ENCLOSURE 1

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

TECHNICAL SPECIFICATION CHANGE REQUEST

PART 1 - SECTION 1.0

March 28, 1996

CONSUMERS POWER COMPANY
Docket 50-255
Request for Change to the Technical Specifications
License DPR-20

1.0 USE AND APPLICATION CHANGE REQUEST

It is requested that Section 1.0, Definitions, of the Technical Specifications contained in the Facility Operating License DPR-20, Docket 50-255, issued to Consumers Power Company on February 21, 1991, for the Palisades Plant be changed as described below:

I. ARRANGEMENT AND CONTENT OF THIS PART OF THE CHANGE REQUEST:

This part of the Technical Specification Change Request (TSCR) proposes changes to those Palisades Technical Specifications addressing Section 1.0, Definitions. These changes are intended to result in requirements which are appropriate for the Palisades plant, but closely emulate those of the Standard Technical Specifications, Combustion Engineering Plants, NUREG 1432, Revision 1.

This discussion and its supporting information frequently refer to three sets of Technical Specifications; the following abbreviations are used for clarity and brevity:

TS - The existing Palisades Technical Specifications,
RTS - The revised Palisades Technical Specifications,
STS - NUREG 1432, Revision 1.

Five attachments are provided to assist the reviewer. Section 1.0 of the TS does not have a corresponding Bases section. Attachments 2 and 5, supplied for other parts of the TSCR, address the Bases for the associated parts of TS. The numbering and content of the remaining attachments is consistent with other parts of the TSCR.

1. Proposed RTS pages
2. Not Applicable to Section 1.0.
3. A line by line comparison of the TS and RTS
4. STS pages marked to show the differences between RTS and STS
5. Not applicable to Section 1.0.
6. A line by line comparison of RTS and STS.

Attachment 3, the line by line comparison of TS and RTS, is presented in a tabular format. The first page contains an explanation of the syntax and abbreviations used. The table is arranged numerically by TS item number. Each requirement in Sections 1 through 4 of TS is listed individually. In some cases, where a single numbered TS requirement contains more than one requirement, each requirement is listed individually under the same number.

In each section of the proposed RTS, new requirements taken from STS have been proposed. Since there is no equivalent requirement in TS, these changes do not appear in Attachment 3. These changes are considered MORE RESTRICTIVE because they add requirements and operating restrictions which do not exist in the current Palisades TS. The new requirements do appear in Attachment 6 where they are identified by an entry of "New" in the third column.

Attachment 3 Provides the Following Information for Each TS Requirement:

Identifying number of TS item,
Identifying number of closest equivalent RTS item,
Identification of TS item as LCO, Action, SR, etc.,
A short paraphrase of requirement,
A description of each proposed change from TS to RTS.

Classification of change as one of the following categories:

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

RELOCATED - A change which only moves requirements, not meeting the 10 CFR 50.36(c)(2)(ii) criteria, from the TS to the FSAR, to the Operating Requirements Manual, or to other documents controlled under 10 CFR 50.59.

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

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

Attachment 6, the line by line comparison of RTS and STS, is also presented in a tabular format. The first page contains an explanation of the syntax and abbreviations used; the second page contains a list of Palisades terminology used in place of the generic STS terminology. The table is arranged numerically by RTS item number. Each requirement in Sections 1 through 3 of RTS or STS is listed individually. Requirements which appear in TS, but not in RTS or STS, do not appear in the Attachment 6 listing.

Attachment 6 Provides the Following Information for Each RTS Requirement:

Identifying number of RTS requirement,
Identifying number of equivalent STS requirement,
Identification of each requirement as LCO, Action, SR, etc.,
Short paraphrase of each requirement,
A description of each difference between STS and RTS.

II. TECHNICAL SPECIFICATION CHANGES PROPOSED:

TS Section 1.0 addresses only definitions of terms used within the TS. RTS and STS Sections 1.0 address definitions, but also include discussions of the proper interpretation of LCO Action Table structure and Completion Times, and of Surveillance Requirement Frequencies. Each proposed change from TS to RTS is discussed in the attachments to this part of the TSCR.

Each proposed change to a requirement in TS is described in Attachment 3.

Those proposed RTS requirements which have no counterpart in TS are described in Attachment 6. These new requirements are identified by the word "New" in the third column of Attachment 6.

The Major Changes From TS to RTS Proposed in This Part of the TSCR are:

1. The definitions of plant operating conditions have been replaced with the operation Mode definitions used in STS. In several instances the name for a TS defined operation condition is the same as that for an STS Mode, but the definition differs. The most significant is the change in definitions for Hot Standby and Hot Shutdown.

   HOT STANDBY:
   TS: $T_{ave} \geq 525°F$, any rods withdrawn, and power < 2%;
   RTS: $T_{ave} \geq 300°F$ and $K_{eff} < 0.99$.

   HOT SHUTDOWN:
   TS: $T_{ave} \geq 525°F$, subcritical, and SDM meeting LCO 3.10.
   RTS: $T_{ave} < 300°F$ and > 200°F, and $K_{eff} < 0.99$.

2. STS Sections 1.2, Logical Connectors; 1.3, Completion Times; and 1.4 Frequency have been included in RTS.

The Major Differences Between RTS and STS in This Part of the TSCR are:

1. Several STS definitions which are inappropriate for Palisades have been replaced by plant specific definitions. Palisades uses Siemens Power Corporation as a fuel vendor; therefore, several STS definitions associated with core physics are inappropriate.

2. The definitions for Response Times have been omitted. Palisades TS do not require response time testing. A review during the Systematic Evaluation Program concluded that addition of response time testing requirements was not necessary.

III. NO SIGNIFICANT HAZARDS ANALYSIS:

Each change proposed for Section 1.0 is classified as ADMINISTRATIVE. ADMINISTRATIVE changes move requirements within the TS or clarify existing TS requirements, without affecting their technical content. Since ADMINISTRATIVE changes do not alter the technical content of any requirements, they cannot involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any previously evaluated, or involve a significant reduction in a margin of safety.

IV. CONCLUSION

The Palisades Plant Review Committee has reviewed this part of the STS conversion Technical Specifications Change Request and has determined that proposing this change does not involve an unreviewed safety question. Further, the change involves no significant hazards consideration. This change has been reviewed by the Nuclear Performance Assessment Department.

ATTACHMENT 1

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

1.0 USE AND APPLICATION PART

Proposed Revised Technical Specifications pages

1.0 USE AND APPLICATION

1.1 Definitions

----------NOTE----------
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
ASSEMBLY RADIAL PEAKING FACTOR (F_r^A)	F_r^A shall be the maximum ratio of the individual assembly power to the core average assembly power integrated over the total core height, including tilt.
AVERAGE DISINTEGRATION ENERGY ($\bar{E}$)	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the primary coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.
AXIAL OFFSET (AO)	AO shall be the ratio of the power generated in the lower half of the core minus the power generated in the upper half of the core, to the sum of those powers (determined using the incore detectors.)
AXIAL SHAPE INDEX (ASI)	ASI shall be the ratio of the power generated in the lower half of the core minus the power generated in the upper half of the core, to the sum of those powers (determined using the excore detectors.)

CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor, alarm, interlock, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated. Neutron detectors may be excluded from CHANNEL CALIBRATIONS.
CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be: a. Analog and bistable channels - the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarms, interlocks, displays, and trip functions; b. Digital computer channels - the use of diagnostic programs to test digital computer hardware and the injection of simulated process data into the channel to verify OPERABILITY, including alarm and trip functions. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.
CORE ALTERATION	CORE ALTERATION shall be the movement or manipulation of any fuel, sources, or control rods, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)	The COLR is the plant specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, 1977, or ICRP 30, Supplement to Part 1, pages 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity".

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except Primary Coolant Pump seal leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known not to interfere with the operation of leakage detection systems and not to be pressure boundary LEAKAGE; or

3. Primary Coolant System (PCS) LEAKAGE through a Steam Generator (SG) to the Secondary System.

LEAKAGE (continued)	b. <u>Unidentified LEAKAGE</u> All LEAKAGE (except Primary Coolant Pump seal leakoff), that is not identified LEAKAGE; c. <u>Pressure Boundary LEAKAGE</u> LEAKAGE (except SG LEAKAGE) through a nonisolable fault in a PCS component body, pipe wall, or vessel wall.
MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average primary coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE - OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Authorized under the provisions of 10 CFR 50.59; or b. Otherwise approved by the Nuclear Regulatory Commission.

