered to be an administrative change since the requirements of the program are unchanged. This change maintains consistency with NUREG-1432.

RAI 3.1-01

New
See INSERT

A.8 CTS 6.6.5b.1 lists, among referenced LCOs, "3.10.1." That item is unnecessary and has been deleted. Neither CTS 3.10.1, nor its ITS replacement reference the COLR. CTS 6.6.5 a. lists the core operating limits that are established and documented in the COLR prior to each core reload. Specifically, these limits are: ASI Limits (CTS 3.1.1), Regulating Group Insertions Limits (CTS 3.10.5), Linear Heat Rate Limits (CTS 3.23.1), and Radial Peaking Factor Limits (CTS 3.23.2). CTS 6.6.5 b. list the documents approved by the NRC that describe the analytical methods used to determine the core operating limits. As part of this listing, cross references are made to the LCOs pertaining to the affected limit (e.g., ASI Limits, Regulating Group Insertion Limits, etc...). In error, CTS 6.6.5 b.1. lists CTS 3.10.1 (Shutdown Margin Requirements) as an LCO related to a document that describes analytical methods used to determine the core operating limits. Since Shutdown Margin is not a cycle dependent limit (the limit is contained in the technical specifications and not in the COLR), referencing CTS 3.10.1 in CTS 6.6.5 b.1 is inappropriate and has been deleted. This change has been characterized as administrative in nature since it does not ~~alter any requirement~~ of the CTS, but simply corrects an administrative oversight.

RAI 3.1-01

INSERT

CTS 6.6.5 a. lists the core operating limits that are established and documented in the COLR prior to each core reload. Specifically, these limits are: ASI Limits (CTS 3.1.1), Regulating Group Insertions Limits (CTS 3.10.5), Linear Heat Rate Limits (CTS 3.23.1), and Radial Peaking Factor Limits (CTS 3.23.2). CTS 6.6.5 b. list the documents approved by the NRC that describe the analytical methods used to determine the core operating limits. As part of this listing, cross references are made to the LCOs pertaining to the affected limit (e.g., ASI Limits, Regulating Group Insertion Limits, etc...). In error, CTS 6.6.5 b.1 lists CTS 3.10.1 (Shutdown Margin Requirements) as an LCO related to a document that describes analytical methods used to determine the core operating limits. However, as part of the conversion to the Improved Technical Specifications, the values for Shutdown Margin were relocated from CTS 3.10.1 to the COLR consistent with NUREG-1432 as modified by TSTF-9. As such, CTS 6.6.5 (proposed ITS 5.6.5) has been revised to include ITS LCO 3.1.1 "Shutdown Margin" as a limit that is established and maintained in the COLR. This change has been characterized as administrative in nature since it does not alter any requirement of the CTS, but simply provides conforming information.

5.6 Reporting Requirements

CTS 6.6.4

5.6.4 Monthly Operating Reports (continued)

power operated relief valves or pressurizer safety valves,] shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report. ①

6.6.5

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

RAI 3.1-01 X

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

LCO 3.1.1 Shutdown MARGIN
LCO 3.1.6 Regulating Rod Group Position Limits
LCO 3.2.1 Linear Heat Rate Limits
LCO 3.2.2 Radial Peaking Factor Limits
LCO 3.2.4 ASI Limits

~~The individual specifications that address core operating limits must be referenced here.~~ ①

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in THE LATEST REVISION OF the following documents: ⑤

<INSERT>

~~Identify the Topical Report(s) by number, title, date, and NRC staff approval document, or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.~~

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

~~5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~

~~a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic~~ ⑳

(continued)

SECTION 5.0

INSERT

RAI 3.1-01

1. XN-75-27(A), "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," and Supplements 1(A), 2(A), 3(P)(A), 4(P)(A), and 5(P)(A); Exxon Nuclear Company. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)
   3.1.1

2. ANF-84-73(P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," and Appendix B(P)(A) and Supplements 1(P)(A), 2(P)(A); Advanced Nuclear Fuels Corporation. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)
   3.1.1

3. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company. (LCOs 3.2.1, 3.2.2, & 3.2.4)

4. ANF-84-093(P)(A), "Steamline Break Methodology for PWRs" and Supplement 1(P)(A); Advanced Nuclear Fuels Corporation. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)
   3.1.1

5. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," and Supplements 1(P)(A), 2(P)(A), 3(P)(A), and 4(P)(A); Exxon Nuclear Company. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

6. EXEM PWR Large Break LOCA Model as defined by:
   (LCOs 3.1.6, 3.2.1, & 3.2.2)

   a) XN-NF-82-20(A), "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," and Supplements 1(P)(A), 2(P)(A), 3(P)(A), and 4(P)(A); Exxon Nuclear Company.

   b) XN-NF-82-07(P)(A), "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company.

***NRC REQUEST*:**

3.1-02 ITS 3.1.3 [STS 3.1.4] Moderator Temperature Coefficient (MTC)
ITS SR 3.1.3.1 and STS SR 3.1.4.1 Note
JFD 11

The STS SR 3.1.4.1 includes a note that the SR need not be performed prior to entry into Mode 2. This note has been excluded in the ITS because the frequency specifies prior to 2% RTP.

***Comment*:** The SR frequency does not negate the applicability of SR 3.0.4; that SRs must be met prior to entry into modes of applicability. In any case, including the note avoids misinterpretation. Recommend including the note.

***Consumers Energy Response*:**

The Note which modifies ISTS SR 3.1.4.1 is intended to avoid a potential SR 3.0.4 conflict. However, the inclusion of this Note in the ISTS is redundant since the Frequency specifies the precise requirement for performing the surveillance. NUMARC 93-03 "Writer's Guide for the Restructured Technical Specification" Section 4.1.7 (Chapter 3 Surveillance Requirements Contents) item "f" states; "To specify the precise requirements for performance of a Surveillance, such that exceptions to SR 3.0.4 would not be necessary, the Frequency may be specified such that it is not [due] until the specific conditions are met. Alternately, the surveillance may be stated as not required [to be met or performed] until a particular event, condition, or time has been reached." The Frequency of proposed ITS SR 3.1.3.1 is specified as a "condition" versus a "Mode". Therefore, a corresponding Note in the SR would have to be stated as a "condition" (i.e., Not required to be performed prior to 2% RTP) to avoid an SR 3.0.4 conflict between the entry conditions of Mode 2 (keff $\geq$ 0.99) and 2% RTP. Since a Note containing this information would be redundant to the Frequency, it was not included in the ITS.

***Affected Submittal Pages*:**

None

NRC REQUEST:

3.1-03 ITS 3.1.3 [STS 3.1.4] Moderator Temperature Coefficient (MTC)
STS SR 3.1.4.2
JFD 6

The STS SR 4.1.4.2 is not included because it is not in the CTS and "... the most negative limit is also assured of being met by design."

Comment: Once the initial MTC measurement is met is it always true that End-Of-Cycle (EOC) measurements will be met for all core loadings? Is this a plant unique feature?

Consumers Energy Response:

Yes, once the value of MTC is verified to be less positive than the technical specification limit at the beginning of core life, the value of MTC will always be less than the technical specification limit at the End-of-Cycle based on current core loading design methodologies. It is believed this feature is not unique to the Palisades plant.

In regards to the change in MTC over core life, ISTS SR 3.1.4.2 requires a verification that MTC is within the lower limit assumed in the safety analysis after reaching 40 EFPD of core burnup, and within 7 EFPD of reaching two-thirds of the expected core burnup. As discussed in JFD 6, the CTS does not contain a requirement to verify MTC is within the lower limit assumed in the safety analysis since this value is assured by core design. That is, the measured value of MTC can be extrapolated using core modeling techniques to determine the value that will exist at the end of core life. The predicted value of MTC is verified to be less negative than the value previously assumed in the safety analysis. Inherent to this process is the assumption that the core continues to behave as designed. This assumption is verified by performing proposed ITS SR 3.1.2.1 which verifies the overall core reactivity balance is within plus or minus 1% of the predicted values every 31 EFPD. Should an anomaly greater than 1% develop between the measured and predicted core reactivity values, an evaluation of the core design and the effects on the safety analyses must be performed.

Affected Submittal Pages:

None

NRC REQUEST:

3.1-04 ITS 3.1.4 [STS 3.1.5] Control Rod Alignment
ITS 3.1.4 Required Action D
Bases LCO section (page B 3.1-26) and Bases ACTIONS (page B 3.1-30)
DOC M.6
JFD 10 and JFD 17

The ITS has added a Required Action D that an immovable but trippable control rod shall be returned to operable status prior to entering Mode 2.

Comment #1: The completion time for the Required Action is prior to entering the LCO's applicability, which is illogical; the condition is not needed.

Consumers Energy Response:

ITS Condition "D" addresses the situation when one full-length control rod is immovable but trippable. As described in DOC M.6, the CTS does not contain an explicit LCO for control rod Operability. Thus, the plant is allowed unrestricted operation when one control rod is inoperable. Since proposed ITS 3.1.4 requires all control rods to be Operable, declaring an immovable but trippable control rod inoperable without a corresponding Required Action, would require entry into Specification 3.0.3. As such, ITS Condition "D" has been incorporated to preclude an unnecessary plant shutdown due to an immovable control rod. Since unlimited continued operation with an inoperable, but trippable, rod is allowed, LCO 3.0.4 would not prohibit MODE changes while in Condition "D." The proposed Completion Time was specified to assure repairs were made prior to the next reactor start-up.

Affected Submittal Pages:

Att 1 ITS 3.1.4, page 3.1.4-2
Att 2 ITS 3.1.1, page B 3.1.1-3
Att 2 ITS 3.1.4, page B 3.1.4-1
Att 2 ITS 3.1.4, page B 3.1.4-4
Att 2 ITS 3.1.4, page B 3.1.4-8
Att 2 ITS 3.1.4, page B 3.1.4-10
Att 2 ITS 3.1.4, page B 3.1.4-11
Att 2 ITS 3.1.4, page B 3.1.5-1
Att 2 ITS 3.1.4, page B 3.1.5-4
Att 2 ITS 3.1.4, page B 3.1.6-1
Att 2 ITS 3.1.4, page B 3.1.6-4
Att 5 NUREG, page B 3.1-34
Att 5 NUREG, B 3.1-36 Insert

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One full-length control rod immovable, but trippable.	D.1 Restore control rod to OPERABLE status.	Prior to entering MODE 2 from MODE 3
E. Required Action and associated Completion Time not met. OR One or more control rods inoperable for reasons other than Condition D. OR Two or more control rods misaligned by > 8 inches. OR Both rod position indication channels inoperable for one or more control rods.	E.1 Be in MODE 3.	6 hours

RAI 3.1-04
X

BASES

APPLICABLE SAFETY ANALYSES (continued)

In addition to the limiting MSLB transient, the SDM requirement for MODES 3 and 4 must also protect against an inadvertent boron dilution; (Ref. 3) and an uncontrolled control rod bank withdrawal from subcritical conditions (Ref. 5).

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the PCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life when critical boron concentrations are highest.

The withdrawal of a control rod bank from subcritical conditions adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The withdrawal of control rod banks also produce a time dependent redistribution of core power.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled control rod banks withdrawal transient is terminated by either a high power trip or a high pressurizer pressure trip. In all cases, power level, PCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2).

LCO

RAI 3.1-01

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value for ~~of the LCO~~. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, then the minimum required time assumed for operator action to terminate dilution may no longer be applicable. X

SDM is a core physics design condition that can be ensured through Full-length control rod positioning (regulating and shutdown rods) and through the soluble boron concentration. RAI 3.1-04 X

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Control Rod Alignment

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and regulating rods is an initial assumption in all safety analyses that assume full-length control rod insertion upon reactor trip. Maximum control rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. RA 3.1-04 X

The Palisades Nuclear Plant design criteria contain the applicable criteria for these reactivity and power distribution design requirements (Ref. 1).

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod ~~inoperability or~~ misalignment may cause increased power peaking, due to the asymmetric reactivity distribution, and a reduction in the total available control rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM. X ed

Limits on control rod alignment and OPERABILITY have been established, and all control rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Control rods are moved by their Control Rod Drive Mechanisms (CRDMs). Each CRDM moves its rod at a fixed rate of approximately 46 inches per minute. Although the ability to move a full-length control rod by its drive mechanism is not an initial assumption used in the safety analyses, it is required to support OPERABILITY. As such, the inability to move a full-length control rod results in that full-length control rod being inoperable. RA 3.1-04 X X X

BASES

APPLICABLE SAFETY ANALYSES (continued)

The most limiting static misalignment occurs when Bank 4 is fully inserted with one rod fully withdrawn (Bank 4 is 99 inches out of alignment with the rated Power Dependent Insertion Limit (PDIL). This event was bounded by the dropped full-length control rod event (Ref. 4).

Since the control rod drop incidents result in the most rapid approach to SAFDLs caused by a control rod misoperation, the accident analysis analyzed a single full-length control rod drop. The most rapid approach to the DNBR SAFDL may be caused by a single full-length control rod drop.

The above control rod misoperations may or may not result in an automatic reactor trip. In the case of the full-length rod drop, a prompt decrease in core average power and a distortion in radial power are initially produced, which, when conservatively coupled, result in a local power and heat flux increase, and a decrease in DNBR parameters.

The results of the control rod misoperation analysis show that during the most limiting misoperation events, no violations of the SAFDLs, fuel centerline temperature, or PCS pressure occur.

Control rod alignment satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2).

LCO

RAI 3.1-04

The limits on shutdown, regulating, and part-length rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on full-length OPERABILITY ensure that upon reactor trip, the control rods will be available and will be inserted to provide enough negative reactivity to shut down the reactor. The OPERABILITY requirements also ensure that the control rod banks maintain the correct ~~power distribution and control rod~~ alignment and that each control rod is capable of being moved by its CRDM. The OPERABILITY requirement for the part-length rods is that they are fully withdrawn. ~~and are capable of being moved by their CRDMs~~.

X

X

X

X

BASES

ACTIONS (continued)

RAI 3.1-04

D.1

Condition D is entered whenever it is discovered that a single control rod can not be moved by its operator yet the control rod is still capable of being tripped. Although the ability to move a control rod is not an initial assumption used in the safety analyses, it does relate to control rod OPERABILITY. The inability to move a control rod by its operator may be indicative of a systemic failure (other than trippability) which could potentially affect other rods. Thus, declaring a control rod inoperable in this instance is conservative since it limits the number of control rods which can not be moved by their operators to only one. The Completion Time to restore an inoperable control rod to OPERABLE status is stated as prior to entering MODE 2 from MODE 3. This Completion Time allows unrestricted operation in MODES 1 and 2 while conservatively preventing a reactor startup with an immovable control rod.

full-length

E.1

If the Required Action or associated Completion Time of Condition A, Condition B, Condition C, or Condition D is not met; one or more control rods are inoperable for reasons other than Condition D; or two or more control rods are misaligned by > 8 inches, or two channels of control rod position indication are inoperable for one or more control rods, the plant is required to be brought to MODE 3. By being brought to MODE 3, the plant is brought outside its MODE of applicability. Continued operation is not allowed in the case of more than one control rod misaligned from any other rod in its group by > 8 inches, or two or more rods inoperable. This is because these cases may be indicative of a loss of SDM and power re-distribution, and a loss of safety function, respectively.

Also, if no rod position indication exists for one or more control rods, continued operation is not allowed because the safety analysis assumptions of rod position cannot be ensured.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.3

Demonstrating the rod position deviation alarm is OPERABLE verifies the alarm is functional. The 92 day Frequency takes into account other information continuously available to the operator in the control room, so that during control rod movement, deviations can be detected.

SR 3.1.4.4

RAI 3.1-04

Full-length

Verifying each full-length control rod is trippable would require that each control rod be tripped. In MODES 1 and 2, tripping each control rod would result in radial or axial power tilts, or oscillations. Therefore, individual full-length control rods are exercised every 92 days to provide increased confidence that all control rods continue to be trippable, even if they are not regularly tripped. A movement of 6 inches is adequate to demonstrate motion without exceeding the alignment limit when only one control rod is being moved. The 92 day Frequency takes into consideration other information available to the operator in the control room and other surveillances being performed more frequently, which add to the determination of OPERABILITY of the control rods. At any time, if a control rod(s) is inoperable, a determination of the trippability of the control rod(s) must be made, and appropriate action taken.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.5

ed

Performance of a CHANNEL CALIBRATION of each control rod position indication channel ensures the channel is OPERABLE and capable of indicating control rod position over the entire length of the control rod's travel with the exception of the secondary rod position indicating channel dead band near the bottom of travel. This dead band exists because the control rod drive mechanism housing seismic support prevents operation of the reed switches. Since this Surveillance must be performed when the reactor is shut down, an 18 month Frequency to be coincident with refueling outage was selected. Operating experience has shown that these components usually pass this Surveillance when performed at a Frequency of once every 18 months. Furthermore, the Frequency takes into account other surveillances being performed at shorter Frequencies, which determine the OPERABILITY of the control rod position indicating systems. X

SR 3.1.4.6

Verification of full-length control rod drop times determines that the maximum control rod drop time is consistent with the assumed drop time used in that safety analysis (Ref. 2). The 2.5 second acceptance criteria is measured from the time the CRDM clutch is deenergized by the reactor protection system or test switch to 90% insertion. This time is bounded by that assumed in the safety analysis (Ref.2). Measuring drop times prior to reactor criticality, after reactor vessel head reinstallation, ensures that reactor internals and CRDMs will not interfere with full-length control rod motion or drop time and that no degradation in these systems has occurred that would adversely affect full-length control rod motion or drop time. Individual full-length control rods whose drop times are greater than safety analysis assumptions are not OPERABLE. This SR is performed prior to criticality, based on the need to perform this Surveillance under the conditions that apply during a plant outage and because of the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

RAI 3.1-04

X X X

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown and Part-Length Rod Group Insertion Limits

BASES

RAI 3.1-04

X

full-length

BACKGROUND

The insertion limits of the shutdown rods are initial assumptions in all safety analyses that assume control rod insertion upon reactor trip. The insertion limits directly affect core power distributions and assumptions of available SDM, ejected rod worth, and initial reactivity insertion rate.

The Palisades Nuclear Plant design criteria (Ref. 1) and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," contain the applicable criteria for these reactivity and power distribution design requirements. Limits on shutdown rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the reactivity limits, ejected rod worth, and SDM limits are preserved.

The shutdown rods are arranged into groups that are radially symmetric. Therefore, movement of the shutdown rod groups does not introduce radial asymmetries in the core power distribution. The shutdown and regulating rod groups provide the required reactivity worth for immediate reactor shutdown upon a reactor trip.

The Palisades Nuclear Plant has four part-length control rods installed. The part-length rods are required to remain completely withdrawn during power operation except during rod exercising performed in conjunction with SR 3.1.4.4. The part-length rods do not insert on a reactor trip.

ed

X

ALL Control

The design calculations are performed with the assumption that the shutdown rod groups are withdrawn prior to the regulating rod groups. The shutdown rods can be fully withdrawn without the core going critical. This provides available negative reactivity for SDM in the event of boration errors. The shutdown rod groups are controlled manually by the control room operator. During normal plant operation, the shutdown rod groups are fully withdrawn. The shutdown rod groups must be completely withdrawn from the core prior to withdrawing any regulating rods during an approach to criticality. The shutdown rod groups are then left in this position until the reactor is shut down.

BASES

LCO (continued)

Maintaining the shutdown rod groups within their insertion limits ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. Maintaining the part-length rod group within its insertion limit ensures that the power distribution envelope is maintained.

APPLICABILITY

The shutdown and part-length rod groups must be within their insertion limits, with the reactor in MODES 1 and 2. In MODE 2 the Applicability begins anytime any regulating rod is withdrawn above 5 inches. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. In MODE 4, 5, or 6, the shutdown rod groups are inserted in the core to at least the lower electrical limit and contribute to the SDM. In MODE 3 the shutdown rod groups may be withdrawn in preparation of a reactor startup. Refer to LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

The Applicability has been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.4 (rod exercise test). Control rod exercising verifies the freedom of the rods to move, and requires the individual shutdown ~~or part-length~~ rods to move below the LCO limits for their group. Only the full-length rods are required to be tested by SR 3.1.4.4. The part-length rods may also ~~need to be periodically exercised to maintain mechanical seal integrity. Therefore, though not required part of SR 3.1.4.4, the part-length control rods may be exercised under the controlled conditions of SR 3.1.4.4.~~ RAI 3.1-04 X X

Positioning of an individual control rod within its group is addressed by LCO 3.1.4, "Control Rod Alignment."

... be moved however, if a Part-length rod is moved below the limit of the associated LCO, the Required Actions of Condition A must be taken.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Regulating Rod Group Position Limits

BASES

εd

X X

RAI 3.1-04

BACKGROUND

groups full-length

The insertion limits of the regulating ~~rods~~ are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power distributions, assumptions of available SDM, and initial reactivity insertion rate. The applicable criteria for these reactivity and power distribution design requirements are contained in the Palisades Nuclear Plant design criteria (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Limits on regulating rod group insertion have been established, and all regulating rod group positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking, ejected rod worth, reactivity insertion rate, and SDM limits are preserved.

The regulating rod groups operate with a predetermined amount of position overlap, in order to approximate a linear relation between rod worth and rod position (integral rod worth). The regulating rod groups are withdrawn and operate in a predetermined sequence. The group sequence and overlap limits are specified in the COLR.

The regulating rods are used for precise reactivity control of the reactor. The positions of the regulating rods are manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 2). Together, LCO 3.1.6; LCO 3.2.3, "QUADRANT POWER TILT (T_q)"; and LCO 3.2.4, "AXIAL SHAPE INDEX (ASI)," provide limits on control component operation and on monitored process variables to ensure the core operates within the linear heat rate (LCO 3.2.1, "Linear Heat Rate (LHR)") and radial peaking factor F_R^T and F_R^A (LCO 3.2.2, "Radial Peaking Factors) limits in the COLR.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Fuel cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result, should an accident occur with simultaneous violation of one or more of these LCOs. Changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

The SDM requirement is ensured by limiting the regulating and shutdown rod group insertion limits, so that the allowable inserted worth of the rods is such that sufficient reactivity is available to shut down the reactor to hot zero power. SDM assumes the maximum worth rod remains fully withdrawn upon trip (Ref. 4).

The most limiting SDM requirements for Mode 1 and 2 conditions at Beginning of Cycle (BOC) are determined by the requirements of several transients, e.g., Loss of Flow, etc. However, the most limiting SDM requirements for MODES 1 and 2 at End of Cycle (EOC) come from just one transient, Main Steam Line Break (MSLB). The requirements of the MSLB event at EOC for the full power and no load conditions are significantly larger than those of any other event at that time in cycle and, also, considerably larger than the most limiting requirements at BOC.

Although the most limiting SDM requirements at EOC are much larger than those at BOC, the available SDMs obtained via full-length tripping the control rods are substantially larger due to the much lower boron concentration at EOC. To verify that adequate SDMs are available throughout the cycle to satisfy the changing requirements, calculations are performed at both BOC and EOC. It has been determined that calculations at these two times in cycle are sufficient since the difference between available SDMs and the limiting SDM requirements are the smallest at these times in cycle. The full-length measurement of a control rod ~~banks~~ bank worth performed as part of the Startup Testing Program demonstrates that the core has the expected shutdown capability. Consequently, adherence to LCO 3.1.5, "Shutdown and Part-Length Rod Group Insertion Limits," and LCO 3.1.6 provides assurance that the available ~~SDS~~ SDM at any time in cycle will exceed the limiting SDM requirements at that time in cycle.

RAI 3.1-04 X
ed X
RAI 3.1-04
ed X

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown Control Element Assembly (CEA) Insertion Limits (Analog) 13
and Part-Length Rod Group 12

RAI 3.1-04

BASES

BACKGROUND

The insertion limits of the shutdown ~~CEAs~~ rods are initial full-length control rod assumptions in all safety analyses that assume ~~CEA~~ insertion upon reactor trip. The insertion limits directly affect core power distributions and assumptions of available SDM, ejected ~~CEA~~ rod worth, and initial reactivity insertion rate. 4

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INSERT 1

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," and GDC 26, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on shutdown ~~CEA~~ rod insertion have been established, and all ~~CEA~~ positions are monitored and controlled during power operation to ensure that the reactivity limits, ejected ~~CEA~~ rod worth, and SDM limits are preserved. 9 4 4

The shutdown ~~CEAs~~ rods are arranged into groups that are radially symmetric. Therefore, movement of the shutdown ~~CEAs does~~ rod groups not introduce radial asymmetries in the core power distribution. The shutdown and regulating ~~CEAs~~ provide the required reactivity worth for immediate reactor shutdown rod groups upon a reactor trip. 4 12

12

INSERT 2

The design calculations rod groups are performed with the assumption that the shutdown ~~CEAs~~ rod groups are withdrawn prior to the regulating ~~CEAs~~ rod groups. The shutdown ~~CEAs~~ can be fully withdrawn without the core going critical. This provides available negative reactivity for SDM in the event of boration errors. ~~The shutdown~~ ~~CEAs~~ All Control rod groups 12 are controlled manually ~~or automatically~~ by the control room operator. During normal ~~unit~~ plant operation, the shutdown ~~CEAs~~ rod groups are fully withdrawn. The shutdown ~~CEAs~~ rod groups must be completely withdrawn from the core prior to withdrawing any regulating ~~CEAs during an approach to~~ rods criticality. The shutdown ~~CEAs~~ rod groups are then left in this position until the reactor is shut down. They affect core power, burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal. 12 10 9 4 12

ed

(continued)

SECTION 3.1

RAI 31-04

INSERT 6

The Applicabilty has been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.4 (rod exercise test). Control rod exercising verifies the freedom of the rods to move, and requires the individual shutdown ~~or part-length~~ rods to move below the LCO limits for their group. Only the full-length rods are required to be tested by SR 3.1.4.4. The part-length rods may also need to be periodically exercised to maintain mechanical seal integrity. Therefore, though not required part of SR 3.1.4.4, the part-length control rods may be exercised under the controlled conditions of SR 3.1.4.4.

Positioning of an individual control rod within its group is addressed by LCO 3.1.4, "Control Rod Alignment."

be moved however, if a part-length rod is moved below the limit of the associated LCO, the Required Action of Condition A must be taken.

NRC REQUEST:

3.1-04 ITS 3.1.4 [STS 3.1.5] Control Rod Alignment
ITS 3.1.4 Required Action D
Bases LCO section (page B 3.1-26) and Bases ACTIONS (page B 3.1-30)
DOC M.6
JFD 10 and JFD 17

The ITS has added a Required Action D that an immovable but trippable control rod shall be returned to operable status prior to entering Mode 2.

Comment #2: The definition for operable control rod is at variance with the STS definition. In the STS control rod operability is equated with trippability, not movability. In the ITS the control rods must be trippable and movable to be operable; for plants converting to the STS this is a plant unique definition, why? Recommend deleting Required Action D.

Consumers Energy Response:

While it is acknowledged the ISTS equates control rod Operability with trippability and not movability, ITS 3.1.4 has retained the CTS requirement that an immovable control rod is inoperable. This was done, in part, to preserve the operational flexibility in the CTS which precludes a forced plant shutdown in the event a single control rod becomes inoperable (immovable). For example; in the ISTS an immovable (but trippable) control rod is considered Operable. Correspondingly, the Bases for ISTS SR 3.1.5.5 explains that an immovable control rod is considered Operable if discovery is made between required performances of SR 3.1.5.5 (an SR 3.0.1 exemption). However, at the time the control rod fails to meet the acceptance criteria for the freedom of movement test (ISTS SR 3.1.5.5), the control rod is declared inoperable and a shutdown to Mode 3 is required. In comparison, an immovable control rod in the ITS is declared inoperable and entry is made into the appropriate Required Actions which allow continuous operation. Since surveillances do not have to be performed on inoperable equipment (SR 3.0.1), restoration of the inoperable control rod is not required until the plant enters Mode 2 from Mode 3.

Affected Submittal Pages:

See NRC Request number 3.1-04 comment #1 for Affected Submittal Pages.

***NRC REQUEST*:**

3.1-04 ITS 3.1.4 [STS 3.1.5] Control Rod Alignment
ITS 3.1.4 Required Action D
Bases LCO section (page B 3.1-26) and Bases ACTIONS (page B 3.1-30)
DOC M.6
JFD 10 and JFD 17

The ITS has added a Required Action D that an immovable but trippable control rod shall be returned to operable status prior to entering Mode 2.

Comment #3: The only element to Part Length Control Rod Operability is that they be fully withdrawn; they do not need to be either Trippable or Moveable.

***Consumers Energy Response*:**

Agree. Since the only element to Part Length control rod Operability is that they must be fully withdrawn, the ITS and ITS Bases have been modified as appropriate to identify the Conditions, Required Actions, and Surveillance Requirements that apply specifically to the full length control rods. This change should help clarify the Operability requirements associated with the Part Length control rods.

The change in wording, between CTS "control rod" and (revised) ITS "full length control rod," was necessitated by the ITS omission of the CTS definition of "Control Rod" which states "CONTROL RODS shall be all full-length shutdown and regulating rods." The words "shutdown and regulating" need not be retained, because there are no other full length control rod types in the Palisades design.

***Affected Submittal Pages*:**

See NRC Request number 3.1-04 comment #1 for Affected Submittal Pages.

NRC REQUEST:

3.1-05 ITS 3.1.4 [STS 3.1.5] Control Rod Alignment
ITS 3.1.4 Required Actions A and B, Completion Times
ITS SR 3.1.4.1 and SR 3.1.4.2
DOC A.4, DOC M.3 and JFD 19

The ITS adds new Required Actions to perform a rod position verification (SR 3.1.4.1) 15 minutes after control rod movement when either a channel of rod position indication is inoperable or when the rod position deviation alarm is inoperable.

Comment #1: The completion times should include a 15 minute requirement for when the inoperability is first discovered (i.e., "15 minutes AND Once within...").

Consumers Energy Response:

An initial performance of rod position verification (ITS SR 3.1.4.1) upon discovery that one channel of rod position indication is inoperable is not warranted based on the following: 1) Operability of the remaining indication channel, 2) knowledge of rod position prior to the loss of the indication channel, and 3) the routine performance of rod position verification every 12 hours. The proposed Completion Time is conservatively appropriate since failure of one of the two rod position channels simply represents a loss of redundancy. In addition, since rod motion is performed manually (i.e., automatic rod control is not used), the remaining indication channel is verified to function as expected each time the affected control rods are moved.

Affected Submittal Pages:

None

***NRC REQUEST*:**

3.1-05 ITS 3.1.4 [STS 3.1.5] Control Rod Alignment
ITS 3.1.4 Required Actions A and B, Completion Times
ITS SR 3.1.4.1 and SR 3.1.4.2
DOC A.4, DOC M.3 and JFD 19

The ITS adds new Required Actions to perform a rod position verification (SR 3.1.4.1) 15 minutes after control rod movement when either a channel of rod position indication is inoperable or when the rod position deviation alarm is inoperable.

Comment #2: Discuss how a rod position verification and a channel check differ.

***Consumers Energy Response*:**

A rod position verification is a verification that the control rods are positioned and aligned as assumed in the safety analysis. For Palisades, this means that each control rod is aligned within 8 inches of all other control rods in its group. Verification of rod position can be obtained from either the primary or secondary rod position indicating channels.

A Channel Check is a assessment of channel behavior and is generally achieved by comparing the output of the primary rod position indication instruments to the output of the secondary rod position indication instruments. Thus, a Channel Check assures the instrumentation used to monitor control rod position is functioning properly.

***Affected Submittal Pages*:**

None

NRC REQUEST:

3.1-06 ITS 3.1.5 [STS 3.1.6] Shutdown and Part Length Rod Group Insertion Limits
ITS 3.1.5 Applicability
JFD 6 and DOC A.6

The ITS applicability differs from both the STS and CTS by equating control rods withdrawn less than 5 inches with fully inserted control rods.

Comment: During startup, are the Regulating Control Rods "bumped" off the bottom < 5" before the Shutdown and Part Length Control Rods are fully withdrawn?

Consumers Energy Response:

During a plant startup the regulating rods may be <5 inches from the bottom of their travel before the shutdown and part length rods are fully withdrawn. This could result from "bumping" the regulating rods prior to an initial startup after a refueling outage or following a reactor trip or, based on the "as left" position of the regulating rods following a mid-cycle shutdown. The control rod drive system is designed with "lower electrical limit switches" which prevent individual control rods from being inserted beyond 3 inches (plus or minus limit switch uncertainties) from the bottom of their mechanical travel. Thus, when the regulating rods are manually inserted using their drive motor (versus from a reactor trip signal), insertion is electrically interrupted approximately 3 inches from the bottom of full (mechanical) rod travel. For an initial startup after a refueling outage or following a reactor trip, the regulating rods may be bumped off their lower mechanical stops (but less the 5 inches) to prevent thermal binding in the control rod drive piston guide tube.

Affected Submittal Pages:

None

NRC REQUEST

3.1-07 ITS 3.1.5 [STS 3.1.6] Shutdown and Part Length Rod Group Insertion Limits
Bases ITS 3.1.5 LCO Section (page B 3.1-36) Insert 2
JFD 8

The ITS 3.1.5 Bases LCO paragraph includes clarifying information provided as Insert 2.

Comment: This information adds clarity and conservatism. Request that a TSTF be provided to incorporate this information into the STS.

Consumers Energy Response:

Upon further review of the information proposed in the Bases of ITS 3.1.5, it was determined that the addition of this information created the potential for a misapplication of the ITS. That is, anytime it is discovered that a control rod can not be moved by its operator the Conditions of ITS 3.1.4 must be entered. Since movement of the shutdown rods is typically limited to the control rod exercise test, the inability to restore a shutdown rod to within the limits of the LCO would be indicative of an inoperable (i.e., immovable) control rod. Initially, ITS 3.1.5 Required Action A.1 allowed 2 hours to restore the shutdown or part-length to within the group limit. This was NOT intended to provide an additional 2 hour delay into the Conditions of ITS 3.1.4 for an inoperable control rod. Furthermore, the Bases stated "declaring a rod which is below its insertion limit, but within 8 inches of all other rods in its group, to be misaligned is acceptable." This statement was ambiguous since the only reason a rod would be misaligned is because it could not be realigned by its motor operator. Therefore, to eliminate potential confusion, ITS 3.1.5 Required Action A.1 has been revised to declare the affected control rod inoperable, and to enter the Conditions and Required Actions of LCO 3.1.4 immediately.

While it is analytically conservative for Palisades to declare a single control rod that is not within its insertion limits inoperable, it is not known whether this interpretation is appropriate for all CE designs. As such, this change has been proposed as a plant specific change.