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the plant specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the power operated relief valve lift settings and enable temperature associated with Low Temperature Overpressure Protection, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these limits is addressed in individual specifications.
QUADRANT POWER TILT (T_q)	T_q shall be the maximum positive ratio of the power generated in any quadrant minus the average quadrant power, to the average quadrant power.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the primary coolant of 2530 MWt.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full length control rods (shutdown and regulating) are fully inserted except for the single rod of highest reactivity worth, which is assumed to be fully withdrawn. However, with all rods verified fully inserted by two independent means, it is not necessary to account for a stuck rod in the SDM calculation. With any rods not capable of being fully inserted, the reactivity worth of these rods must be accounted for in the determination of SDM; b. In MODES 1 and 2, the fuel and moderator temperatures are changed to 532°F; and c. There is no change in part length rod position.

STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the primary coolant.
TOTAL RADIAL PEAKING FACTOR (F_r^T)	F_r^T shall be the maximum product of the ratio of the individual assembly power to the core average assembly power times the local peaking factor for that assembly integrated over the total core height, including tilt. The local peaking factor is defined as the maximum ratio of an individual fuel rod power to the assembly average rod power.

Table 1.1-1
MODES

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

(a) Excluding decay heat.

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

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

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES The following examples illustrate the use of logical connectors.

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

EXAMPLES
(continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Align . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE

The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

BACKGROUND

Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the plant. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the plant is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the plant is not within the LCO Applicability.

If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.

Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will <u>not</u> result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

DESCRIPTION (continued)

However, when a subsequent train, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the first inoperability; and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate reentry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this reentry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ." Example 1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Conditions A and B in Example 1.3-3 may not be extended.

EXAMPLES The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

EXAMPLES
(continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

EXAMPLES
(continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO
C. One Function X train inoperable. AND One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. OR C.2 Restore Function Y train to OPERABLE status.	12 hours 12 hours

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

EXAMPLES
(continued)

EXAMPLE 1.3-3

The Completion Times of Conditions A and B are modified by a logical connector, with a separate 10 day Completion Time measured from the time it was discovered the LCO was not met. In this example, without the separate Completion Time, it would be possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. The separate Completion Time modified by the phrase "from discovery of failure to meet the LCO" is designed to prevent indefinite continued operation while not meeting the LCO. This Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock." In this instance, the Completion Time "time zero" is specified as commencing at the time the LCO was initially not met, instead of at the time the associated Condition was entered.

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 30 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (including the extension) expires while one or more valves are still inoperable, Condition B is entered.

EXAMPLES
(continued)

EXAMPLE 1.3-5

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 30 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

EXAMPLES
(continued)

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered.

The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

IMMEDIATE COMPLETION ACTION — When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE

The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be performed in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "Specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

EXAMPLES
(continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the plant is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the plant is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the plant is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition. Failure to do so would result in a violation of SR 3.0.4.

EXAMPLES
(continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

EXAMPLES
(continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------NOTE---------- Not required to be performed until 12 hours after ≥ 25% RTP. ----------	
Perform channel adjustment.	7 days

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." The interval continues, whether or not the plant operation is < 25% RTP between performances. Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.

Once the plant reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

ATTACHMENT 2

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

1.0 USE AND APPLICATION PART

Not Applicable

to

This Section

ATTACHMENT 3

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

1.0 USE AND APPLICATION PART

Comparison of Existing and Revised Technical Specifications

Palisades Tech Spec Requirement List. Corrected through Amendment 170

A list of the existing Palisades Tech Specs (TS) correlated to Palisades Revised Technical Specifications (RTS).

First Column; Existing Palisades Tech Spec (TS) number

Each numbered TS item is listed in the left-most column. Items which contain more than one requirement are listed once for each requirement.

Second Column; Palisades Revised Tech Spec (RTS) number

The nearest corresponding numbered RTS item is listed in the second column. If the item does not appear in RTS, it is noted as 'Deleted' or 'Relocated.'

Deleted is used where an item has been eliminated as a tech spec, ie deleting, iaw GL 84-15, the requirement to test a D.G. when an ECCS pump in the opposite train becomes inoperable.

Relocated is used where an item has been moved to a controlled program or document because it does not meet the "Criteria" of 10 CFR 50.36(2)(c)(ii).

Where an item is relocated or deleted, the number of the associated RTS section has been added to allow sorting the list by section number. Relocated items, such as heavy load restrictions, which are not associated with any particular RTS section are arbitrarily assigned the number 5.0.

Third Column; TS Item Description

An abbreviation of the TS requirement appears in the third column. Each item is identified as: LCO, ACTION, SR, Admin, Exception, etc. Some items are implied, rather than explicit, ie a LCO is implied when an ACTION exists without a stated LCO.

Description Key; TS requirement type:	Column 3 syntax:
Safety Limit	SL: Safety limit; Applicable conditions
Surveillance Requirement	SR: Equipment to be tested; Test description; Frequency
Limiting Safety Setting	LSS: RPS Trip Channel & required setting
Limiting Condition for Operation	LCO: Equipment to be operable; Applicable conditions
Action	ACTN: Condition requiring action; Required action; Completion time
Administrative Requirement	ADMN: Administrative requirement
Permitted Instrument Bypass	Byps: Bypassable component; conditions when bypass permitted
Defined Term	DEF: Name of defined item
Exception to other Requirement	XCPT: Excepted spec or condition; Applicable conditions
Descriptive material	DESC: Subject matter
Table	TBL: Table

Forth Column; Classification of Changes:

Each change is identified as ADMINISTRATIVE, RELOCATED, MORE RESTRICTIVE, or LESS RESTRICTIVE.

Fifth Column; Discussion of Changes:

Each change is discussed briefly.

TS Number	RTS Number	TS requirement description		Classification and Description of Changes	
1.0	1.1		Definitions		
New	1.1(1)	DEF:	Actions	ADMINISTRATIVE:	Added STS definition.
1.1(1)	1.1(2)	DEF:	Assembly Radial Peaking Factor (F_r^A)	ADMINISTRATIVE:	Definition was reworded slightly to agree more closely with the analytical documents which it supports. There is no change in intent or effect.
1.1(2)	1.1(3)	DEF:	$\bar{E}$ - Average Disintegration Energy	ADMINISTRATIVE:	Requirement unchanged.
1.1(3)	1.1(4)/1.1(5)	DEF:	Axial Offset or Axial Shape Index	ADMINISTRATIVE:	Redefined as separate definitions to agree with actual plant usage.
1.1(4)	1.1(6)	DEF:	Channel Calibration	ADMINISTRATIVE:	Requirement unchanged.
1.1(5)	1.1(7)	DEF:	Channel Check	ADMINISTRATIVE:	Changed to agree with STS.
1.1(6)	1.1(8)	DEF:	Channel Functional Test	ADMINISTRATIVE:	Changed to agree more closely with STS.
1.1(7)	1.1 (Tbl 1.1-1)	DEF:	Cold Shutdown; Shutdown Boron & <210°F	ADMINISTRATIVE:	Changed to "MODE 5"; <200°F, iaw STS.
1.1(8)	1.1 Deleted	DEF:	Containment Integrity	ADMINISTRATIVE:	Not a defined term in STS.
1.1(9)	1.1 Deleted	DEF:	Control Rods; Full-length rods	ADMINISTRATIVE:	Not a defined term in STS.
1.1(10)	1.1(10)	DEF:	COLR; Core Op Limits Report	ADMINISTRATIVE:	Requirement unchanged.
1.1(11)	1.1(11)	DEF:	Dose Equivalent I-131	ADMINISTRATIVE:	Requirement unchanged.
1.1(12)	1.1 (Tbl 1.1-1)	DEF:	Hot Shutdown; Sub crit, SDM >LCO 3.10, >525°F	ADMINISTRATIVE:	Changed to "MODE 3"; K_{eff} <0.99 & >300°F, iaw STS.
1.1(13)	1.1 (Tbl 1.1-1)	DEF:	Hot Standby; >525°F, any rods out, <2% RTP	ADMINISTRATIVE:	Changed to "MODE 2"; K_{eff} >0.99 & <5%, iaw STS.
New	1.1(13)	DEF:	MODE; see Table 1.1-1	ADMINISTRATIVE:	Added STS definition.
1.1(14)	1.1(15)	DEF:	Low Power Physics testing	ADMINISTRATIVE:	Changed to "PHYSICS TESTS", iaw STS.
1.1(15)	1.1(14)	DEF:	Operable; capable of its required function	ADMINISTRATIVE:	Changed to agree with STS.
1.1(16)	1.1 (Tbl 1.1-1)	DEF:	Power Operation; >2% RTP	ADMINISTRATIVE:	Changed to "MODE 1"; >5%, iaw STS.
1.1(17)	1.1(17)	DEF:	Quadrant Power Tilt (T_q)	ADMINISTRATIVE:	Requirement unchanged.
1.1(18)	1.1(18)	DEF:	Rated Power; 2530 Mwt	ADMINISTRATIVE:	Changed to "RATED THERMAL POWER"; used STS definition.
1.1(19)	1.1 Deleted	DEF:	Reactor Critical; >10^{-4}%	ADMINISTRATIVE:	Not a defined term in STS.