Affected Submittal Pages:

Att 1 ITS 3.1.5, page 3.1.5-1
Att 2 ITS 3.1.5, page B 3.1.5-3
Att 2 ITS 3.1.5, page B 3.1.5-5
Att 3 CTS, page 3-53 (ITS 3.1.5, page 1 of 3)
Att 3 DOC 3.1.5, page 3 of 5
Att 3 DOC 3.1.5, page 4 of 5
Att 5 NUREG, page 3.1-13
Att 5 NUREG, page B 3.1-36 Insert
Att 5 NUREG, page B 3.1-37
Att 5 NUREG, page B 3.1-37 Insert
Att 6 JFD 3.1.6, page 4 of 4

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Shutdown and Part-Length Control Rod Group Insertion Limits

ed

X

LCO 3.1.5 All shutdown and part-length rod groups shall be withdrawn to ≥ 128 inches.

APPLICABILITY: MODE 1,
MODE 2 with any regulating rod withdrawn above 5 inches.

-----NOTE-----
This LCO is not applicable while performing SR 3.1.4.4 (rod exercise test).

ACTIONS

RAI 3.1-07

X X X

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more shutdown or part-length rods ~~groups~~ not within limit.	A.1 ~~Restore shutdown and part-length rod groups to within limit.~~	Immediately ~~2 hours~~ Declare affected Control rod(s) inoperable and enter the applicable Conditions and Required Actions of LCO 3.1.4.
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.5.1 Verify each shutdown and part-length rod group is withdrawn ≥ 128 inches.	12 hours

BASES

APPLICABLE SAFETY ANALYSES (continued)

The acceptance criteria for addressing shutdown rods as well as regulating rod insertion limits and inoperability or misalignment are that:

a. There be no violation of:

   1. Specified acceptable fuel design limits, or

   2. Primary Coolant System pressure boundary damage; and

b. The core remains subcritical after accident transients.

cd

As such, the shutdown and part-length rod group insertion limits affect safety analyses involving core reactivity, control ejected rod worth, and SDM (Ref. 2). The part-length rods have the potential to cause power distribution envelopes to be exceeded if inserted while the reactor is critical. Therefore, they must remain withdrawn in accordance with the limits of the LCO (Ref. 3). X

The shutdown and part-length rod group insertion limits satisfy Criterion 2 of 10 CFR 50.36(c)(2).

LCO

RAI 3.1-07

Continued from next pg.

The shutdown and part-length rod groups must be within their insertion limits any time the reactor is critical or approaching criticality. For a control rod group to be considered above its insertion limit, all rods in that group ~~(other than misaligned rods addressed by LCO 3.1.4, "Control Rod Alignment")~~ must be above the insertion limit.

~~If only one rod in a group is below the insertion limit, the group may be considered to be above the limit if that rod is considered to be misaligned, and the appropriate condition of LCO 3.1.4 is entered. Since LCO 3.1.4 would not allow continued operation with more than one rod misaligned, declaring a rod which is below its group's insertion limit, but within 8 inches of all other rods in its group, to be misaligned is acceptable. This action may only be taken if all other control rods are properly aligned.~~

BASES

ACTIONS

A.1

Prior to entering this condition, the shutdown and part-length rod groups were fully withdrawn. If a shutdown rod group is then inserted into the core, its potential negative reactivity is added to the core as it is inserted.

RAI 3.1-07

X

INSERT 1

~~If the shutdown or part-length rod groups are not within limits, then 2 hours is allowed for restoring the rod groups to within limits. The 2 hour total Completion Time allows the operator adequate time to adjust the rod groups in an orderly manner and is consistent with the required Completion Times in LCO 3.1.4.~~

B.1

When Required Action A.1 cannot be met or completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown and part-length rod groups are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown rods will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. Verification that the part-length rod groups are within their insertion limits ensures that they do not adversely affect power distribution requirements. This SR and Frequency ensure that the shutdown and part-length rod groups are withdrawn before the regulating rods are withdrawn during a plant startup.

SECTION 3.1

INSERT 1

RAI 3.1-07

If one or more shutdown or part-length rods are not within limits, the affected rod(s) must be declared inoperable and the applicable Conditions and Required Actions of LCO 3.1.4 entered immediately. This Required Action is based on the recognition that the shutdown and part-length rods are normally withdrawn beyond their insertion limits and are capable of being moved by their control rod drive mechanism. Although the requirements of this LCO are not applicable during performance of the control rod exercise test, the inability to restore a control rod to within the limits of the LCO following rod exercising would be indicative of a problem affecting the OPERABILITY of the control rod. Therefore, entering the applicable Conditions and Required Actions of LCO 3.1.4 is appropriate since they provide the applicable compensatory measures commensurate with the inoperability of the control rod.

3.~~10~~ 1 ~~CONTROL ROD AND POWER DISTRIBUTION LIMITS~~ Reactivity Control | A.1

~~3.10.6~~ Shutdown Rod Limits and Part Length Group Insertion

Applicability/LCO ~~a.~~ All shutdown rods shall be withdrawn before any regulating rods are withdrawn above 5 inches A.6 | A.7

~~b. The shutdown rods shall not be withdrawn until normal water level is established in the pressurizer.~~ LA.1

Applicability/LCO ~~c.~~ The shutdown rods shall not be inserted below their exercise limit until all regulating rods are inserted. | A.7

~~3.10.7 Low Power Physics Testing~~ A.1

APPLIC Note

Sections 3.10.1.a, 3.10.1.b, 3.10.3, 3.10.4.b, 3.10.5 and 3.10.6 may be deviated from during ~~low power physics testing and CPDM~~ exercises if necessary to perform a test but only for the time necessary to perform the test. A.8 (See 3.1.1, 3.1.4, 3.1.6) (See 3.1.7)

~~**Basis**~~

~~Sufficient CONTROL RODS shall be withdrawn at all times to assure that the reactivity decrease from a reactor trip provides adequate shutdown margin. The available worth of withdrawn rods must include the reactivity defect of power and the failure of the withdrawn rod of highest worth to insert. The requirement for a shutdown margin of 2.0% in reactivity with 4-pump operation, and of 3.75% in reactivity with less than 4-pump operation, is consistent with the assumptions used in the analysis of accident conditions (including steam line break) as reported in Reference 1 and additional analysis. Requiring the boron concentration to be at cold shutdown boron concentration at less than hot shutdown assures adequate shutdown margin exists to ensure a return to power does not occur if an unanticipated cooldown accident occurs. This requirement applies to normal operating situations and not during emergency conditions where it is necessary to perform operations to mitigate the consequences of an accident. By imposing a minimum shutdown cooling pump flow rate of 2810 gpm, sufficient time is provided for the operator to terminate a boron dilution under asymmetric conditions. For operation with no primary coolant pumps operating and a recirculating flow rate less than 2810 gpm the increased shutdown margin and controls on charging pump operability or alternately the surveillance of the charging pumps will ensure that the acceptance criteria, for an inadvertent boron dilution event will not be violated.[1] The change in insertion limit with reactor power insures that the shutdown requirements for 4-pump operation is met at all power levels. The 2.5-second drop time specified for the CONTROL RODS is the drop time used in the transient analysis.[1]~~ A.2

RAI 3.1-07

< ADD RA A.1 & B.1 > ~~M.1~~ A.9

X

Amendment No. ~~31~~, ~~54~~, ~~57~~, ~~68~~, ~~118~~, ~~137~~, ~~162~~, 169
July 26, 1995

A.7 CTS 3.10.6a states "All shutdown rods shall be withdrawn before any regulating rods are withdrawn." CTS 3.10.6c states "The shutdown rods shall not be inserted below their exercise limit until all regulating rods are inserted." The proposed ITS 3.1.5 LCO states "All shutdown and part/length rod groups shall be withdrawn to ≥ 128 inches." The Applicability for LCO 3.1.5 is MODE 1, MODE 2 with any regulating rod withdrawn above 5 inches. The proposed ITS wording for the LCO and Applicability is equivalent to the CTS wording in 3.10.6b. In the ITS, the shutdown rods must be withdrawn ≥ 128 inches by the LCO before the regulating rods are withdrawn above 5 inches (see DOC A.6 for discussion on 5 inches criteria). In addition, the CTS 3.10.6c requirement that the shutdown rods cannot be inserted below their exercise limit is also maintained in the ITS. This is because the shutdown rods cannot be inserted, except for rod exercising allowed by Applicability note, until out of the MODE of Applicability which required the regulating rods to be ≤ 5 inches withdrawn. Therefore, the CTS and the proposed ITS are equivalent.

A.8 CTS 3.10.7 includes an exception which allows a deviation from the requirement for shutdown rod limits during performance of CRDM exercises. The exception contains a qualifying statement which reads "if necessary to perform a test but only for the time necessary to perform the test." The Applicability Note for proposed ITS 3.1.5 which also provides an exception from the requirement for shutdown rod limits during performance of CRDM exercise does not contain this same qualifier since these type details are governed by the usage rules for the ITS. Therefore, deletion of this information is considered administrative in nature. This change is consistent with NUREG-1432.

A.9 New See next page (formerly DOC M.1) RAI 3.1-07 move to Pg 3 of 5

RAI 3.1-07

MORE RESTRICTIVE CHANGES (M)

There were no "More Restrictive" changes associated with this Specification.

A.9 M.1 CTS 3.10.3 and CTS 3.10.6 stipulate the requirement for rod position on an individual rod basis (i.e., all shutdown and part-length rod must be fully withdrawn). In addition, CTS 3.4.10.4a requires that a control rod must be aligned within 8 inches from the remainder of the bank. The CTS does not specify rod positions on a group basis, and does not contain actions when controls rods are misaligned from their groups by less than 8 inches. Proposed ITS 3.1.5 establishes insertion limits for the shutdown and part-length rod groups by requiring them to be withdrawn ≥ 128 inches. Required Action A.1 of ITS 3.1.5 requires that any shutdown or part-length rod group that is not within its group insertion limit be declared inoperable and the Conditions of ITS 3.1.4 entered immediately ~~restored to within limits within 2 hours.~~ If the Required Action and associated Completion Time are not met, Required Action B.1 requires the plant to be in Mode ~~4~~ 3 within 6 hours. To ensure compliance with the requirements of LCO 3.1.5, for a control rod group to be considered above its insertion limit, all rods in that group ~~(other than misaligned rods addressed by LCO 3.1.4, "Control Rod Alignment")~~ must be above the insertion limit. ~~If only one rod in a group is below the insertion limit, the group may be considered to be above the limit if that rod is considered to be misaligned, and the appropriate condition of LCO 3.1.4 is entered. Since LCO 3.1.4 would not allow continued operation with more than one rod misaligned, declaring a rod which is below its group's insertion limit, but within 8 inches of all other rods in its group, to be misaligned is acceptable. The Required Actions of the ITS are more restrictive than the CTS since the ITS limits the number of control rods with can be misaligned from their group by less than 8 inches to only one rod. Therefore, the~~ addition of ITS Required Actions A.1 and B.1 is characterized as ~~a more restrictive~~ an administrative change, since the action taken when a shutdown or part-length rod exceed its insertion limit is consistent with the CTS actions for an inoperable control rod.

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

LA.1 CTS 3.10.6b states "The shutdown rods shall not be withdrawn until normal water level is established in the pressurizer." This requirement was included to help assure an inadvertent criticality will not occur with the PCS water solid. This statement is more appropriate for being addressed in plant procedures and is not included in the proposed ITS. Changes to plant procedures are made in accordance with the plant procedure change process. This change maintains consistency with NUREG-1432.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.~~8~~ 5 Shutdown ~~Control Element Assembly (CEA)~~ Control and Part Length Rod Group 12 Insertion Limits ~~(Analog)~~ 13

LCO 3.1.~~8~~ 5 All shutdown ~~CEAs~~ 12 and part-length rod groups shall be withdrawn to ≥ ~~[129]~~ 128 1 inches.

APPLICABILITY: MODE 1,
MODE 2 with any regulating ~~CEA not fully inserted~~ rod withdrawn above 5 inches 6

-----NOTE-----
This LCO is not applicable while performing SR 3.1.~~8~~ 4.~~6~~ 4 (rod exercise test). 2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more shutdown ~~CEAs~~ 12 or part-length rod groups not within limit.	~~A.1.1 Verify SDM ≥ [4.5]% Δk/k.~~ ~~OR~~ ~~A.1.2 Initiate boration to restore SDM to within limit.~~ ~~AND~~	~~1 hour~~ ~~1 hour~~
TSTF-67 16	A.~~2~~ 1 Restore shutdown ~~CEA(s)~~ and part-length rod groups 12 to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
17	Declare affected control rod(s) inoperable and enter the applicable Conditions and Required Actions of LCO 3.1.4.	Immediately

RAI 3.1-07

SECTION 3.1

INSERT 1

The part-length rods have the potential to cause power distribution envelopes to be exceeded if inserted while the reactor is critical. Therefore, they must remain withdrawn in accordance with the limits of the LCO (Ref. 3).

INSERT 2

RAI 3.1-07

For a control rod group to be considered above its insertion limit, all rods in that group ~~(other than misaligned rods addressed by LCO 3.1.4, "Control Rod Alignment")~~ must be above the insertion limit. ~~If only one rod in a group is below the insertion limit, the group may be considered to be above the limit if that rod is considered to be misaligned, and the appropriate condition of LCO 3.1.4 is entered. Since LCO 3.1.4 would not allow continued operation with more than one rod misaligned, declaring a rod which is below its group's insertion limit, but within 8 inches of all other rods in its group, to be misaligned is acceptable. This action may only be taken if all other control rods are properly aligned.~~

Maintaining the shutdown rod groups within their insertion limits...

INSERT 3

Maintaining the part length rod group within its insertion limit ensures that the power distribution envelope is maintained.

INSERT 4

In MODE 2, the Applicability begins anytime any regulating rod is withdrawn above 5 inches.

INSERT 5

...to at least the lower electrical limit, and contribute to the SDM. In MODE 3, the shutdown rod groups may be withdrawn in preparation for a reactor startup.

BASES

ACTIONS (16)

A.1 ~~.1, A.1.2, and A.2~~ (continued)

TSTF-67 (16)

core as it is inserted. ~~If boron concentration is not changed at this time, SDM should not change. This, however, is verified within 1 hour, or boration is initiated to bring the SDM to within limit, if the CEA(s) is not restored to within limits prior to this time.~~

RAI 3.1-07

Shutdown or Part-length rod group

~~If the CEA(s) is not restored to within limits within 1 hour and the SDM is within limit, then an additional 1 hours is allowed for restoring the CEA(s) to within limits. The 2 hour total Completion Time allows the operator adequate time to adjust the CEA(s) in an orderly manner and is consistent with the required Completion Times in LCO 3.1.5 4. "Control Element Assembly (CEA) Alignment."~~ (12) (4)

INSERT 1

B.1

When Required Action A.1 ~~or A.2~~ cannot be met or completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

and part length rod groups

Verification that the shutdown CEAs are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown CEAs rods will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown CEAs are withdrawn before the regulating CEAs rods are withdrawn during a unit plant startup. (12) (4)

(12) INSERT 2

Con the and part length rod group

ed X

rod group

~~Since the shutdown CEAs are positioned manually by the control room operator, verification of shutdown CEA position at a Frequency of 12 hours is adequate to ensure that the shutdown CEAs are within their insertion limits. Also, the 12 hour Frequency takes into account other information available to the operator in the control room for the purpose of monitoring the status of the shutdown CEAs.~~ X (12)

(continued)

SECTION 3.1

RAI 3.1-07

INSERT 1

If one or more shutdown or part-length rods are not within limits, the affected rod(s) must be declared inoperable and the applicable Conditions and Required Actions of LCO 3.1.4 entered immediately. This Required Action is based on the recognition that the shutdown and part-length rods are normally withdrawn beyond their insertion limits and are capable of being moved by their control rod drive mechanism. Although the requirements of this LCO are not applicable during performance of the control rod exercise test, the inability to restore a control rod to within the limits of the LCO following rod exercising would be indicative of a problem affecting the OPERABILITY of the control rod. Therefore, entering the applicable Conditions and Required Actions of LCO 3.1.4 is appropriate since they provide the applicable compensatory measures commensurate with the inoperability of the control rod.

INSERT 2

Verification that the part-length rod groups are within their insertion limits ensures that they do not adversely affect power distribution requirements.

Change	Discussion

14. The NUREG-1432 Bases in the Applicability section states "In MODE 3, 4, 5, or 6, the shutdown CEAs are fully inserted in the core and contribute to the SDM. In the proposed ITS, MODE 3 was deleted from this sentence and another sentence added to state "In MODE 3, the shutdown rod groups are not always fully inserted. In addition, the term "fully inserted" is changed in the proposed ITS to state "to at least the lower electrical limit." This change is made to remove confusion with respect to what constitutes "full inserted." For the Palisades control rod design, the lower electrical limit corresponds to the point where electrical rod insertion ceases, and is about 3 inches from the bottom of full rod travel. The reactivity level in this region is negligible. These changes are plant specific changes to provide clarification of the requirements for shutdown rod groups.

15. To reflect the incorporation of TSTF-136 which consolidates ISTS 3.1.1 and ISTS 3.1.2, the specification number for ISTS 3.1.6, "Shutdown CEA Insertion Limits," has been changed to ITS 3.1.5 and conforming changes have been made to the Bases. These changes are consistent with NUREG-1432 as modified by TSTF-136.

16. The definition of Shutdown Margin was revised in NUREG-1432 to clarify that changes in fuel and moderator temperature are included in the determination of the Control Element Assembly Power Dependent Insertion Limits which are used to ensure adequate Shutdown Margin in MODEs 1 and 2. As a result of this change, ISTS 3.1.6 Required Action A.1.1 (verify SDM) and Required Action A.1.2 (initiate boration) have been deleted since they are no longer necessary to ensure adequate Shutdown Margin. Therefore, these Required Actions and associated Bases discussions are not included in proposed ITS 3.1.5. This change is consistent with NUREG-1432 as modified by TSTF-67.

RAI 3.1-07

17. New - See INSERT

INSERT

17. ISTS 3.1.6 Required Action A.1 (as modified by TSTF-67) allows 2 hours to restore out-of-limit shutdown rods to within the limit of the LCO. Proposed ITS 3.1.5 Required Action A.1 requires out-of-limit shutdown (and part-length) rods to be declared inoperable and the Conditions and Required Actions of ITS 3.1.4 entered immediately. Anytime it is discovered that a control rod can not be moved by its operator the control rod must be considered inoperable. Since movement of the shutdown rods is typically limited to the control rod exercise test, the inability to restore a shutdown rod to within the limits of the LCO would be indicative of an inoperable (i.e., immovable) control rod. Therefore, the Required Actions for a shutdown rod outside its specified limit has been changed to be consistent with the Required Actions for an inoperable control rod.

NRC REQUEST

3.1-08 ITS 3.1.6 [STS 3.1.7] Regulating Rod Group Position Limits
ITS 3.1.6 Required Action B, Completion Time
ITS SR 3.1.6.1
DOC A.4 and JFD 5

The ITS adds a new Required Action to perform a rod position verification (SR 3.1.6.1) 15 minutes after control rod movement when either the PDIL Alarm Circuit or the CROOS Alarm Circuit are inoperable.

Comment: The completion times should include a 15 minute requirement for when the inoperability is first discovered (i.e., "15 minutes AND Once within...").

Consumers Energy Response:

An initial performance of group position verification (ITS SR 3.1.6.1) upon discovery that the PDIL or CROOS alarm circuit is inoperable is not warranted based on the following; 1) violation of the power dependent insertion limit or the mis-sequence control rod groups can only occurs as a result of control rod movement, 2) knowledge of rod group position prior to the loss of the indication channel, and 3) the routine performance of rod group position verification every 12 hours. The proposed Completion Time is appropriate since rod positioning is performed manually (i.e., automatic rod control is not used), and verification of rod group position is performed within 15 minutes following rod motion. This Completion Time is also consistent with the CTS.

Affected Submittal Pages:

None

NRC REQUEST

3.1-09 ITS 3.1.6 [STS 3.1.7] Regulating Rod Group Position Limits
ITS 3.1.6 LCO and Required Action B
ITS SR 3.1.6.1
DOC M.1 and JFD 10

The ITS includes explicit sequence and overlap requirements in the LCO, Required Actions and in SR 3.1.6.1.

Comment: This information adds clarity and conservatism. Request that a TSTF be provided to incorporate this information into the STS.

Consumers Energy Response:

Neither NUREG-1432 (ISTS for CE Plants) nor NUREG-0212 (STS for CE Plants) contain a requirement for control rod group "overlap." During the development of NUREG-1432 the subject of an overlap requirement was discussed with the participating CE plants. At that time it was felt that an overlap requirement was not needed.

Palisades will propose a generic change to NUREG-1432 at the next meeting of the CE Owners Group Licensing Subcommittee to include an explicit rod group overlap requirement in the LCO for ISTS 3.1.7.

Affected Submittal Pages:

None

NRC REQUEST

3.1-10 ITS 3.1.6 [STS 3.1.7] Regulating Rod Group Position Limits
Bases ITS 3.1.6 LCO Section (page B 3.1-42) Insert 2
JFD 13

The ITS 3.1.6 Bases LCO paragraph includes clarifying information provided as Insert 2.

Comment: This information adds clarity and conservatism. Request that a TSTF be provided to incorporate this information into the STS.

Consumers Energy Response:

Consistent with the response to NRC Comment 3.1-07, the Bases of ITS 3.1.6 has been revised to eliminate information that was found to be ambiguous. The revised Bases still clarifies that all rods in a given group must be above the insertion limits in order for the group to be considered within its insertion limits. Unlike the shutdown rods discussed in ITS 3.1.5, the regulating rods are moved as a group in response to changing plant conditions. As such, violation of the insertion limits on a group basis is possible. Thus, maintaining a 2 hour restoration period (consistent with the CTS and ISTS) is appropriate.

While it is analytically conservative for Palisades to declare a single control rod that is not within its insertion limits inoperable, it is not know whether this interpretation is appropriate for all CE designs. As such, this change has been proposed as a plant specific change.

Affected Submittal Pages:

Att 2 ITS 3.1.6, page B 3.1.6-5
Att 5 NUREG, page B 3.1-42 Insert
Att 6 JFD 3.1.7, page 4 of 5

BASES

APPLICABLE SAFETY ANALYSES (continued)

Operation at the insertion limits or ASI limits may approach the maximum allowable linear heat generation rate or peaking factor, with the allowed T_q present. Operation at the insertion limit may also indicate the maximum ejected rod worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected rod worth.

The regulating and shutdown rod insertion limits ensure that safety analyses assumptions for reactivity insertion rate, SDM, ejected rod worth, and power distribution peaking factors are preserved.

The regulating rod group position limits satisfy Criterion 2 of 10 CFR 50.36(c)(2).

LCO

The limits on regulating rod group sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is maintained, and ensuring adequate negative reactivity insertion on trip. The overlap between regulating rod groups provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during regulating rod group motion.

For a control rod group to be considered above its insertion limit, all rods in that group ~~(other than misaligned rods addressed by LCO 3.1.4, "Control Rod Alignment")~~ must be above the insertion limit. ~~If only one rod in a group is below the insertion limit, the group may be considered to be above the limit if that rod is considered to be misaligned, and the appropriate condition of LCO 3.1.4 is entered. Since LCO 3.1.4 would not allow continued operation with more than one rod misaligned, declaring a rod which is below its group's insertion limit, but within 8 inches of all other rods in its group, to be misaligned is acceptable. This action may only be taken if all other control rods are properly aligned.~~

RAI 3.1.10

SECTION 3.1

INSERT 1

The most limiting SDM requirements for Mode 1 and 2 conditions at (Beginning of Cycle (BOC) are determined by the requirements of several transients, e.g., Loss of Flow, etc. However, the most limiting SDM requirements for Modes 1 and 2 at End of Cycle (EOC) come from just one transient, Main Steam Line Break (MSLB). The requirements of the MSLB event at EOC for the full power and no load conditions are significantly larger than those of any other event at that time in cycle and, also, considerably larger than the most limiting requirements at BOC.

Although the most limiting SDM requirements at EOC are much larger than those at BOC, the available SDMs obtained via tripping the control rods are substantially larger due to the much lower boron concentration at EOC. To verify that adequate SDMs are available throughout the cycle to satisfy the changing requirements, calculations are performed at both BOC and EOC. It has been determined that calculations at these two times in cycle are sufficient since the difference between available SDMs and the limiting SDM requirements are the smallest at these times in cycle. The measurement of control rod bank~~s~~ worth performed as part of the Startup Testing Program demonstrates that the core has the expected shutdown capability. Consequently, adherence to LCO 3.1.5, "Shutdown and Part-Length Rod Group Insertion Limits," and LCO 3.1.6 provides assurance that the available SD~~s~~ M at any time in cycle will exceed the limiting SDM requirements at that time in cycle.

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ed
m

RAI 3.1-10

INSERT 2

For a control rod group to be considered above its insertion limit, all rods in that group ~~(other than misaligned rods addressed by LCO 3.1.4, "Control Rod Alignment")~~ must be above the insertion limit. If only one rod in a group is below the insertion limit, the group may be considered to be above the limit if that rod is considered to be misaligned, and the appropriate condition of LCO 3.1.4 is entered. Since LCO 3.1.4 would not allow continued operation with more than one rod misaligned, declaring a rod which is below its group's insertion limit, but within 8 inches of all other rods in its group, to be misaligned is acceptable. This action may only be taken if all other control rods are properly aligned.

X

Change	**Discussion**

12. The Palisades Nuclear Plant analysis does not model separate insertion limits for transient and steady state conditions as specified in Conditions A, B and C of NUREG-1432. The Palisades Nuclear Plant PDIL limits specify the regulating rod group position limits which account for anticipated power maneuvers and transient mitigation. Therefore, the proposed Palisades ITS removes the steady state and transient insertion limit discussion, where appropriate, and provides a discussion of the Palisades Nuclear Plant insertion limits. This is a plant specific change to reflect the Palisades CTS and analysis.

RAI 3.1-10

13. A discussion has been added in the Bases under the LCO section to clarify that ~~if any individual regulating rod does not meet the alignment requirements of LCO 3.1.4, "Control Rod Alignment," then LCO 3.1.4 may be entered as long as the remainder of the group is above its insertion limits. This discussion was added to help avoid confusion since LCO 3.1.6 is written to address regulating rods on a group basis and LCO 3.1.4 addresses individual rod misalignments.~~ This is a plant specific change to reflect the Palisades control rod design and CTS requirements.

14. To reflect the incorporation of TSTF-136 which consolidates ISTS 3.1.1 and ISTS 3.1.2, the specification number for ISTS 3.1.7, "Shutdown CEA Insertion Limits," has been changed to ITS 3.1.6 and conforming changes have been made to the Bases. These changes are consistent with NUREG-1432 as modified by TSTF-136.

15. The definition of Shutdown Margin was revised in NUREG-1432 to clarify that changes in fuel and moderator temperature are included in the determination of the Control Element Assembly Power Dependent Insertion Limits which are used to ensure adequate Shutdown Margin in MODES 1 and 2. As a result of this change, ISTS 3.1.7 Required Action A.1.1 (verify SDM) and Required Action A.1.2 (initiate boration) have been deleted since they are no longer necessary to ensure adequate Shutdown Margin. Therefore, these Required Actions and associated Bases discussions are not included in proposed ITS 3.1.6. An expanded discussion has been incorporated in the Applicable Safety Analyses portion of the Bases to clarify the requirements for SDM as it applies to control rod position. These change are consistent with NUREG-1432 as modified by TSTF-67.

for a control rod group to be considered above its insertion limit, all rods in that group must be above the insertion limit.

NRC REQUEST

3.1-11 ITS 3.1.7 [STS 3.1.9] Special Test Exception
ITS 3.1.7 LCO Requirements
JFD 17

The ITS changes the STS SDM requirement to "$\geq$ 1% shutdown reactivity...."

Comment: What is the value of "1% shutdown reactivity" based upon?

Consumers Energy Response:

The value of "1% shutdown reactivity" is based on engineering judgement and is intended to provide adequate negative reactivity to shut down and maintain the reactor subcritical during Physics Testing and includes margin for calculational uncertainties.

Affected Submittal Pages:

None

NRC REQUEST

3.1-12 ITS 3.1.7 [STS 3.1.9] Special Test Exception
ITS 3.1.7 Required Actions B and D
JFD 14 and JFD 15

The ITS revises the STS Required Actions making them more logical.

Comment: This information adds clarity. Request that a TSTF be provided to incorporate this information into the STS.

Consumers Energy Response:

Palisades will propose a generic change to NUREG-1432 at the next meeting of the CE Owners Group Licensing Subcommittee to revise the Required Actions associated with ISTS 3.1.9 to make them more logical.

Affected Submittal Pages:

None

NRC REQUEST:

3.2-01 ITS 3.2.1 Linear Heat Rate (LHR)
ITS 3.2.1 LCO
JFD 8

The ITS 3.2.1 LCO adds, "as determined by an OPERABLE Incore Alarm System or by an OPERABLE Excore Monitoring System," which is neither in the CTS nor the STS.

Comment: The wording of the LCO precludes the Condition A option of "OR LHR, as determined by manual incore readings, not within limits...." Suggest that the LCO be reworded to add the straight forward requirement that the Incore Alarm System and the Excore Monitoring System shall both be operable.

Consumers Energy Response:

The CTS allows LHR to be monitored by either the Incore Alarm System, or the Excore Monitoring System. If the Incore Alarm System is inoperable and the Excore Monitoring System is not being used to monitor LHR, operations are allowed to continue provided power is reduced to 85% Rated Power and incore readings are manually recorded. Specifying an Operable Incore Alarm System or Excore Monitoring System in the LCO is necessary to support the structure of the ITS while maintaining the flexibility provided in the CTS. That is, the Incore Alarm System and the Excore Monitoring System must be inoperable (proposed ITS Condition B) before reliance is placed on the manual method of verifying LHR. The third entry in Condition A (LHR, as determined by manual incore detector reading, not within limits specified in the COLR) is necessary to ensure the LHR limits are not violated.

It was correctly identified by the NRC reviewer that the LCO wording was not straight forward. As such, the LCO wording has been revised to clearly require the Incore Alarm System or Excore Monitoring System to be Operable for monitoring LHR. In addition, the term "Incore Monitoring System" has been replaced with the term "Incore Alarm System" throughout Specification 3.2.1 and its associated Bases to eliminate the ambiguity of the LCO requirement. Conforming changes have also been made to the supporting documents.

Affected Submittal Pages

Att 1 ITS 3.2.1, pg 3.2.1-1
Att 1 ITS 3.2.1, pg 3.2.1-2
Att 1 ITS 3.2.1, pg 3.2.1-3
Att 2 ITS 3.2.1, pg B 3.2.1-2
Att 2 ITS 3.2.1, pg B 3.2.1-3
Att 2 ITS 3.2.1, pg B 3.2.1-5
Att 2 ITS 3.2.1, pg B 3.2.1-6
Att 2 ITS 3.2.1, pg B 3.2.1-7
Att 2 ITS 3.2.1, pg B 3.2.1-8
Att 2 ITS 3.2.1, pg B 3.2.1-9
Att 3 DOC 3.2.1, pg 2 of 7
Att 3 DOC 3.2.1, pg 4 of 7
Att 5 NUREG 3.2.1, pg 3.2-1
Att 5 NUREG 3.2.1, pg 3.2-3
Att 5 NUREG 3.2.1, pg B 3.2-4 insert
Att 5 NUREG 3.2.1, pg B 3.2-5 insert
Att 6 JFD 3.2.1, pg 1 of 5
Att 6 JFD 3.2.1, pg 3 of 5
Att 6 JFD 3.2.1, pg 4 of 5

RAI 3.2-01

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Linear Heat Rate (LHR)

LHR shall be within the limits specified in the COLR, and the Incore Alarm System OR Excore Monitoring System shall be OPERABLE to monitor LHR.

LCO 3.2.1 ~~LHR, as determined by an OPERABLE Incore Monitoring System or by an OPERABLE Excore Monitoring System, shall be within the limits specified in the COLR.~~ X

APPLICABILITY: MODE 1 with THERMAL POWER > 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LHR, as determined by the automatic Incore ~~Monitoring~~ Alarm System, not within limits specified in the COLR, as indicated by four or more coincident incore channels. OR LHR, as determined by the Excore Monitoring System, not within limits specified in the COLR. OR LHR, as determined by manual incore detector readings, not within limits specified in the COLR.	A.1 Restore LHR to within limits.	1 hour

X

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Incore Alarm and Excore Monitoring Systems inoperable for monitoring LHR.	B.1 Reduce THERMAL POWER to ≤ 85% RTP. AND B.2 Verify LHR is within limits ~~Determine LHR~~ using manual incore readings.	2 hours 4 hours AND Once per 2 hours thereafter
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER to ≤ 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	-------------------NOTE------------------- Only required when Incore Alarm ~~Monitoring~~ System is being used to monitor LHR. -- Verify LHR is within the limits specified in the COLR.	 12 hours

RAI 3.2-01

X

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
		RAI 3.2-01
SR 3.2.1.2	--------NOTE-------- Only required when Incore ~~Monitoring~~ Alarm System is being used to monitor LHR. ---------------- Adjust incore alarm setpoints based on a measured power distribution.	Prior to operation > 50% RTP after each fuel loading AND 31 EFPD thereafter
SR 3.2.1.3	--------NOTE-------- Only required when Excore Monitoring System is being used to monitor LHR. ---------------- Verify measured ASI has been within 0.05 of target ASI for last 24 hours.	Prior to each initial use of Excore Monitoring System to monitor LHR
SR 3.2.1.4	--------NOTE-------- Only required when Excore Monitoring System is being used to monitor LHR. ---------------- Verify THERMAL POWER is less than the APL.	1 hour

X

BASES

BACKGROUND (continued)

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions.

The limits on LHR, Assembly Radial Peaking Factor (F_r^A), Total Radial Peaking Factor (F_r^T), QUADRANT POWER TILT (T_q), and AXIAL SHAPE INDEX (ASI), which are obtained directly from the core reload analysis, ensure compliance with the safety limits on LHR and Departure from Nucleate Boiling Ratio (DNBR).

RAI 3.2-01

Either of the two core power distribution monitoring systems, the Incore ~~Monitoring~~ System or the Excore Monitoring System, provides adequate monitoring of the core power distribution and is capable of verifying that the LHR is within its limits. The Incore ~~Monitoring~~ System performs this function by continuously monitoring the local power at many points throughout the core and comparing the measurements to predetermined setpoints above which the limit on LHR could be exceeded. The Excore Monitoring System performs this function by providing comparison of the measured core ASI with predetermined ASI limits based on incore measurements. An Excore Monitoring System Allowable Power Level (APL), which may be less than RATED THERMAL POWER, and an additional restriction on T_q, are applied when using the Excore Monitoring System to ensure that the ASI limits adequately restrict the LHR to less than the limiting values.

Alarm

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In conjunction with the use of the Excore Monitoring System for monitoring LHR and in establishing ASI limits, the following assumptions are made:

a. The control rod insertion limits of LCO 3.1.5, "Shutdown and Part-Length Rod Group Insertion Limits," and LCO 3.1.6, "Regulating Rod Group Position Limits," are satisfied;

b. The additional T_q restriction of SR 3.2.1.6 is satisfied; and

c. Radial Peaking Factors, F_r^A and F_r^T, do not exceed the limits of LCO 3.2.2.

BASES

BACKGROUND (continued)

The limitations on the Radial Peaking Factors provided in the COLR ensure that the assumptions used in the analysis for establishing the LHR limits and Limiting Safety System Settings (LSSS) remain valid during operation at the various allowable control rod group insertion limits. RAI 3.2-01

The Incore ~~Monitoring~~ Alarm System continuously provides a direct measure of the LHR and the Radial Peaking factors. It also provides alarms that have been established for the individual incore detector segments, ensuring that the peak LHRs are maintained within the limits specified in the COLR. The setpoints for these alarms include tolerances, set in conservative directions, for: X

a. A measurement calculational uncertainty factor (as identified in the COLR);

b. An engineering uncertainty factor of 1.03; and

c. A THERMAL POWER measurement uncertainty factor of 1.02.

The measurement uncertainties associated with LHR, F_r^A and F_r^T are based on a statistical analysis performed on power distribution benchmarking results. The COLR includes the applicable measurement uncertainties for fresh and depleted incore detector usage. The engineering and THERMAL POWER uncertainties are incorporated in the power distribution calculation performed by the fuel vendor.

The excore power distribution monitoring system consists of Power Range Channels 5 through 8. The power range channels monitor neutron flux from 0 to 125 percent full power. They are arranged symmetrically around the reactor core to provide information on the radial and axial flux distributions.

The power range detector assembly consists of two uncompensated ion chambers for each channel. One detector extends axially along the lower half of the core while the other, which is located directly above it, monitors flux from the upper half of the core. The DC current signal from each of the ion chambers is fed directly to the control room drawer assembly without pre-amplification. Each excore detector supplies data to a Thermal Margin Monitor (TMM). Each TMM uses these excore signals to calculate Axial Shape Index (ASI) on a continuous basis.

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm; and

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d. The full-length control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

X

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. 4). This is accomplished by maintaining the power distribution and primary coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by accident analyses (Ref. 1), with due regard for the correlations between measured quantities, the power distribution, and uncertainties in determining the power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate so that the peak cladding temperature does not exceed 2200°F (Ref. 4). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and the Primary Coolant System Operation ensure that these criteria are met as long as the core is operated within the LHR, ASI, F_r^A, F_r^T, and T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses.

Fuel cladding damage does not necessarily occur while the plant is operating at conditions outside the limits of these LCOs during normal operation. Fuel cladding damage could result, however, if an accident occurs from initial conditions outside the limits of these LCOs. The potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and can correspondingly increase local LHR. RAI 3.2-01

The Incore ~~Monitoring~~ System provides for monitoring of LHR, radial peaking factors, and QUADRANT POWER TILT to ensure that fuel design conditions and safety analysis assumptions are maintained. The Incore ~~Monitoring~~ System is also utilized to determine the target AXIAL OFFSET (AO) and to determine the Allowable Power Level (APL) when using the excore detectors. X

Alarm

X

BASES

APPLICABLE SAFETY ANALYSES (continued)

The Excore Monitoring System provides for monitoring of ASI and QUADRANT POWER TILT to ensure that fuel design conditions and safety analysis assumptions are maintained.

The LHR satisfies Criterion 2 of 10 CFR 50.36(c)(2).

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The power distribution LCO limits, except T_q, are provided in the COLR. The limitation on the LHR in the peak power fuel rod at the peak power elevation Z ensures that, in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F.

RAI 3.2.01

maintained within the limits specified in the COLR

be OPERABLE to monitor LHR

The LCO requires that LHR be ~~monitored by~~ and either the ~~an OPERABLE~~ Incore ~~Monitoring~~ Alarm System or ~~an OPERABLE~~ Excore Monitoring System. When using the Incore ~~Monitoring~~ System, the LHR is not considered to be out of limits until there are four or more incore detectors simultaneously in alarm. When using the Excore Monitoring System, LHR is considered within limits when the conditions are acceptable for use of the Excore Monitoring System and the associated ASI and T_q limits specified in the SRs are met.

To be considered OPERABLE, the incore Alarm ~~monitoring~~ system must have at least 160 of the 215 possible incore detectors detectors ~~OPERABLE and 2 incores~~ per axial level per core quadrant OPERABLE. ~~For the LHR monitoring (automatic alarming) function of the incore monitoring system to be considered OPERABLE, the required alarm setpoints must be entered into the Plant Computer.~~

In addition, the Plant Process Computer must be OPERABLE and

To be considered OPERABLE, the Excore Monitoring System must have been calibrated with OPERABLE incore detectors, the ASI must not have been out of limits for the last 24 hours, and THERMAL POWER must be less than the APL.

BASES

APPLICABILITY

In MODE 1 with THERMAL POWER > 25% RTP, power distribution must be maintained within the limits assumed in the accident analysis to ensure that fuel damage does not result following an AOO. In MODE 1 with THERMAL POWER ≤ 25% RTP, and in other MODES, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on the core power distribution, and because ample thermal margin exists to ensure that the fuel integrity is not jeopardized and safety analysis assumptions remain valid.

ACTIONS

A.1

RAI 3.2-01

There are three acceptable methods for verifying that LHR is within limits. The LCO requires monitoring by either an OPERABLE Incore Alarm ~~Monitoring~~ System or an OPERABLE Excore Monitoring System. When both of the required systems are inoperable, Condition B allows for monitoring by taking manual readings of the incore detectors. Any of these three methods may indicate that the LHR is not within limits. With the LHR exceeding its limit, excessive fuel damage could occur following an accident. In this Condition, prompt action must be taken to restore the LHR to within the specified limits. One hour to restore the LHR to within its specified limits is reasonable and ensures that the core does not continue to operate in this Condition. The 1 hour Completion Time also allows the operator sufficient time for evaluating core conditions and for initiating proper corrective actions.

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BASES

ACTIONS (continued)

B.1 and B.2

RAI 3.2-01

Alarm

With the Incore ~~Monitoring~~ System inoperable for monitoring LHR and the Excore Monitoring System inoperable for monitoring LHR, THERMAL POWER must be reduced to ≤ 85% RTP within 2 hours. Operation at ≤ 85% RTP ensures that ample thermal margin is maintained. A 2 hour Completion Time is adequate to achieve the required plant condition without challenging plant systems. Additionally, with the Incore Alarm and Excore Monitoring Systems inoperable, LHR must be verified to be within limits within 4 hours, and every 2 hours thereafter by manually collecting incore detector readings at the terminal blocks in the control room utilizing a suitable signal detector. The manual readings shall be taken on a minimum of 10 individual detectors per quadrant (to include a total of 160 detectors in a 10 hour period). The time interval of 2 hours and the minimum of 10 detectors per quadrant are sufficient to maintain adequate surveillance of the power distribution to detect significant changes until the monitoring systems are returned to service.

C.1

If the Required Action and associated Completion Time are not met, THERMAL POWER must be reduced to ≤ 25% RTP. This reduced power level ensures that the core is operating within its thermal limits and places the core in a conservative condition. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reach ≤ 25% RPT from full power MODE 1 conditions in an orderly manner and without challenging plant systems.

RAI 3.2-01

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

The Incore ~~Monitoring~~ System provides continuous monitoring of LHR through the plant computer. The plant computer is used to generate alarm setpoints that are based on measured margin to allowed LHR. As the incore detectors are read by the plant computer, they are continuously compared to the alarm setpoints. If the Incore ~~Monitoring~~ System LHR monitoring function is inoperable, excore detectors or manual recordings of the incore detector readings may be used to monitor LHR. Periodically monitoring LHR ensures that the assumptions made in the Safety Analysis are maintained. This SR is modified by a Note that states that the SR is only applicable when the Incore ~~Monitoring~~ System is being used to monitor LHR. The 12 hour Frequency is consistent with an SR which is to be performed each shift.

Alarm

SR 3.2.1.2

Continuous monitoring of the LHR is provided by the Incore ~~Monitoring~~ System which provides adequate monitoring of the core power distribution and is capable of verifying that the LHR does not exceed its specified limits.

Performance of this SR verifies the Incore ~~Monitoring~~ System can accurately monitor LHR by ensuring the alarm setpoints are based on a measured power distribution. Therefore, they are only applicable when the Incore ~~Monitoring~~ System is being used to determine the LHR.

The alarm setpoints must be initially adjusted following each fuel loading prior to operation above 50% RTP, and periodically adjusted every 31 Effective Full Power Days (EFPD) thereafter. A 31 EFPD Frequency is consistent with the historical testing frequency of the reactor monitoring system. The SR is modified by a Note which allows the SR to be performed only when the Incore ~~Monitoring~~ System is being used to determine LHR.

A.4 CTS 3.23.1 provides actions when the LHR is being monitored by the excore monitoring system but the system is no longer appropriate for monitoring LHR as indicated by an Axial Offset (AO) of more than 0.05 (ACTION 2). The actions include both "discontinue using the excore monitoring system for monitoring LHR" and "follow the procedure in ACTION 3 below.". Inherent in entry into CTS 3.23.1 ACTION 2 is that the normally used Incore ~~Monitoring~~ ALARM System is inoperable. Therefore, this situation is one with both the Incore ~~Monitoring~~ ALARM System and the excore monitoring system inoperable for the purpose of monitoring LHR. This is included as ITS 3.2.1 Condition B. The specific direction to enter this Condition is not included in ITS since this is the normal use and application of the improved STS format. Therefore, this omission is considered an administrative change. RAI 3.2-01 X X

A.5 CTS 3.23.1 provides actions when the LHR is indicated as not within the limits specified in the COLR by four or more coincident incore alarms (ACTION 1), and when the manually recorded incore readings indicate a local power level greater than the alarm setpoints (ACTION 3). However, no specific action is provided in the CTS for when the LHR is not within limits as monitored by the excore monitoring system. The ITS includes a second entry condition for ITS 3.2.1 Condition A specifically for when the LHR is determined to be not within limits using the excore monitoring system, Since the appropriate action is the same regardless of the method used to determine that LHR is not within limits, the addition of a specific Required Action, entry condition for "LHR, as determined by the Excore Monitoring System, not within limits specified in the COLR" is considered an administrative change.

A.6 CTS 3.23.1 ACTION 3 indicates that when the LHR is indicated as not within the limits specified in the COLR by the manually recorded incore readings "the action specified in ACTION 1 above shall be taken." The ITS includes a third entry condition for ITS 3.2.1 Condition A specifically for when the LHR is determined to be not within limits using the manual incore readings, Since these are only different formats to require the same action, the addition of a specific Required Action, entry condition for "LHR, as determined by manual incore readings, not within limits specified in the COLR" is considered an administrative change.

M.2 CTS does not include specific surveillance requirements to verify that LHR remains within limits. Such an SR is included as ITS SR 3.2.1.1. This SR is necessary to provide direct verification that the LCO requirements are met when using the Incore Alarm ~~Monitoring~~ System for monitoring LHR. Consistent with the NUREG, verification that an OPERABLE Incore ~~Monitoring~~ System does not indicate LHR out of limits is sufficient to fulfill this SR. This is an additional restriction on plant operation. RAI 3.2-01

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

LA.1 CTS 3.23.1 contains specific details regarding the requirements for monitoring of the LHR, i.e., "in the peak power fuel rod at the peak power elevation Z." This information is not required to be provided in NUREG LCO 3.2.1. These details describe elements of the LHR which are addressed by the methodology for determining LHR and are not directly a part of the actual requirement, i.e., Limiting Condition for Operation. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in the LCO Bases of ITS 3.2.1 provides adequate assurance that they will be maintained. The Bases are controlled by the Bases Control Program in Chapter 5 of the proposed Technical Specifications. This change is consistent with NUREG-1432.

LA.2 CTS 3.23.1 ACTION 3 contains specific details regarding the requirements for monitoring of LHR by manual readings of the incore detection system when the incore LHR alarm system is inoperable, i.e., "readings shall be taken on a minimum of 10 individual detectors per quadrant (to include a total number of 160 detectors in a 10-hour period)." This information is not provided in NUREG LCO 3.2.1. These details describe elements of the incore detection system requirements which are addressed by the methodology for proper use of the system and are not directly a part of the actual requirement, i.e., Limiting Condition for Operation. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in the Bases of ITS 3.2.1 provides adequate assurance that they will be maintained. The Bases are controlled by the Bases Control Program in Chapter 5 of the proposed Technical Specifications. This change is consistent with NUREG-1432.

RAI 3.2-01

(4) and the Incore Alarm System or Excore Monitoring System shall be OPERABLE to monitor LHR.

LHR ~~(Analog)~~ 3.2.1

3.2 POWER DISTRIBUTION LIMITS

(4) 3.2.1 Linear Heat Rate (LHR) ~~(Analog)~~

(2) 3.23.1 LCO (8) **LCO 3.2.1** LHR, as determined by an OPERABLE Incore Alarm System or by an OPERABLE Excore Monitoring System, shall ~~not exceed~~ be within the limits specified in the COLR. X X

3.23.1 APPL (5) **APPLICABILITY:** MODE 1 with THERMAL POWER > 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
(4) 3.23.1 Act 1 (2) (6) Automatic A. LHR, as determined by the Incore ~~Detector~~ Alarm ~~Monitoring~~ System, ~~exceeds the~~ not within limits ~~of Figure 3.2.1-1 of~~ specified in the COLR, as indicated by four or more coincident incore channels. OR new (4') (2) LHR, as determined by the Excore ~~Detector~~ Monitoring System, ~~exceeds the limits as indicated by the ASI outside the power dependent control limits as~~ not within limits specified in ~~Figure 3.2.1-2 of~~ the COLR. OR LHR, as determined by manual incore readings, not within limits specified in the COLR. (6) (7) 3.23.1 Act 3 (5) INSERT	A.1 Restore LHR to within limits.	1 hour X
3.23.1 Act 2/3 3.23.1 Act 3 (5) ~~B~~ C. Required Action and associated Completion Time not met.	~~B~~ C.1 ~~Be in MODE 2.~~ Reduce THERMAL POWER to ≤ 25% RTP.	~~6~~ 4 hours

SECTION 3.2

INSERT

3.23.1 ACT 2	B. Incore Alarm and Excore Monitoring Systems inoperable for monitoring LHR.	B.1	Reduce THERMAL POWER to ≤ 85% RTP.	2 hours
		<u>AND</u>		
3.23.1 ACT 3		B.2	~~Determine~~ Verify LHR is within limits using manual incore readings.	4 hours <u>AND</u> Once per 2 hours thereafter

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SURVEILLANCE REQUIREMENTS (continued)

RAI 3.2-01

SURVEILLANCE		FREQUENCY
SR 3.2.1.~~3~~ 2	Alarm ---NOTE~~S~~--- ~~1.~~ Only applicable when the Incore ~~Detector Monitoring~~ System is being used to determine LHR. ~~2. Not required to be performed below 20% RTP.~~	Prior to operation >50% RTP after each fuel loading AND 31 ~~days~~ EFPD thereafter
	Adjust ~~Verify~~ incore ~~detector local power density~~ alarm setpoints ~~are less than or equal to the limits specified in the COLR.~~ based on a measured power distribution.	

4.19.1.1

9

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9

INSERT

4.19.1.2.a
4.19.1.2.b
4.19.1.2.c
4.19.1.2.d

X

20-o

SECTION 3.2

INSERT A

The Incore Alarm System provides for monitoring of LHR, radial peaking factors, and QUADRANT POWER TILT to ensure that fuel design conditions and safety analysis assumptions are maintained. The Incore Alarm System is also utilized to determine the target AXIAL OFFSET (AO) and to determine the Allowable Power Level (APL) when using the excore detectors.

The Excore Monitoring System provides for monitoring of ASI and QUADRANT POWER TILT to ensure that fuel design conditions and safety analysis assumptions are maintained. RAI 3.2-01

INSERT B

maintained within the limits specified in the COLR and the

The LCO requires that LHR be ~~monitored by~~ either ~~an OPERABLE~~ Incore Alarm System or ~~an OPERABLE~~ Excore Monitoring System. When using the Incore Alarm System, the LHR is not considered to be out of limits until there are four or more incore detectors simultaneously in alarm. When using the Excore Monitoring System, LHR is considered within limits when the conditions are acceptable for use of the Excore Monitoring System and the associated ASI and T_q limits specified in the SRs are met. X X

be OPERABLE to monitor LHR

To be considered OPERABLE, the Incore ~~monitoring~~ Alarm System must have at least 160 of the 215 possible incore detectors OPERABLE and 2 incore detectors per axial level per core quadrant OPERABLE. For the LHR Monitoring (automatic alarming) function of the incore monitoring system to be considered OPERABLE, the required alarm setpoints must be entered into the Plant computer. X X

To be considered OPERABLE, the Excore Monitoring System must have been calibrated with OPERABLE incore detectors, the ASI must not have been out of limits for the last 24 hours, and THERMAL POWER must be less than the APL.

In addition, the plant process computer must be OPERABLE and

SECTION 3.2

INSERT A

B.1 and B.2.

With the Incore Alarm System inoperable for monitoring LHR and the Excore Monitoring System inoperable for monitoring LHR, THERMAL POWER must be reduced to ≤ 85% RTP within 2 hours. Operation at ≤ 85% RTP ensures that ample thermal margin is maintained. A 2 hour Completion Time is adequate to achieve the required unit condition without challenging plant systems. Additionally, with the Incore Alarm and Excore Monitoring Systems inoperable, LHR must be verified to be within limits within 4 hours, and every 2 hours thereafter by manually collecting incore detector readings at the terminal blocks in the control room utilizing a suitable signal detector. The manual readings shall be taken on a minimum of 10 individual detectors per quadrant (to include a total of 160 detectors in a 10 hour period). The time interval of 2 hours and the minimum of 10 detectors per quadrant are sufficient to maintain adequate surveillance of the power distribution to detect significant changes until the monitoring systems are returned to service.

INSERT B

RAI 3.2-01

The Incore ~~Monitoring~~ Alarm System provides continuous monitoring of LHR through the plant computer. The plant computer is used to generate alarm setpoints that are based on measured margin to allowed LHR. As the incore detectors are read by the plant computer, they are continuously compared to the alarm setpoints. If the Incore Alarm System LHR monitoring function is inoperable, excore detectors or manual recordings of the incore detector readings may be used to monitor LHR. Periodically monitoring LHR ensures that the assumptions made in the Safety Analysis are maintained. This SR is modified by a Note that states that the SR is only applicable when the Incore Alarm System is being used to monitor LHR. X

Change	Discussion
Note:	This attachment provides a brief discussion of the deviations from NUREG-1432 that were made to support the development of the Palisades Nuclear Plant ITS. The Change Numbers correspond to the respective deviation shown on the "NUREG MARKUPS." The first five justifications were used generically throughout the markup of the NUREG. Not all generic justifications are used in each specification.
1.	The brackets have been removed and the proper plant specific information or value has been provided.
2.	Deviations have been made for clarity, grammatical preference, or to establish consistency within the Improved Technical Specifications. These deviations are editorial in nature and do not involve technical changes or changes of intent.
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 specification. These include an ITS 3.2.1 Applicability less restrictive than the NUREG and the addition of an ACTION for determination of LHR using manual readings when both the Incore ~~Monitoring~~ Alarm System and the excore monitoring system are inoperable for determining LHR. With power reduced to below 85% RTP (per ITS 3.2.1, Required Action B.1), the manual readings of the incore monitors provide an adequate indication that LHR is within limits. This is consistent with CTS as approved in Amendment 68. Additionally, the proposed Applicability for ITS 3.2.1 is actually more restrictive than CTS 3.23.1 which is applicable only above 50% RTP. An ITS 3.2.1 Applicability of "MODE 1 > 25% RTP" is consistent with the Applicability for the other Power Distribution Limit specifications, and provides for incore adjustments based on power distribution maps prior to exceeding 25% which is consistent with Quadrant Power Tilt needs for incore adjustments. RAI 3.2-01 X

RAI 3.2-01

Change	Discussion
8.	An addition to the LCO in incorporated which requires that the LHR be determined by an OPERABLE Incore Alarm ~~Monitoring~~ System or by an OPERABLE excore monitoring system. Such an LCO requirement is consistent with the NUREG SR Note which requires that the LHR be determined by either the incore detector monitoring system or the excore detector monitoring system. However, incorporating the requirement into the LCO provides a more direct indication that the LCO is not met when both the incore LHR alarm function and the excore LHR monitoring function are inoperable (which results in entry into ITS Condition B, as discussed in JFD 5). X
9.	The Surveillance Requirements (SRs) for LHR are revised consistent with the current licensing basis. The NUREG SR Note is inappropriate for Palisades Nuclear Plant because manual reading of the incore monitors is also allowed for determining LHR to be within limits. This is corrected by incorporating the SR Note requirements directly into the LCO (see JFD 8) and adding an ACTION for use of the manual incore readings (see JFDs 5 and 7). The NUREG SRs are also inappropriate for all plants since failure of the alarms or setpoints to be properly set does not mean that the LHR is not within limits. However, SR 3.0.1 would require that the LCO be considered not met when any of these SRs are not met . This is not consistent with the format and content intent of the improved STS NUREGs, is considered overly conservative, and is not adopted.
	ITS SR 3.2.1.1 specifically requires the verification that LHR is within the limits specified in the COLR. This SR is a direct verification that the LCO is being met (which is missing from the NUREG). However, since the LHR is normally automatically monitored and alarmed by the incore power distribution monitoring system, the SR is only required to be performed when the Incore Alarm ~~Monitoring~~ System is being used to determine LHR, and is met by administrative verification that the Incore Alarm ~~monitoring~~ System is OPERABLE for monitoring LHR, and that the Incore Alarm ~~monitoring~~ System does not indicate LHR is not within limits. X xxx xxx

Change	Discussion

RAI 3.2-01

9. (continued)

NUREG SR 3.2.1.2 and SR 3.2.1.3 requirements for incore alarms are combined and revised to reflect CTS 4.19.1. ITS SR 3.2.1.2 requires that the incore alarm setpoints be adjusted (i.e., the alarms be set) based on a measured power distribution. This Surveillance provides adequate assurance that the Incore ~~Monitoring~~ Alarm System is providing accurate monitoring of the LHR. This change is consistent with CTS 4.19.1 requirements for adjustments of incore alarm settings. Y

ITS SR 3.2.1.3, SR 3.2.1.4, SR 3.2.1.5, and SR 3.2.1.6 require the verification of parameters that similarly indicate the LHR is within the limits specified in the COLR when using the excore monitoring system. These SRs also provide verification that the parameters are appropriate for use of the excore monitoring system to monitor LHR and that the LCO is being met (which is missing from the NUREG). However, since the LHR is normally automatically monitored and alarmed by the Incore ~~monitoring~~ Alarm System, these SRs are only required to be met when the excore monitoring system is being used to determine LHR. These SRs are generally consistent with the requirements of CTS 4.19.1.2a, b, c, and d. KAK

10. The periodic Frequency of NUREG SR 3.2.1.3 is revised to 31 EFPD. CTS 4.19.1.1 provides requirements to adjust the incore alarm settings based on a measured power distribution on a periodic Frequency of "7 days of power operation." Although the CTS Frequency is based on days of power operation, this is inconsistent with the Frequency of ITS Section 3.1 SRs which are based on EFPD, inconsistent with NUREGs for other vendors (e.g., NUREG-1430 and NUREG-1431) for Power Distribution Limit SRs which are based on EFPD, and inconsistent with preferred methods for tracking this Frequency since EFPD is already required to be tracked to for numerous calculations related to burnup and other fuel status parameters. When the plant is operating steadily at full power there is no difference in the NUREG SR 3.2.1.3 periodic Frequency of "31 days" and the proposed "31 EFPD." However, when the 31 days includes operation at less than full power the "31 EFPD" is longer than the NUREG would allow. Still, the revision to the SR Frequency is acceptable since the Frequency continues to be sufficient to assure the incore alarm settings are appropriately since any change is a slow process.

NRC REQUEST:

3.2-02 ITS 3.2.1 Linear Heat Rate (LHR)
ITS 3.2.1 Surveillance Requirements
JFD 9

The STS SRs have been changed in the ITS to be consistent with the CTS.

Comment #1: The ITS SR 3.2.1.1 Note incorrectly references LCO 3.2.5 and LCO 3.2.6. What is the purpose of this note? Recommend deleting note.

Comment #2: Provide ITS SR 3.2.1.1 an appropriate specific frequency.

Consumers Energy Response:

The markup of ISTS SR 3.2.1.1 (Attachment 5 NUREG page 3.2-2 Insert) contains a Note which inappropriately references LCO 3.2.5 and LCO 3.2.6. The markup also inappropriately specified a Frequency of "as required by applicable specification.". The Note was intended to state "Only Required when the Incore Monitoring System is being used to monitor LHR" and to specify a Frequency of "12 hours." The intended version of this Note was correctly presented in the clean typed copy of Specification 3.2.1 and its associated Bases found in Attachments 1 and 2, respectively, and appropriately justified in JFD 9 (Attachment 6).

Affected Submittal Pages

Att 5 NUREG, pg 3.2-2 insert

SECTION 3.2

INSERT

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	--------------NOTE-------------- ~~Only required to be performed when specified by LCO 3.2.5, "Incore Monitoring System," or by LCO 3.2.6, "Excore Monitoring System."~~ --------------------------------- Verify LHR is within the limits specified in the COLR. Only required when Incore Alarm System is being used to monitor LHR.	RAI 3.2-02 12 hours ~~As required by applicable Specification~~

NRC REQUEST:

3.2-02 ITS 3.2.1 Linear Heat Rate (LHR)
ITS 3.2.1 Surveillance Requirements
JFD 9

The STS SRs have been changed in the ITS to be consistent with the CTS.

Comment #3: ITS SR 3.2.1.3 and ITS SR 3.2.1.5 should appear with the ASI specification; recommend moving to ITS 3.2.4. (4) ITS SR 3.2.1.6 should appear with the T_q specification; recommend moving to ITS 3.2.3.

Comment #4: ITS SR 3.2.1.6 should appear with the T_q specification; recommend moving to ITS 3.2.3.

Consumers Energy Response:

ITS SR 3.2.1.3, SR 3.2.1.5, and SR 3.2.1.6 ensure the conditions related to core power distribution are acceptable before using the Excore Monitoring System to monitor LHR, and to ensure LHR remains within limit. The Excore Monitoring System does not determine LHR directly. However, if more restrictive limits are placed on both ASI and T_q, Excore readings may be used to assure LHR is within limits. These more restrictive limits are only necessary when the Incore Alarm System is unavailable. The limits imposed by these SRs are more restrictive than the limits imposed in their respective specifications (i.e., ITS 3.2.4, "Axial Shape Index", and ITS 3.2.3, "Quadrant Power Tilt"). Since failure to meet an SR would be failure to meet the LCO (SR 3.0.1), placing the more restrictive SRs in their respective specifications would invoke inappropriate Required Actions in the event an SR has failed. Therefore, to ensure the appropriate Required Actions are taken when LHR is not within limits as determined by the Excore Core Monitoring, SR 3.2.1.3, SR 3.2.1.5, and SR 3.2.1.6 must be retained in ITS 3.2.1.

Affected Submittal Pages

None

NRC REQUEST:

3.2-03 ITS 3.2.1 Linear Heat Rate (LHR)
ITS SR 3.2.1.2 Frequency
JFD 10

The frequency for ITS SR 3.2.1.2 has been changed, from 7 days in the CTS and 31 days in the STS, to 31 EFPD; a beyond scope change.

Comment: Recommend retaining the STS frequency of 31 days for ITS SR 3.2.1.2.

Consumers Energy Response:

The Frequency of proposed ITS SR 3.2.1.2 was changed from units of "days" to "EFPD" (Effective Full Power Days) to be consistent with proposed SR 3.2.2.1 (Specification 3.2.1, DOC L.4). Aligning the Frequency of these two SRs is logical since the input to SR 3.2.1.2 is based on the results of SR 3.2.2.1. As noted in NRC Request 3.2-05, the Frequency of SR 3.2.2.1 was changed from units of "days of accumulated operation in Mode 1" to "EFPD" (Specification 3.2.2, DOC L.2). These changes were made to establish consistency with the methods generally accepted to track core parameters that are sensitive to fuel burnup. These methods are deemed acceptable since power distribution changes are relatively slow over a 31 day period. In addition, nearly all "power operation" is at the full power condition, and when the plant is operating at full power there is no difference in a Frequency of 31 days and 31 EFPD.

Although these Frequency changes represent a deviation from NUREG-1432, they are consistent with similar type power distribution Frequencies in NUREG-1430 (B&W plants) and NUREG-1431 (Westinghouse plants) previously found acceptable by the NRC. As such, Palisades would like to retain the Frequency units of EFPD in SR 3.2.1.2 and SR 3.2.2.1 on the basis it is consistent with the Improved Standard Technical Specifications for power distribution related surveillances.

Affected Submittal Pages

None

NRC REQUEST:

3.2-04 ITS 3.2.2 Radial Peaking Factors
ITS 3.2.2 Required Action B
DOC M.1, JFD 5 and JFD 8

The CTS requires going to Hot Shutdown (similar to Mode 3) in 6 hours if peaking factors are not within limits, with Power < 50% RTP. The ITS requires going to ≤ 25% RTP in 4 hours if peaking factors are not returned to within limits in 6 hours.

Comment #1: In the STS, when a radial peaking factor is not within limit the first action is to reduce power. The ITS allows 6 hours delay prior to reducing power. The ITS should more closely reflect the STS actions.

Consumers Energy Response:

When radial peaking factors are not within limit, the Required Actions of both the ISTS and ITS allow 6 hours to establish compliance with the LCO. The Required Actions of the ISTS are more prescriptive than the ITS since they include the method for restoring compliance with the LCO. Neither the CTS, nor the ITS provide this same level of detail but simply require the peaking factors be restored to within limits without specifying the method used to accomplish the restoration. Although restoration would typically include a reduction in thermal power, such a reduction may not always be necessary. Alternatively, correcting the source of the peaking may be the optimum method for restoration. The method to restore peaking factors prescribed in the ISTS is by reducing thermal power while withdrawing the CEAs to or beyond their long term steady state insertion limit. Since only one set of insertion limits is used at Palisades, actions to be taken if rods are inserted beyond the insertion limits are specified in LCO 3.1.5 or 3.1.6. These actions have a 2 hour Completion Time. Therefore, a prescriptive Required Action to reduce thermal power may not always be appropriate. As such, Palisades would like to maintain the operational flexibility that exists in the CTS for restoring radial peaking factors to within limits.

Affected Submittal Pages

None

NRC REQUEST:

3.2-04 ITS 3.2.2 Radial Peaking Factors
ITS 3.2.2 Required Action B
DOC M.1, JFD 5 and JFD 8

The CTS requires going to Hot Shutdown (similar to Mode 3) in 6 hours if peaking factors are not within limits, with Power < 50% RTP. The ITS requires going to ≤ 25% RTP in 4 hours if peaking factors are not returned to within limits in 6 hours.

Comment #2: This change to the CTS is less restrictive and needs to be appropriately justified.

Consumers Energy Response:

DOC L.1 has been revised to enhance the justification which provides four additional hours to exit the mode of applicability when radial peaking factors can not be restored within limits.

Affected Submittal Pages

Att 3 DOC 3.2.2, pg 3 of 4
Att 4 NSHC 3.2.2, pg 1 of 5

LA.2 CTS 4.19.2.1 provides Surveillance Requirements (SRs) for the Radial Peaking Factors. However, it contains specific details for monitoring of the peaking factors, i.e., that the SR is performed by verifying the "measured" radial peaking factors "obtained by using the incore detection system." This information is not provided in NUREG SR 3.2.2.1. These details describe elements of the radial peaking factor verification which are addressed by the methodology and are not directly a part of the actual requirement, i.e., Surveillance Requirement. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in the Bases of ITS SR 3.2.2.1 provides adequate assurance that they will be maintained. The Bases are controlled by the Bases Control Program in Chapter 5 of the proposed Technical Specifications. This change is consistent with NUREG-1432.

LESS RESTRICTIVE CHANGES (L)

L.1 ~~CTS 3.23.2 provides~~ actions for peaking factors exceeding their limits based on power level. The first of these actions is for P (power) < 50%, and requires the plant to be in at least hot shutdown, i.e., subcritical, within 6 hours. ITS 3.2.2 Required Action A.1 provides 6 hours to attempt restoration of the peaking factors to within limits, and if the Required Action and its associated Completion Time is not met, then Required Action B.1 requires that THERMAL POWER be reduced to ≤ 25% RTP. This change is less restrictive in two ways. First, six hours is provided to attempt restoration of the peaking factors to within limits that is not provided in the CTS. Second, the default action requires only that the plant to be reduced to ≤ 25% RTP, rather than subcritical.

The ITS Required Actions are appropriate for the conditions and assure the plant will not operate for an extended period with the peaking factors not within limits. The Completion Time provides a reasonable time for determining the proper method, power level, and associated limits for restoration, and for the restoration of the plant to within limits, and a reasonable time to remove the plant from the applicable conditions in an orderly manner and without challenging plant systems. This change is consistent with NUREG-1432 as modified for plant specific parameters and analysis.

See INSERT

RAI 3.2-04

INSERT

CTS 3.23.2 provides actions for peaking factors exceeding their limits based on power level. The first of these actions is for P (power) < 50%, and requires the plant to be in at least hot shutdown (i.e., subcritical) within 6 hours. ITS 3.2.2 Required Action A.1 provides 6 hours to attempt restoration of the peaking factors to within limits, and if the Required Action and its associated Completion Time is not met, then Required Action B.1 requires that THERMAL POWER be reduced to ≤ 25% RTP. This change is less restrictive in two ways. First, 6 hours is provided to attempt restoration of the peaking factors to within limits that is not provided in the CTS. Second, the default action requires only that the plant to be reduced to ≤ 25% RTP, rather than subcritical, in the subsequent 4 hours.