Comparison of existing Palisades Tech Specs and Proposed Palisades Tech Specs. (03/28/96)

TS Number	RTS Number	TS requirement description	Classification and Description of Changes	
1.1(20)	1.1 Deleted	DEF: Refueling Boron Concentration; K-eff <0.95	ADMINISTRATIVE:	Not a defined term in STS.
1.1(21)	1.1(9)	DEF: Refueling Operation; Moving core parts	ADMINISTRATIVE:	Changed to "CORE ALTERATION", iaw STS.
1.1(22)	1.1 (Tbl 1.1-1)	DEF: Refueling Shutdown; >Ref Boron & <210°F	ADMINISTRATIVE:	Changed to "MODE 6"; Rx vessel head not tensioned, iaw STS.
1.1(23)	1.1 Deleted	DEF: Shutdown Boron Concentration; K-eff <0.98	ADMINISTRATIVE:	Not a defined term in STS.
1.1(24)	1.1(19)	DEF: Shutdown Margin; Amount subcritical W/Rx trip	ADMINISTRATIVE:	Changed definition slightly to agree W/STS.
New	1.1(20)	DEF: Staggered Test Basis	ADMINISTRATIVE:	Added STS definition.
New	1.1(21)	DEF: Thermal Power	ADMINISTRATIVE:	Added STS definition.
1.1(25)	1.1(22)	DEF: Total Radial Peaking Factor	ADMINISTRATIVE:	Definition was reworded slightly to agree more closely with the analytical documents which it supports. There is no change in intent or effect.

ATTACHMENT 4

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

1.0 USE AND APPLICATION PART

STS Pages Marked to Show the Differences Between RTS and STS

1.0 USE AND APPLICATION

1.1 Definitions

NOTE

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
ASSEMBLY RADIAL PEAKING FACTOR (F_r^A)	F_r^A shall be the maximum ratio of the individual assembly power to the core average assembly power integrated over the total core height, including tilt.
AVERAGE DISINTEGRATION ENERGY ($\bar{E}$)	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the primary coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > [15] minutes, making up at least 95% of the total noniodine activity in the coolant.
~~AXIAL SHAPE INDEX (ASI)~~	~~ASI shall be the power generated in the lower half of the core less the power generated in the upper half of the core, divided by the sum of the power generated in the lower and upper halves of the core.~~ $$ASI = \frac{lower - upper}{lower + upper}$$
AXIAL OFFSET (AO)	AO shall be the ratio of the power generated in the lower half of the core minus the power generated in the upper half of the core, to the sum of those powers (determined using the incore detectors.)

(continued)

1.1 Definitions

AXIAL SHAPE INDEX (ASI)	ASI shall be the ratio of the power generated in the lower half of the core minus the power generated in the upper half of the core, to the sum of those powers (determined using the excore detectors.)
~~AZIMUTHAL POWER TILT (T_q) Digital~~	~~AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.~~
~~AZIMUTHAL POWER TILT (T_q) Analog~~	~~AZIMUTHAL POWER TILT shall be the maximum of the difference between the power generated in any core quadrant (upper or lower) (P_{quad}) and the average power of all quadrants (P_{avg}) in that half (upper or lower) of the core, divided by the average power of all quadrants in that half (upper or lower) of the core.~~ $$T_q = \text{Max} \quad \frac{P_{quad} - P_{avg}}{P_{avg}}$$
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with~~in~~ the necessary range and accuracy to known values of the parameter ~~that~~ which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel, including the ~~required~~ sensor, alarm, ~~display~~ interlock, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. ~~Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.~~ The CHANNEL CALIBRATION may be performed by ~~means of~~ any series of sequential, overlapping, or total channel steps ~~so~~ such that the entire channel is calibrated. Neutron detectors may be excluded from CHANNEL CALIBRATIONS.

(continued)

1.1 Definitions

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be: a. Analog and bistable channels—the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarms, interlocks, displays, and trip functions; b. Digital computer channels—the use of diagnostic programs to test digital computer hardware and the injection of simulated process data into the channel to verify OPERABILITY, including alarm and trip functions. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.
CORE ALTERATION	CORE ALTERATION shall be the movement or manipulation of any fuel, sources, or ~~reactivity control components~~ control rods ~~[excluding control element assemblies (CEAs) withdrawn into the upper guide structure]~~, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the ~~unit~~ plant specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.

(continued)

1.1 Definitions (continued)

DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ~~[~~Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, ~~NRC,~~ 1977, or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity~~]~~.
~~$\bar{E}$ AVERAGE DISINTEGRATION ENERGY~~	~~$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half-lives > [15] minutes, making up at least 95% of the total noniodine activity in the coolant.~~
~~ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME~~	~~The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.~~
~~L_a~~	~~The maximum allowable containment leakage rate, L_a, shall be [0.25]% of containment air weight per day at the calculated peak containment pressure (P_a).~~

(continued)

1.1 Definitions (continued)

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except ~~reactor~~ Primary coolant pump ~~(RCP)~~ seal water ~~injection or~~ leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known ~~either~~ not to interfere with the operation of leakage detection systems ~~or~~ and not to be pressure boundary LEAKAGE; or

3. ~~Reactor~~ Primary Coolant System ~~(RCS)~~ (PCS) LEAKAGE through a steam generator (SG) to the Secondary System.

b. Unidentified LEAKAGE

All LEAKAGE (except primary coolant pump seal leakoff), that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an ~~RCS~~ PCS component body, pipe wall, or vessel wall.

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average ~~reactor~~ primary coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

(continued)

1.1 Definitions (continued)

OPERABLE—OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: ~~a. Described in Chapter [14, Initial Test Program] of the FSAR;~~ ~~b~~a. Authorized under the provisions of 10 CFR 50.59; or ~~c~~b. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the ~~unit~~ plant specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the power operated relief valve lift settings and enable temperature associated with Low Temperature Overpressure Protection, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these ~~operating~~ limits is addressed ~~in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."~~ individual specifications.
QUADRANT POWER TILT (T_q)	T_q shall be the maximum positive ratio of the power generated in any quadrant minus the average quadrant power, to the average quadrant power.

(continued)

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the ~~reactor~~ primary coolant of ~~[3410]~~ 2530 MWt.
~~REACTOR PROTECTIVE SYSTEM (RPS) RESPONSE TIME~~	~~The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power to the CEAs drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.~~
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full length ~~CEAs~~ control rods (shutdown and regulating) are fully inserted except for the single ~~CEA~~ rod of highest reactivity worth, which is assumed to be fully withdrawn. However, with all ~~CEAs~~ rods verified fully inserted by two independent means, it is not necessary to account for a stuck ~~CEA~~ rod in the SDM calculation. With any ~~CEAs~~ rods not capable of being fully inserted, the reactivity worth of these ~~CEAs~~ rods must be accounted for in the determination of SDM; b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the [nominal zero power design level][; and] c. There is no change in part length ~~CEA~~ rod position.

(continued)

1.1 Definitions (continued)

STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the ~~reactor~~ primary coolant.
TOTAL RADIAL PEAKING FACTOR (F_r^T)	F_r^T shall be the maximum product of the ratio of the individual assembly power to the core average assembly power times the local peaking factor for that assembly integrated over the total core height, including tilt. The local peaking factor is defined as the maximum ratio of an individual fuel rod power to the assembly average rod power.

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE ~~REACTOR~~ PRIMARY COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ ~~[350]~~ 300
4	Hot Shutdown[b]	< 0.99	NA	~~[350] > T_{avg} > [200]~~ 300 > T_{ave} > 200
5	Cold Shutdown[b]	< 0.99	NA	≤ ~~[200]~~
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.

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

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

1.2 Logical Connectors

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES The following examples illustrate the use of logical connectors.

1.2 Logical Connectors

EXAMPLES EXAMPLE 1.2-1
(continued)
ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . .	
	OR	
	A.2.1 Verify . . .	
	AND	
	A.2.2.1 Reduce . . .	
	OR	
	A.2.2.2 Perform . . .	
	OR	
	A.3 Align . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.3 Completion Times

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE

The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

BACKGROUND

Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the ~~unit~~ plant. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the ~~unit~~ plant is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the ~~unit~~ plant is not within the LCO Applicability.

If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.

Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will <u>not</u> result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

1.3 Completion Times

DESCRIPTION (continued)

However, when a subsequent train, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the first inoperability; and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ." Example 1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Conditions A and B in Example 1.3-3 may not be extended.