The ITS Required Action to restore the radial peaking factors to the within limits specified in the COLR assure the plant will not operate for an extended period with the peaking factors not within limits. The Completion Time of 6 hours provides a reasonable time for evaluating core conditions, calculating a reduced power level at which the peaking factors would be within limits, determining the proper method for the power reduction (e.g., rod positioning and/or boration) and, completing the reduction in power. In the event the peaking factors are not restored to within limits, an additional 4 hours is provided to remove the plant from the mode of applicability. Although CTS 3.23.2 requires the plant to be placed in hot shutdown, terminating the power reduction anywhere below 25% is permissible since CTS LCO 3.0.1 only requires compliance with an LCO during the plant condition specified in that LCO. Thus, the default action of proposed ITS Required Action B.1 is consistent with the shutdown action for CTS 3.23.2. A Completion Time of 4 hours is reasonable to reduce thermal power below 25% in an orderly manner and without challenging plant systems.

LESS RESTRICTIVE CHANGE L.1

CTS 3.23.2 provides actions for peaking factors exceeding their limits based on power level. The first of these actions is for P (power) < 50%, and requires the plant to be in at least hot shutdown, i.e., subcritical, within 6 hours. ITS 3.2.2 Required Action A.1 provides 6 hours to attempt restoration of the peaking factors to within limits, and if the Required Action and its associated Completion Time is not met, then Required Action B.1 requires that THERMAL POWER be reduced to ≤ 25% RTP. This change is less restrictive in two ways. First, six hours is provided to attempt restoration of the peaking factors to within limits that is not provided in the CTS. Second, the default action requires only that the plant to be reduced to ≤ 25% RTP, rather than subcritical.

The ITS Required Actions are appropriate for the conditions and assure the plant will not operate for an extended period with the peaking factors not within limits. The Completion Times provides a reasonable time for determining the proper method, power level, and associated limits for restoration, and for the restoration of the plant to within limits, and a reasonable time to remove the plant from the applicable conditions in an orderly manner and without challenging plant systems. This change is consistent with NUREG-1432 as modified for plant specific parameters and analysis.

See INSERT

RAI 3.2-04

INSERT

CTS 3.23.2 provides actions for peaking factors exceeding their limits based on power level. The first of these actions is for P (power) < 50%, and requires the plant to be in at least hot shutdown (i.e., subcritical) within 6 hours. ITS 3.2.2 Required Action A.1 provides 6 hours to attempt restoration of the peaking factors to within limits, and if the Required Action and its associated Completion Time is not met, then Required Action B.1 requires that THERMAL POWER be reduced to ≤ 25% RTP. This change is less restrictive in two ways. First, 6 hours is provided to attempt restoration of the peaking factors to within limits that is not provided in the CTS. Second, the default action requires only that the plant to be reduced to ≤ 25% RTP, rather than subcritical, in the subsequent 4 hours.

The ITS Required Action to restore the radial peaking factors to the within limits specified in the COLR assure the plant will not operate for an extended period with the peaking factors not within limits. The Completion Time of 6 hours provides a reasonable time for evaluating core conditions, calculating a reduced power level at which the peaking factors would be within limits, determining the proper method for the power reduction (e.g., rod positioning and/or boration) and, completing the reduction in power. In the event the peaking factors are not restored to within limits, an additional 4 hours is provided to remove the plant from the mode of applicability. Although CTS 3.23.2 requires the plant to be placed in hot shutdown, terminating the power reduction anywhere below 25% is permissible since CTS LCO 3.0.1 only requires compliance with an LCO during the plant condition specified in that LCO. Thus, the default action of proposed ITS Required Action B.1 is consistent with the shutdown action for CTS 3.23.2. A Completion Time of 4 hours is reasonable to reduce thermal power below 25% in an orderly manner and without challenging plant systems.

NRC REQUEST:

3.2-05 ITS 3.2.2 Radial Peaking Factors
ITS SR 3.2.2.1 Frequency
DOC L.2 and JFD 9

The frequency for ITS SR 3.2.2.1 has been changed, from 7 days in the CTS and 31 days in the STS, to 31 EFPD; a beyond scope change.

Comment: Recommend retaining the STS frequency of 31 days for ITS SR 3.2.2.1.

Consumers Energy Response:

Please see the response to NRC Request 3.2-03.

Affected Submittal Pages

None

NRC REQUEST:

3.2-06 ITS 3.2.3 [STS 3.2.4] Power Tilt (T_q)
ITS 3.2.3 [STS 3.2.4] LCO and Required Actions
JFD 1 and JFD 5

STS 3.2.4 has been rewritten to reflect CTS limits in ITS 3.2.3.

Comment #1: The ITS has not retained STS Required Action C.3 to restore T_q to < [0.03] prior to increasing thermal power (if T_q is no longer >[0.10]); submit TSTF for change to STS. NRC to review.

Comment #2: The ITS has not retained the STS Notes to the Required Action C and the related Completion Times, though similar requirements are retained in administrative controls; submit TSTF for change to STS. NRC to review.

Consumers Energy Response:

Palisades will propose a generic change to NUREG-1432 at the next meeting of the CE Owners Group Licensing Subcommittee to delete ISTS 3.2.4, Required Action C.3 and its associated Completion Time.

Affected Submittal Pages

None

NRC REQUEST:

3.2-07 ITS 3.2.3 [STS 3.2.4] Power Tilt (T_q)
ITS 3.2.3 [STS 3.2.4] Bases to Required Actions
(STS pages B 3.2-23 & B 3.2-24)
JFD 5

The STS Bases contains two paragraphs addressing STS Required Actions C.1, C.2 and C.3, that have been deleted in the ITS Bases.

Comment: Recommend retaining this information tailored for Palisades.

Consumers Energy Response:

The deleted paragraphs were reviewed for information that is appropriate for inclusion in the ITS Bases. While much of the deleted information deals with the omitted Actions C.1, C.2, and C.3 the appropriate information has been added to the Bases for ITS Action B.1.

Affected Submittal Pages

Att 2, ITS 3.2.3, page B 3.2.3-2
Att 5, NUREG 3.2.4, page B 3.2-23

BASES

ACTIONS

A.1

If the measured T_q is > 0.05, T_q must be restored within 2 hours or F_r^A and F_r^T must be determined to be within the limits of LCO 3.2.2, and determined to be within these limits every 8 hours thereafter, as long as T_q is out of limits. Two Four hours is sufficient time to allow the operator to reposition control rods, and significant radial xenon redistribution cannot occur within this time. The 8 hour Completion Time ensures changes in F_r^A and F_r^T can be identified before the limits of LCO 3.2.2 are exceeded. ed

B.1

RAI 3.2-07

With the measured T_q > 0.10, power must be reduced to < 50% RTP within 4 hours, and F_r^A and F_r^T must be within their specified limits to ensure that acceptable flux peaking factors are maintained as required by Condition A (which continues to be applicable). ~~Based on operating experience, 4 hours is sufficient time for evaluation of these factors~~. If F_r^A and F_r^T are within limits, operation at < 50% RTP may proceed while attempts are made to restore T_q to within its limit.

X

C.1

If T_q is > 0.15, or if Required Actions and associated Completion Times are not met, THERMAL POWER must be reduced to ≤ 25% RTP. This requirement ensures that the core is operating within its thermal limits and places the core in a conservative condition. Four hours is a reasonable time to reach 25% RTP in an orderly manner and without challenging plant systems.

If the tilt is generated due to a control rod misalignment, continued operation at < 50% RTP allows for realignment; if the cause is other than control rod misalignment, continued operation may be necessary to discover the cause of the tilt. Reducing THERMAL POWER to < 50% RTP, and the more frequent measurement of peaking factors required by Action A.1, provide conservative protection from potential increased peaking due to xenon redistribution.

If the tilt is generated due to a control rod misalignment, continued operation at < 50% RTP allows for realignment; if the cause is other than control rod misalignment, continued operation may be necessary to discover the cause of the tilt. Reducing THERMAL POWER to < 50% RTP, and the more frequent measurement of peaking factors required by Action A.1, provide conservative protection from potential increased peaking due to xenon redistribution.

RAI 3.2-07

ACTIONS X Ed

A.1 ~~and A.2~~

6 7 8 9 4 8 If the measured T_q is > 0.05, ~~[0.03] and < 0.10, the calculation of F_{xy} may be nonconservative~~. T_q must be restored within 2 hours or F_r^A ~~F_{xy}~~ and F_r^T must be determined to be within the limits of LCO 3.2.2 ~~and LCO 3.2.3~~, and determined to be within these limits every 8 hours thereafter, as long as T_q is out of limits. Two ~~Four~~ hours is sufficient time to allow the operator to reposition control rods ~~CEAs~~, and significant radial xenon redistribution cannot occur within this time. The 8 hour Completion Time ensures changes in F_r^A ~~F_{xy}~~ and F_r^T can be identified before the limits of LCO 3.2.2 ~~and LCO 3.2.3, respectively,~~ are exceeded.

C ~~B~~.1

5 If T_q is > 0.15, or if Required Actions and associated Completion Times ~~of Condition A~~ are not met, THERMAL POWER must be reduced to ≤ 25 ~~50%~~ RTP. This requirement ensures that the core is operating within its thermal limits and places the core in a conservative condition. Four hours is a reasonable time to reach 25 ~~50%~~ RTP in an orderly manner and without challenging plant systems.

~~B.1 and B.2~~

~~C.1, C.2, and C.3~~

2 6 4 5 With the measured T_q > 0.10, power must be reduced to < 50% RTP within 4 hours, and F_r^A ~~F_{xy}~~ and F_r^T must be within their specified limits to ensure that acceptable flux peaking factors are maintained. Based on operating experience, 4 hours ~~1 hour~~ is sufficient time for evaluation of ~~the operator to evaluate~~ these factors. If F_r^A ~~F_{xy}~~ and F_r^T are within limits, operation may proceed ~~for a total of 2 hours after the Condition is entered~~ while attempts are made to restore T_q to within its limit. AS REQUIRED BY CONDITION A, WHICH CONTINUES TO BE APPLICABLE

5 ~~If T_q ≤ 0.10 cannot be achieved, power must be reduced to ≤ 50% RTP within 2 hours. If the tilt is generated due to a CEA misalignment, operating at ≤ 50% RTP allows for the recovery of the CEA. Except as a result of CEA misalignment, T_q > 0.10 is not expected; if it occurs, continued operation of the reactor may be necessary to discover the cause of the tilt. If this procedure is followed, operation is restricted to only those conditions required to identify the cause of the tilt. It is necessary to account explicitly for power asymmetries because the~~

5

(continued)

5

NRC REQUEST:

3.2-08 ITS 3.2.3 [STS 3.2.4] Power Tilt (T_q)
ITS 3.2.3 [STS 3.2.4] Required Actions
DOC L.1 and DOC M.3

The CTS required action if T_q is > 0.15 is to go to Hot Shutdown in 12 hours while the ITS required action is to decrease power to ≤ 25% RTP in 4 hours.

Comment: Confirm that the ITS Applicability is appropriate and that the CTS required action is overly restrictive. When T_q becomes too large shutting down may be the appropriate action.

Consumers Energy Response:

The Applicability for the ITS T_q LCO is unchanged from CTS. The Applicability for T_q in both the CTS and ITS is > 25% Rated Power. Although CTS 3.23.3 requires the plant be placed in Hot Standby whenever T_q is > 0.15, terminating the power reduction anywhere below 25% is permissible since CTS LCO 3.0.1 only requires compliance with an LCO during the plant conditions specified in that LCO. At power levels < 25% there is insufficient Thermal Power to require a limit on core power distribution. In addition, ample thermal margin exists to ensure fuel integrity is not jeopardized and safety analysis assumptions remain valid. As such, requiring the plant to be placed in hot standby when T_q is not within limit is overly restrictive.

Affected Submittal Pages

None

NRC REQUEST:

3.2-09 ITS 3.2.3 [STS 3.2.4] Power Tilt (T_q)
ITS SR 3.2.1.6
CTS SR 4.19.1.1.b

Proposed ITS SR 3.2.1.6 (CTS SR 4.19.1.1.b) imposes a limit on T_q of 0.03.

Comment: This limit does not appear anywhere in any ITS LCO limit; why not? Should the STS limits be adopted?

Consumers Energy Response:

The 3% limit for T_q does not appear in any ITS LCO since it is specified as a surveillance requirement associated with the LCO for LHR. This surveillance requirement ensures the conditions related to core power distribution are acceptable before using the Excore Monitoring System to monitor LHR, and to ensure LHR remains within limit.

Adopting the ISTS limits (i.e., specifying the 3% limit on T_q in the Quadrant Power Tilt specification) would not be appropriate since the 3% restriction on T_q is only applicable when the Excore Monitoring System is being used to monitor LHR. If T_q were to exceed the 3% limit when the Excore Monitoring System is being used to monitor LHR, the appropriate Required Actions would be those actions associated with the LHR LCO, not the T_q LCO.

Also see response to RAI 3.2-02.

Affected Submittal Pages

None

NRC REQUEST:

3.2-10 ITS 3.2.3 [STS 3.2.4] Power Tilt (T_q)
ITS 3.2.3 Required Action C
DOC M.2

ITS Condition C (default actions if required action not met) is an addition to CTS actions.

Comment: Wouldn't the CTS actions implicitly required a similar action? How is this more restrictive; is this an administrative change?

Consumers Energy Response:

If the Actions of CTS 3.23.3 could not be met, then LCO 3.0.3 would require the plant to be placed in hot standby within 7 hours. The "more restrictive" aspect of adding ITS Condition C is the shorter time for completing the shutdown (i.e., 4 hours in ITS versus 7 hours in CTS).

Affected Submittal Pages

None

ENCLOSURE 2

CONSUMERS ENERGY COMPANY
PALISADES PLANT
DOCKET 50-255

CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS
RESPONSE TO DECEMBER 4, 1998
REQUEST FOR ADDITIONAL INFORMATION

EDITORIAL CHANGES

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Alignment

LCO 3.1.4 All control rods, including their rod position indication channels, shall be OPERABLE and aligned to within 8 inches of all other rods in their respective group, and the control rod position deviation alarm shall be OPERABLE. ed

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One channel of rod position indication inoperable for one or more control rods.	A.1	Perform SR 3.1.4.1 (rod position verification).	Once within 15 minutes following any rod motion in that group
B. Rod position deviation alarm inoperable.	B.1	Perform SR 3.1.4.1 (rod position verification).	Once within 15 minutes of movement of any control rod
C. One control rod misaligned by > 8 inches ~~from any other rod in its group~~.	C.1	Perform SR 3.2.2.1 (peaking factor verification).	2 hours
	OR		
	C.2	Reduce THERMAL POWER to ≤ 75% RTP.	2 hours

ed
Consistency with Condition 'E'.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify the position of each control rod to be within 8 inches of all other control rods in its group.	12 hours
SR 3.1.4.2	Perform a CHANNEL CHECK of the control rod position indication channels.	12 hours
SR 3.1.4.3	Verify the rod position deviation alarm is OPERABLE.	92 days
SR 3.1.4.4	Verify control rod freedom of movement by moving each individual full-length control rod that is not fully inserted into the reactor core ≥ 6 inches in either direction.	92 days
SR 3.1.4.5	Perform a CHANNEL CALIBRATION of the control rod position indication channels.	18 months
SR 3.1.4.6	Verify each full-length control rod drop time is ≤ 2.5 seconds.	Prior to reactor criticality, after each reinstallation of the reactor head

BASES

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption that the control rod of highest reactivity worth is fully withdrawn following a reactor trip. For MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (Departure from Nucleate Boiling Ratio (DNBR), fuel centerline temperature limit AOOs, and ≤ 280 cal/gm energy deposition for the control rod ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements are based on a Main Steam Line Break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected Steam Generator (SG), and consequently the PCS. This results in a reduction of the primary coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. ~~As PCS temperature decreases, the severity of an MSLB decreases until the MODE 5 value is reached~~. The most limiting MSLB, with respect to potential fuel damage ~~before a reactor trip occurs,~~ is a guillotine break of a main steam line ~~inside containment~~ initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating PCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1. ~~The full power MSLB analysis bounds the results for Hot Zero Power~~.

ed

BASES

BACKGROUND (continued)

The control rods are arranged into groups that are radially symmetric. Therefore, movement of the control rod groups does not introduce radial asymmetries in the core power distribution. The shutdown and regulating rods provide the required reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating rods also provide reactivity (power level) control during normal operation and transients.

The axial position of shutdown and regulating rods is indicated by two separate and independent systems, which are 1) synchro based position indication system, and 2) the reed switch based position indication system.

The synchro based position indication system measures the phase angle of a synchro geared to the CRDM rack. Full control rod travel corresponds to less than 1 turn of the synchro. Each control rod has its own synchro. The Primary Information Processor (PIP) node scans and converts synchro outputs into inches of control rod withdrawal. The resolution of this system is approximately 0.5 inches. Each synchro also has ~~a~~ cam operated limit switches which can provide positive indication of control rod position.

The reed switch based position indication system is referred to as the Secondary Position Indication (SPI) system. This system provides a highly accurate indication of actual control rod position, but at a lower precision than the synchros. The reed switches are wired so that the voltage read across the reed switch stack is proportional to rod position. The reed switches are spaced along a tube with a center to center spacing distance of 1.5 inches. The resolution of the SPI reed switch stacks is 1.5 inches. The reed switches also provide input to the matrix indication lights which provide control rod status indication for various key positions. To increase the reliability of the system, there are redundant ~~overlapping~~ reed switches ~~levels~~ which prevent false ~~rod drop~~ indication in the event ~~a~~ an individual reed switch fails. ~~to close~~.

A control rod position deviation alarm is provided to alert the operator when any two control rods in the same group are more than 8 inches apart. This helps to ensure any control rod misalignments are minimized. The alarm can be generated by either the SPI system or PIP node since the SPI system, in conjunction with the host computer, is redundant to the PIP node in the task of control rod measurement, control rod monitoring, and limit processing.

BASES

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Refs. 3 and 4). The accident analysis defines control rod misoperation as any event, with the exception of sequential group withdrawals, which could result from a single malfunction in the reactivity control systems. For example, control rod misalignment may be caused by a malfunction of the Rod Control System, or by operator error. A stuck rod may be caused by mechanical jamming. Inadvertent withdrawal of a single control rod may be caused by an electrical or mechanical failure in the Rod Control System. A dropped control rod could be caused by an electrical or ~~mechancial~~ mechanical failure in the CRDM. X Ed

The acceptance criteria for addressing control rod inoperability/misalignment are that:

a. There shall be no violations of:

   1. Specified Acceptable Fuel Design Limits (SAFDL), or

   2. Primary Coolant System (PCS) pressure boundary integrity; and

b. The core must remain subcritical after accident transients.

Three types of misoperations are discussed in the safety analysis (Ref. 4). During movement of a group, one control rod may stop moving while the other control rods in the group continue. This condition may cause excessive power peaking. The second type of misoperations occurs if one control rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the remaining control rods to meet the SDM requirement with the maximum worth rod stuck fully withdrawn. If a control rod is stuck in the fully withdrawn position, its worth is added to the SDM requirement, since the safety analysis does not take two stuck rods into account. The third type of misoperations occurs when one rod drops partially or fully into the reactor core. This event causes an initial power reduction followed by a return towards the original power, due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in excessive local Linear Heat Rates (LHRs).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The most limiting static misalignment occurs when Bank 4 is fully inserted with one rod fully withdrawn (Bank 4 is 99 inches out of alignment with the rated Power Dependent Insertion Limit (PDIL). This event was bounded by the dropped full-length control rod event (Ref. 4).

Since the control rod drop incidents result in the most rapid approach to SAFDLs caused by a control rod misoperation, the accident analysis analyzed a single full-length control rod drop. ~~The most rapid approach to the DNBR SAFDL may be caused by a single full-length control rod drop~~.

cd
Redundant statement

The above control rod misoperations may or may not result in an automatic reactor trip. In the case of the full-length rod drop, a prompt decrease in core average power and a distortion in radial power are initially produced, which, when conservatively coupled, result in a local power and heat flux increase, and a decrease in DNBR parameters.

The results of the control rod misoperation analysis show that during the most limiting misoperation events, no violations of the SAFDLs, fuel centerline temperature, or PCS pressure occur.

Control rod alignment satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2).

LCO

The limits on shutdown, regulating, and part-length rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the control rods will be available and will be inserted to provide enough negative reactivity to shut down the reactor. The OPERABILITY requirements also ensure that the control rod banks maintain the correct power distribution and control rod alignment and that each control rod is capable of being moved by its CRDM. The OPERABILITY requirement for the part-length rods is that they are fully withdrawn and are capable of being moved by their CRDMs.

generated by either the PIP node or the SPI system,

BASES

LCO (continued)

The requirement is to maintain the control rod alignment to within 8 inches between any control rod and all other rods in its group. To ~~This~~ help~~s~~ ensure this requirement is met, the control rod position deviation alarm must be OPERABLE and provide an alarm when any control rod becomes misaligned > 8 inches from any other rod in its group. The safety analysis assumes a total misalignment from fully withdrawn to fully inserted. This case bounds the safety analysis for a single rod in any intermediate position. ed X

The primary rod position indication system is considered OPERABLE, for purposes of this specification, if the digital position readout, the PPC display, or the cam operated position indication lights give positive indication of rod position. The secondary rod position indication system is considered OPERABLE if the magnetically operated reed switches are providing positive indication of rod position either via the plant process computer or taking direct readings of the output from the magnetic reed switches.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDM, any of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on control rod OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (e.g., trippability) and alignment of control rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and regulating rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the PCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5, and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

BASES

ACTIONS

E.1 (continued)

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual control rod positions are within 8 inches of all other control rods in the group at a 12 hour Frequency allows the operator to detect a control rod that is beginning to deviate from its expected position. The specified Frequency takes into account other control rod position information that is continuously available to the operator in the control room, so that during control rod movement, deviations can be detected. Also protection can be provided by the control rod deviation alarm.

SR 3.1.4.2

OPERABILITY of two control rod position indicator channels is required to determine control rod positions, and thereby ensure compliance with the control rod alignment and insertion limits. Performance of a CHANNEL CHECK on the primary and secondary control rod position indication channels provides confidence in the accuracy of the rod position indication systems. The control rod "full in" and "full out" lights, which correspond to the lower electrical limit and the upper electrical limit respectively, provide an additional means for determining the control rod positions when the control rods are at either their fully inserted or fully withdrawn positions.

The 12 hour Frequency takes into consideration other information continuously available to the operator in the control room, so that during control rod movement, deviations can be detected, and protection can be provided by the control rod deviation alarm.

BASES

BACKGROUND (continued)

They affect core power, burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

APPLICABLE SAFETY ANALYSES

Accident analysis assumes that the shutdown rod groups are fully withdrawn any time the reactor is critical. This ensures that:

a. The minimum SDM is maintained; and

b. The potential effects of a control rod ejection accident are limited to acceptable limits.

Control rods are considered fully withdrawn at 128 inches, since this position places them in an very insignificant reactivity worth region of the integral worth curve for each bank. ed X

On a reactor trip, all full-length control rods (shutdown and regulating), except the most reactive rod, are assumed to insert into the core. The shutdown and regulating rod groups shall be at or above their insertion limits and available to insert the required amount of negative reactivity on a reactor trip signal. The regulating rods may be partially inserted in the core as allowed by LCO 3.1.6, "Regulating Rod Group Position Limits." The shutdown rod group insertion limit is established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)) following a reactor trip from full power. The combination of regulating rod and shutdown rods (less the most reactive rod, which is assumed to remain fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 2). The shutdown rod group insertion limit also limits the reactivity worth of an ejected shutdown rod.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1 (continued) Ed

Since ~~the shutdown and part-length~~ Control rod groups are positioned manually by the control room operator, verification of shutdown and part-length rod group position at a Frequency of 12 hours is adequate to ensure that the shutdown and part-length rod groups are within their insertion limits. Also, the 12 hour Frequency takes into account other information available to the operator in the control room for the purpose of monitoring the status of the shutdown and part-length rod groups. X

REFERENCES

1. FSAR, Section 5.1
2. FSAR, Section 14.2
3. FSAR, Section 14.6

BASES

LCO (continued)

The Power Dependent Insertion Limit (PDIL) alarm circuit is required to be OPERABLE for notification that the regulating rod groups are outside the required insertion limits. The Control Rod Out Of Sequence (CROOS) alarm circuit is required to be OPERABLE for notification that the rods are not within the required sequence and overlap limits. When the PDIL or the CROOS alarm circuit is inoperable, the verification of rod group positions is increased to ensure improper rod alignment is identified before unacceptable flux distribution occurs. ed X

APPLICABILITY

The regulating rod group sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES. SDM is preserved in MODES 3, 4, and 5 by adjustments to the soluble boron concentration.

The Applicability has been modified by a Note indicating the LCO requirement is suspended SR 3.1.4.4 (rod exercise test). Control rod exercising verifies the freedom of the rods to move, and requires the individual regulating rods to move below the LCO limits which ~~would normally~~ could violate the LCO for their group. ed X

ACTIONS

A.1 and A.2

Operation beyond the insertion limit may result in a loss of SDM and excessive peaking factors. The insertion limit should not be violated during normal operation; this violation, however, may occur during transients when the operator is manually controlling the regulating rods in response to changing plant conditions.

The PDIL and CROOS alarms can be generated by either the synchro based Primary Indication Processor (PIP) node, or the reed switch based Secondary Position Indication (SPI) system since the SPI system, in conjunction with the host computer, is redundant to the PIP node in the task of control rod measurement, control rod monitoring, and limit processing.

ENCLOSURE 3

CONSUMERS ENERGY COMPANY
PALISADES PLANT
DOCKET 50-255

CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS
RESPONSE TO DECEMBER 4, 1998
REQUEST FOR ADDITIONAL INFORMATION

REVISED PAGES FOR CHAPTER 2.0

CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS
RESPONSE TO DECEMBER 4, 1998 REQUEST FOR ADDITIONAL INFORMATION
REVISED PAGES FOR CHAPTER 2.0

Page Change Instructions

Revise the Palisades submittal for conversion to Improved Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by date and contain vertical lines in the margin indicating the areas of change.

REMOVE PAGES	INSERT PAGES	REV DATE	NRC COMMENT#
ATTACHMENT 1 TO ITS CONVERSION SUBMITTAL			
No page changes			
ATTACHMENT 2 TO ITS CONVERSION SUBMITTAL			
B 2.1.2-3	B 2.1.2-3	02/05/99	RAI 2.0-01
ATTACHMENT 3 TO ITS CONVERSION SUBMITTAL			
No page changes			
ATTACHMENT 4 TO ITS CONVERSION SUBMITTAL			
No page changes			
ATTACHMENT 5 TO ITS CONVERSION SUBMITTAL			
B 2.0-8	B 2.0-8	02/05/99	RAI 2.0-01
ATTACHMENT 6 TO ITS CONVERSION SUBMITTAL			
No page changes			

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. In MODE 6 with the reactor vessel head installed and the reactor vessel head closure bolts less than fully tensioned the potential for an over pressurization event still exists. Although overpressurization of the PCS is impossible once the reactor vessel head is removed, the requirements of this SL apply as long as fuel is in the reactor. Once all the fuel has been removed from the reactor, the requirements of SL 2.1.2 no longer apply.

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 (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 fully tightened, making it unlikely that the RCS can be pressurized.~~ 7

In Mode 6 with the reactor head installed and the reactor vessel closure bolts less than fully tensioned, the potential for an overpressurization event still exists. Although overpressurization of the PCS is impossible once the reactor vessel head is removed, the requirements of this SL apply as long as fuel is in the reactor. Once all the fuel has been removed from the reactor, the requirements of SL 2.1.2 no longer apply.

SAFETY LIMIT VIOLATIONS

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

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. 4

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 than specified in SL 2.1.2 exceeds 110% of the PCS 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 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. 4 4

2.2.2.2 4

If the PCS 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 PCS 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 7 4

(continued)

ENCLOSURE 4

CONSUMERS ENERGY COMPANY
PALISADES PLANT
DOCKET 50-255

CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS
RESPONSE TO DECEMBER 4, 1998
REQUEST FOR ADDITIONAL INFORMATION

REVISED PAGES FOR SECTION 3.1

CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS
RESPONSE TO DECEMBER 4, 1998 REQUEST FOR ADDITIONAL INFORMATION
REVISED PAGES FOR SECTION 3.1

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REMOVE PAGES	INSERT PAGES	REV DATE	NRC COMMENT#
ATTACHMENT 1 TO ITS CONVERSION SUBMITTAL			
ITS 3.1.1-1	ITS 3.1.1-1	02/05/99	RAI 3.1-01
ITS 3.1.1-2	- - - - - -		RAI 3.1-01
ITS 3.1.4-1	ITS 3.1.4-1	02/05/99	editorial
ITS 3.1.4-2	ITS 3.1.4-2	02/05/99	RAI 3.1-04
ITS 3.1.4-3	ITS 3.1.4-3	02/05/99	editorial
ITS 3.1.5-1	ITS 3.1.5-1	02/05/99	RAI 3.1-07
- - - - - -	ITS 3.1.5-2	02/05/99	RAI 3.1-07
ITS 3.1.6-1	ITS 3.1.6-1	02/05/99	Tech change
ATTACHMENT 2 TO ITS CONVERSION SUBMITTAL			
ITS B 3.1.1-2	ITS B 3.1.1-2	02/05/99	editorial
ITS B 3.1.1-3	ITS B 3.1.1-3	02/05/99	RAI 3.1-01 RAI 3.1-04
ITS B 3.1.1-5	ITS B 3.1.1-5	02/05/99	RAI 3.1-01
ITS B 3.1.1-6	ITS B 3.1.1-6	02/05/99	RAI 3.1-01
ITS B 3.1.4-1	ITS B 3.1.4-1	02/05/99	RAI 3.1-04
ITS B 3.1.4-2	ITS B 3.1.4-2	02/05/99	editorial
ITS B 3.1.4-3	ITS B 3.1.4-3	02/05/99	editorial
ITS B 3.1.4-4	ITS B 3.1.4-4	02/05/99	RAI 3.1-04
ITS B 3.1.4-5	ITS B 3.1.4-5	02/05/99	editorial
ITS B 3.1.4-8	ITS B 3.1.4-8	02/05/99	RAI 3.1-04
ITS B 3.1.4-9	ITS B 3.1.4-9	02/05/99	editorial
ITS B 3.1.4-10	ITS B 3.1.4-10	02/05/99	RAI 3.1-04
ITS B 3.1.4-11	ITS B 3.1.4-11	02/05/99	RAI 3.1-04
ITS B 3.1.5-1	ITS B 3.1.5-1	02/05/99	RAI 3.1-04
ITS B 3.1.5-2	ITS B 3.1.5-2	02/05/99	editorial
ITS B 3.1.5-3	ITS B 3.1.5-3	02/05/99	RAI 3.1-07
ITS B 3.1.5-4	ITS B 3.1.5-4	02/05/99	RAI 3.1-04
ITS B 3.1.5-5	ITS B 3.1.5-5	02/05/99	RAI 3.1-07
ITS B 3.1.5-6	ITS B 3.1.5-6	02/05/99	editorial
ITS B 3.1.6-1	ITS B 3.1.6-1	02/05/99	RAI 3.1-04
ITS B 3.1.6-4	ITS B 3.1.6-4	02/05/99	RAI 3.1-04
ITS B 3.1.6-5	ITS B 3.1.6-5	02/05/99	RAI 3.1-10
ITS B 3.1.6-6	ITS B 3.1.6-6	02/05/99	editorial
ITS B 3.1.6-7	ITS B 3.1.6-7	02/05/99	editorial
ITS B 3.1.6-8	ITS B 3.1.6-8	02/05/99	editorial

CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS
RESPONSE TO DECEMBER 4, 1998 REQUEST FOR ADDITIONAL INFORMATION
REVISED PAGES FOR SECTION 3.1

ATTACHMENT 3 TO ITS CONVERSION SUBMITTAL

CTS 3.1.1, pg 3-50	CTS 3.1.1, pg 3-50	02/05/99	RAI 3.1-01
CTS 3.1.4, pg 3-52	CTS 3.1.4, pg 3-52	02/05/99	Tech change
CTS 3.1.4, pg 4-11	CTS 3.1.4, pg 4-11	02/05/99	Tech change
CTS 3.1.5, pg 3-53	CTS 3.1.5, pg 3-53	02/05/99	RAI 3.1-07
DOC 3.1.1, pg 1 of 6 through DOC 3.1.1, pg 6 of 6	DOC 3.1.1, pg 1 of 5 through DOC 3.1.1, pg 5 of 5	02/05/99	RAI 3.1-01
DOC 3.1.4, pg 4 of 7	DOC 3.1.4, pg 4 of 7	02/05/99	Tech change
DOC 3.1.5, pg 1 of 5 through DOC 3.1.5, pg 5 of 5	DOC 3.1.5, pg 1 of 4 through DOC 3.1.5, pg 4 of 4	02/05/99	RAI 3.1-07

ATTACHMENT 4 TO ITS CONVERSION SUBMITTAL

NSHC 3.1.1, pg 1 of 3 through NSHC 3.1.1, pg 3 of 3	NSHC 3.1.1, pg 1 of 1	02/05/99	RAI 3.1-01

ATTACHMENT 5 TO ITS CONVERSION SUBMITTAL

NUREG 3.1-1	NUREG 3.1.1	02/05/99	RAI 3.1-01
NUREG 3.1-1 insert	- - - - -	02/05/99	RAI 3.1-01
NUREG 3.1-8	NUREG 3.1-8	02/05/99	editorial
NUREG 3.1-11 insert	NUREG 3.1.11 insert	02/05/99	editorial
NUREG 3.1-12	NUREG 3.1-12	02/05/99	editorial
NUREG 3.1-13	NUREG 3.1-13	02/05/99	RAI 3.1-07
NUREG 3.1-15	NUREG 3.1-15	02/05/99	Tech change
NUREG B 3.1-2	NUREG B 3.1-2	02/05/99	editorial
NUREG B 3.1-3	NUREG B 3.1-3	02/05/99	editorial
NUREG B 3.1-4	NUREG B 3.1-4	02/05/99	RAI 3.1-01
NUREG B 3.1-5	NUREG B 3.1-5	02/05/99	RAI 3.1-01
NUREG B 3.1-6 insert	NUREG B 3.1-6 insert	02/05/99	RAI 3.1-01
NUREG B 3.1-24 insert	NUREG B 3.1-24 insert	02/05/99	editorial
NUREG B 3.1-26	NUREG B 3.1-26	02/05/99	editorial
NUREG B 3.1-26 insert	NUREG B 3.1-26 insert	02/05/99	editorial
NUREG B 3.1-31 insert	NUREG B 3.1-31 insert	02/05/99	editorial
NUREG B 3.1-32	NUREG B 3.1-32	02/05/99	Tech change
NUREG B 3.1-34	NUREG B 3.1-34	02/05/99	RAI 3.1-04
NUREG B 3.1-34 insert	NUREG B 3.1-34 insert	02/05/99	Tech change
NUREG B 3.1-35	NUREG B 3.1-35	02/05/99	editorial
NUREG B 3.1-36 insert (#1 - 6, 2 pgs)	NUREG B 3.1-36 insert (#1-6, 1 pg)	02/05/99	RAI 3.1-04 RAI 3.1-07

ATTACHMENT 5 TO ITS CONVERSION SUBMITTAL (continued)

NUREG B 3.1-37	NUREG B 3.1-37	02/05/99	RAI 3.1-07
NUREG B 3.1-37 insert	NUREG B 3.1-37 insert	02/05/99	RAI 3.1-07
NUREG B 3.1-39	NUREG B 3.1-39	02/05/99	editorial
NUREG B 3.1-42 insert	NUREG B 3.1-42 insert	02/05/99	RAI 3.1-10
NUREG B 3.1-43	NUREG B 3.1-43	02/05/99	editorial

ATTACHMENT 6 TO ITS CONVERSION SUBMITTAL

JFD 3.1.1, pg 2 of 3	JFD 3.1.1, pg 2 of 3	02/05/99	RAI 3.1-01
JFD 3.1.5, pg 2 of 7	JFD 3.1.5, pg 2 of 7	02/05/99	Tech change
JFD 3.1.6, pg 2 of 4	JFD 3.1.6, pg 2 of 4	02/05/99	Tech change
JFD 3.1.6, pg 4 of 4	JFD 3.1.6, pg 4 of 4	02/05/99	RAI 3.1-07
JFD 3.1.7, pg 4 of 5	JFD 3.1.7, pg 4 of 5	02/05/99	RAI 3.1-10

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be within the limits specified in the COLR.

APPLICABILITY: MODE 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify SDM to be within limits.	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Alignment

LCO 3.1.4 All control rods, including their position indication channels, shall be OPERABLE and aligned to within 8 inches of all other rods in their respective group, and the control rod position deviation alarm shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel of rod position indication inoperable for one or more control rods.	A.1 Perform SR 3.1.4.1 (rod position verification).	Once within 15 minutes following any rod motion in that group
B. Rod position deviation alarm inoperable.	B.1 Perform SR 3.1.4.1 (rod position verification).	Once within 15 minutes of movement of any control rod
C. One control rod misaligned by > 8 inches.	C.1 Perform SR 3.2.2.1 (peaking factor verification). <u>OR</u> C.2 Reduce THERMAL POWER to ≤ 75% RTP.	2 hours 2 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One full-length control rod immovable, but trippable.	D.1 Restore control rod to OPERABLE status.	Prior to entering MODE 2 from MODE 3
E. Required Action and associated Completion Time not met. <u>OR</u> One or more control rods inoperable for reasons other than Condition D. <u>OR</u> Two or more control rods misaligned by > 8 inches. <u>OR</u> Both rod position indication channels inoperable for one or more control rods.	E.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify the position of each control rod to be within 8 inches of all other control rods in its group.	12 hours
SR 3.1.4.2	Perform a CHANNEL CHECK of the control rod position indication channels.	12 hours
SR 3.1.4.3	Verify control rod freedom of movement by moving each individual full-length control rod that is not fully inserted into the reactor core $\geq$ 6 inches in either direction.	92 days
SR 3.1.4.4	Verify the rod position deviation alarm is OPERABLE.	18 months
SR 3.1.4.5	Perform a CHANNEL CALIBRATION of the control rod position indication channels.	18 months
SR 3.1.4.6	Verify each full-length control rod drop time is $\leq$ 2.5 seconds.	Prior to reactor criticality, after each reinstallation of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Shutdown and Part-Length Control Rod Group Insertion Limits

LCO 3.1.5 All shutdown and part-length rod groups shall be withdrawn to ≥ 128 inches.

APPLICABILITY: MODE 1,
MODE 2 with any regulating rod withdrawn above 5 inches.

---------------------------NOTE---------------------------
This LCO is not applicable while performing SR 3.1.4.3 (rod exercise test).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more shutdown or part-length rods not within limit.	A.1 Declare affected control rod(s) inoperable and enter the applicable Conditions and Required Actions of LCO 3.1.4.	Immediately
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each shutdown and part-length rod group is withdrawn ≥ 128 inches.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Regulating Rod Group Position Limits

LCO 3.1.6 The Power Dependent Insertion Limit (PDIL) alarm circuit and the Control Rod Out Of Sequence (CROOS) alarm circuit shall be OPERABLE, and the regulating rod groups shall be limited to the withdrawal sequence, overlap, and insertion limits specified in the COLR.

APPLICABILITY: MODES 1 and 2.

---------------------------NOTE---------------------------
This LCO is not applicable while performing SR 3.1.4.3 (rod exercise test).

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Regulating rod groups inserted beyond the insertion limit.	A.1	Restore regulating rod groups to within limits.	2 hours
	<u>OR</u>		
	A.2	Reduce THERMAL POWER to less than or equal to the fraction of RTP allowed by the regulating rod group position and insertion limits specified in the COLR.	2 hours

BASES

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption that the control rod of highest reactivity worth is fully withdrawn following a reactor trip. For MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (Departure from Nucleate Boiling Ratio (DNBR), fuel centerline temperature limit AOOs, and $\leq$ 280 cal/gm energy deposition for the control rod ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements are based on a Main Steam Line Break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected Steam Generator (SG), and consequently the PCS. This results in a reduction of the primary coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. The most limiting MSLB with respect to potential fuel damage is a guillotine break of a main steam line initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating PCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

BASES

APPLICABLE SAFETY ANALYSES (continued)

In addition to the limiting MSLB transient, the SDM requirement for MODES 3 and 4 must also protect against an inadvertent boron dilution; (Ref. 3) and an uncontrolled control rod bank withdrawal from subcritical conditions (Ref. 5).

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the PCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life when critical boron concentrations are highest.

The withdrawal of a control rod bank from subcritical conditions adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The withdrawal of control rod banks also produce a time dependent redistribution of core power.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled control rod banks withdrawal transient is terminated by either a high power trip or a high pressurizer pressure trip. In all cases, power level, PCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2).

LCO

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the value for SDM. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, then the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

SDM is a core physics design condition that can be ensured through full-length control rod positioning (regulating and shutdown rods) and through the soluble boron concentration.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

SDM is verified by a reactivity balance calculation, considering the listed reactivity effects:

a. PCS boron concentration;

b. Control rod positions;

c. PCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration; and

f. Isothermal Temperature Coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical and the fuel temperature will be changing at the same rate as the PCS.

Samarium is not considered in the reactivity analysis since the analysis assumes that the negative reactivity due to Samarium is offset by the positive reactivity of Plutonium built in.

SR 3.1.1.1 requires SDM to be within the limits specified in the COLR. This SDM value ensures the consequences of an MSLB, will be acceptable as a result of a cooldown of the PCS which adds positive reactivity in the presence of a negative moderator temperature coefficient as well as the other events described in the Applicable Safety Analysis. As such, the requirements of this SR must be met whenever the plant is in MODES 3, 4, and 5.

The Frequency of 24 hours for the verification of SDM is based on the generally slow change in required boron concentration, and also allows sufficient time for the operator to collect the required data, which may include performing a boron concentration analysis, and completing the calculation.

BASES

REFERENCES

1. FSAR, Section 5.1
2. FSAR, Section 14.14
3. FSAR, Section 14.3
4. 10 CFR 100
5. FSAR, Section 14.2

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Control Rod Alignment

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and regulating rods is an initial assumption in all safety analyses that assume full-length control rod insertion upon reactor trip. Maximum control rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The Palisades Nuclear Plant design criteria contain the applicable criteria for these reactivity and power distribution design requirements (Ref. 1).

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod misalignment may cause increased power peaking, due to the asymmetric reactivity distribution, and a reduction in the total available control rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment and OPERABILITY have been established, and all control rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Control rods are moved by their Control Rod Drive Mechanisms (CRDMs). Each CRDM moves its rod at a fixed rate of approximately 46 inches per minute. Although the ability to move a full-length control rod by its drive mechanism is not an initial assumption used in the safety analyses, it is required to support OPERABILITY. As such, the inability to move a full-length control rod results in that full-length control rod being inoperable.

BASES

BACKGROUND (continued)

The control rods are arranged into groups that are radially symmetric. Therefore, movement of the control rod groups does not introduce radial asymmetries in the core power distribution. The shutdown and regulating rods provide the required reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating rods also provide reactivity (power level) control during normal operation and transients.

The axial position of shutdown and regulating rods is indicated by two separate and independent systems, which are 1) synchro based position indication system, and 2) the reed switch based position indication system.

The synchro based position indication system measures the phase angle of a synchro geared to the CRDM rack. Full control rod travel corresponds to less than 1 turn of the synchro. Each control rod has its own synchro. The Primary Information Processor (PIP) node scans and converts synchro outputs into inches of control rod withdrawal. The resolution of this system is approximately 0.5 inches. Each synchro also has cam operated limit switches which can provide positive indication of control rod position.

The reed switch based position indication system is referred to as the Secondary Position Indication (SPI) system. This system provides a highly accurate indication of actual control rod position, but at a lower precision than the synchros. The reed switches are wired so that the voltage read across the reed switch stack is proportional to rod position. The reed switches are spaced along a tube with a center to center spacing distance of 1.5 inches. The resolution of the SPI reed switch stacks is 1.5 inches. The reed switches also provide input to the matrix indication lights which provide control rod status indication for various key positions. To increase the reliability of the system, there are redundant reed switches which prevent false indication in the event an individual reed switch fails.

A control rod position deviation alarm is provided to alert the operator when any two control rods in the same group are more than 8 inches apart. This helps to ensure any control rod misalignments are minimized. The alarm can be generated by either the SPI system or PIP node since the SPI system, in conjunction with the host computer, is redundant to the PIP node in the task of control rod measurements, control rod monitoring, and limit processing.

BASES

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Refs. 3 and 4). The accident analysis defines control rod misoperation as any event, with the exception of sequential group withdrawals, which could result from a single malfunction in the reactivity control systems. For example, control rod misalignment may be caused by a malfunction of the Rod Control System, or by operator error. A stuck rod may be caused by mechanical jamming. Inadvertent withdrawal of a single control rod may be caused by an electrical or mechanical failure in the Rod Control System. A dropped control rod could be caused by an electrical or mechanical failure in the CRDM.

The acceptance criteria for addressing control rod inoperability/misalignment are that:

a. There shall be no violations of:

    1. Specified Acceptable Fuel Design Limits (SAFDL), or

    2. Primary Coolant System (PCS) pressure boundary integrity; and

b. The core must remain subcritical after accident transients.

Three types of misoperations are discussed in the safety analysis (Ref. 4). During movement of a group, one control rod may stop moving while the other control rods in the group continue. This condition may cause excessive power peaking. The second type of misoperations occurs if one control rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the remaining control rods to meet the SDM requirement with the maximum worth rod stuck fully withdrawn. If a control rod is stuck in the fully withdrawn position, its worth is added to the SDM requirement, since the safety analysis does not take two stuck rods into account. The third type of misoperations occurs when one rod drops partially or fully into the reactor core. This event causes an initial power reduction followed by a return towards the original power, due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in excessive local Linear Heat Rates (LHRs).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The most limiting static misalignment occurs when Bank 4 is fully inserted with one rod fully withdrawn (Bank 4 is 99 inches out of alignment with the rated Power Dependent Insertion Limit (PDIL). This event was bounded by the dropped full-length control rod event (Ref. 4).

Since the control rod drop incidents result in the most rapid approach to SAFDLs caused by a control rod misoperation, the accident analysis analyzed a single full-length control rod drop.

The above control rod misoperations may or may not result in an automatic reactor trip. In the case of the full-length rod drop, a prompt decrease in core average power and a distortion in radial power are initially produced, which, when conservatively coupled, result in a local power and heat flux increase, and a decrease in DNBR parameters.

The results of the control rod misoperation analysis show that during the most limiting misoperation events, no violations of the SAFDLs, fuel centerline temperature, or PCS pressure occur.

Control rod alignment satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2).

LCO

The limits on shutdown, regulating, and part-length rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the full-length control rods will be available and will be inserted to provide enough negative reactivity to shut down the reactor. The OPERABILITY requirements also ensure that the control rod banks maintain the correct alignment and that each full-length control rod is capable of being moved by its CRDM. The OPERABILITY requirement for the part-length rods is that they are fully withdrawn.

BASES

LCO (continued)

The requirement is to maintain the control rod alignment to within 8 inches between any control rod and all other rods in its group. To help ensure this requirement is met, the control rod position deviation alarm generated by either the PIP node or the SPI system, must be OPERABLE and provide an alarm when any control rod becomes misaligned > 8 inches from any other rod in its group. The safety analysis assumes a total misalignment from fully withdrawn to fully inserted. This case bounds the safety analysis for a single rod in any intermediate position.

The primary rod position indication system is considered OPERABLE, for purposes of this specification, if the digital position readout, the PPC display, or the cam operated position indication lights give positive indication of rod position. The secondary rod position indication system is considered OPERABLE if the magnetically operated reed switches are providing positive indication of rod position either via the plant process computer or taking direct readings of the output from the magnetic reed switches.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDM, any of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on control rod OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (e.g., trippability) and alignment of control rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and regulating rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the PCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5, and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

BASES

ACTIONS (continued)

D.1

Condition D is entered whenever it is discovered that a single full-length control rod can not be moved by its operator yet the control rod is still capable of being tripped. Although the ability to move a full-length control rod is not an initial assumption used in the safety analyses, it does relate to full-length control rod OPERABILITY. The inability to move a full-length control rod by its operator may be indicative of a systemic failure (other than trippability) which could potentially affect other rods. Thus, declaring a full-length control rod inoperable in this instance is conservative since it limits the number of full-length control rods which can not be moved by their operators to only one. The Completion Time to restore an inoperable control rod to OPERABLE status is stated as prior to entering MODE 2 from MODE 3. This Completion Time allows unrestricted operation in MODES 1 and 2 while conservatively preventing a reactor startup with an immovable full-length control rod.

E.1

If the Required Action or associated Completion Time of Condition A, Condition B, Condition C, or Condition D is not met; one or more control rods are inoperable for reasons other than Condition D; or two or more control rods are misaligned by > 8 inches, or two channels of control rod position indication are inoperable for one or more control rods, the plant is required to be brought to MODE 3. By being brought to MODE 3, the plant is brought outside its MODE of applicability. Continued operation is not allowed in the case of more than one control rod misaligned from any other rod in its group by > 8 inches, or two or more rods inoperable. This is because these cases may be indicative of a loss of SDM and power re-distribution, and a loss of safety function, respectively.

Also, if no rod position indication exists for one or more control rods, continued operation is not allowed because the safety analysis assumptions of rod position cannot be ensured.

BASES

ACTIONS

E.1 (continued)

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual control rod positions are within 8 inches of all other control rods in the group at a 12 hour Frequency allows the operator to detect a control rod that is beginning to deviate from its expected position. The specified Frequency takes into account other control rod position information that is continuously available to the operator in the control room, so that during control rod movement, deviations can be detected. Also protection can be provided by the control rod deviation alarm.

SR 3.1.4.2

OPERABILITY of two control rod position indicator channels is required to determine control rod positions, and thereby ensure compliance with the control rod alignment and insertion limits. Performance of a CHANNEL CHECK on the primary and secondary control rod position indication channels provides confidence in the accuracy of the rod position indication systems. The control rod "full in" and "full out" lights, which correspond to the lower electrical limit and the upper electrical limit respectively, provide an additional means for determining the control rod positions when the control rods are at either their fully inserted or fully withdrawn positions.

The 12 hour Frequency takes into consideration other information continuously available to the operator in the control room, so that during control rod movement, deviations can be detected, and protection can be provided by the control rod deviation alarm.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.3

Verifying each full-length control rod is trippable would require that each full-length control rod be tripped. In MODES 1 and 2, tripping each full-length control rod would result in radial or axial power tilts, or oscillations. Therefore, individual full-length control rods are exercised every 92 days to provide increased confidence that all full-length control rods continue to be trippable, even if they are not regularly tripped. A movement of 6 inches is adequate to demonstrate motion without exceeding the alignment limit when only one control rod is being moved. The 92 day Frequency takes into consideration other information available to the operator in the control room and other surveillances being performed more frequently, which add to the determination of OPERABILITY of the control rods. At any time, if a control rod(s) is inoperable, a determination of the trippability of the control rod(s) must be made, and appropriate action taken.

SR 3.1.4.4

Demonstrating the rod position deviation alarm is OPERABLE verifies the alarm is functional. The 92 day Frequency takes into account other information continuously available to the operator in the control room, so that during control rod movement, deviations can be detected.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.5

Performance of a CHANNEL CALIBRATION of each control rod position indication channel ensures the channel is OPERABLE and capable of indicating control rod position over the entire length of the control rod's travel with the exception of the secondary rod position indicating channel dead band near the bottom of travel. This dead band exists because the control rod drive mechanism housing seismic support prevents operation of the reed switches. Since this Surveillance must be performed when the reactor is shut down, an 18 month Frequency to be coincident with refueling outage was selected. Operating experience has shown that these components usually pass this Surveillance when performed at a Frequency of once every 18 months. Furthermore, the Frequency takes into account other surveillances being performed at shorter Frequencies, which determine the OPERABILITY of the control rod position indicating systems.

SR 3.1.4.6

Verification of full-length control rod drop times determines that the maximum control rod drop time is consistent with the assumed drop time used in that safety analysis (Ref. 2). The 2.5 second acceptance criteria is measured from the time the CRDM clutch is deenergized by the reactor protection system or test switch to 90% insertion. This time is bounded by that assumed in the safety analysis (Ref.2). Measuring drop times prior to reactor criticality, after reactor vessel head reinstallation, ensures that reactor internals and CRDMs will not interfere with full-length control rod motion or drop time and that no degradation in these systems has occurred that would adversely affect full-length control rod motion or drop time. Individual full-length control rods whose drop times are greater than safety analysis assumptions are not OPERABLE. This SR is performed prior to criticality, based on the need to perform this Surveillance under the conditions that apply during a plant outage and because of the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown and Part-Length Rod Group Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown rods are initial assumptions in all safety analyses that assume full-length control rod insertion upon reactor trip. The insertion limits directly affect core power distributions and assumptions of available SDM, ejected rod worth, and initial reactivity insertion rate.

The Palisades Nuclear Plant design criteria (Ref. 1) and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," contain the applicable criteria for these reactivity and power distribution design requirements. Limits on shutdown rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the reactivity limits, ejected rod worth, and SDM limits are preserved.

The shutdown rods are arranged into groups that are radially symmetric. Therefore, movement of the shutdown rod groups does not introduce radial asymmetries in the core power distribution. The shutdown and regulating rod groups provide the required reactivity worth for immediate reactor shutdown upon a reactor trip.

The Palisades Nuclear Plant has four part-length control rods installed. The part-length rods are required to remain completely withdrawn during power operation except during rod exercising performed in conjunction with SR 3.1.4.3. The part-length rods do not insert on a reactor trip.

The design calculations are performed with the assumption that the shutdown rod groups are withdrawn prior to the regulating rod groups. The shutdown rods can be fully withdrawn without the core going critical. This provides available negative reactivity for SDM in the event of boration errors. All control rod groups are controlled manually by the control room operator. During normal plant operation, the shutdown rod groups are fully withdrawn. The shutdown rod groups must be completely withdrawn from the core prior to withdrawing any regulating rods during an approach to criticality. The shutdown rod groups are then left in this position until the reactor is shut down.

BASES

BACKGROUND (continued)	They affect core power, burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

APPLICABLE SAFETY ANALYSES

Accident analysis assumes that the shutdown rod groups are fully withdrawn any time the reactor is critical. This ensures that:

a. The minimum SDM is maintained; and

b. The potential effects of a control rod ejection accident are limited to acceptable limits.

Control rods are considered fully withdrawn at 128 inches, since this position places them in an insignificant reactivity worth region of the integral worth curve for each bank.

On a reactor trip, all full-length control rods (shutdown and regulating), except the most reactive rod, are assumed to insert into the core. The shutdown and regulating rod groups shall be at or above their insertion limits and available to insert the required amount of negative reactivity on a reactor trip signal. The regulating rods may be partially inserted in the core as allowed by LCO 3.1.6, "Regulating Rod Group Position Limits." The shutdown rod group insertion limit is established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)) following a reactor trip from full power. The combination of regulating rod and shutdown rods (less the most reactive rod, which is assumed to remain fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 2). The shutdown rod group insertion limit also limits the reactivity worth of an ejected shutdown rod.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The acceptance criteria for addressing shutdown rods as well as regulating rod insertion limits and inoperability or misalignment are that:

a. There be no violation of:

   1. Specified acceptable fuel design limits, or

   2. Primary Coolant System pressure boundary damage; and

b. The core remains subcritical after accident transients.

As such, the shutdown and part-length rod group insertion limits affect safety analyses involving core reactivity, ejected rod worth, and SDM (Ref. 2). The part-length control rods have the potential to cause power distribution envelopes to be exceeded if inserted while the reactor is critical. Therefore, they must remain withdrawn in accordance with the limits of the LCO (Ref. 3).

The shutdown and part-length rod group insertion limits satisfy Criterion 2 of 10 CFR 50.36(c)(2).

LCO

The shutdown and part-length rod groups must be within their insertion limits any time the reactor is critical or approaching criticality. For a control rod group to be considered above its insertion limit, all rods in that group must be above the insertion limit. Maintaining the shutdown rod groups within their insertion limits ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. Maintaining the part-length rod group within its insertion limit ensures that the power distribution envelope is maintained.

BASES

APPLICABILITY

The shutdown and part-length rod groups must be within their insertion limits, with the reactor in MODES 1 and 2. In MODE 2 the Applicability begins anytime any regulating rod is withdrawn above 5 inches. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. In MODE 4, 5, or 6, the shutdown rod groups are inserted in the core to at least the lower electrical limit and contribute to the SDM. In MODE 3 the shutdown rod groups may be withdrawn in preparation of a reactor startup. Refer to LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

The Applicability has been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.3 (rod exercise test). Control rod exercising verifies the freedom of the rods to move, and requires the individual shutdown rods to move below the LCO limits for their group. Only the full-length rods are required to be tested by SR 3.1.4.3. The part-length rods may also be moved however, if a part-length rod is moved below the limit of the associated LCO, the Required Actions of Condition A must be taken.

Positioning of an individual control rod within its group is addressed by LCO 3.1.4, "Control Rod Alignment."

BASES

ACTIONS

A.1

Prior to entering this condition, the shutdown and part-length rod groups were fully withdrawn. If a shutdown rod group is then inserted into the core, its potential negative reactivity is added to the core as it is inserted.

If one or more shutdown or part-length rods are not within limits, the affected rod(s) must be declared inoperable and the applicable Conditions and Required Actions of LCO 3.1.4 entered immediately. This Required Action is based on the recognition that the shutdown and part-length rods are normally withdrawn beyond their insertion limits and are capable of being moved by their control rod drive mechanism. Although the requirements of this LCO are not applicable during performance of the control rod exercise test, the inability to restore a control rod to within the limits of the LCO following rod exercising would be indicative of a problem affecting the OPERABILITY of the control rod. Therefore, entering the applicable Conditions and Required Actions of LCO 3.1.4 is appropriate since they provide the applicable compensatory measures commensurate with the inoperability of the control rod.

B.1

When Required Action A.1 cannot be met or completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown and part-length rod groups are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown rods will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. Verification that the part-length rod groups are within their insertion limits ensures that they do not adversely affect power distribution requirements. This SR and Frequency ensure that the shutdown and part-length rod groups are withdrawn before the regulating rods are withdrawn during a plant startup.

Since control rod groups are positioned manually by the control room operator, verification of shutdown and part-length rod group position at a Frequency of 12 hours is adequate to ensure that the shutdown and part-length rod groups are within their insertion limits. Also, the 12 hour Frequency takes into account other information available to the operator in the control room for the purpose of monitoring the status of the shutdown and part-length rod groups.

REFERENCES

1. FSAR, Section 5.1
2. FSAR, Section 14.2
3. FSAR, Section 14.6

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Regulating Rod Group Position Limits

BASES

BACKGROUND

The insertion limits of the regulating rod groups are initial assumptions in all safety analyses that assume full-length rod insertion upon reactor trip. The insertion limits directly affect core power distributions, assumptions of available SDM, and initial reactivity insertion rate. The applicable criteria for these reactivity and power distribution design requirements are contained in the Palisades Nuclear Plant design criteria (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Limits on regulating rod group insertion have been established, and all regulating rod group positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking, ejected rod worth, reactivity insertion rate, and SDM limits are preserved.

The regulating rod groups operate with a predetermined amount of position overlap, in order to approximate a linear relation between rod worth and rod position (integral rod worth). The regulating rod groups are withdrawn and operate in a predetermined sequence. The group sequence and overlap limits are specified in the COLR.

The regulating rods are used for precise reactivity control of the reactor. The positions of the regulating rods are manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 2). Together, LCO 3.1.6; LCO 3.2.3, "QUADRANT POWER TILT (T_q)"; and LCO 3.2.4, "AXIAL SHAPE INDEX (ASI)," provide limits on control component operation and on monitored process variables to ensure the core operates within the linear heat rate (LCO 3.2.1, "Linear Heat Rate (LHR)") and radial peaking factor F_R^T and F_R^A (LCO 3.2.2, "Radial Peaking Factors) limits in the COLR.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Fuel cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result, should an accident occur with simultaneous violation of one or more of these LCOs. Changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

The SDM requirement is ensured by limiting the regulating and shutdown rod group insertion limits, so that the allowable inserted worth of the rods is such that sufficient reactivity is available to shut down the reactor to hot zero power. SDM assumes the maximum worth rod remains fully withdrawn upon trip (Ref. 4).

The most limiting SDM requirements for Mode 1 and 2 conditions at Beginning of Cycle (BOC) are determined by the requirements of several transients, e.g., Loss of Flow, etc. However, the most limiting SDM requirements for MODES 1 and 2 at End of Cycle (EOC) come from just one transient, Main Steam Line Break (MSLB). The requirements of the MSLB event at EOC for the full power and no load conditions are significantly larger than those of any other event at that time in cycle and, also, considerably larger than the most limiting requirements at BOC.

Although the most limiting SDM requirements at EOC are much larger than those at BOC, the available SDMs obtained via tripping the full-length control rods are substantially larger due to the much lower boron concentration at EOC. To verify that adequate SDMs are available throughout the cycle to satisfy the changing requirements, calculations are performed at both BOC and EOC. It has been determined that calculations at these two times in cycle are sufficient since the difference between available SDMs and the limiting SDM requirements are the smallest at these times in cycle. The measurement of full-length control rod bank worth performed as part of the Startup Testing Program demonstrates that the core has the expected shutdown capability. Consequently, adherence to LCO 3.1.5, "Shutdown and Part-Length Rod Group Insertion Limits," and LCO 3.1.6 provides assurance that the available SDM at any time in cycle will exceed the limiting SDM requirements at that time in cycle.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Operation at the insertion limits or ASI limits may approach the maximum allowable linear heat generation rate or peaking factor, with the allowed T_q present. Operation at the insertion limit may also indicate the maximum ejected rod worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected rod worth.

The regulating and shutdown rod insertion limits ensure that safety analyses assumptions for reactivity insertion rate, SDM, ejected rod worth, and power distribution peaking factors are preserved.

The regulating rod group position limits satisfy Criterion 2 of 10 CFR 50.36(c)(2).

LCO

The limits on regulating rod group sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is maintained, and ensuring adequate negative reactivity insertion on trip. The overlap between regulating rod groups provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during regulating rod group motion. For a control rod group to be considered above its insertion limit, all rods in that group must be above the insertion limit.

The Power Dependent Insertion Limit (PDIL) alarm circuit is required to be OPERABLE for notification that the regulating rod groups are outside the required insertion limits. The Control Rod Out Of Sequence (CROOS) alarm circuit is required to be OPERABLE for notification that the rods are not within the required sequence and overlap limits. When the PDIL or the CROOS alarm circuit is inoperable, the verification of rod group positions is increased to ensure improper rod alignment is identified before unacceptable flux distribution occurs. The PDIL and CROOS alarms can be generated by either the synchro based Primary Indication Processor (PIP) node, or the reed switch based Secondary Position Indication (SPI) system since the SPI system, in conjunction with the host computer, is redundant to the PIP node in the task of control rod measurement, control rod monitoring and limit processing.

BASES

APPLICABILITY

The regulating rod group sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES. SDM is preserved in MODES 3, 4, and 5 by adjustments to the soluble boron concentration.

The Applicability has been modified by a Note indicating the LCO requirement is suspended SR 3.1.4.3 (rod exercise test). Control rod exercising verifies the freedom of the rods to move, and requires the individual regulating rods to move below the LCO limits which could violate the LCO for their group.

ACTIONS

A.1 and A.2

Operation beyond the insertion limit may result in a loss of SDM and excessive peaking factors. The insertion limit should not be violated during normal operation; this violation, however, may occur during transients when the operator is manually controlling the regulating rods in response to changing plant conditions.

When the regulating groups are inserted beyond the insertion limits, actions must be taken to either withdraw the regulating groups beyond the limits or to reduce THERMAL POWER to less than or equal to that allowed for the actual rod group position limit. Two hours provides a reasonable time to accomplish this, allowing the operator to deal with current plant conditions while limiting peaking factors to acceptable levels.

BASES

ACTIONS (continued)

B.1

Operating outside the regulating rod group sequence and overlap limits specified in the COLR may result in excessive peaking factors. If the sequence and overlap limits are exceeded, the regulating rod groups must be restored to within the appropriate sequence and overlap. Two hours provides adequate time for the operator to restore the regulating rod group to within the appropriate sequence and overlap limits.

C.1

When the PDIL or the CROOS alarm circuit is inoperable, performing SR 3.1.6.1 once within 15 minutes following any rod motion ensures improper rod alignments are identified before unacceptable flux distributions occur.

D.1

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

With the PDIL alarm circuit OPERABLE, verification of each regulating rod group position every 12 hours is sufficient to detect rod positions that may approach the acceptable limits, and to provide the operator with time to undertake the Required Action(s) should the sequence or insertion limits be found to be exceeded.

The 12 hour Frequency also takes into account the indication provided by the PDIL alarm circuit and other information about rod group positions available to the operator in the control room.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.6.2

Demonstrating the PDIL alarm circuit OPERABLE verifies that the PDIL alarm circuit is functional. The 31 day Frequency takes into account other Surveillances being performed at shorter Frequencies that identify improper control rod alignments.

SR 3.1.6.3

Demonstrating the CROOS alarm circuit OPERABLE verifies that the CROOS alarm circuit is functional. The 31 day Frequency takes into account other Surveillances being performed at shorter Frequencies that identify improper control rod alignment.

REFERENCES

1. FSAR, Section 5.1
2. 10 CFR 50.46
3. FSAR, Section 14.16
4. FSAR, Section 14.4

REACTIVITY CONTROL SYSTEMS (A.1)

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

~~Applicability~~

~~Applies to operation of CONTROL RODS and hot channel factors during operation.~~ (A.1)

~~Objective~~

~~To specify limits of CONTROL ROD movement to assure an acceptable power distribution during power operation, limit worth of individual rods to values analyzed for accident conditions, maintain adequate shutdown margin after a reactor trip and to specify acceptable power limits for power tilt conditions.~~

(A.6)

Specifications

3.1.1 Shutdown Margin Requirements LA.1 (A.2)

a. With four primary coolant pumps in operation at hot shutdown and above, the shutdown margin shall be ≥ 2%.