(continued)

1.3 Completion Times

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-2 (continued)

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO
C. One Function X train inoperable. AND One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. OR C.2 Restore Function Y train to OPERABLE status.	~~72~~12 hours ~~72~~12 hours

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 (continued)

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

The Completion Times of Conditions A and B are modified by a logical connector, with a separate 10 day Completion Time measured from the time it was discovered the LCO was not met. In this example, without the separate Completion Time, it would be possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. The separate Completion Time modified by the phrase "from discovery of failure to meet the LCO" is designed to prevent indefinite continued operation while not meeting the LCO. This Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock." In this instance, the Completion Time "time zero" is specified as commencing at the time the LCO was initially not met, instead of at the time the associated Condition was entered.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours ~~12~~30 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (including the extension) expires while one or more valves are still inoperable, Condition B is entered.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-5

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours ~~12~~30 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-5 (continued)

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

~~EXAMPLE 1.3-6~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One channel inoperable.~~	~~A.1 Perform SR 3.x.x.x.~~	~~Once per 8 hours~~
	~~OR~~	
	~~A.2 Reduce THERMAL POWER to ≤ 50% RTP.~~	~~8 hours~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~	~~6 hours~~

(continued)

1.3 Completion Times

~~EXAMPLES~~ ~~EXAMPLE 1.3-6 (continued)~~

~~Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.~~

~~If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.~~

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-~~7~~6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered.

(continued)

1.3 Completion Times

EXAMPLES	EXAMPLE 1.3-~~7~~6 (continued) The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.
IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

(continued)

1.4 Frequency

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE

The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be ~~met~~ performed in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the ~~unit~~ plant is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the ~~unit~~ plant is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the ~~unit~~ plant is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition. Failure to do so would result in a violation of SR 3.0.4.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------NOTE---------- Not required to be performed until 12 hours after ≥ 25% RTP. ----------	
Perform channel adjustment.	7 days

~~The interval continues, whether or not the unit operation is < 25% RTP between performances.~~

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." The interval continues, whether or not the plant operation is < 25% RTP between performances. Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.

Once the ~~unit~~ plant reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

ATTACHMENT 5

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

1.0 USE AND APPLICATION PART

Not Applicable

to

This Section

ATTACHMENT 6

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

1.0 USE AND APPLICATION PART

Comparison of Revised and Standard Technical Specifications

Palisades Revised Tech Spec Requirement List. (03/28/96)

A listing of the proposed Palisades Revised Tech Specs (RTS) correlated to the CE Standard Tech Specs (STS).

First Column; Proposed Palisades Revised Tech Spec (RTS) number
Each RTS item is listed in the left-most column.
If a STS item has been omitted from RTS, the word 'Omitted' is used.

Second Column; CE Standard Tech Spec (STS) number
The corresponding STS item is listed in the second column.
If a RTS item does not appear in STS, it is noted as 'Added'.

Third Column; Existing Palisades Tech Spec (TS) number
The closest TS item is listed in the third column.
If a RTS item does not appear in TS, it is noted as 'New'.

Fourth Column; RTS Item Description
An abbreviation of the RTS item appears in the third column.
Each item is identified as: LCO, ACTION, SR, ADMIN, Exception, etc.
In cases where a STS item was omitted from RTS, the description is of the STS item.

Description Key: RTS requirement type:	Column 4 syntax:
Safety Limit	SL: Safety limit; Applicable conditions
Limiting Condition for Operation	LCO: LCO Description; Applicable conditions
Condition	COND: Description of non-conforming condition
Action	ACTN: Required action; Completion time
Surveillance Requirement	SR: Test description; Frequency
Table	TABL: Title
Administrative Requirement	ADMN: Administrative requirement
Defined Term	DEF: Name of defined term

Fifth Column; Comments and Explanations of Differences between RTS and STS.
A brief explanation of differences between RTS and STS is provided in the fifth column.

Other abbreviations used in the listing are:

NA:	Not Applicable
CFT:	Channel Functional Test
CHNL:	Channel

RTS Number	STS Number	TS Number	RTS (STS) requirement Description	Explanation of Differences

Global differences between the proposed Palisades Technical Specifications and the Standard Technical Specifications for CE plants, Nureg 1432:

The following changes are not discussed in the explanation of differences for each TS requirement.

1) Bracketed values have been replaced with appropriate values for Palisades. Typically, the basis for these values is provided in the bases document.

2) Each required action of the form "Perform SR X.X.X.X . . ." has been altered by a parenthetical summary of the SR requirements. This change allows a reader to understand the required actions without constantly turning pages to locate the referenced SR.

3) Terminology has been changed to reflect Palisades usage:

"RWT"	becomes	"SIRWT"	Safety Injection Refueling Water Tank
"CEA"	becomes	"Control Rod" or "Rod"	Palisades uses cruciform control rods rather than the multifingered "Control Element Assemblies" of later CE plants.
"RCS"	becomes	"PCS"	Palisades terminology is "Primary Coolant System" rather than "Reactor Coolant System"
"SIAS"	becomes	"SIS"	Palisades terminology is "Safety Injection Signal" rather than "Safety Injection Actuation Signal"
"AC Vital bus"	becomes	"Preferred AC bus"	Palisades terminology.
"PAMI"	becomes	"AMI"	Accident Monitoring Instrumentation, Palisades terminology
"ESFAS"	becomes	"ESF Instrumentation"	There is no stand-alone ESFAS system or cabinet at Palisades; ESF instruments actuate the ESF functions
"DG LOVS"	becomes	"DG UV Start"	Palisades Terminology
"Remote Shutdown System"	becomes	"Alternate Shutdown System"	Palisades Terminology
"Power Rate of Change-High"	becomes	"High Startup Rate"	Palisades Terminology

RTS Number	STS Number	TS Number	RTS (STS) requirement Description	Explanation of Differences
1.1	1.1	1.1	DEFINITION SECTION	Note: Several definitions relating to physics parameters differ from the standard since Palisades is not only an older CE design than that modeled in the standard TS, but uses a different fuel vendor.
1.1 (1)	1.1 (1)	New	DEF: ACTIONS	Unchanged.
1.1 (2)	Added	1.1.(14)	DEF: F_r^A	Palisades specific definition
1.1 (3)	Added	1.1.(16)	DEF: AO	Palisades specific definition
1.1 (4)	1.1 (2)	1.1.(16)	DEF: ASI	Palisades specific definition
1.1 (5)	1.1 (11)	1.4.(6)	DEF: $\bar{E}$	Unchanged. Arranged alphabetically by name, rather than by abbreviation in order to be consistent with other entries in this section.
Omitted	1.1 (3)	NA	DEF: (AZIMUTHAL POWER TILT (Digital))	NA Palisades.
Omitted	1.1 (4)	NA	DEF: (AZIMUTHAL POWER TILT (Analog))	NA Palisades.
1.1 (6)	1.1 (5)	1.3.(3)	DEF: CHANNEL CALIBRATION	Retained existing Channel Calibration definition. STS definition contains new requirements in sentences discussing RTDs and thermocouples, and inplace cross calibrations. The existing TS and proposed RTS contain explicit requirements for calibration of thermocouples. The remainder of the Palisades definition is very similar to the STS definition.
1.1 (7)	1.1 (6)	1.3.(1)	DEF: CHANNEL CHECK	Unchanged.
1.1 (8)	1.1 (7)	1.3.(2)	DEF: CHANNEL FUNCTIONAL TEST	Unchanged.
1.1 (9)	1.1 (8)	1.1(8)	DEF: CORE ALTERATION	Unchanged from Rev 0 of NUREG. Change BWR-05-C3 is inappropriate for Palisades. Removal of the Upper Guide Structure, at Palisades, must be considered a CORE ALTERATION due to the possibility of a fuel assembly remaining attached. The original wording, with its broader implications, is more appropriate.
1.1 (10)	1.1 (9)	NA	DEF: COLR	Unchanged other than using the term "plant" instead of "unit".
1.1 (11)	1.1 (10)	1.4(5)	DEF: DOSE EQUIVALENT I-131	Unchanged.

RTS Number	STS Number	TS Number	RTS (STS) requirement Description	Explanation of Differences
Omitted	1.1 (11)	NA	DEF: (ESF RESPONSE TIME)	Palisades does not use this term. An SER issued during SEP found it unnecessary to perform response time testing at Palisades. This is discussed further in the discussion of deleting the response time SRs.
Omitted	1.1 (12)	NA	DEF: (L_a)	The definition of L_a was moved to the Containment Leak Testing Program in accordance with the NRC/NEI model Tech Specs implementing Appendix J, Option B testing.
1.1 (12)	1.1 (13)	New	DEF: LEAKAGE	Changed to use Palisades specific terminology and to make the second half of paragraph a.2 an "and" rather than an "or".
1.1 (13)	1.1 (14)	New	DEF: MODE	Unchanged except for the use of Palisades specific terminology.
1.1 (14)	1.1 (15)	1.4(1)	DEF: OPERABLE	Unchanged.
1.1 (15)	1.1 (16)	1.1(10)	DEF: PHYSICS TESTS	Item a. deleted. It is not applicable to Palisades.
1.1 (16)	1.1 (17)	New	DEF: PTLR	PTLR definition changed to reflect the wording of Tech Spec Task Force change TSTF-4 (WOG-1.3).
1.1 (17)	Added	1.1(13)	DEF: T_q	Palisades specific definition.
1.1 (18)	1.1 (18)	1.1(1)	DEF: RTP	Unchanged, except for use of Palisades specific terminology.
Omitted	1.1 (19)	NA	DEF: (RPS RESPONSE TIME)	Palisades does not use this term. An SER issued during SEP found it un-necessary to perform response time testing at Palisades. This is discussed further in the discussion of deleting the response time SRs.
1.1 (19)	1.1 (20)	1.1(9)	DEF: SDM	Unchanged, except for use of Palisades specific terminology.
1.1 (20)	1.1 (21)	New	DEF: STAGGERED TEST BASIS	Unchanged.
1.1 (21)	1.1 (22)	New	DEF: THERMAL POWER	Unchanged.
1.1 (22)	Added	1.1(15)	DEF: F_r^T	Palisades specific definition.
Omitted	1.1 (23)	NA	DEF: (UNRODDED PEAKING FACTOR (Analog))	NA Palisades
Omitted	1.1 (24)	NA	DEF: (UNRODDED PEAKING FACTOR (Digital))	NA Palisades
1.1-1	1.1-1	New	Tabl: MODE Definitions	Unchanged.
1.1-1(1)	1.1-1(1)	1.1(3)	DEF: MODE 1 (Power Operation)	Unchanged

Palisades RTS Cross Reference to STS. (03/28/96).

RTS Number	STS Number	TS Number	RTS (STS) requirement Description	Explanation of Differences
1.1-1(2)	1.1-1(2)	1.1(4)	DEF: MODE 2 (Startup)	Unchanged.
1.1-1(3)	1.1-1(3)	1.1(5)	DEF: MODE 3 (Hot Standby)	Unchanged.
1.1-1(4)	1.1-1(4)	New	DEF: MODE 4 (Hot Shutdown)	Unchanged.
1.1-1(5)	1.1-1(5)	1.1(7)	DEF: MODE 5 (Cold Shutdown)	Unchanged.
1.1-1(6)	1.1-1(6)	1.1(8)	DEF: MODE 6 (Refueling)	Unchanged.
1.2	1.2	New	ADMN: Logical Connector discussion	Unchanged.
1.3	1.3	New	ADMN: Completion Time discussion	The completion times of Example 1.3-3 were changed to reflect items actually contained in proposed RTS (LCO 3.8.1). The equivalent the of example in STS does not appear in the Palisades proposed RTS. Completion times in examples 1.3-4 and 1.3-5 have been changed to reflect the Palisades 30 hours for reaching MODE 4. Example 1.3-6 was deleted and 1.3-7 renumbered. Example 1.3-6 represents a Completion Time form not used in the proposed RTS.
1.4	1.4	New	ADMN: Frequency Discussion	The word "met," in the first paragraph of the description section, has been changed to "performed" to better fit the context. The text following example 1.4-3 has been re-arranged and an introductory sentence added; the published text appears to have been disarranged.

ATTACHMENT 5

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

3.2 POWER DISTRIBUTION LIMITS

STS Bases Pages Marked to Show the Differences Between RTS and STS

B 3.2 POWER DISTRIBUTION LIMITS

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

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the ~~reactor~~ primary coolant in the event of a Loss Of Coolant Accident (LOCA), loss of flow accident, ejected ~~control element assembly (CEA)~~ control rod accident, or other postulated accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using ~~CEAs~~ control rods to alter the axial power distribution;

b. Decreasing ~~CEA~~ control rod insertion by boration, thereby improving the radial power distribution; and

c. Correcting off optimum conditions (e.g., a ~~CEA~~ control rod drop or misoperation of the ~~unit~~ plant) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., ~~CEA~~ control rod insertion and alignment limits), the power distribution satisfies this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of ~~a~~Anticipated ~~o~~Operational ~~o~~Occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on ~~l~~Linear ~~h~~Heat ~~r~~Rate (LHR) and ~~d~~Departure from ~~n~~Nucleate ~~b~~Boiling Ratio (~~DNB~~DNBR).

The limits on LHR, ~~Total Planar Radial Peaking Factor (F_{xy}^T), Total Integrated Radial Peaking Factor (F_r^T)~~ Assembly Radial Peaking F_r^T, Total Radial Peaking F_r^A, T_q, and ASI represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis.

Either of the two core power distribution monitoring systems, the Excore ~~Detector~~ Monitoring System or the Incore ~~Detector~~ Monitoring System, provides adequate monitoring of the core power distribution and is capable of verifying that the LHR is within its limits. The Excore ~~Detector~~ Monitoring System performs this function by continuously monitoring ASI with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the ASI is maintained within the allowable limits specified in the COLR.

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

a. The ~~CEA~~ control rod insertion limits of LCO 3.1.6, "Shutdown ~~CEA~~ and Part Length Rod Insertion Limits," and LCO 3.1.7, "Regulating ~~CEA~~ Rod Insertion Limits," are satisfied;

b. The T_q restrictions of LCO 3.2.4~~2~~3 are satisfied; and

c. ~~F_{xy}^T is within the limits of LCO 3.2.2.~~ Radial Peaking factors F_r^A and F_r^T are within limits of LCO 3.2.2

The Incore ~~Detector~~ Monitoring System continuously provides a direct measure of the peaking factors and 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 m~~Measurement ~~calculational~~ uncertainty ~~factor of 1.062~~ (See COLR Table 2.3);

b. ~~An e~~Engineering uncertainty ~~factor of 1.03~~ (3%);

c. ~~An allowance of 1.002 for axial fuel densification and thermal expansion; and~~

~~d. A~~ THERMAL POWER ~~measurement~~ uncertainty ~~factor of 1.02~~ (2%).

BASES

The measurement uncertainties associated with LHR, F_r^A and F_r^T are based on a statistical analysis performed on Palisades Incore Detector Algorithm (PIDAL) power distribution benchmarking results. These values are included in the approved NRC submittal for Palisades use of PIDAL methodology. Table 2.3 of the COLR illustrates 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.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) and AOOs (Condition 2) (Ref. 3, GDC 10). The power distribution and ~~CEA~~ control rod insertion and alignment LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 4);

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 3, GDC 10).

c. During an ejected rod accident, the fission energy input to the fuel must not exceed ~~280~~200 cal/gm (Ref. ~~[~~5~~<~~]); and

d. The 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, GDC 26).

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 ~~reactor~~ 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.

BASES

The LCOs governing LHR, ASI, and the ~~Reactor~~ Primary Coolant System ensure that these criteria are met as long as the core is operated within the ASI, ~~F^T_{xy} F^A_r, and~~ F^T_r ~~, and T_q~~ limits specified in the COLR. 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 occur while the ~~unit~~ 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 LHR satisfies Criterion 2 of the NRC Policy Statement.~~

LCO

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

APPLICABILITY

In MODE 1, with THERMAL POWER ≥ 50% RTP, the 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 other MODES~~ Below 50% RTP and all other MODES, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on the core power distribution.

ACTIONS

A.1

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.

B.1.1 ~~and~~ B 1.3

~~If the LHR cannot be returned to within its specified limits, THERMAL POWER must be reduced. The change to MODE 2 ensures that the core is operating within its thermal limits and places the core in a conservative condition. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power MODE 1 conditions in an orderly manner and without challenging plant systems.~~

If the Incore Monitoring System is inoperable, verification must be performed to ensure that the excore monitoring system is operable for LHR monitoring. The THERMAL POWER must also be restricted to the excore Allowable Power Level (APL), within 2 hours. This ensures the analysis inputs to the safety analysis are not violated.

Verifying the following parameters:

$T_q \leq 3\%$, THERMAL POWER $\leq$ APL and ASI within 5% of target AO further ensures that the excore monitoring system will give accurate results for LHR monitoring. Operating within these limits bounds inputs used in the safety analysis and ensures that the LHR is within the limits used in the safety analysis. A four hour time interval is adequate to ensure the LHR is always within limits.

B 2.1 and 2.2

If the incore alarm system is inoperable and the excore monitoring system is not being used to monitor LHR, operation at less than or equal to 85% RTP may continue, provided LHR is verified through manual incore readings. Operation at $\leq$ 85% RTP ensures that ample thermal margin is maintained. These readings shall be manually collected at the terminal blocks in the control room utilizing a suitable signal detector. 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) within 4 hours and at least every 2 hours thereafter. The time interval of 2 hours and the minimum of 10 detectors per quadrant are sufficient to maintain adequate surveillance of the core power distribution to detect significant changes until the monitoring systems are returned to service.

BASES

C.1

If the required action and associated completion time is not met the plant must be < 50% RTP within 2 hours. At this power level ample THERMAL MARGIN exists to ensure that the fuel integrity is not jeopardized and safety analysis assumptions used are still valid. A 2 hour time period is ample to place the plant in conservative state < 50% RTP.