LA.1

b. With less than four primary coolant pumps in operation at hot shutdown and above, boration shall be immediately initiated to increase and maintain the shutdown margin at ≥3.75%. (A.5) RA A.1

LA.1

(A.3) c. At less than the hot shutdown condition, with at least one primary coolant pump in operation or at least one shutdown cooling pump in operation, with a flow rate ≥2810 gpm, the boron concentration shall be greater than the cold shutdown boron concentration for normal cooldowns and heatups, ie, non-emergency conditions. (A.4)

~~During non-emergency conditions, at less than the hot shutdown condition with no operating primary coolant pumps and a primary system recirculating flow rate < 2810 gpm but ≥ 650 gpm, then within one hour either:~~

~~1. (a) Establish a shutdown margin of ≥ 3.5% and~~

~~(b) Assure two of the three charging pumps are electrically disabled.~~

~~OR~~

~~2. At least every 15 minutes verify that no charging pumps are operating. If one or more charging pumps are determined to be operating in any 15 minute surveillance period, terminate charging pump operation and insure that the shutdown margin requirements are met and maintained.~~

(See 3.4)

(A.6) — < ADD LCO & Applicability >

(M.1) — < ADD RA.A.1 >

(M.2) - < ADD SR 3.1.1.1 and FREQ >

Amendment No. ~~31, 43, 57, 68, 70, 118~~, 162
October 26, 1994

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02/05/99

A.1

3.~~10~~1 ~~CONTROL ROD AND POWER DISTRIBUTION LIMITS~~ Reactivity Control System

3.~~10~~.4 ~~Misaligned or Inoperable CONTROL ROD or Part-Length Rod~~ Control Rod Alignment

LCO 3.1.4

a. A CONTROL ROD or a part-length rod is considered misaligned if it is out of position from the remainder of the bank by more than 8 inches, and the rod position deviation alarm shall be Operable. M.1

b. A CONTROL ROD is considered inoperable if it cannot be moved by its operator or if it cannot be tripped. A part-length rod is considered inoperable if it is not fully withdrawn from the core and cannot be moved by its operator. LA.2 If more than one CONTROL ROD or part-length rod becomes misaligned or inoperable, the reactor shall be placed in the ~~hot shutdown~~ MODE 3 condition within ~~12~~ 6 hours. M.2

RA.E

c. If a CONTROL ROD or a part-length rod is misaligned, hot channel factors must promptly be shown to be within design limits or reactor power shall be reduced to 75% or less of rated power within two hours. A.6 ~~In addition, shutdown margin and individual rod worth limits must be met. Individual rod worth calculations will consider the effects of xenon redistribution and reduced fuel burnup in the region of the misaligned CONTROL ROD or part-length rod.~~ A.3

RA. C.1 C.2

~~3.10.5 Regulating Group Insertion Limits~~

~~a. The regulating groups shall be limited to the withdrawal sequence, overlap, and insertion limits specified in the COLR.~~

~~b. With any regulating group inserted beyond its limit,~~

~~1. Restore all regulating groups to within insertion limit within 2 hours.~~

(See 3.1.6)

A.2 —— <ADD Applicability>

M.3 —— <ADD RA B>

M.4 <ADD SR 3.1.4.4>

M.6 <ADD RA D>

Amendment No. ~~21~~, ~~31~~, ~~162~~, 169
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4.2 EQUIPMENT SAMPLING AND TESTS

A.1

Table 4.2.2

Minimum Frequencies for Equipment Tests

		Test	Frequency	FSAR Section REFERENCE
SR3.1.4.6	1. CONTROL RODS	Drop Times of All Full Length Rods	Prior to reactor Criticality after each removal of the ~~Refueling~~ reactor head (A.1)	~~7.6.1.3~~ (A.1)
SR 3.1.4.3	2. CONTROL RODS	Partial Movement of all Rods (Minimum of 6 In)	Every 92 Days	~~7.6.1.3~~
(See 3.4)	~~3. Pressurizer Safety Valves~~	~~Set Point~~	~~One Each Refueling~~	~~4.3.7~~
(See 3.7)	~~4. Main Steam Safety Valves~~	~~Set Point~~	~~Five Each Refueling~~	~~4.3.4~~
(See 3.9)	~~5. Refueling System Interlocks~~	~~Functioning~~	~~Prior to Refueling Operations~~	~~9.11.4~~
(See 3.7)	~~6. Service Water System Valve Actuation on SIS and RAS~~	~~Functioning~~	~~Refueling~~	~~9.1.2~~
(See 3.4)	~~7. Primary System Leakage~~	~~Evaluate~~	~~Daily~~	~~4.7.1~~
A.1	~~8. Deleted~~			
R	9. Boric Acid Heat Tracing	Verify proper temperature readings.	Daily	
(See 3.5)	~~10. Safety Injection Tank Level and Pressure~~	~~Verify that level and pressure indication is between independent high high/low alarms for level and pressure.~~	~~Each Shift~~	

Amendment No. ~~12~~, ~~81~~, ~~133~~, ~~152~~, ~~155~~, ~~157~~, ~~162~~,

"ELEC CHANGES"

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3.~~10~~1 ~~CONTROL ROD AND POWER DISTRIBUTION LIMITS~~ Reactivity Control A.1

~~3.10.8~~ Shutdown Rod Limits and Part Length Group Insertion

Applicability/LCO ~~a~~. All shutdown rods shall be withdrawn before any regulating rods are withdrawn above 5 inches. A.6 A.7

b. The shutdown rods shall not be withdrawn until normal water level is established in the pressurizer. LA.1

Applicability/LCO ~~c~~ The shutdown rods shall not be inserted below their exercise limit until all regulating rods are inserted. A.7

3.10.7 / Low Power Physics Testing A.1 (See 3.1.1 3.1.4 3.1.6)

Applic Note Sections 3.10.1.a, 3.10.1.b, 3.10.3, 3.10.4.b, 3.10.5 and 3.10.6 may be deviated from during low power physics testing and CPDM exercises if necessary to perform a test but only for the time necessary to perform the test. (See 3.1.7) A.8

Basis

Sufficient CONTROL RODS shall be withdrawn at all times to assure that the reactivity decrease from a reactor trip provides adequate shutdown margin. The available worth of withdrawn rods must include the reactivity defect of power and the failure of the withdrawn rod of highest worth to insert. The requirement for a shutdown margin of 2.0% in reactivity with 4-pump operation, and of 3.75% in reactivity with less than 4-pump operation, is consistent with the assumptions used in the analysis of accident conditions (including steam line break) as reported in Reference 1 and additional analysis. Requiring the boron concentration to be at cold shutdown boron concentration at less than hot shutdown assures adequate shutdown margin exists to ensure a return to power does not occur if an unanticipated cooldown accident occurs. This requirement applies to normal operating situations and not during emergency conditions where it is necessary to perform operations to mitigate the consequences of an accident. By imposing a minimum shutdown cooling pump flow rate of 2810 gpm, sufficient time is provided for the operator to terminate a boron dilution under asymmetric conditions. For operation with no primary coolant pumps operating and a recirculating flow rate less than 2810 gpm the increased shutdown margin and controls on charging pump operability or alternately the surveillance of the charging pumps will ensure that the acceptance criteria, for an inadvertent boron dilution event will not be violated.[1] The change in insertion limit with reactor power insures that the shutdown requirements for 4-pump operation is met at all power levels. The 2.5-second drop time specified for the CONTROL RODS is the drop time used in the transient analysis.[1] A2

< ADD RA A.1 & B.1 > A.9

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 3.10.1a and 3.10.1b specify requirements for SHUTDOWN MARGIN in terms of "...at hot shutdown and above." In the proposed ITS, MODE 3 is essentially equivalent to the CTS "HOT SHUTDOWN" as specified in the Discussion of Changes for Section 1.0. The "...and above" portion of the CTS "...at hot shutdown and above" applies up through the CTS "HOT STANDBY" and "POWER OPERATIONS" which corresponds to the ITS MODES 1 and 2. The insertion limit requirements of CTS 3.10.5, Shutdown Rod Limits, and CTS 3.10.6, Regulating Group Insertion Limits, ensure that adequate SHUTDOWN MARGIN exists when the plant is at power. Therefore, in the proposed ITS, the required amount of SHUTDOWN MARGIN in MODES 1 and 2 is verified through the Palisades ITS 3.1.5, Shutdown and Part-Length Rod Group Insertion Limits, and 3.1.6, Regulating Rod Group Position Limits. Since the requirements of the CTS are maintained and only restructured to meet the ITS format, these changes are considered to be administrative changes. These changes are consistent with NUREG-1432.

A.3 CTS 3.10.1c specifies SHUTDOWN MARGIN requirements at "less than the hot shutdown condition" (below 525°F). In the proposed ITS this corresponds to MODE 3 <525°F, MODE 4, and MODE 5. The requirements for the refueling condition (MODE 6) are addressed in proposed ITS 3.9.1. This is an administrative change to reflect the NUREG-1432 defined MODES. This change is consistent with the intent of NUREG-1432.

A.4 CTS 3.10.1c includes the statement "...with at least one primary coolant pump in operation or at least one shutdown cooling pump in operation, with a flow rate ≥ 2810 gpm, the boron concentration shall be greater than the cold shutdown boron concentration." In the proposed ITS for operation with Tave < 525°F, SHUTDOWN MARGIN (SDM) will be within the limits specified in the COLR regardless of the primary system flow rate and throughout the temperature range as a cooldown occurs. Overall, this is considered to be an administrative change since the "cold shutdown boron concentration" requirement is replaced by the requirement to have SDM within the limits specified in the COLR throughout the temperature range. This change could be more or less restrictive depending on a particular primary coolant temperature evaluated, however, overall the requirement is considered an administrative "substitution" of one requirement for another while still preserving the SDM requirements.

A.5 CTS 3.10.1b states in part that "...boration shall be immediately initiated to increase and maintain the shutdown margin at...." In the proposed ITS this statement becomes Action A and the term "immediately" is changed to 15 minutes. In the proposed NUREG-1432, the time frame of 15 minutes is used in lieu of "immediately" to specify a specific time in which an action must be started. The terminology conveys the same meaning in the CTS in that quick action must be taken. In NUREG-1432, a Completion Time of "immediately" is defined in Section 1.3 as "pursue continuously in a controlled manner without delay." Therefore, while a Completion Time of "15 minutes" is used in the proposed ITS as compared to the CTS "Immediately" the effective meaning is the same. Therefore, this is considered to be an Administrative Change. This change is consistent with NUREG-1432.

A.6 CTS 3.10.1a, CTS 3.1.10.1b and CTS 3.1.10c contain the requirements for SHUTDOWN MARGIN. The amount of required SHUTDOWN MARGIN is dependent on the plant operating conditions (e.g., above or below hot shutdown) and the number of primary coolant pumps in operation. To establish consistency with the format and style of the ITS, the values of the required SHUTDOWN MARGIN have been moved to the COLR including the plant specific operating conditions and pump configurations (See DOC LA.1) A new LCO statement has been added which states that the SHUTDOWN MARGIN must be within the limits specified in the COLR, and an Applicability of MODES 3, 4, and 5 stipulated. These changes do not alter the actual CTS requirement for SHUTDOWN MARGIN, nor do they impose any additional requirements. These changes merely present the same information in a different format necessary to convert to the ITS. As such, these changes are considered administrative in nature.

MORE RESTRICTIVE CHANGES (M)

M.1 CTS 3.10.1a specifies "With four primary coolant pumps in operation at hot shutdown and above, the shutdown margin shall be 2%." However there is no action specified in the CTS if the shutdown margin is found to be less than 2% and so the plant would have to enter LCO 3.0.3. In the proposed ITS, if the SHUTDOWN MARGIN is found to be below the limit, boration must be initiated within 15 minutes. This is similar to the restoration action specified in CTS 3.10.1b which specifies if shutdown margin is below the required amount that "boration shall be immediately initiated to increase and maintain the shutdown margin." Since in the CTS, LCO 3.0.3 would be have to be entered if the SHUTDOWN MARGIN was found to be below the 2% limit, the 15 minutes to initiate boration is considered to be a more restrictive change. Initiating boration to restore the required amount of SHUTDOWN MARGIN is the appropriate action to take in this situation to return the plant to a safe condition. Furthermore, CTS 3.10.1c does not specify actions to take if flow is ≥ 2810 and the shutdown margin requirements (boron concentration greater than the cold shutdown boron concentration) have not been met. Therefore, if the SHUTDOWN MARGIN was not met, and the plant was above the CTS Cold Shutdown (210°F) then the plant would have to be shutdown in accordance with LCO 3.0.3. In the proposed ITS, ACTION A requires that if the SHUTDOWN MARGIN (SDM) requirement is not within limit, then boration must be initiated within 15 minutes to restore SDM to within limit. Therefore, since the proposed ITS requires that action be taken with 15 minutes, it is considered to be a more restrictive action. This change is consistent with NUREG-1432.

M.2 The Palisades Nuclear Plant CTS does not contain an explicit surveillance requirement for SHUTDOWN MARGIN even though there was a requirement that the limits be met as specified in 3.10.1. Proposed ITS 3.1.1 adds SR 3.1.1.1 to verify SHUTDOWN MARGIN "every 24 hours." Since the requirement to verify SHUTDOWN MARGIN was not explicitly required in the CTS, the addition of the proposed Frequency is considered a "more restrictive" change. This change is consistent with NUREG-1432.

M.3 CTS 3.10.7 includes an exception which allows a deviation from the requirement for shutdown margin during performance of CRDM exercises. Proposed ITS 3.1.1 does not contain this same exception since violation of the LCO is not expected during the performance of the control rod drive exercise surveillance (SR 3.1.4.4). During the performance of SR 3.1.4.4, control rods will be exercised between 6 inches and 8 inches. The change in reactivity as a result of this movement is small due to the relative worth of the control rods which is largely determined by their position in the core at the time this SR is performed. This small change in reactivity is not enough to cause a violation of the Shutdown Margin requirements of ITS 3.1.1. Thus, reliance on the exception contained in CTS 3.10.7 is not needed. This change is consistent with NUREG-1432.

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

LA.1 CTS 3.10.1 contains the requirements for Shutdown Margin including specific values based on plant conditions and configuration. This proposed change relocates the values for Shutdown Margin to the COLR in order to provide core design and operational flexibility that can be used for improved fuel management and to solve plant specific issues. Placing the Shutdown Margin values in the COLR allows the core design to be finalized after shutdown when the actual end of cycle burnup is known. This would save redesign efforts if the actual burnup differs from the projected value. Current reload design efforts and the resolution of plant specific issues are restricted by the guidelines to not change the Shutdown Margin since it would result in a License Amendment Request. Although the actual value of Shutdown Margin is not derived through calculations, it is assumed to be an initial input in the plant safety analyses. As such, a change in Shutdown Margin must be evaluated for its impact on the safety analyses to determine if the revised value results in an unreviewed safety question. Placing the Shutdown Margin limits in the COLR does not result in a significant impact on plant safety since changes to the safety analyses (including a change in Shutdown Margin limits) are done in accordance with NRC approved methodologies.

LESS RESTRICTIVE CHANGES (L)

There were no "Less Restrictive" changes associated with this specification.

M.3 If the rod position deviation alarm is inoperable, Condition B of the proposed ITS requires that SR 3.1.4.1 (rod position verification) be performed within 15 minutes of movement of any control rod. This action ensures that the rods are maintained within their alignment limits and is consistent with other action times in the Palisades CTS for verifying rod position indication such as when a channel of rod position indication is lost. The addition of this requirement is considered a more restrictive change since the CTS does not address requirements for the rod position deviation alarm. This change is consistent with NUREG-1432 with the exception of the Completion Time which is consistent for other CTS Completion Times for performing the rod position verification.

M.4 The proposed ITS includes SR 3.1.4.4 which verifies that the rod position deviation alarm is OPERABLE every 18 months. This surveillance frequency is adequate for ensuring that the rod position deviation alarm remains OPERABLE given the other indications available to the operator of rod position to detect if a deviation has occurred. The addition of this requirement is considered a more restrictive change since the CTS does not address requirements for the rod position deviation alarm.

M.5 CTS 3.10.1d ("Shutdown Margin Requirements") states if a control rod cannot be tripped, shutdown margin shall be increased by boration as necessary to compensate for the worth of the withdrawn inoperable control rod. In addition, CTS 3.10.4b ("Misaligned or Inoperable Control Rod or Part Length Rod") states, in part, that if more than one control rod becomes inoperable, the reactor shall be placed in the hot shutdown condition within 12 hours. The intent of these two CTS requirements is to provide compensatory measures to allow continued plant operations with one inoperable (untrippable) control rod, and to provide the necessary required actions when more than one control rod is inoperable. Although the CTS allows unrestricted operations with an untrippable control rod, this allowance is inconsistent with the assumptions used in the safety analysis. Therefore, ITS 3.1.4 has been proposed to place the plant in Mode 3 whenever one or more control rods are inoperable for reasons other than a single control rod being immovable. This change is consistent with NUREG-1432.

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 The Bases of the current Technical Specifications for this section have been completely replaced by the revised Bases that reflect the format and applicable content consistent with NUREG-1432. The revised Bases are shown in the proposed Technical Specification Bases..

A.3 CTS 3.10.3, Part-Length Control Rods, specifies that "The part-length control rods will be completely withdrawn from the core..." In the proposed ITS, the part-length control rods are required to be ≥ 128 inches as opposed to "completely withdrawn." Requiring the part-length rods to be withdrawn ≥ 128 inches has the same effect as completely withdrawn in that the rods are removed from the active region of the core. This is consistent with NUREG-1432 in that the requirement for rods to be withdrawn is specified in terms of inches withdrawn. This is considered to be an administrative change.

A.4 CTS 3.10.3 specifies that the part-length controls will be completely withdrawn from the core "(except for the control rod exercises and physics test)." The exception for control rod exercises is addressed as part of the Applicability Note. The physics tests exceptions are no longer needed because the part-length rods are not required to be moved during PHYSICS TESTS. These changes are considered to be administrative changes since no requirements have changed. These changes maintain consistency with NUREG-1432.

A.5 CTS Table 4.17.6 Item 2 requires that the Rod Position Indication have a CHANNEL CHECK performed every 12 hours. This requirement becomes SR 3.1.5.1 in the proposed ITS. Proposed SR 3.1.5.1 requires "Verify each shutdown and part length rod is withdrawn ≥ 128 inches every 12 hours." The surveillance in the proposed ITS functions to perform the same verifications as that intended in the CTS "CHANNEL CHECK" since the CTS definition of "CHANNEL CHECK" includes the statement "A CHANNEL CHECK shall include verification that the monitored parameter is within the limits imposed by the Technical Specifications." CTS 3.10.6 requires that the shutdown rods shall be withdrawn before any regulating rods are withdrawn. CTS 3.10.4b in part states that a part-length rod is considered inoperable if it is not fully withdrawn. CTS 3.10.3 requires that the part-length rods be completely withdrawn. Therefore, the proposed surveillance performs this by proposed ITS SR 3.1.5.1 ensuring that the shutdown and part-length rods are withdrawn ≥ 128 inches. This is considered to be an administrative change since the requirements have not changed but have been reformatted in accordance with NUREG-1432.

A.6 CTS 3.10.6a states "All shutdown rods shall be withdrawn before any regulating rods are withdrawn." In the proposed ITS, the phrase "above 5 inches" is added to clarify what is intended by "withdrawn." Allowing the regulating rods to be withdrawn up to 5 inches facilities normal operation of the control rod drive motors which are "bumped" to bring the rods off the bottom before they are withdrawn. This area of the core is very insignificant with respect to the integral worth of the rod. This also corresponds to the Shutdown Rod Insertion interlock which prevents the shutdown rods from being inserted once the regulating rods are withdrawn greater than 5 inches. This change is a clarification to define what "withdrawn" means with respect to the regulating rods.

A.7 CTS 3.10.6a states "All shutdown rods shall be withdrawn before any regulating rods are withdrawn." CTS 3.10.6c states "The shutdown rods shall not be inserted below their exercise limit until all regulating rods are inserted." The proposed ITS 3.1.5 LCO states "All shutdown and part/length rod groups shall be withdrawn to ≥ 128 inches." The Applicability for LCO 3.1.5 is MODE 1, MODE 2 with any regulating rod withdrawn above 5 inches. The proposed ITS wording for the LCO and Applicability is equivalent to the CTS wording in 3.10.6b. In the ITS, the shutdown rods must be withdrawn ≥ 128 inches by the LCO before the regulating rods are withdrawn above 5 inches (see DOC A.6 for discussion on 5 inches criteria). In addition, the CTS 3.10.6c requirement that the shutdown rods cannot be inserted below their exercise limit is also maintained in the ITS. This is because the shutdown rods cannot be inserted, except for rod exercising allowed by Applicability note, until out of the MODE of Applicability which required the regulating rods to be ≤ 5 inches withdrawn. Therefore, the CTS and the proposed ITS are equivalent.

A.8 CTS 3.10.7 includes an exception which allows a deviation from the requirement for shutdown rod limits during performance of CRDM exercises. The exception contains a qualifying statement which reads "if necessary to perform a test but only for the time necessary to perform the test." The Applicability Note for proposed ITS 3.1.5 which also provides an exception from the requirement for shutdown rod limits during performance of CRDM exercise does not contain this same qualifier since these type details are governed by the usage rules for the ITS. Therefore, deletion of this information is considered administrative in nature. This change is consistent with NUREG-1432.

A.9 CTS 3.10.3 and CTS 3.10.6 stipulate the requirement for rod position on an individual rod basis (i.e., all shutdown and part-length rod must be fully withdrawn). In addition, CTS 3.4.10.4a requires that a control rod must be aligned within 8 inches from the remainder of the bank. The CTS does not specify rod positions on a group basis, and does not contain actions when controls rods are misaligned from their groups by less than 8 inches. Proposed ITS 3.1.5 establishes insertion limits for the shutdown and part-length rod groups by requiring them to be withdrawn ≥ 128 inches. Required Action A.1 of ITS 3.1.5 requires that any shutdown or part-length rod group that is not within its group insertion limit be declared inoperable and the Conditions of ITS 3.1.4 entered immediately. If the Required Action and associated Completion Time are not met, Required Action B.1 requires the plant to be in Mode 3 within 6 hours. To ensure compliance with the requirements of LCO 3.1.5, for a control rod group to be considered above its insertion limit, all rods in that group must be above the insertion limit. The addition of ITS Required Actions A.1 and B.1 is characterized as an administrative change since the action taken when a shutdown or part-length rod exceed its insertion limit is consistent with the CTS actions for an inoperable control rod.

MORE RESTRICTIVE CHANGES (M)

There were no "More Restrictive" changes associated with this specification.

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

LA.1 CTS 3.10.6b states "The shutdown rods shall not be withdrawn until normal water level is established in the pressurizer." This requirement was included to help assure an inadvertent criticality will not occur with the PCS water solid. This statement is more appropriate for being addressed in plant procedures and is not included in the proposed ITS. Changes to plant procedures are made in accordance with the plant procedure change process. This change maintains consistency with NUREG-1432.

LESS RESTRICTIVE CHANGES (L)

There were no "Less Restrictive" changes associated with this specification.

LESS RESTRICTIVE CHANGES L.1

There were no "Less Restrictive" changes associated with this Specification.

6

TSTF-136

7

TSTF-9

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM) ~~— T_{avg} > 200°F (Analog)~~

LCO 3.1.1 SDM shall be ~~≥ [4.5]% Δk/k.~~ within the limits provided in the COLR.

APPLICABILITY: MODES 3, ~~and~~ 4, and 5

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

TSTF-9

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify SDM ~~is ≥ [4.5]% Δk/k.~~ to be within limits.	24 hours

7

7

9

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3.1 REACTIVITY CONTROL SYSTEMS

Rod

3.1.~~8~~ 4 Control ~~Element Assembly (CEA)~~ Alignment ~~(Analog)~~

CTS 3.10.4a
TBL 3.17.6
ITEM 2
DEV Alm New

Control rods, including their position indication channels, [8]

LCO 3.1.~~5~~ 4 All ~~CEAs~~ shall be OPERABLE and aligned to within ~~[7]~~ 8 inches ~~(indicated position)~~ of their respective group ~~s~~, ~~and the CEA motion inhibit~~ and the ~~CEA~~ Control rod position deviation ~~circuit~~ alarm shall be OPERABLE]. 1 4

all other rods in

APPLICABILITY: MODES 1 and 2.

CTS 3.17.6.2

ACTIONS

19

INSERT →

CONDITION	REQUIRED ACTION	COMPLETION TIME
C ~~A~~. One ~~or more~~ ~~regulating CEAs trippable and~~ control rod misaligned ~~from its group~~ by > ~~[7]~~ 8 inches ~~and ≤ [15 inches]. OR One regulating CEA trippable and misaligned from its group by > [15 inches].~~	C.2 ~~A.1~~ Reduce THERMAL POWER to ≤ ~~70%~~ 75% RTP.	2 ~~8~~ hours 5
	~~AND A.2.1 Verify SDM is ≥ [4.5]% Δk/k.~~	~~1 hour~~
	~~OR A.2.2 Initiate boration to restore SDM to within limit.~~	~~1 hour~~
	~~AND A.3.1 Restore the misaligned CEA(s) to within [7 inches] (indicated position) of its group. OR~~	~~2 hours~~
	C.1 Perform SR 3.2.2.1 (peaking factor verification) OR	2 hours 5

(continued)

22
1 9

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23

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Not Used

TSTF-143
Not Used 7

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SECTION 3.1

INSERT 3

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. .... OR Both rod position indication channels inoperable for one or more control rods.	E.1	....

INSERT 4

Perform a CHANNEL CHECK of the control rod position indication channels.

TSTF-127

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY	
new 6	SR 3.1.~~5~~.~~4~~ 4 Verify the ~~CEA~~ rod position deviation ~~circuit~~ alarm is OPERABLE.	~~31 days~~ 18 months	4
Table 4.2.2 Item 2	SR 3.1.~~5~~.~~5~~ 4 3 Verify ~~CEA~~ control rod freedom of movement ~~(trippability)~~ by moving each individual full-length control rod ~~CEA~~ that is not fully inserted into the reactor core ~~[5]~~ 6 inches in either direction. 2 | 4 | 4 | 4 | 1	92 days	
Table 4.17.6 Item 2 CHANNEL CALIBRATION	SR 3.1.~~5~~.~~6~~ 4 5 Perform a ~~CHANNEL FUNCTIONAL TEST~~ CHANNEL CALIBRATION of the ~~reed switch~~ control rod position ~~transmitter~~ indication channel~~s~~.	18 months	11
Table 4.2.2 Item 1	SR 3.1.~~5~~.~~7~~ 4 6 Verify each full-length control rod ~~CEA~~ drop time is ≤ ~~[3.1]~~ 2.5 seconds. | 4 | 1	Prior to reactor criticality, after each reinstallation ~~removal~~ of the reactor head	2

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12

3.1 REACTIVITY CONTROL SYSTEMS

3.1.~~8~~ 5 Shutdown ~~Control Element Assembly (CEA)~~ and Part Length Rod Group Insertion Limits ~~(Analog)~~ 13

12

LCO 3.1.~~8~~ 5 All shutdown ~~CEAs~~ and part-length rod groups shall be withdrawn to ≥ ~~[129]~~ 128 inches. 12 1

APPLICABILITY: MODE 1,
MODE 2 with any regulating ~~CEA not fully inserted~~ rod withdrawn above 5 inches. 6

-----------------------------NOTE-----------------------------
This LCO is not applicable while performing SR 3.1.~~5~~4.~~6~~4 (rod exercise test). 4 2

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more shutdown ~~CEAs~~ or part-length rod groups not within limit. 12	~~A.1.1~~	~~Verify SDM ≥ [4.5]% Δk/k.~~	~~1 hour~~
	~~OR~~		
	~~A.1.2~~	~~Initiate boration to restore SDM to within limit.~~	~~1 hour~~
	~~AND~~		
TSTF-67 16	~~A.2~~	~~Restore shutdown CEA(s) to within limit.~~	~~2 hours~~
		Declare affected control rod(s) inoperable and enter the applicable Conditions and Required Actions of LCO 3.1.4.	Immediately
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours

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3.1 REACTIVITY CONTROL SYSTEMS

CTS 3.10.5a
CTS Table 3.17.6 Item 13,18

3.1.~~7~~ 6 Regulating ~~Control Element Assembly (CEA) Insertion~~ Rod Group Position Limits ~~(Analog)~~ 11 7

LCO 3.1.~~7~~ 6 The ~~power dependent insertion limit~~ (PDIL) alarm circuit 2 and the Control Rod Out Of Sequence (CROOS) alarm circuit 8 shall be OPERABLE, and the regulating ~~CEA~~ rod groups shall be 4 limited to the withdrawal sequence, overlap, 10 and ~~to the~~ insertion limits specified in the COLR.

APPLICABILITY: MODES 1 and 2.

----------------------------NOTE----------------------------
This LCO is not applicable while performing SR 3.1.~~5.5~~ 4.3 2 ~~[or during reactor power cutback operation]~~ (rod exercise test) 4. 3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
CTS 3.10.5b.1 4 A. Regulating ~~CEA~~ rod groups inserted beyond the ~~transient~~ 12 insertion limit.	~~A.1.1 Verify SDM ≥ [4.5]% Δk/k.~~ ~~OR~~ ~~A.1.2 Initiate boration to restore SDM to within limit.~~ ~~AND~~	~~1 hour~~ ~~1 hour~~
TSTF-67 15	A.~~2~~.1 Restore regulating ~~CEA~~ rod groups to within limits. OR	2 hours 4
		(continued)

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BASES (continued)

APPLICABLE SAFETY ANALYSES

FSAR 14.14.2.2

TSTF-136 (7), (4)

INSERT

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth CEA stuck out following a reactor trip.

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limit AOOs, and ≤ 280 cal/gm energy deposition for the CEA control rod ejection accident); and (4)

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

FSAR 14.14

FSAR 14.14.1

The most limiting accident for the SDM requirements are based on a main steam line break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam (2) generator (SG), and consequently the RCS. This results in a (2) (4) reduction of the reactor primary coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As RCS (5) temperature decreases, the severity of an MSLB decreases until the MODE 5 value is reached. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to (12)

(continued)

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BASES

APPLICABLE SAFETY ANALYSES (continued)

TSTF-67 (14)

~~power, and~~ THERMAL POWER does not violate the Safety Limit (12) (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirement for MODES 3 and 4 must also protect against: an

a. Inadvertent boron dilution; (Ref. 3) and (4)

b. ~~An~~ uncontrolled control rod bank ~~CEA~~ withdrawal from ~~a~~ subcritical ~~or low power~~ conditions (Ref. 5). (4)

(15)

~~c. Startup of an inactive reactor coolant pump (RCP); and~~

~~d. CEA ejection.~~

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the PCS ~~RCS~~ and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life when critical boron concentrations are highest.

TSTF-67 (14)

The withdrawal of a control rod bank ~~CEAs~~ from subcritical ~~or low power~~ conditions adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The withdrawal of ~~CEAs~~ control rod banks also produces a time dependent redistribution of core power.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled ~~CEA~~ withdrawal transient is terminated by either a high power trip or a high pressurizer pressure trip. In all cases, power level, PCS ~~RCS~~ pressure, linear heat rate, and the DNBR do not exceed allowable limits.

(15)

~~The startup of an inactive RCP will not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. The maximum positive reactivity addition that can occur due to an inadvertent RCP start is less than half the minimum required~~

(continued)

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BASES

APPLICABLE SAFETY ANALYSES (continued)

SDM. An ~~idle~~ RCP cannot, ~~therefore~~, produce a ~~return~~ to power from ~~the~~ hot standby ~~condition~~.

SDM satisfies Criterion 2 of ~~the NRC Policy Statement~~. 10 CFR 50.36(c)(2) (2)

LCO

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value for of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, then the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

SDM is a control rod core physics design condition that can be ensured through CEA positioning (regulating and shutdown CEA rods) and through the soluble boron concentration. (4)

APPLICABILITY

In MODES 3 and 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5, ~~"Shutdown Control Element Assembly (CEA) Insertion Limits,"~~ and LCO 3.1.6. ~~If the insertion limits of LCO 3.1.6 or LCO 3.1.7 are not being complied with, SDM is not automatically violated. The SDM must be calculated by performing a reactivity balance calculation (considering the listed reactivity effects in Bases Section SR 3.1.1.1). In MODE 5, SDM is addressed by LCO 3.1.2, "SHUTDOWN MARGIN (SDM)—T_{avg} ≤ 200°F."~~ In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." (4)

TSTF-67 (14)
TSTF-136 (7)

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

(continued)

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BASES

ACTIONS A.1 (continued)

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid storage tank or the borated water storage tank. The operator should borate with the best source available for the plant conditions.

injection flow

concentrated

In determining the boration flow rate, the time core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle, when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1% Δk/k must be recovered and a boration flow rate of [35] gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 25 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% Δk/k. These boration parameters of [35] gpm and [100] ppm represent typical values and are provided for the purpose of offering a specific example.

1.0 E-4 Δρ/ppm

SURVEILLANCE REQUIREMENTS SR 3.1.1.1

SDM is verified by ~~performing~~ a reactivity balance calculation, considering the listed reactivity effects:

a. RCS boron concentration;

b. CEA positions;

Control rod

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

(continued)

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SECTION 3.1

INSERT 1

Samarium is not considered in the reactivity analysis since the analysis assumes that the negative reactivity due to samarium is offset by the positive reactivity of plutonium build in.

INSERT 2

SR 3.1.1.1 requires SDM to be within the limits provided in the COLR. This SDM value ensures the consequences of an MSLB will be acceptable as a result of a cooldown of the PCS which adds positive reactivity in the presence of a negative moderator temperature coefficient, as well as the other events described in the Applicable Safety Analysis. As such, the requirements of this SR must be met whenever the plant is in MODES 3, 4, and 5.

SECTION 3.1

INSERT

The synchro based position indication system measures the phase angle of a synchro geared to the CRDM rack. Full control rod travel corresponds to less than 1 turn of the synchro. Each control rod has its own synchro. The Primary Information Processor (PIP) node scans and converts synchro outputs into inches of control rod withdrawal. The resolution of this system is approximately 0.5 inches. Each synchro also has cam operated limit switches which can provide positive indication of control rod position.

The reed switch based position indication system is referred to as the Secondary Position Indication (SPI) system. This system provides a highly accurate indication of actual control rod position, but at a lower precision than the synchros. The reed switches are wired so that the voltage read across the reed switch stack is proportional to rod position. The reed switches are spaced along a tube with a center to center spacing distance of 1.5 inches. The resolution of the SPI reed switch stacks is 1.5 inches. The reed switches also provide input to the matrix indication lights which provide control rod status indication for various key positions. To increase the reliability of the system, there are redundant reed switches which prevent false indication in the event an individual reed switch fails.

A control rod position deviation alarm is provided to alert the operator when any two control rods in the same group are more than 8 inches apart. This helps to ensure any control rod misalignments are minimized. The alarm can be generated by either the SPI system or PIP node since the SPI system, in conjunction with the host computer, is redundant to the PIP node in the task of control rod measurement, control rod monitoring, and limit processing.

BASES

APPLICABLE SAFETY ANALYSES (continued)

determine that the required SDM is met with the maximum worth CEA also fully withdrawn (Ref. 5).

Since the CEA drop incidents result in the most rapid approach to ~~specified acceptable fuel design limits~~ (SAFDLs) caused by a CEA misoperation, the accident analysis analyzed a single full length CEA drop. ~~The most rapid approach to the DNBR SAFDL may be caused by a single full length CEA drop or a CEA subgroup drop, depending upon initial conditions.~~ 4 2 15

Control rod

Control rod may or may not

~~All of~~ the above CEA misoperations ~~will~~ result in an automatic reactor trip. In the case of the full length CEA rod drop, a prompt decrease in core average power and a distortion in radial power are initially produced, which, when conservatively coupled, result in a local power and heat flux increase, and a decrease in DNBR parameters. 2 4

Control rod

The results of the CEA misoperation analysis show that during the most limiting misoperation events, no violations of the SAFDLs, fuel centerline temperature, or RCS pressure occur. 4

Control rod

CEA alignment satisfies Criteria 2 and 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36(c)(2). 2

LCO

and part-length rod

The limits on shutdown, ~~and~~ regulating, CEA alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the CEAs will be available and will be inserted to provide enough negative reactivity to shut down the reactor. The OPERABILITY requirements also ensure that the CEA banks maintain the correct power distribution and CEA alignment.

Control rod

17 INSERT

18, 9 4

~~The requirement is to maintain the CEA alignment to within [7 inches] between any CEA and its group. The minimum misalignment assumed in safety analysis is [15 inches], and in some cases, a total misalignment from fully withdrawn to fully inserted is assumed.~~

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable

(continued)

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SECTION 3.1

INSERT

.....and that each control rod is capable of being moved by its CRDM. The OPERABILITY requirement for the part-length rods is that they are fully withdrawn.

The requirement is to maintain the control rod alignment to within 8 inches between any control rod and all other rods in its group. To help ensure this requirement is met, the control rod position deviation alarm generated by either the PIP node or SPI system, must be OPERABLE and provide an alarm when any control rod becomes misaligned > 8 inches from any other rod in its group. The safety analysis assumes a total misalignment from fully withdrawn to fully inserted. This case bounds the safety analysis for a single rod in any intermediate position.

The primary rod position indication system is considered OPERABLE, for purposes of this specification, if the digital position readout, the PPC display, or the cam operated position indication lights give positive indication of rod position. The secondary rod position indication system is considered OPERABLE if the magnetically operated reed switches are providing positive indication of rod position either via the plant process computer or taking direct readings of the output from the magnetic reed switches.

SECTION 3.1

INSERT 1

Performance of a CHANNEL CHECK on the primary and secondary control rod position indication channels provides confidence in the accurracy of the rod position indication systems.