SURVEILLANCE REQUIREMENTS

A Note was added to the SRs to require LHR to be determined by either the Excore ~~Detector~~ Monitoring System or the Incore ~~Detector~~ Monitoring System.

~~SR 3.2.1.1~~

~~Performance of this SR verifies that the Excore Detector Monitoring System can accurately monitor the LHR. Therefore, this SR is only applicable when the Excore Detector Monitoring System is being used to determine the LHR. The 31 day Frequency is appropriate for this SR because it is consistent with the requirements of SR 3.3.1.3 for calibration of the excore detectors using the incore detectors.~~

~~The SR is modified by a Note that states that the SR is only applicable when the Excore Detection Monitoring System is being used to determine LHR. The reason for the Note is that the excore detectors input neutron flux information into the ASI calculation.~~

~~SR 3.2.1.2 and SR 3.2.1.3~~

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

~~Performance of these SRs verifies that the Incore Detector Monitoring System can accurately monitor LHR. Therefore, they are only applicable when the Incore Detector Monitoring System is being used to determine the LHR.~~

BASES

~~A 31 day Frequency is consistent with the historical testing frequency of the reactor monitoring system. The SRs are modified by two Notes. Note 1 allows the SRs to be performed only when the Incore Detector Monitoring System is being used to determine LHR. Note 2 states that the SRs are not required to be performed when THERMAL POWER is < 20% RTP. The accuracy of the neutron flux information from the incore detectors is not reliable at THERMAL POWER < 20% RTP.~~

SR 3.2.1.1

This SR is only applicable while the excore system is being used for LHR monitoring. The verification of measured ASI being within 0.05 of the target ASI value for at least 3 of 4, 2 of 3, and 2 of 2 operable channels ensures that ASI is being monitored adequately. ASI remaining within stated limits ensures that radial peaking factors and LHR limits are not exceeded. Operating within these limits allows the power distribution to be monitored adequately and with a high level of confidence. ASI monitored by the stated operable number of channels ensures that the minimum number of operable channels are being used to determine accurate ASI readings. Two channels is the minimum number of operable channels permitted to achieve accurate ASI values. The frequency of 15 minutes is adequate to safely monitor ASI while using the excore system. This time period is short enough to catch any significant changes in ASI. Changes in operating conditions that could cause a flux redistribution, i.e., boration/dilution, rod motion, and Xenon redistribution are either delayed response or flagged by other alarming functions within the control room.

SR 3.2.1.2

This SR ensures that the incore monitoring system is available for LHR monitoring. The incore system compares incoming incore data to an alarm data-base to determine if LHR is within alarm setpoints. This data is also fed to the PIDAL program that calculates T_q, AO, F_r^A, F_r^T, and LHR. Both of these programs run within the plant process computer environment. In the event that the incore system has failed a message would be flagged on the control room PC terminal that says, "Host Communication lost to Reach-Mon." It is more than adequate for an operator to verify that this message does not exist on the PC terminal every 12 hours.

SR 3.2.1.3

During normal plant operations, the incore alarm system supplies continuous monitoring of LHR. As data flows in from the incore detectors LHR is calculated and compared to the alarm setpoints continuously and automatically. If the excore detectors or the incore manual readings are used to monitor LHR then the time period allotted to perform LHR verification is specified by each particular condition. Monitoring LHR ensures that the assumptions made in the Safety Analysis are maintained.

SR 3.2.1.4

The Incore Monitoring System provides a detailed picture of core operating parameters. The Palisades Incore Detector Algorithm (PIDAL) provides LHR, ASI, T_q, F_r^A and F_r^T information for the entire core. The PIDAL program is used to generate alarm setpoints that are correct for fuel depletion. The frequency of 31 days is consistent with the historical testing frequency of the reactor monitoring system. The intent of updating Incore alarm setpoints is to compensate for fuel depletion effects on the core power distribution. This time period is adequate to capture the long term effects of fuel depletion. After each refueling, the incore alarm setpoints should be updated prior to reaching 50% RTP. These values would be adjusted once again once the excore monitoring system is calibrated with the incore alarm system at RTP conditions.

SR 3.2.1.5

This surveillance requirements ensures that the excore monitoring system is operable for use in LHR monitoring. This surveillance requirement can be satisfied by verifying values obtained from a current SR 3.2.1.1 performance. This Surveillance requirement is only applicable when the excore monitoring system is used for LHR Monitoring. The 31 day frequency ensures that these parameters are consistent for use of the excore monitoring system for LHR monitoring.

SR 3.2.1.6

This surveillance ensures that the incore alarm channels are functioning as designed. The channel calibration ensures that the setpoint value will initiate an incore alarm. The 18 month frequency is consistent with refueling outage scheduling. This is an adequate time period to perform SR 3.1.1.3 with a high level of confidence.

BASES

REFERENCES

1. FSAR, Chapter ~~[15]~~14
2. FSAR, Chapter ~~[6]~~
3. 10 CFR 50, Appendix A
4. 10 CFR 50.46
5. FSAR, Chapter 14, Section 14

~~B 3.2 POWER DISTRIBUTION LIMITS~~

~~B 3.2.2 Total Planar Radial Peaking Factor (F^T_{XY}) (Analog)~~

~~BASES~~

~~BACKGROUND~~ ~~The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO decreases or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow accident, ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System trip function. This LCO limits damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.~~

~~Methods of controlling the power distribution include:~~

~~a. Using CEAs to alter the axial power distribution;~~

~~b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and~~

~~c. Correcting off optimum conditions (e.g., a CEA drop or mis operation of the unit) that cause margin degradations.~~

~~The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings (LSSS) and this LCO are based on accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences (AOOs) and the limits of acceptable consequences are not exceeded for other postulated accidents.~~

~~Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.~~

~~Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the linear heat rate (LHR) and departure from nucleate boiling (DNB).~~

~~BASES~~

~~The limits on LHR, F^T_{XY}, Total Integrated Radial Peaking Factor (F^T_r), T_q, and ASI represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis.~~

~~Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System or the Incore Detector Monitoring System, provides adequate monitoring of the core power distribution and is capable of verifying that the LHR does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the ASI with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the ASI is maintained within the allowable limits specified in the COLR.~~

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

~~a. The CEA insertion limits of LCO 3.1.6, "Shutdown CEA Insertion Limits," and LCO 3.1.7, "Regulating CEA Insertion Limits," are satisfied;~~

~~b. The T_q restrictions of LCO 3.2.4 are satisfied; and~~

~~c. F^T_{XY} does not exceed the limits of this LCO.~~

~~The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors, and the alarms that have been established for the individual incore detector segments ensure 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 of 1.062;~~

~~c. An allowance of 1.002 for axial fuel densification and thermal expansion; and~~

~~d. A THERMAL POWER measurement uncertainty factor of 1.02.~~

~~BASES~~

~~APPLICABLE SAFETY ANALYSES~~ ~~The fuel cladding must not sustain damage as a result of normal operation (Condition 1) or AOOs (Condition 2) (Ref. 3, GDC 10). The Power Distribution and CEA Insertion and Alignment LCOs preclude core power distributions that violate the following fuel design criteria:~~

~~a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 4);~~

~~b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 3, GDC 10);~~

~~c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. [ ]); and~~

~~d. The 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, GDC 26).~~

~~The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. 4). This limiting is accomplished by maintaining the power distribution and reactor coolant conditions such that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations between measured quantities, the power distribution, and the uncertainties in the determination of 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 Reactor Coolant System ensure that these criteria are met as long as the core is operated within the ASI, F^T_{XY}, (F^T_r), and T_q limits specified in the COLR. 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.~~

~~BASES~~

~~Fuel cladding damage does not occur while at conditions outside the limits of these LCOs during normal operation. Fuel cladding damage could result, however, should an accident occur from initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local LHR.~~

~~F^T_{XY} satisfies Criterion 2 of the NRC Policy Statement.~~

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

~~APPLICABILITY~~ ~~In MODE 1, power distribution must be maintained within the limits assumed in the accident analyses to ensure that fuel damage does not result following an AOO. In other MODES, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on the core power distribution.~~

~~ACTIONS~~ ~~A.1 and A.2~~

~~A Note modifies Condition A to require Required Actions A.1 and A.2 to be completed if the Condition is entered. This ensures that corrective action is taken prior to unrestricted operation.~~

~~The limitations on F^T_{XY} provided in the COLR ensure that the assumptions used in the analysis for establishing the LHR, LCO, and LSSS remain valid during operation at the various allowable CEA group insertion limits. If F^T_{XY} exceeds its basic limitation, operation may continue under the additional restrictions imposed by these Required Actions (reducing THERMAL POWER and withdrawing CEAs to or beyond the long term steady state insertion limits of LCO 3.1.7), because these additional restrictions adequately ensure that the assumptions used in establishing the LHR, LCO, and LSSS remain valid (Ref. 3). Six hours to return F^T_{XY} to within its limit is reasonable and ensures that all CEAs meet the long term steady state insertion limits of LCO 3.1.7.~~

~~BASES~~

~~B.1~~

~~If F^T_{XY} cannot be returned to within its limit, THERMAL POWER must be reduced. A change to MODE 2 ensures that the core is operating within its thermal limits and places the core in a conservative condition. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.~~

~~SURVEILLANCE REQUIREMENTS~~

~~SR 3.2.2.1~~

~~The periodic Surveillance to determine the calculated F^T_{XY} ensures that F^T_{XY} remains within the range assumed in the analysis throughout the fuel cycle. Determining the measured F^T_{XY} after each fuel loading prior to the reactor exceeding 70% RTP ensures that the core is properly loaded.~~

~~Performance of the Surveillance every 31 days of accumulated operation in MODE 1 ensures that unacceptable changes in the F^T_{XY} are promptly detected.~~

~~The power distribution map can only be obtained after THERMAL POWER exceeds 20% RTP because the incore detectors are not reliable below 20% RTP.~~

~~The SR is modified by a Note that requires that SR 3.2.2.2 and SR 3.2.2.3 be completed each time SR 3.2.1.1 is completed. (Values computed by these SRs are required to perform SR 3.2.2.1.) The Note also requires that the incore detectors be used to determine F^T_{XY} by using them to obtain a power distribution map with all full length CEAs above the long term steady state insertion limits, as specified in the COLR.~~

~~SR 3.2.2.2 and SR 3.2.2.3~~

~~Measuring the value of F_{XY} and T_q each time a calculated value of F^T_{XY} is required ensures that the calculated value of F^T_{XY} accurately reflects the condition of the core.~~

~~The Frequency for these Surveillance is in accordance with the Frequency requirements of SR 3.2.2.1, because these SRs provide information to complete SR 3.2.2.1.~~

~~BASES~~

~~REFERENCES~~

~~1. FSAR, Chapter [15]~~

~~2. FSAR, Chapter [6]~~

~~3. 10 CFR 50, Appendix A~~

~~4. 10 CFR 50.46~~

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.~~3~~2 ~~Total Integrated Radial Peaking Factor (F_r^T) (Analog)~~ Radial Peaking Factors F_r^A and F_r^T

BASES

BACKGROUND

~~The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a Loss Of Coolant Accident (LOCA), loss of flow accident, ejected Control Element Assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.~~

~~Methods of controlling the power distribution include:~~

~~a. The use of CEAs to alter the axial power distribution;~~

~~b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and~~

~~c. Correcting off optimum conditions (e.g., a CEA drop or mis-operation of the unit) that cause margin degradations.~~

~~The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The Limiting Safety System Settings (LSSS) and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.~~

~~Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.~~

~~Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the linear heat rate (LHR) and departure from nucleate boiling (DNB).~~

~~The limits on LHR, Total Planar Radial Peaking Factor (F_{xy}^T), F_r^T, T_q, and ASI represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis.~~

~~Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System or the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the LHR does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the ASI with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the ASI is maintained within the allowable limits specified in the COLR.~~

~~In conjunction with the use of the Excore Detector Monitoring System and in establishing the ASI limits, the following conditions are assumed:~~

~~a. The CEA insertion limits of LCO 3.1.6, "Shutdown CEA Insertion Limits," and LCO 3.1.7, "Regulating CEA Insertion Limits," are satisfied;~~

~~b. The T_q restrictions of LCO 3.2.4 are satisfied; and~~

~~c. F_{xy}^T does not exceed the limits of LCO 3.2.2.~~

~~The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors, and the alarms established for the individual incore detector segments ensure 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 of 1.062;~~

~~b. An engineering uncertainty factor of 1.03;~~

~~c. An allowance of 1.002 for axial fuel densification and thermal expansion; and~~

~~d. A THERMAL POWER measurement uncertainty factor of 1.02.~~

A detailed BACKGROUND description for LCO 3.2.2, "Radial Peaking Factors F_r^A and F_r^T" is included in the description of LCO 3.2.1, "Linear Heat Rates (LHR)."

BASES

APPLICABLE SAFETY ANALYSES

~~The fuel cladding must not sustain damage as a result of normal operation (Condition 1) and AOOs (Condition 2) (Ref. 3, GDC 10). The power distribution and CEA insertion and alignment LCOs preclude core power distributions that violate the following fuel design criteria:~~

~~a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 4);~~

~~b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 3, GDC 10);~~

~~c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. [ ]); and~~

~~d. The 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, GDC 26).~~

~~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 reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1), with due regard for the correlations between measured quantities, the power distribution, and uncertainties in the determination of 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 Reactor Coolant System ensure that these criteria are met as long as the core is operated within the ASI, F_{xy}^T, and F_r^T limits specified in the COLR, and within the 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 range used in the accident analysis.~~

BASES

~~Fuel cladding damage does not occur while 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. This potential for fuel cladding damage exists because changes in the power distribution cause increased power peaking and correspondingly increased local LHR.~~

~~F_r^T satisfies Criterion 2 of the NRC Policy Statement.~~
A detailed APPLICABLE SAFETY ANALYSES description for LCO 3.2.2, "Radial Peaking Factors F_q^A and F_q^T is included in the description of LCO 3.2.1, "Linear Heat Rates (LHR)."

LCO

~~The LCO limits for power distribution are based on correlations between power peaking and measured variables used as inputs to LHR and departure from nucleate boiling ratio operating limits. The LCO limits for power distribution, except T_q, are provided in the COLR. The limitation on the LHR ensures that, in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F.~~
A detailed LCO description for LCO 3.2.2, "Radial Peaking Factors F_q^A and F_q^T" is included in the description of LCO 3.2.1, "Linear Heat Rates (LHR)."

APPLICABILITY

In MODE 1~~,~~ ≥ 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 other MODES, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on the core power distribution.

ACTIONS

A.1~~, A.2, and A.3~~

~~A Note modifying Condition A requires Required Actions A.1, A.2, and A.3 to be completed if the Condition is entered. This ensures that corrective action is taken prior to unrestricted operation.~~

~~The limitations on F_r^T provided in the COLR ensure that the assumptions used in the analysis for establishing the ASI, LCO, and LSSS remain valid during operation at the various allowable CEA group insertion limits. If F_r^T exceeds its basic limitation, operation may continue under the additional restrictions imposed by the Required Actions (reducing THERMAL POWER, withdrawing CEAs to or beyond the long term steady state insertion limits of LCO 3.1.7, and establishing a revised upper THERMAL POWER limit) because these additional restrictions provide adequate provisions to ensure that the assumptions used in establishing the LHR, LCO, and LSSS remain valid. Six hours to return F_r^T to within its limits is reasonable and ensures that all CEAs meet the long term steady state insertion limits of LCO 3.1.7.~~

In the event F_r^A and/or F_r^T are not within specified limits, THERMAL POWER must be reduced to ensure the core is operating within the assumptions made in the safety analysis. The acceptable power level is specified by equation #1 below:

EQ #1

$$P = \left[1 - 3.33 \left(\frac{F_r}{F_t} - 1\right) (RTP)\right]$$

Where;

F_r = Measured valve of either F_r^A or F_r^T

F_t = Corresponding limit stated in the COLR

Operating at or below this power level ensures that the assumptions made in the safety analysis are preserved. The completion time of 6 hours is an adequate period of time to perform power level adjustments.

BASES

B.1

If F_r^T and/or F_r^A cannot be returned to within its limit, THERMAL POWER must be reduced. ~~A change to MODE 2 ensures that the core is operating within its thermal limits and places the core in a conservative condition. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.~~ Operation at or below 25% RTP ensures ample THERMAL MARGIN exists to meet applicable fuel design limits. A completion time of 6 hours is sufficient to complete a safe and orderly power reduction.

SURVEILLANCE REQUIREMENTS

SR 3.2.~~3~~2.1

The periodic Surveillance to determine the calculated F_r^T and F_r^A ensures that ~~F_r^T~~ radial peaking remains within the range assumed in the safety analysis throughout the fuel cycle. Determining the measured ~~F_r^T~~ radial peaking factors once after each fuel loading prior to exceeding ~~70~~50% RTP ensures that the core is properly loaded.

Performance of the Surveillance every 31 days of accumulated operation in MODE 1 ensures that unacceptable changes in the ~~F_r^T~~ radial power distribution are promptly detected.

The power distribution map can only be obtained after THERMAL POWER exceeds ~~20~~25% RTP because the incore detectors are not reliable below ~~20~~25% RTP.