INSERT 2

... which correspond to the lower electrical limit and the upper electrical limit respectively,

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.1.5.3 (continued)~~

~~can be detected, and protection can be provided by the CEA deviation circuits.~~ (3)

TSTF-127 (6)

(24) (2) SR 3.1.5.4 3

Demonstrating the ~~CEA~~ rod position deviation ~~circuit~~ alarm is OPERABLE verifies the ~~circuit~~ alarm is functional. The ~~31~~ 92 day Frequency takes into account other information continuously available to the operator in the control room, so that during ~~CEA~~ control rod movement, deviations can be detected~~, and protection can be provided by the CEA motion inhibit~~. (4) (3)

(24) (2) SR 3.1.5.6 4

Verifying each ~~CEA~~ full length control rod is trippable would require that each ~~CEA~~ control rod be tripped. In MODES 1 and 2, tripping each ~~CEA~~ control rod would result in radial or axial power tilts, or oscillations. Therefore, individual ~~CEAs~~ full-length control rods are exercised every 92 days to provide increased confidence that all ~~CEAs~~ control rods continue to be trippable, even if they are not regularly tripped. A movement of (1) (6) ~~[5]~~ inches is adequate to demonstrate motion without exceeding the alignment limit when only one ~~CEA~~ control rod is being moved. The 92 day Frequency takes into consideration other information available to the operator in the control room and other surveillances being performed more frequently, which add to the determination of OPERABILITY of the CEAs. ~~Between required performances of SR 3.1.5.5, if a CEA(s) is discovered to be immovable, but remains trippable and aligned, the CEA is considered to be OPERABLE.~~ (10) (4) At any time, if a ~~CEA(s)~~ control rod is immovable, a determination of the trippability ~~(OPERABILITY)~~ of the ~~CEA(s)~~ control rod must be made, and appropriate action taken. (4)

(24) (2) SR 3.1.5.6 4 5

Performance of a CHANNEL ~~FUNCTIONAL TEST~~ CALIBRATION of each ~~reed switch~~ rod position ~~transmitter~~ indication channel ensures the channel is OPERABLE and capable of indicating ~~CEA~~ control rod position over the entire length of the ~~CEA's~~ control rod's travel. Since this Surveillance must be performed when the reactor is shut down, an 18 month Frequency to be coincident with refueling outage was (11) (4)

(4) INSERT

(continued)

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.~~6~~ 5 Shutdown ~~Control Element Assembly (CEA)~~ Insertion Limits ~~(Analog)~~ (13)
and Part-Length Rod Group (12)

BASES

BACKGROUND

The insertion limits of the shutdown ~~CEAs~~ rods are initial assumptions in all safety analyses that assume ~~CEA~~ full-length control rod insertion upon reactor trip. The insertion limits directly affect core power distributions and assumptions of available SDM, ejected ~~CEA~~ rod worth, and initial reactivity insertion rate. (4)

(9) INSERT 1

~~t~~The applicable criteria for these reactivity and power distribution design requirements ~~are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," and GDC 26, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors"~~ (Ref. ~~2~~ 1). Limits on shutdown ~~CEA~~ rod insertion have been established, and all ~~CEA~~ rod positions are monitored and controlled during power operation to ensure that the reactivity limits, ejected ~~CEA~~ rod worth, and SDM limits are preserved. (9) (4)

The shutdown ~~CEAs~~ rods are arranged into groups that are radially symmetric. Therefore, movement of the shutdown ~~CEAs does~~ rod groups does not introduce radial asymmetries in the core power distribution. The shutdown and regulating ~~CEAs~~ rod groups provide the required reactivity worth for immediate reactor shutdown upon a reactor trip. (4) (12)

(12) INSERT 2

The design calculations are performed with the assumption that the shutdown ~~CEAs~~ rod groups are withdrawn prior to the regulating ~~CEAs~~ rod groups. The shutdown ~~CEAs~~ can be fully withdrawn without the core going critical. This provides available negative reactivity for SDM in the event of boration errors. ~~The shutdown CEAs~~ All control rod groups (12) are controlled manually ~~or automatically~~ (10) by the control room operator. During normal ~~unit~~ plant operation, the shutdown ~~CEAs~~ rod groups are fully withdrawn. The shutdown ~~CEAs~~ rod groups must be completely withdrawn from the core prior to withdrawing any regulating ~~CEAs~~ rods during an approach to criticality. The shutdown ~~CEAs~~ rod groups are then left in this position until the reactor is shut down. They affect core power, burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal. (12) (4) (12) (4) (12)

(continued)

Revised
02/05/99

SECTION 3.1

INSERT 1

The Palisades Nuclear Plant design criteria (Ref. 1) and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling System for Light Water Nuclear Power Reactors," contains..

INSERT 2

The Palisades Nuclear Plant has four part-length control rods installed. The part length rods are required to remain completely withdrawn during power operations, except during rod exercising performed in conjunction with SR 3.1.4.3. The part-length rods do not insert on a reactor trip.

BASES (continued)

APPLICABLE SAFETY ANALYSES

rod groups

Accident analysis assumes that the shutdown CEAs are fully withdrawn any time the reactor is critical. This ensures that: (12)

a. The minimum SDM is maintained; and

control rod

b. The potential effects of a CEA ejection accident are limited to acceptable limits. (4)

Control rods CEAs are considered fully withdrawn at 128 (8) inches, since this position places them outside the active region of the core. in a insignificant reactivity worth region of the integral worth curve for each bank (4)

Full length control rods On a reactor trip, all CEAs (shutdown and regulating), except the most reactive CEA rod, are assumed to insert into the core. The shutdown and regulating CEAs shall be at their rod groups insertion limits or above and available to insert the maximum required amount of negative reactivity on a reactor trip signal. The regulating CEAs rods may be partially inserted in the core as allowed by LCO 3.1.7 6 "Regulating Control Element Assembly Rod (CEA) Insertion Group Position Limits." The shutdown CEA rod group insertion limit is established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM) — T_{avg} > 200°F") following a reactor trip from full power. The combination of regulating CEAs and shutdown rods CEAs rods (less the most reactive CEA, which is assumed to be remain fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 3 2). The shutdown CEA insertion limit also limits the reactivity worth of an ejected shutdown CEA rod. (4) (11) (12) (15) TSTF-136

rod group rod group

The acceptance criteria for addressing shutdown CEA as well as regulating CEA insertion limits and inoperability or misalignment are that: rod group (12)

a. There be no violation of:

1. specified acceptable fuel design limits, or
2. Reactor Primary Coolant System pressure boundary damage; and (4)

b. The core remains subcritical after accident transients.

(continued)

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02/05/99

SECTION 3.1

INSERT 1

The part-length rods have the potential to cause power distribution envelopes to be exceeded if inserted while the reactor is critical. Therefore, they must remain withdrawn in accordance with the limits of the LCO (Ref. 3).

INSERT 2

For a control rod group to be considered above its insertion limit, all rods in that group must be above the insertion limit.

Maintaining the shutdown rod groups within their insertion limits...

INSERT 3

Maintaining the part length rod group within its insertion limit ensures that the power distribution envelope is maintained.

INSERT 4

In MODE 2, the Applicability begins anytime any regulating rod is withdrawn above 5 inches.

INSERT 5

...to at least the lower electrical limit, and contribute to the SDM. In MODE 3, the shutdown rod groups may be withdrawn in preparation for a reactor startup.

INSERT 6

The Applicabilty has been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.3 (rod exercise test). Control rod exercising verifies the freedom of the rods to move, and requires the individual shutdown rods to move below the LCO limits for their group. Only the full-length rods are required to be tested by SR 3.1.4.3. The part-length rods may also be moved however, if a part-length rod is moved below the limit of the associated LCO the Required Action of Condition A must be taken.

Positioning of an individual control rod within its group is addressed by LCO 3.1.4, "Control Rod Alignment."

BASES

ACTIONS (16) A.1 ~~.1, A.1.2, and A.2~~ (continued)

TSTF-67 (16)

core as it is inserted. ~~If boron concentration is not changed at this time, SDM should not change. This, however, is verified within 1 hour, or boration is initiated to bring the SDM to within limit, if the CEA(s) is not restored to within limits prior to this time.~~

~~If the CEA(s) is not restored to within limits within 1 hour and the SDM is within limit, then an additional 1 hour is allowed for restoring the CEA(s) to within limits. The 2 hour total Completion Time allows the operator adequate time to adjust the CEA(s) in an orderly manner and is consistent with the required Completion Times in LCO 3.1.5, "Control Element Assembly (CEA) Alignment."~~ (12) (4)

INSERT 1

B.1

When Required Action A.1 ~~or A.2~~ cannot be met or completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Verification that the shutdown CEAs and part length rod groups are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown rods CEAs will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown CEAs are withdrawn before the regulating CEAs rods are withdrawn during a unit plant startup. (12) (4)

(12) INSERT 2

Control and part length rod group

~~Since the shutdown CEAs are positioned manually by the control room operator, verification of shutdown CEA position at a Frequency of 12 hours is adequate to ensure that the shutdown CEAs are within their insertion limits. Also, the 12 hour Frequency takes into account other information available to the operator in the control room for the purpose of monitoring the status of the shutdown CEAs.~~ rod group (12)

(continued)

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SECTION 3.1

INSERT 1

If one or more shutdown or part-length rods are not within limits, the affected rod(s) must be declared inoperable and the applicable Conditions and Required Actions of LCO 3.1.4 entered immediately. This Required Action is based on the recognition that the shutdown and part-length rods are normally withdrawn beyond their insertion limits and are capable of being moved by their control rod drive mechanism. Although the requirements of this LCO are not applicable during performance of the control rod exercise test, the inability to restore a control rod to within the limits of the LCO following rod exercising would be indicative of a problem affecting the OPERABILITY of the control rod. Therefore, entering the applicable Conditions and Required Actions of LCO 3.1.4 is appropriate since they provide the applicable compensatory measures commensurate with the inoperability of the control rod.

INSERT 2

Verification that the part-length rod groups are within their insertion limits ensures that they do not adversely affect power distribution requirements.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.~~8~~ 6 Regulating ~~Control Element Assembly (CEA) Insertion~~ Limits ~~(Analog)~~ (11)

Rod Group Position (7)

BASES

BACKGROUND

The insertion limits of the regulating ~~CEAs~~ rod group are initial assumptions in all safety analyses that assume ~~CEA~~ control rod insertion upon reactor trip. The insertion limits directly affect core power distributions, assumptions of available SDM, and initial reactivity insertion rate. The applicable criteria for these reactivity and power distribution design requirements are ~~10 CFR 50, Appendix A, GDC 10, "Reactor Design," and GDC 26, "Reactivity Limits"~~ contained in the Palisades Nuclear Plant design criteria (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). (4) (9)

Limits on regulating ~~CEA~~ rod group insertion have been established, and all ~~CEA~~ regulating rod group positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking, ejected ~~CEA~~ rod worth, reactivity insertion rate, and SDM limits are preserved. (4) (4)

The regulating ~~CEA~~ rod groups operate with a predetermined amount of position overlap, in order to approximate a linear relation between ~~CEA~~ rod worth and ~~CEA~~ rod position (integral ~~CEA~~ rod worth). The regulating ~~CEA~~ rod groups are withdrawn and operate in a predetermined sequence. The group sequence and overlap limits are specified in the COLR. (4) (4)

The regulating ~~CEAs~~ rods are used for precise reactivity control of the reactor. The positions of the regulating ~~CEAs~~ rods are manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting). (4)

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 2). Together, LCO 3.1.~~7~~6, ~~"Regulating Control Element Assembly (CEA) Insertion Limits"~~; LCO 3.2.~~4~~3, "~~AZIMUTHAL~~ Quadrant POWER TILT (T_q)"; and LCO 3.2.~~5~~4 "AXIAL SHAPE INDEX (ASI)," provide limits on control component operation and on monitored process variables to ensure the core operates within the linear heat rate (LCO 3.2.1, "Linear Heat Rate (LHR)")~~,~~ ~~total planar~~ and radial peaking factor ~~(F_{xy})~~ F_R^T and F_R^A (2) (2)

(continued)

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SECTION 3.1

INSERT 1

The most limiting SDM requirements for Mode 1 and 2 conditions at (Beginning of Cycle (BOC) are determined by the requirements of several transients, e.g., Loss of Flow, etc. However, the most limiting SDM requirements for Modes 1 and 2 at End of Cycle (EOC) come from just one transient, Main Steam Line Break (MSLB). The requirements of the MSLB event at EOC for the full power and no load conditions are significantly larger than those of any other event at that time in cycle and, also, considerably larger than the most limiting requirements at BOC.

Although the most limiting SDM requirements at EOC are much larger than those at BOC, the available SDMs obtained via tripping the full-length control rods are substantially larger due to the much lower boron concentration at EOC. To verify that adequate SDMs are available throughout the cycle to satisfy the changing requirements, calculations are performed at both BOC and EOC. It has been determined that calculations at these two times in cycle are sufficient since the difference between available SDMs and the limiting SDM requirements are the smallest at these times in cycle. The measurement of full-length control rod bank worth performed as part of the Startup Testing Program demonstrates that the core has the expected shutdown capability. Consequently, adherence to LCO 3.1.5, "Shutdown and Part-Length Rod Group Insertion Limits," and LCO 3.1.6 provides assurance that the available SDM at any time in cycle will exceed the limiting SDM requirements at that time in cycle.

INSERT 2

For a control rod group to be considered above its insertion limit, all rods in that group must be above the insertion limit.

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BASES (continued)

APPLICABILITY

rod group

The regulating CEA sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits must be maintained, since they preserve the assumed power distribution, ejected CEA worth, SDM, and reactivity rate insertion assumptions. Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected CEA worth assumptions would be exceeded in these MODES. SDM is preserved in MODES 3, 4, and 5 by adjustments to the soluble boron concentration. (7) rod (4) (4)

Applicability rod 4.3 (rod exercise test).

This LCO has been modified by a Note indicating the LCO requirement is suspended during SR 3.1.5.5. This SR verifies the freedom of the CEAs to move, and requires the regulating CEAs to move below the LCO limits, which would normally violate the LCO. The Note also allows the LCO to be not applicable during reactor power cutback operation, which inserts a selected CEA group (usually group 5) during loss of load events. (2) Control rod exercise (4) (3)

rods rods individual for their group

ACTIONS

A.1.1 and A.1.2, A.2.1, and A.2.2

Operation beyond the transient insertion limit may result in a loss of SDM and excessive peaking factors. If the regulating CEA insertion limits are not met, then SDM must be verified by performing a reactivity balance calculation, considering the listed reactivity effects in Bases Section SR 3.1.1.1. One hour is sufficient time for conducting the calculation and commencing boration if the SDM is not within limits. The transient insertion limit should not be violated during normal operation; this violation, however, may occur during transients when the operator is manually controlling the CEAs in response to changing plant conditions. When the regulating groups are inserted beyond the transient insertion limits, actions must be taken to either withdraw the regulating groups beyond the limits or to reduce THERMAL POWER to less than or equal to that allowed for the actual CEA insertion limit. Two hours provides a reasonable time to accomplish this, allowing the operator to deal with current plant conditions while limiting peaking factors to acceptable levels. (12) (12) (4) (12) (7)

TSTF-67 (15)

regulating rods

rod group position

(continued)

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Change	**Discussion**
7.	ISTS Change Traveler TSTF-136 combines ISTS 3.1.1 and ISTS 3.1.2 into a single specification in order to eliminate unnecessary and confusing duplication, and renumbers the remaining specifications in Section 3.1. The impetus for this change was the approval of TSTF-9 which allowed the values for shutdown margin to be moved to the COLR. As a result of TSTF-9, the LCO, Actions, and Surveillance Requirements of ISTS 3.1.1 and ISTS 3.1.2 were the same. Palisades has relocated the shutdown margin values to the COLR in accordance with TSTF-9 and has consolidated ISTS 3.1.1 and ISTS 3.1.2 into a single specification. Proposed ITS 3.1.1 address the plant conditions encompassed in MODEs 3, 4, and 5 as a result of this consolidation.
8.	The Palisades plant was designed prior to issuance of the General Design Criteria (GDC) in 10 CFR 50. Therefore, reference to the GDCs is omitted and appropriately replaced by reference to "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 7, 1971. It was this updated discussion, including the identified exemptions, which formed the original plant Licensing Basis for future compliance with the GDCs.
9.	TSTF-9 permits relocation of the shutdown margin values specified in ISTS 3.1.1 and ISTS 3.1.2 to the COLR. Palisades has elected to exercise this option in the ITS. The appropriate justification for this change is provided in DOC LA.1 for ITS 3.1.1.
10.	Samarium is not considered in the Palisades Nuclear Plant reactivity balance due to the fact the that Palisades Nuclear Plant fuel vendor does not account for Samarium in fuel design calculations. The vendor assumes that the negative reactivity defect due to Samarium is offset by the positive reactivity of Plutonium build in. Plutonium build in and Samarium are equally competing reactivity effects that are accounted for in fuel design calculations. Therefore, including Samarium into the SDM calculation would not be correct for the Palisades Nuclear Plant.

Change **Discussion**

6. The Frequency for the rod position deviation alarm surveillance (ISTS SR 3.1.5.4) is being decreased from 31 days to 18 months. Verification of that alarm's operability involves misaligning each control rod group until the alarm actuates. This involves both exceeding the LCO 3.1.4 group alignment limits and moving part length rods. Neither of these actions is desired during power operation. The CTS neither requires this alarm to be Operable nor includes any associated surveillance requirement. Since Palisades rods are manually controlled, and rod group alignments are verified after moving rods, the alarm is not as significant as in a plant with automatic rod control.

7. The NUREG-1432 Action A (regulating rods), and Action B (shutdown rods) requirements to restore the misaligned rod to within 7 inches of its group or to restore the group to within 7 inches of the misaligned rod (Action A only) were consolidated into one Action as a result of TSTF-143. Since the CTS does not require misaligned rods to be restored, ISTS Required Actions A.3.1 and A.3.2, as modified by TSTF-143, are not included in the ITS. In addition, neither the CTS, nor the ITS make a distinguish between misaligned shutdown rods or misaligned regulating rods. Therefore, ISTS Condition B is not required.

8. NUREG-1432 LCO 3.1.5 requires that control rods must be aligned to within a certain amount of inches "(indicated position)" of their respective group. Including the term "indicated position" is not appropriate for the Palisades plant. The NUREG was based on plants which use magnetic jacks as the mechanism for moving the control rods. These type of mechanisms typically have a demand position and an indicated position. A "demand" is placed on the magnetic jack to move a certain amount and this is reflected in the control rod "demand counter" whether or not the control rod actually moved. The term "indicated position" would refer to the position indication system which is actually monitoring control rod travel. The design at the Palisades plant uses a primary and secondary rod position indicating system with both systems actually indicating "actual" rod position since there is no "demand" position. Therefore, the term "indicated position" is not included in the Palisades ITS.

9. NUREG-1432, Condition A is modeled after plants which have an analysis which have varying amounts of rod misalignment. The Palisades CTS only assumes that a control rod is either within limits or is misaligned. There are no actions or supporting analysis for differing amounts of misalignment. Therefore, NUREG-1432 is revised, in the applicable portions, to only discuss misalignments greater than 8 inches in the Palisades Nuclear Plant proposed ITS.

Change **Discussion**

7. NUREG-1432 has a Note in the Applicability which modifies the LCO by stating "This LCO is not applicable while performing SR 3.1.5.5." In NUREG-1432, SR 3.1.5.5 is the rod exercise test which corresponds to SR 3.1.4.3 in the proposed Palisades ITS. The Bases discussion for the Applicability Note has been modified to clarify the requirement of rod testing as it relates to part-length rods. Part-length control rods do not have to tested by SR 3.1.4.3 since they are not trippable. Periodically, the part-length rods may need to be moved to help restore the mechanical seal integrity of the control rod drive mechanism. Performing part-length rod exercising in conjunction with SR 3.1.4.3 ensures it is performed under controlled conditions. CTS 3.10.3, Part-Length Control Rods states that "The part-length control rods will be completely withdrawn from the core (except for control rod exercises and physics tests)." As such, the Applicability Note in ITS 3.1.5 as clarified by the Bases is consistent with the current licensing basis.

8. A discussion has been added in the Bases under the LCO section to clarify that if an individual shutdown or part-length rod does not meet the insertion limit requirement, then LCO 3.1.4, "Control Rod Alignment," may be entered as long as the remainder of the group is above its insertion limits. This discussion was added to help avoid confusion since LCO 3.1.5 is written to address shutdown and part-length rods on a group basis and LCO 3.1.4 addresses individual rod misalignments. This is a plant specific change to reflect the Palisades control rod design and CTS requirements.

9. The Palisades plant was designed prior to issuance of the General Design Criteria (GDC) in 10 CFR 50. Therefore, reference to the GDCs is omitted and appropriately replaced by reference to the "Palisades Nuclear Plant design." The Palisades Nuclear Plant design was compared to the GDCs as they appeared in 10 CFR 50 Appendix A on July 7, 1971. It was this updated discussion, including the identified exemptions, which formed the original plant Licensing Basis for future compliance with the GDCs.

10. The Palisades plant always runs with the rod control system in manual. The automatic feature of the rod control system has been disabled. Therefore, references to automatic rod control have been deleted. This is a plant specific change to reflect the Palisades design and operating practices.

Change **Discussion**

14. The NUREG-1432 Bases in the Applicability section states "In MODE 3, 4, 5, or 6, the shutdown CEAs are fully inserted in the core and contribute to the SDM. In the proposed ITS, MODE 3 was deleted from this sentence and another sentence added to state "In MODE 3, the shutdown rod groups are not always fully inserted. In addition, the term "fully inserted" is changed in the proposed ITS to state "to at least the lower electrical limit." This change is made to remove confusion with respect to what constitutes "full inserted." For the Palisades control rod design, the lower electrical limit corresponds to the point where electrical rod insertion ceases, and is about 3 inches from the bottom of full rod travel. The reactivity level in this region is negligible. These changes are plant specific changes to provide clarification of the requirements for shutdown rod groups.

15. To reflect the incorporation of TSTF-136 which consolidates ISTS 3.1.1 and ISTS 3.1.2, the specification number for ISTS 3.1.6, "Shutdown CEA Insertion Limits," has been changed to ITS 3.1.5 and conforming changes have been made to the Bases. These changes are consistent with NUREG-1432 as modified by TSTF-136.

16. The definition of Shutdown Margin was revised in NUREG-1432 to clarify that changes in fuel and moderator temperature are included in the determination of the Control Element Assembly Power Dependent Insertion Limits which are used to ensure adequate Shutdown Margin in MODEs 1 and 2. As a result of this change, ISTS 3.1.6 Required Action A.1.1 (verify SDM) and Required Action A.1.2 (initiate boration) have been deleted since they are no longer necessary to ensure adequate Shutdown Margin. Therefore, these Required Actions and associated Bases discussions are not included in proposed ITS 3.1.5. This change is consistent with NUREG-1432 as modified by TSTF-67.

17. ISTS 3.1.6 Required Action A.1 (as modified by TSTF-67) allows 2 hours to restore out-of-limit shutdown rods to within the limit of the LCO. Proposed ITS 3.1.5 Required Action A.1 requires out-of-limit shutdown (and part-length) rods to be declared inoperable and the Conditions and Required Actions of ITS 3.1.4 entered immediately. Anytime it is discovered that a control rod can not be moved by its operator the control rod must be considered inoperable. Since movement of the shutdown rods is typically limited to the control rod exercise test, the inability to restore a shutdown rod to within the limits of the LCO would be indicative of an inoperable (i.e., immovable) control rod. Therefore, the Required Actions for a shutdown rod outside its specified limit has been changed to be consistent with the Required Actions for an inoperable control rod.

Change	Discussion

12. The Palisades Nuclear Plant analysis does not model separate insertion limits for transient and steady state conditions as specified in Conditions A, B and C of NUREG-1432. The Palisades Nuclear Plant PDIL limits specify the regulating rod group position limits which account for anticipated power maneuvers and transient mitigation. Therefore, the proposed Palisades ITS removes the steady state and transient insertion limit discussion, where appropriate, and provides a discussion of the Palisades Nuclear Plant insertion limits. This is a plant specific change to reflect the Palisades CTS and analysis.

13. A discussion has been added in the Bases under the LCO section to clarify that for a control rod group to be considered above its insertion limit, all rods in that group must be above the insertion limit. This is a plant specific change to reflect the Palisades control rod design and CTS requirements.

14. To reflect the incorporation of TSTF-136 which consolidates ISTS 3.1.1 and ISTS 3.1.2, the specification number for ISTS 3.1.7, "Shutdown CEA Insertion Limits," has been changed to ITS 3.1.6 and conforming changes have been made to the Bases. These changes are consistent with NUREG-1432 as modified by TSTF-136.

15. The definition of Shutdown Margin was revised in NUREG-1432 to clarify that changes in fuel and moderator temperature are included in the determination of the Control Element Assembly Power Dependent Insertion Limits which are used to ensure adequate Shutdown Margin in MODES 1 and 2. As a result of this change, ISTS 3.1.7 Required Action A.1.1 (verify SDM) and Required Action A.1.2 (initiate boration) have been deleted since they are no longer necessary to ensure adequate Shutdown Margin. Therefore, these Required Actions and associated Bases discussions are not included in proposed ITS 3.1.6. An expanded discussion has been incorporated in the Applicable Safety Analyses portion of the Bases to clarify the requirements for SDM as it applies to control rod position. These change are consistent with NUREG-1432 as modified by TSTF-67.

ENCLOSURE 5

CONSUMERS ENERGY COMPANY
PALISADES PLANT
DOCKET 50-255

CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS
RESPONSE TO DECEMBER 4, 1998
REQUEST FOR ADDITIONAL INFORMATION

REVISED PAGES FOR SECTION 3.2

CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS
RESPONSE TO DECEMBER 4, 1998 REQUEST FOR ADDITIONAL INFORMATION
REVISED PAGES FOR SECTION 3.2

Page Change Instructions

Revise the Palisades submittal for conversion to Improved Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by date and contain vertical lines in the margin indicating the areas of change.

REMOVE PAGES	INSERT PAGES	REV DATE	NRC COMMENT#
ATTACHMENT 1 TO ITS CONVERSION SUBMITTAL			
ITS 3.2.1-1	ITS 3.2.1-1	02/05/99	RAI 3.2-01
ITS 3.2.1-2	ITS 3.2.1-2	02/05/99	RAI 3.2-01
ITS 3.2.1-3	ITS 3.2.1-3	02/05/99	RAI 3.2-01
ATTACHMENT 2 TO ITS CONVERSION SUBMITTAL			
ITS B 3.2.1-2	ITS B 3.2.1-2	02/05/99	RAI 3.2-01
ITS B 3.2.1-3	ITS B 3.2.1-3	02/05/99	RAI 3.2-01
ITS B 3.2.1-5	ITS B 3.2.1-5	02/05/99	RAI 3.2-01
ITS B 3.2.1-6	ITS B 3.2.1-6	02/05/99	RAI 3.2-01
ITS B 3.2.1-7	ITS B 3.2.1-7	02/05/99	RAI 3.2-01
ITS B 3.2.1-8	ITS B 3.2.1-8	02/05/99	RAI 3.2-01
ITS B 3.2.1-9	ITS B 3.2.1-9	02/05/99	RAI 3.2-01
ITS B 3.2.3-2	ITS B 3.2.3-2	02/05/99	RAI 3.2-07
ATTACHMENT 3 TO ITS CONVERSION SUBMITTAL			
DOC 3.2.1, pg 2 of 7	DOC 3.2.1, pg 2 of 7	02/05/99	RAI 3.2-01
DOC 3.2.1, pg 4 of 7	DOC 3.2.1, pg 4 of 7	02/05/99	RAI 3.2-01
DOC 3.2.2, pg 3 of 4	DOC 3.2.2, pg 3 of 4	02/05/99	RAI 3.2-04
ATTACHMENT 4 TO ITS CONVERSION SUBMITTAL			
NSHC 3.2.2, pg 1 of 5	NSHC 3.2.2, pg 1 of 5	02/05/99	RAI 3.2-04
ATTACHMENT 5 TO ITS CONVERSION SUBMITTAL			
NUREG 3.2-1	NUREG 3.2-1	02/05/99	RAI 3.2-01
NUREG 3.2-1 insert	NUREG 3.2-1 insert	02/05/99	editorial
NUREG 3.2-2 insert	NUREG 3.2-2 insert	02/05/99	RAI 3.2-02
NUREG 3.2-3	NUREG 3.2-3	02/05/99	RAI 3.2-01
NUREG B 3.2-4 insert	NUREG B 3.2-4 insert	02/05/99	RAI 3.2-01
NUREG B 3.2-5 insert	NUREG B 3.2-5 insert	02/05/99	RAI 3.2-01
NUREG B 3.2-23	NUREG B 3.2-23	02/05/99	RAI 3.2-07
ATTACHMENT 6 TO ITS CONVERSION SUBMITTAL			
JFD 3.2.1, pg 1 of 5	JFD 3.2.1, pg 1 of 5	02/05/99	RAI 3.2-01
JFD 3.2.1, pg 3 of 5	JFD 3.2.1, pg 3 of 5	02/05/99	RAI 3.2-01
JFD 3.2.1, pg 4 of 5	JFD 3.2.1, pg 4 of 5	02/05/99	RAI 3.2-01

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Linear Heat Rate (LHR)

LCO 3.2.1 LHR shall be within the limits specified in the COLR, and the Incore Alarm System or Excore Monitoring System shall be OPERABLE to monitor LHR.

APPLICABILITY: MODE 1 with THERMAL POWER > 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LHR, as determined by the automatic Incore Alarm System, not within limits specified in the COLR, as indicated by four or more coincident incore channels. <u>OR</u> LHR, as determined by the Excore Monitoring System, not within limits specified in the COLR. <u>OR</u> LHR, as determined by manual incore detector readings, not within limits specified in the COLR.	A.1 Restore LHR to within limits.	1 hour

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Incore Alarm and Excore Monitoring Systems inoperable for monitoring LHR.	B.1 Reduce THERMAL POWER to ≤ 85% RTP. AND B.2 Verify LHR is within limits using manual incore readings.	2 hours 4 hours AND Once per 2 hours thereafter
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER to ≤ 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	-------------------NOTE-------------------- Only required when Incore Alarm System is being used to monitor LHR. --- Verify LHR is within the limits specified in the COLR.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	-------------------NOTE------------------- Only required when Incore Alarm System is being used to monitor LHR. -- Adjust incore alarm setpoints based on a measured power distribution.	Prior to operation > 50% RTP after each fuel loading AND 31 EFPD thereafter
SR 3.2.1.3	-------------------NOTE------------------- Only required when Excore Monitoring System is being used to monitor LHR. -- Verify measured ASI has been within 0.05 of target ASI for last 24 hours.	Prior to each initial use of Excore Monitoring System to monitor LHR
SR 3.2.1.4	-------------------NOTE------------------- Only required when Excore Monitoring System is being used to monitor LHR. -- Verify THERMAL POWER is less than the APL.	1 hour

BASES

BACKGROUND (continued)

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions.

The limits on LHR, Assembly Radial Peaking Factor (F_r^A), Total Radial Peaking Factor (F_r^T), QUADRANT POWER TILT (T_q), and AXIAL SHAPE INDEX (ASI), which are obtained directly from the core reload analysis, ensure compliance with the safety limits on LHR and Departure from Nucleate Boiling Ratio (DNBR).

Either of the two core power distribution monitoring systems, the Incore Alarm System or the Excore Monitoring System, provides adequate monitoring of the core power distribution and is capable of verifying that the LHR is within its limits. The Incore Alarm System performs this function by continuously monitoring the local power at many points throughout the core and comparing the measurements to predetermined setpoints above which the limit on LHR could be exceeded. The Excore Monitoring System performs this function by providing comparison of the measured core ASI with predetermined ASI limits based on incore measurements. An Excore Monitoring System Allowable Power Level (APL), which may be less than RATED THERMAL POWER, and an additional restriction on T_q, are applied when using the Excore Monitoring System to ensure that the ASI limits adequately restrict the LHR to less than the limiting values.

In conjunction with the use of the Excore Monitoring System for monitoring LHR and in establishing ASI limits, the following assumptions are made:

a. The control rod insertion limits of LCO 3.1.5, "Shutdown and Part-Length Rod Group Insertion Limits," and LCO 3.1.6, "Regulating Rod Group Position Limits," are satisfied;

b. The additional T_q restriction of SR 3.2.1.6 is satisfied; and

c. Radial Peaking Factors, F_r^A and F_r^T, do not exceed the limits of LCO 3.2.2.

BASES

BACKGROUND (continued)

The limitations on the Radial Peaking Factors provided in the COLR ensure that the assumptions used in the analysis for establishing the LHR limits and Limiting Safety System Settings (LSSS) remain valid during operation at the various allowable control rod group insertion limits.

The Incore Alarm System continuously provides a direct measure of the LHR and the Radial Peaking factors. It also provides alarms that have been established for the individual incore detector segments, ensuring that the peak LHRs are maintained within the limits specified in the COLR. The setpoints for these alarms include tolerances, set in conservative directions, for:

a. A measurement calculational uncertainty factor (as identified in the COLR);

b. An engineering uncertainty factor of 1.03; and

c. A THERMAL POWER measurement uncertainty factor of 1.02.

The measurement uncertainties associated with LHR, F_r^A and F_r^T are based on a statistical analysis performed on power distribution benchmarking results. The COLR includes the applicable measurement uncertainties for fresh and depleted incore detector usage. The engineering and THERMAL POWER uncertainties are incorporated in the power distribution calculation performed by the fuel vendor.

The excore power distribution monitoring system consists of Power Range Channels 5 through 8. The power range channels monitor neutron flux from 0 to 125 percent full power. They are arranged symmetrically around the reactor core to provide information on the radial and axial flux distributions.

The power range detector assembly consists of two uncompensated ion chambers for each channel. One detector extends axially along the lower half of the core while the other, which is located directly above it, monitors flux from the upper half of the core. The DC current signal from each of the ion chambers is fed directly to the control room drawer assembly without pre-amplification. Each excore detector supplies data to a Thermal Margin Monitor (TMM). Each TMM uses these excore signals to calculate Axial Shape Index (ASI) on a continuous basis.

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm; and

d. The full-length control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. 4). This is accomplished by maintaining the power distribution and primary coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by accident analyses (Ref. 1), with due regard for the correlations between measured quantities, the power distribution, and uncertainties in determining the power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate so that the peak cladding temperature does not exceed 2200°F (Ref. 4). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and the Primary Coolant System Operation ensure that these criteria are met as long as the core is operated within the LHR, ASI, F_r^A, F_r^T, and T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses.

Fuel cladding damage does not necessarily occur while the plant is operating at conditions outside the limits of these LCOs during normal operation. Fuel cladding damage could result, however, if an accident occurs from initial conditions outside the limits of these LCOs. The potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and can correspondingly increase local LHR.

The Incore Alarm System provides for monitoring of LHR, radial peaking factors, and QUADRANT POWER TILT to ensure that fuel design conditions and safety analysis assumptions are maintained. The Incore Alarm System is also utilized to determine the target AXIAL OFFSET (AO) and to determine the Allowable Power Level (APL) when using the excore detectors.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The Excore Monitoring System provides for monitoring of ASI and QUADRANT POWER TILT to ensure that fuel design conditions and safety analysis assumptions are maintained.

The LHR satisfies Criterion 2 of 10 CFR 50.36(c)(2).

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The power distribution LCO limits, except T_q, are provided in the COLR. The limitation on the LHR in the peak power fuel rod at the peak power elevation Z ensures that, in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F.

The LCO requires that LHR be maintained within the limits specified in the COLR and either the Incore Alarm System or Excore Monitoring System be OPERABLE to monitor LHR. When using the Incore Alarm System, the LHR is not considered to be out of limits until there are four or more incore detectors simultaneously in alarm. When using the Excore Monitoring System, LHR is considered within limits when the conditions are acceptable for use of the Excore Monitoring System and the associated ASI and T_q limits specified in the SRs are met.

To be considered OPERABLE, the Incore Alarm System must have at least 160 of the 215 possible incore detectors OPERABLE and 2 incore detectors per axial level per core quadrant OPERABLE. In addition, the plant process computer must be OPERABLE and the required alarm setpoints entered into the plant computer.

To be considered OPERABLE, the Excore Monitoring System must have been calibrated with OPERABLE incore detectors, the ASI must not have been out of limits for the last 24 hours, and THERMAL POWER must be less than the APL.

BASES

APPLICABILITY

In MODE 1 with THERMAL POWER > 25% RTP, power distribution must be maintained within the limits assumed in the accident analysis to ensure that fuel damage does not result following an AOO. In MODE 1 with THERMAL POWER ≤ 25% RTP, and in other MODES, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on the core power distribution, and because ample thermal margin exists to ensure that the fuel integrity is not jeopardized and safety analysis assumptions remain valid.

ACTIONS

A.1

There are three acceptable methods for verifying that LHR is within limits. The LCO requires monitoring by either an OPERABLE Incore Alarm System or an OPERABLE Excore Monitoring System. When both of the required systems are inoperable, Condition B allows for monitoring by taking manual readings of the incore detectors. Any of these three methods may indicate that the LHR is not within limits. With the LHR exceeding its limit, excessive fuel damage could occur following an accident. In this Condition, prompt action must be taken to restore the LHR to within the specified limits. One hour to restore the LHR to within its specified limits is reasonable and ensures that the core does not continue to operate in this Condition. The 1 hour Completion Time also allows the operator sufficient time for evaluating core conditions and for initiating proper corrective actions.

BASES

ACTIONS (continued)

B.1 and B.2

With the Incore Alarm System inoperable for monitoring LHR and the Excore Monitoring System inoperable for monitoring LHR, THERMAL POWER must be reduced to ≤ 85% RTP within 2 hours. Operation at ≤ 85% RTP ensures that ample thermal margin is maintained. A 2 hour Completion Time is adequate to achieve the required plant condition without challenging plant systems. Additionally, with the Incore Alarm and Excore Monitoring Systems inoperable, LHR must be verified to be within limits within 4 hours, and every 2 hours thereafter by manually collecting incore detector readings at the terminal blocks in the control room utilizing a suitable signal detector. The manual readings shall be taken on a minimum of 10 individual detectors per quadrant (to include a total of 160 detectors in a 10 hour period). The time interval of 2 hours and the minimum of 10 detectors per quadrant are sufficient to maintain adequate surveillance of the power distribution to detect significant changes until the monitoring systems are returned to service.

C.1

If the Required Action and associated Completion Time are not met, THERMAL POWER must be reduced to ≤ 25% RTP. This reduced power level ensures that the core is operating within its thermal limits and places the core in a conservative condition. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reach ≤ 25% RPT from full power MODE 1 conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

The Incore Alarm System provides continuous monitoring of LHR through the plant computer. The plant computer is used to generate alarm setpoints that are based on measured margin to allowed LHR. As the incore detectors are read by the plant computer, they are continuously compared to the alarm setpoints. If the Incore Alarm System LHR monitoring function is inoperable, excore detectors or manual recordings of the incore detector readings may be used to monitor LHR. Periodically monitoring LHR ensures that the assumptions made in the Safety Analysis are maintained. This SR is modified by a Note that states that the SR is only applicable when the Incore Alarm System is being used to monitor LHR. The 12 hour Frequency is consistent with an SR which is to be performed each shift.

SR 3.2.1.2

Continuous monitoring of the LHR is provided by the Incore Alarm System which provides adequate monitoring of the core power distribution and is capable of verifying that the LHR does not exceed its specified limits.

Performance of this SR verifies the Incore Alarm System can accurately monitor LHR by ensuring the alarm setpoints are based on a measured power distribution. Therefore, they are only applicable when the Incore Alarm System is being used to determine the LHR.

The alarm setpoints must be initially adjusted following each fuel loading prior to operation above 50% RTP, and periodically adjusted every 31 Effective Full Power Days (EFPD) thereafter. A 31 EFPD Frequency is consistent with the historical testing frequency of the reactor monitoring system. The SR is modified by a Note which allows the SR to be performed only when the Incore Alarm System is being used to determine LHR.

BASES

ACTIONS

A.1

If the measured T_q is > 0.05, T_q must be restored within 2 hours or F_r^A and F_r^T must be determined to be within the limits of LCO 3.2.2, and determined to be within these limits every 8 hours thereafter, as long as T_q is out of limits. Two hours is sufficient time to allow the operator to reposition control rods, and significant radial xenon redistribution cannot occur within this time. The 8 hour Completion Time ensures changes in F_r^A and F_r^T can be identified before the limits of LCO 3.2.2 are exceeded.

B.1

With the measured T_q > 0.10, power must be reduced to < 50% RTP within 4 hours, and F_r^A and F_r^T must be within their specified limits to ensure that acceptable flux peaking factors are maintained as required by Condition A (which continues to be applicable). Based on operating experience, 4 hours is sufficient time for evaluation of these factors. If F_r^A and F_r^T are within limits, operation may proceed while attempts are made to restore T_q to within its limit. If the tilt is generated due to a control rod misalignment, continued operation at < 50% RTP allows for realignment; if the cause is other than control rod misalignment, continued operation may be necessary to discover the cause of the tilt. Reducing THERMAL POWER to < 50% RTP, and the more frequent measurement of peaking factors required by ACTION A.1, provide conservative protection from potential increased peaking due to xenon redistribution.

C.1

If T_q is > 0.15, or if Required Actions and associated Completion Times are not met, THERMAL POWER must be reduced to ≤ 25% RTP. This requirement ensures that the core is operating within its thermal limits and places the core in a conservative condition. Four hours is a reasonable time to reach 25% RTP in an orderly manner and without challenging plant systems.

A.4 CTS 3.23.1 provides actions when the LHR is being monitored by the excore monitoring system but the system is no longer appropriate for monitoring LHR as indicated by an Axial Offset (AO) of more than 0.05 (ACTION 2). The actions include both "discontinue using the excore monitoring system for monitoring LHR" and "follow the procedure in ACTION 3 below." Inherent in entry into CTS 3.23.1 ACTION 2 is that the normally used Incore Alarm System is inoperable. Therefore, this situation is one with both the Incore Alarm System and the excore monitoring system inoperable for the purpose of monitoring LHR. This is included as ITS 3.2.1 Condition B. The specific direction to enter this Condition is not included in ITS since this is the normal use and application of the improved STS format. Therefore, this omission is considered an administrative change.

A.5 CTS 3.23.1 provides actions when the LHR is indicated as not within the limits specified in the COLR by four or more coincident incore alarms (ACTION 1), and when the manually recorded incore readings indicate a local power level greater than the alarm setpoints (ACTION 3). However, no specific action is provided in the CTS for when the LHR is not within limits as monitored by the excore monitoring system. The ITS includes a second entry condition for ITS 3.2.1 Condition A specifically for when the LHR is determined to be not within limits using the excore monitoring system, Since the appropriate action is the same regardless of the method used to determine that LHR is not within limits, the addition of a specific Required Action, entry condition for "LHR, as determined by the Excore Monitoring System, not within limits specified in the COLR" is considered an administrative change.

A.6 CTS 3.23.1 ACTION 3 indicates that when the LHR is indicated as not within the limits specified in the COLR by the manually recorded incore readings "the action specified in ACTION 1 above shall be taken." The ITS includes a third entry condition for ITS 3.2.1 Condition A specifically for when the LHR is determined to be not within limits using the manual incore readings, Since these are only different formats to require the same action, the addition of a specific Required Action, entry condition for "LHR, as determined by manual incore readings, not within limits specified in the COLR" is considered an administrative change.

M.2 CTS does not include specific surveillance requirements to verify that LHR remains within limits. Such an SR is included as ITS SR 3.2.1.1. This SR is necessary to provide direct verification that the LCO requirements are met when using the Incore Alarm System for monitoring LHR. Consistent with the NUREG, verification that an OPERABLE Incore Alarm System does not indicate LHR out of limits is sufficient to fulfill this SR. This is an additional restriction on plant operation.

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

LA.1 CTS 3.23.1 contains specific details regarding the requirements for monitoring of the LHR, i.e., "in the peak power fuel rod at the peak power elevation Z." This information is not required to be provided in NUREG LCO 3.2.1. These details describe elements of the LHR which are addressed by the methodology for determining LHR and are not directly a part of the actual requirement, i.e., Limiting Condition for Operation. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in the LCO Bases of ITS 3.2.1 provides adequate assurance that they will be maintained. The Bases are controlled by the Bases Control Program in Chapter 5 of the proposed Technical Specifications. This change is consistent with NUREG-1432.

LA.2 CTS 3.23.1 ACTION 3 contains specific details regarding the requirements for monitoring of LHR by manual readings of the incore detection system when the incore LHR alarm system is inoperable, i.e., "readings shall be taken on a minimum of 10 individual detectors per quadrant (to include a total number of 160 detectors in a 10-hour period)." This information is not provided in NUREG LCO 3.2.1. These details describe elements of the incore detection system requirements which are addressed by the methodology for proper use of the system and are not directly a part of the actual requirement, i.e., Limiting Condition for Operation. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in the Bases of ITS 3.2.1 provides adequate assurance that they will be maintained. The Bases are controlled by the Bases Control Program in Chapter 5 of the proposed Technical Specifications. This change is consistent with NUREG-1432.

LA.2 CTS 4.19.2.1 provides Surveillance Requirements (SRs) for the Radial Peaking Factors. However, it contains specific details for monitoring of the peaking factors, i.e., that the SR is performed by verifying the "measured" radial peaking factors "obtained by using the incore detection system." This information is not provided in NUREG SR 3.2.2.1. These details describe elements of the radial peaking factor verification which are addressed by the methodology and are not directly a part of the actual requirement, i.e., Surveillance Requirement. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in the Bases of ITS SR 3.2.2.1 provides adequate assurance that they will be maintained. The Bases are controlled by the Bases Control Program in Chapter 5 of the proposed Technical Specifications. This change is consistent with NUREG-1432.

LESS RESTRICTIVE CHANGES (L)

L.1 CTS 3.23.2 provides actions for peaking factors exceeding their limits based on power level. The first of these actions is for P (power) < 50%, and requires the plant to be in at least hot shutdown (i.e., subcritical) within 6 hours. ITS 3.2.2 Required Action A.1 provides 6 hours to attempt restoration of the peaking factors to within limits, and if the Required Action and its associated Completion Time is not met, then Required Action B.1 requires that THERMAL POWER be reduced to ≤ 25% RTP. This change is less restrictive in two ways. First, 6 hours is provided to attempt restoration of the peaking factors to within limits that is not provided in the CTS. Second, the default action requires only that the plant to be reduced to ≤ 25% RTP, rather than subcritical, in the subsequent 4 hours.

The ITS Required Action to restore the radial peaking factors to the within limits specified in the COLR assure the plant will not operate for an extended period with the peaking factors not within limits. The Completion Time of 6 hours provides a reasonable time for evaluating core conditions, calculating a reduced power level at which the peaking factors would be within limits, determining the proper method for the power reduction (e.g., rod positioning and/or boration) and, completing the reduction in power. In the event the peaking factors are not restored to within limits, an additional 4 hours is provided to remove the plant from the mode of applicability. Although CTS 3.23.2 requires the plant to be placed in hot shutdown, terminating the power reduction anywhere below 25% is permissible since CTS LCO 3.0.1 only requires compliance with an LCO during the plant condition specified in that LCO. Thus, the default action of proposed ITS Required Action B.1 is consistent with the shutdown action for CTS 3.23.2. A Completion Time of 4 hours is reasonable to reduce thermal power below 25% in an orderly manner and without challenging plant systems.

LESS RESTRICTIVE CHANGE L.1

CTS 3.23.2 provides actions for peaking factors exceeding their limits based on power level. The first of these actions is for P (power) < 50%, and requires the plant to be in at least hot shutdown (i.e., subcritical) within 6 hours. ITS 3.2.2 Required Action A.1 provides 6 hours to attempt restoration of the peaking factors to within limits, and if the Required Action and its associated Completion Time is not met, then Required Action B.1 requires that THERMAL POWER be reduced to ≤ 25% RTP. This change is less restrictive in two ways. First, 6 hours is provided to attempt restoration of the peaking factors to within limits that is not provided in the CTS. Second, the default action requires only that the plant to be reduced to ≤ 25% RTP, rather than subcritical, in the subsequent 4 hours.

The ITS Required Action to restore the radial peaking factors to the within limits specified in the COLR assure the plant will not operate for an extended period with the peaking factors not within limits. The Completion Time of 6 hours provides a reasonable time for evaluating core conditions, calculating a reduced power level at which the peaking factors would be within limits, determining the proper method for the power reduction (e.g., rod positioning and/or boration) and, completing the reduction in power. In the event the peaking factors are not restored to within limits, an additional 4 hours is provided to remove the plant from the mode of applicability. Although CTS 3.23.2 requires the plant to be placed in hot shutdown, terminating the power reduction anywhere below 25% is permissible since CTS LCO 3.0.1 only requires compliance with an LCO during the plant condition specified in that LCO. Thus, the default action of proposed ITS Required Action B.1 is consistent with the shutdown action for CTS 3.23.2. A Completion Time of 4 hours is reasonable to reduce thermal power below 25% in an orderly manner and without challenging plant systems.

(4)

3.2 POWER DISTRIBUTION LIMITS

(4) 3.2.1 Linear Heat Rate (LHR) ~~(Analog)~~

(2) 3.23.1 LCO (8) LCO 3.2.1 LHR shall ~~not exceed~~ be within the limits specified in the COLR. and the INCore Alarm System or Excore Monitoring System shall be OPERABLE to Monitor LHR

3.23.1 APPL (5) APPLICABILITY: MODE 1 with THERMAL POWER > 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
(4) 3.23.1 Act 1 (2) (6) A. LHR, as determined by the Incore ~~Detector~~ Alarm ~~Monitoring~~ System, ~~exceeds the~~ not within limits ~~of~~ specified in ~~Figure 3.2.1-1 of~~ the COLR, as indicated by four or more coincident incore channels. OR new (4) (2) LHR, as determined by the Excore ~~Detector~~ Monitoring System, ~~exceeds the limits as indicated by the ASI outside the power dependent control limits as~~ not within specified in ~~Figure 3.2.1-2 of~~ the COLR. (6) (7) 3.23.1 Act 3 (5) INSERT OR LHR, as determined by manual incore readings, not within limits specified in the COLR.	A.1 Restore LHR to within limits.	1 hour
3.23.1 Act 2/3 3.23.1 Act 3 (5) ~~B~~ C. Required Action and associated Completion Time not met.	~~B~~ C.1 ~~Be in MODE 2.~~ Reduce THERMAL POWER to ≤ 25% RTP.	~~6~~ 4 hours

Revised 02/05/99

SECTION 3.2

INSERT

3.23.1 ACT 2	B. Incore Alarm and Excore Monitoring Systems inoperable for monitoring LHR.	B.1 Reduce THERMAL POWER to ≤ 85% RTP. AND	2 hours
3.23.1 ACT 3		B.2 Verify LHR is within limits using manual incore readings.	4 hours AND Once per 2 hours thereafter

Revised
02/05/99

SECTION 3.2

INSERT

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	--------------NOTE--------------- Only required when Incore Alarm System is being used to monitor LHR. --------------------------------- Verify LHR is within the limits specified in the COLR.	 12 hours

(4)

LHR ~~(Analog)~~
3.2.1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.~~8~~ (2)	Alarm ~~NOTES~~ ~~1.~~ Only applicable when the Incore ~~Detector Monitoring~~ System is being used to determine LHR. ~~2. Not required to be performed below 20% RTP.~~ Adjust ~~Verify~~ incore ~~detector local power density~~ alarm setpoints ~~are less than or equal to the limits specified in the COLR.~~ based on a measured power distribution.	Prior to operation >50% RTP after each fuel loading AND 31 ~~days~~ EFPD thereafter

4.19.1.1

(9)

(10)

(9) INSERT

4.19.1.2.a
4.19.1.2.b
4.19.1.2.c
4.19.1.2.d

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02/05/99

SECTION 3.2

INSERT A

The Incore Alarm System provides for monitoring of LHR, radial peaking factors, and QUADRANT POWER TILT to ensure that fuel design conditions and safety analysis assumptions are maintained. The Incore Alarm System is also utilized to determine the target AXIAL OFFSET (AO) and to determine the Allowable Power Level (APL) when using the excore detectors.

The Excore Monitoring System provides for monitoring of ASI and QUADRANT POWER TILT to ensure that fuel design conditions and safety analysis assumptions are maintained.

INSERT B

The LCO requires that LHR be maintained within the limits specified in the COLR and either the Incore Alarm System or Excore Monitoring System be OPERABLE to monitor LHR. When using the Incore Alarm System, the LHR is not considered to be out of limits until there are four or more incore detectors simultaneously in alarm. When using the Excore Monitoring System, LHR is considered within limits when the conditions are acceptable for use of the Excore Monitoring System and the associated ASI and T_q limits specified in the SRs are met.

To be considered OPERABLE, the Incore Alarm System must have at least 160 of the 215 possible incore detectors OPERABLE and 2 incore detectors per axial level per core quadrant OPERABLE. In addition, the plant process computer must be OPERABLE and the required alarm setpoints must be entered into the plant computer.

To be considered OPERABLE, the Excore Monitoring System must have been calibrated with OPERABLE incore detectors, the ASI must not have been out of limits for the last 24 hours, and THERMAL POWER must be less than the APL.

SECTION 3.2

INSERT A

B.1 and B.2.

With the Incore Alarm System inoperable for monitoring LHR and the Excore Monitoring System inoperable for monitoring LHR, THERMAL POWER must be reduced to ≤ 85% RTP within 2 hours. Operation at ≤ 85% RTP ensures that ample thermal margin is maintained. A 2 hour Completion Time is adequate to achieve the required unit condition without challenging plant systems. Additionally, with the Incore Alarm and Excore Monitoring Systems inoperable, LHR must be verified to be within limits within 4 hours, and every 2 hours thereafter by manually collecting incore detector readings at the terminal blocks in the control room utilizing a suitable signal detector. The manual readings shall be taken on a minimum of 10 individual detectors per quadrant (to include a total of 160 detectors in a 10 hour period). The time interval of 2 hours and the minimum of 10 detectors per quadrant are sufficient to maintain adequate surveillance of the power distribution to detect significant changes until the monitoring systems are returned to service.

INSERT B

The Incore Alarm System provides continuous monitoring of LHR through the plant computer. The plant computer is used to generate alarm setpoints that are based on measured margin to allowed LHR. As the incore detectors are read by the plant computer, they are continuously compared to the alarm setpoints. If the Incore Alarm System LHR monitoring function is inoperable, excore detectors or manual recordings of the incore detector readings may be used to monitor LHR. Periodically monitoring LHR ensures that the assumptions made in the Safety Analysis are maintained. This SR is modified by a Note that states that the SR is only applicable when the Incore Alarm System is being used to monitor LHR.

BASES (continued)

ACTIONS

A.1 ~~and A.2~~

0.05,

If the measured T_q is > ~~[0.03] and < 0.10, the calculation of T_q may be nonconservative~~ T_q must be restored within 2 hours or ~~F_{xy}~~ F_r^A and F_r^T must be determined to be within the limits of LCO 3.2.2 ~~and LCO 3.2.3~~, and determined to be within these limits every 8 hours thereafter, as long as T_q is out of limits. ~~Two~~ Four hours is sufficient time to allow the operator to reposition ~~CEAs~~ control rods, and significant radial xenon redistribution cannot occur within this time. The 8 hour Completion Time ensures changes in ~~F_{xy}~~ F_r^A and F_r^T can be identified before the limits of LCO 3.2.2 ~~and LCO 3.2.3, respectively,~~ are exceeded.

6 7 8 9 4 8

~~B.1~~ C.1

If T_q is > 0.15, or if Required Actions and associated Completion Times ~~of Condition A~~ are not met, THERMAL POWER must be reduced to ≤ ~~50~~ 25% RTP. This requirement ensures that the core is operating within its thermal limits and places the core in a conservative condition. Four hours is a reasonable time to reach ~~50~~ 25% RTP in an orderly manner and without challenging plant systems.

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B.1 and B.2

~~C.1, C.2, and C.3~~

With the measured T_q > 0.10, power must be reduced to < 50% RTP within 4 hours, and ~~F_{xy}~~ F_r^A and F_r^T must be within their specified limits to ensure that acceptable flux peaking factors are maintained. Based on operating experience, ~~1 hour~~ 4 hours at < 50% RTP is sufficient time for ~~the operator to evaluate~~ evaluation of these factors. If ~~F_{xy}~~ F_r^A and F_r^T are within limits, operation may proceed ~~for a total of 2 hours after the Condition is entered~~ while attempts are made to restore T_q to within its limit, as required by Condition A, which continues to be applicable.

2 6 4 5 4

~~If T_q ≤ 0.10 cannot be achieved, power must be reduced to ≤ 50% RTP within 2 hours. If the tilt is generated due to a CEA misalignment, operating at ≤ 50% RTP allows for the recovery of the CEA. Except as a result of CEA misalignment, T_q > 0.10 is not expected; if it occurs, continued operation of the reactor may be necessary to discover the cause of the tilt. If this procedure is followed, operation is restricted to only those conditions required to identify the cause of the tilt. It is necessary to account explicitly for power asymmetries because the~~

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(continued)

5 If the tilt is due to control rod misalignment, continued operation at < 50% RTP allows for realignment; if the cause is other than control rod misalignment, continued operation may be necessary to discover the cause of the tilt. Reducing THERMAL POWER to < 50% RTP, and the more frequent measurement of peaking factors required by Action A.1 provide conservative protection from potential increased peaking due to xenon distribution.

Revised
02/05/99

Change	Discussion
Note:	This attachment provides a brief discussion of the deviations from NUREG-1432 that were made to support the development of the Palisades Nuclear Plant ITS. The Change Numbers correspond to the respective deviation shown on the "NUREG MARKUPS." The first five justifications were used generically throughout the markup of the NUREG. Not all generic justifications are used in each specification.
1.	The brackets have been removed and the proper plant specific information or value has been provided.
2.	Deviations have been made for clarity, grammatical preference, or to establish consistency within the Improved Technical Specifications. These deviations are editorial in nature and do not involve technical changes or changes of intent.
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 specification. These include an ITS 3.2.1 Applicability less restrictive than the NUREG and the addition of an ACTION for determination of LHR using manual readings when both the Incore Alarm System and the excore monitoring system are inoperable for determining LHR. With power reduced to below 85% RTP (per ITS 3.2.1, Required Action B.1), the manual readings of the incore monitors provide an adequate indication that LHR is within limits. This is consistent with CTS as approved in Amendment 68. Additionally, the proposed Applicability for ITS 3.2.1 is actually more restrictive than CTS 3.23.1 which is applicable only above 50% RTP. An ITS 3.2.1 Applicability of "MODE 1 > 25% RTP" is consistent with the Applicability for the other Power Distribution Limit specifications, and provides for incore adjustments based on power distribution maps prior to exceeding 25% which is consistent with Quadrant Power Tilt needs for incore adjustments.

Change	Discussion
8.	An addition to the LCO in incorporated which requires that the LHR be determined by an OPERABLE Incore Alarm System or by an OPERABLE excore monitoring system. Such an LCO requirement is consistent with the NUREG SR Note which requires that the LHR be determined by either the incore detector monitoring system or the excore detector monitoring system. However, incorporating the requirement into the LCO provides a more direct indication that the LCO is not met when both the incore LHR alarm function and the excore LHR monitoring function are inoperable (which results in entry into ITS Condition B, as discussed in JFD 5).
9.	The Surveillance Requirements (SRs) for LHR are revised consistent with the current licensing basis. The NUREG SR Note is inappropriate for Palisades Nuclear Plant because manual reading of the incore monitors is also allowed for determining LHR to be within limits. This is corrected by incorporating the SR Note requirements directly into the LCO (see JFD 8) and adding an ACTION for use of the manual incore readings (see JFDs 5 and 7). The NUREG SRs are also inappropriate for all plants since failure of the alarms or setpoints to be properly set does not mean that the LHR is not within limits. However, SR 3.0.1 would require that the LCO be considered not met when any of these SRs are not met . This is not consistent with the format and content intent of the improved STS NUREGs, is considered overly conservative, and is not adopted. ITS SR 3.2.1.1 specifically requires the verification that LHR is within the limits specified in the COLR. This SR is a direct verification that the LCO is being met (which is missing from the NUREG). However, since the LHR is normally automatically monitored and alarmed by the incore power distribution monitoring system, the SR is only required to be performed when the Incore Alarm System is being used to determine LHR, and is met by administrative verification that the Incore Alarm System is OPERABLE for monitoring LHR, and that the Incore Alarm System does not indicate LHR is not within limits.

Change	Discussion
9.	(continued)

NUREG SR 3.2.1.2 and SR 3.2.1.3 requirements for incore alarms are combined and revised to reflect CTS 4.19.1. ITS SR 3.2.1.2 requires that the incore alarm setpoints be adjusted (i.e., the alarms be set) based on a measured power distribution. This Surveillance provides adequate assurance that the Incore Alarm System is providing accurate monitoring of the LHR. This change is consistent with CTS 4.19.1 requirements for adjustments of incore alarm settings.

ITS SR 3.2.1.3, SR 3.2.1.4, SR 3.2.1.5, and SR 3.2.1.6 require the verification of parameters that similarly indicate the LHR is within the limits specified in the COLR when using the excore monitoring system. These SRs also provide verification that the parameters are appropriate for use of the excore monitoring system to monitor LHR and that the LCO is being met (which is missing from the NUREG). However, since the LHR is normally automatically monitored and alarmed by the Incore Alarm System, these SRs are only required to be met when the excore monitoring system is being used to determine LHR. These SRs are generally consistent with the requirements of CTS 4.19.1.2a, b, c, and d.

10. The periodic Frequency of NUREG SR 3.2.1.3 is revised to 31 EFPD. CTS 4.19.1.1 provides requirements to adjust the incore alarm settings based on a measured power distribution on a periodic Frequency of "7 days of power operation." Although the CTS Frequency is based on days of power operation, this is inconsistent with the Frequency of ITS Section 3.1 SRs which are based on EFPD, inconsistent with NUREGs for other vendors (e.g., NUREG-1430 and NUREG-1431) for Power Distribution Limit SRs which are based on EFPD, and inconsistent with preferred methods for tracking this Frequency since EFPD is already required to be tracked to for numerous calculations related to burnup and other fuel status parameters. When the plant is operating steadily at full power there is no difference in the NUREG SR 3.2.1.3 periodic Frequency of "31 days" and the proposed "31 EFPD." However, when the 31 days includes operation at less than full power the "31 EFPD" is longer than the NUREG would allow. Still, the revision to the SR Frequency is acceptable since the Frequency continues to be sufficient to assure the incore alarm settings are appropriately since any change is a slow process.

ENCLOSURE 6

CONSUMERS ENERGY COMPANY
PALISADES PLANT
DOCKET 50-255

CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS
RESPONSE TO DECEMBER 4, 1998
REQUEST FOR ADDITIONAL INFORMATION

REVISED PAGES FOR CHAPTER 5.0

CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS
RESPONSE TO DECEMBER 4, 1998 REQUEST FOR ADDITIONAL INFORMATION
REVISED PAGES FOR CHAPTER 5.0

Page Change Instructions

Revise the Palisades submittal for conversion to Improved Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by date and contain vertical lines in the margin indicating the areas of change.

REMOVE PAGES	INSERT PAGES	REV DATE	NRC COMMENT#
ATTACHMENT 1 TO ITS CONVERSION SUBMITTAL			
ITS 5.0-25	ITS 5.0-25	02/05/99	RAI 3.1-01
ATTACHMENT 2 TO ITS CONVERSION SUBMITTAL			
No page changes			
ATTACHMENT 3 TO ITS CONVERSION SUBMITTAL			
CTS 5.0, pg 6-20	CTS 5.0, pg 6-20	02/05/99	RAI 3.1-01
ATTACHMENT 4 TO ITS CONVERSION SUBMITTAL			
No page changes			
ATTACHMENT 5 TO ITS CONVERSION SUBMITTAL			
NUREG 5.6, pg 5.0-21	NUREG 5.6, pg 5.0-21	02/05/99	RAI 3.1-01
ATTACHMENT 6 TO ITS CONVERSION SUBMITTAL			
No page changes			

5.6 Reporting Requirements

5.6.4 Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the NRC no later than the fifteenth of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

3.1.1	Shutdown Margin
3.1.6	Regulating Rod Group Position Limits
3.2.1	Linear Heat Rate Limits
3.2.2	Radial Peaking Factor Limits
3.2.4	ASI Limits

b. The analytical methods used to determine the core operating limits shall be those approved by the NRC, specifically those described in the latest approved revision of the following documents:

1. XN-75-27(A), "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," and Supplements 1(A), 2(A), 3(P)(A), 4(P)(A), and 5(P)(A); Exxon Nuclear Company. (LCOs 3.1.1, 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

2. ANF-84-73(P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," and Appendix B(P)(A) and Supplements 1(P)(A), 2(P)(A); Advanced Nuclear Fuels Corporation. (LCOs 3.1.1, 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

3. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company. (LCOs 3.2.1, 3.2.2, & 3.2.4)

4. ANF-84-093(P)(A), "Steamline Break Methodology for PWRs," and Supplement 1(P)(A); Advanced Nuclear Fuels Corporation. (LCOs 3.1.1, 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

5. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," and Supplements 1(P)(A), 2(P)(A), 3(P)(A), and 4(P)(A); Exxon Nuclear Company. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

6.0 ADMINISTRATIVE CONTROLS

5.6.5 ~~6.6.5~~ Core Operating Limits Report (COLR)

RAI 3.1-01

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following: A.8 X

3.1.1 Shutdown Margin
3.2.4 ~~3.1.1~~ ASI Limits, Rod Position
3.1.6 ~~3.10.5~~ Regulating Group ~~Insertion~~ Limits
3.2.1 ~~3.23.1~~ Linear Heat Rate ~~(LHR)~~ Limits
3.2.2 ~~3.23.2~~ Radial Peaking Factor Limits

A.1

b. The analytical methods used to determine the core operating limits shall be those approved by the NRC, specifically those described in the latest approved revision of the following documents:

1. XN-75-27(A), "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," and Supplements 1(A), 2(A), 3(P)(A), 4(P)(A), and 5(P)(A); Exxon Nuclear Company. (LCOs ~~3.1.1~~ 3.2.4, ~~3.10.1~~ 3.1.1, ~~3.10.5~~ 3.1.6, ~~3.23.1~~ 3.2.1, & ~~3.23.2~~ 3.2.2) X A.1 A.8

2. ANF-84-73(P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," and Appendix B(P)(A) and Supplements 1(P)(A), 2(P)(A); Advanced Nuclear Fuels Corporation. (LCOs 3.1.1, ~~3.1.1~~ 3.2.4, ~~3.10.5~~ 3.1.6, ~~3.23.1~~ 3.2.1, & ~~3.23.2~~ 3.2.2) A.1 A.8

3. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company. (LCOs ~~3.1.1~~ 3.2.4, ~~3.23.1~~ 3.2.1 & ~~3.23.2~~ 3.2.2) A.1

4. ANF-84-093(P)(A), "Steamline Break Methodology for PWRs," and Supplement 1(P)(A); Advanced Nuclear Fuels Corporation. (LCOs 3.1.1, ~~3.10.1~~ 3.2.4, ~~3.10.5~~ 3.1.6, ~~3.23.1~~ 3.2.1, & ~~3.23.2~~ 3.2.2) A.1 A.8

5. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," and Supplements 1(P)(A), 2(P)(A), 3(P)(A), and 4(P)(A); Exxon Nuclear Company. (LCOs ~~3.1.1~~ 3.2.4, ~~3.10.5~~ 3.1.6, ~~3.23.1~~ 3.2.1, & ~~3.23.2~~ 3.2.2) A.1

6. EXEM PWR Large Break LOCA Model as defined by: (LCOs ~~3.10.5~~ 3.1.6, ~~3.23.1~~ 3.2.1, & ~~3.23.2~~ 3.2.2) A.1

a) XN-NF-82-20(A), "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," and Supplements 1(P)(A), 2(P)(A), 3(P)(A), and 4(P)(A); Exxon Nuclear Company.

b) XN-NF-82-07(P)(A), "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company.

c) XN-NF-81-58(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," and Supplements 1(P)(A), 2(P)(A), 3(P)(A), and 4(P)(A); Exxon Nuclear Company.

5.6 Reporting Requirements

CTS 6.6.4

5.6.4 Monthly Operating Reports (continued)

power operated relief valves or pressurizer safety valves,] shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report. (1)

6.6.5

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

RAI 3.1-01

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

LCO 3.1.1 Shutdown MARGIN
LCO 3.1.6 Regulating Rod Group Position Limits
LCO 3.2.1 Linear Heat Rate Limits
LCO 3.2.2 Radial Peaking Factor Limits
LCO 3.2.4 ASI Limits

~~[The individual specifications that address core operating limits must be referenced here.]~~ (1)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in THE LATEST REVISION OF the following documents: (5)

<INSERT>

~~[Identify the Topical Report(s) by number, title, date, and NRC staff approval document, or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]~~

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

~~5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~

~~a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic~~ (20)

(continued)