~~The SR is modified by a Note that requires SR 3.2.3.2 and SR 3.2.3.3 be completed each time SR 3.2.3.1 is completed. This procedure is required because the values computed by these SRs are required to perform this SR.~~

~~SR 3.2.3.2 and SR 3.2.3.3~~

~~Measuring the values of F_r^T and T_q each time a value of F_r^T is calculated ensures that the calculated value of F_r^T accurately reflects the condition of the core.~~

~~The Frequency for these Surveillance is in accordance with the requirements of SR 3.2.3.1 because these SRs provide information to complete SR 3.2.2.1.~~

BASES

REFERENCES

~~1. FSAR, Chapter [15]~~

~~2. FSAR, Chapter [6]~~

~~3. 10 CFR 50, Appendix A~~

~~4. 10 CFR 50.46~~

None

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.43 ~~AZIMUTHAL~~ Quadrant Power Tilt (T_q) ~~(Analog)~~

BASES

BACKGROUND

~~The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow accident, ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.~~

~~Methods of controlling the power distribution include:~~

~~a. Using CEAs to alter the axial power distribution;~~

~~b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and~~

~~c. Correcting off optimum conditions (e.g., a CEA drop or mis-operation of the unit) that cause margin degradations.~~

~~The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.~~

~~Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.~~

~~Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits for Linear Heat Rate (LHR) and Departure from Nucleate Boiling (DNB).~~

BASES

~~The limits on LHR, Total Planar Radial Peaking Factor (F^T_{xy}), Total Integrated Radial Peaking Factor (F^T_r), T_q, and ASI represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis.~~

~~Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System or the Incore Detector Monitoring System, provides adequate monitoring of the core power distribution and is capable of verifying that the LCO limits are not exceeded. The Excore Detector Monitoring System performs this function by continuously monitoring ASI with OPERABLE quadrant symmetric excore neutron detectors and by verifying ASI is maintained within the limits specified in the COLR.~~

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

~~a. The CEA insertion limits of LCO 3.1.6, "Shutdown CEA Insertion Limits," and LCO 3.1.7, "Regulating CEA Insertion Limits," are satisfied;~~

~~b. The T_q restrictions of LCO 3.2.4 are satisfied; and~~

~~c. F^T_{xy} does not exceed the limits of LCO 3.2.2.~~

~~The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors, and the alarms that have been established for the individual incore detector segments ensure 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 of 1.062;~~

~~b. An engineering uncertainty factor of 1.03;~~

~~c. An allowance of 1.002 for axial fuel densification and thermal expansion; and~~

~~d. A THERMAL POWER measurement uncertainty factor of 1.02.~~

A detailed BACKGROUND description for LCO 3.2.3, "Quadrant Power Tilt (T_q)" is included in the description of LCO 3.2.1, "Linear Heat Rates (LHR)."

BASES

APPLICABLE SAFETY ANALYSES

~~The fuel cladding must not sustain damage as a result of normal operation (Condition 1) or AOOs (Condition 2) (Ref. 3, GDC 10). The power distribution and CEA insertion and alignment LCOs preclude core power distributions that violate the following fuel design criteria:~~

~~a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 4);~~

~~b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 3, GDC 10);~~

~~c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. [ ]); and~~

~~d. The 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, GDC 26).~~

~~The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. 4). This process is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analysis (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 (LHGR) 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 Reactor Coolant System ensure that these criteria are met as long as the core is operated within the ASI, F_{xy}^T, and F_r^T limits specified in the COLR, and within the 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 range used in the accident analyses.~~

~~Fuel cladding damage does not occur while the reactor is operating at conditions outside these LCOs during otherwise normal operation. Fuel cladding damage could result, however, if an accident occurs from initial conditions outside the limits of these LCOs. Changes in the power distribution cause increased power peaking and correspondingly increased local LHRs.~~

BASES

APPLICABLE SAFETY ANALYSES

~~The T_q satisfies Criterion 2 of the NRC Policy Statement.~~
A detailed APPLICABLE SAFETY ANALYSES description for LCO 3.2.3, "Quadrant Power Tilt (T_q)" is included in the description of LCO 3.2.1, "Linear Heat Rates (LHR)."

LCO

~~The power distribution LCO limits are based on correlations between power peaking and the measured variables used as inputs to the LHR and departure from nucleate boiling ratio operating limits. The power distribution LCO limits, except T_q, are provided in the COLR. The limits on LHR ensure that in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F.~~
A detailed LCO description for LCO 3.2.3, "Quadrant Power Tilt (T_q)" is included in the description of LCO 3.2.1, "Linear Heat Rates (LHR)."

APPLICABILITY

In MODE 1 ~~with THERMAL POWER~~ > ~~50~~25% RTP, power distribution must be maintained within the limits assumed in accident analysis to ensure that fuel damage does not result following an AOO. In other MODES, this LCO does not apply because THERMAL POWER is not sufficient to require a limit on core power distribution.

ACTIONS

A.1 ~~and A.2~~

~~If the measured T_q is > [0.03] and < 0.10, the calculation of T_q may be non conservative. T_q must be restored within 2 hours or F_{xy}^T 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 hours is sufficient time to allow the operator to reposition CEAs, and significant radial xenon redistribution cannot occur within this time. The 8 hour Completion Time ensures changes in F_{xy}^T and F_r^T can be identified before the limits of LCO 3.2.2 and LCO 3.2.3, respectively, are exceeded.~~

If the measured quadrant power tilt is > 5% but ≤ 10%, SR 3.2.2.1 (Peaking factor verification) must be performed within 2 hours. Two hours is sufficient time to allow the operator to verify radial peaking is within limits, and significant radial Xenon redistribution cannot occur within this time. This ensures the core is operated within the limits assumed in the safety analysis.

~~B.1~~

~~If Required Actions and associated Completion Times of Condition A are not met, THERMAL POWER must be reduced to ≤ 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 50% RTP in an orderly manner and without challenging plant systems.~~

B.1 and B.2

If Quadrant Power Tilt (T_q) exceeds 10% then THERMAL POWER must be reduced to ≤ 50% RTP within four hours. The radial peaking factors must also be verified by performing SR 3.2.2.1 within 4 hours and once per 8 hours thereafter. This requirement ensures that the core is operating within its thermal limits and places the core in a conservative condition. A completion time of 4 hours is a reasonable time to reach 50% RTP and verify radial peaking in an orderly manner without challenging plant systems.

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

~~With T_q > 0.10, F^T_{xy} and F^T_r must be within their specified limits to ensure that acceptable flux peaking factors are maintained. Based on operating experience, 1 hour is sufficient time for the operator to evaluate these factors. If F^T_{xy} and F^T_r 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.~~

~~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 radial power peaking factors used in core power distribution calculations are based on an un-tilted power distribution.~~

~~If T_q is not restored to within its limits, the reactor continues to operate with an axial power distribution mismatch. Continued operation in this configuration may induce an axial xenon oscillation that causes increased LHGRs when the xenon redistributes. If T_q cannot be restored to within its limits within 2 hours, reactor power must be reduced. Reducing THERMAL POWER to ≤ 50% RTP within 2 hours provides conservative protection from increased peaking due to potential xenon redistribution.~~

C.1

If Quadrant Power Tilt (T_q) is > 15% or required action and associated completion times are not met the plant shall be < 25% RTP within 12 hours. Except as a result of control rod misalignment a T_q > 15% is not expected; if it occurs, continued operation of the reactor may be necessary to discover the cause of the tilt. 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 radial peaking factors used in core power distribution calculation are based on an untitled power distribution.

If T_q is not restored to within limits, the reactor continues to operate with an axial power distribution mismatch. Continued operation in this configuration may induce an axial xenon oscillation that causes increased LHGRs when the xenon redistributes.

If T_q cannot be restored to within limits, reactor power must be reduced to < 25% RTP within 12 hours. This provides conservative protection from increased peaking due to potential xenon redistributions.

The Required Actions are modified by a Note that requires all subsequent actions to be performed once power reduction commences after entering the Condition if T_q is not restored to < ~~0.10~~15%. This procedure ensures corrective action is taken before unrestricted power operation resumes. Following THERMAL POWER reduction to ≤ ~~50~~25% RTP, T_q must be restored to ≤ ~~[0.03]~~3% before THERMAL POWER is increased~~ (Required Action C.3)~~. This ~~Required~~ action prevents the operator from increasing THERMAL POWER above the conservative limit when the Condition, T_q outside its limits, has existed but allows the ~~unit~~ plant to continue operation for diagnostic purposes. The Completion Time of Required Action C.~~3~~1 is modified with a Note to indicate that the cause of the out of limit condition must be corrected prior to increasing THERMAL POWER.

This Note also indicates that subsequent power operation above ~~50~~25% RTP may proceed provided that the measured T_q is verified ≤ ~~[0.03]~~3% at least once per hour for 12 hours, or until verified at 95% RTP. This ensures that the power distribution is responding as predicted. The Completion Time of 12 hours is a historical value that allows an acceptable exit from the LCO after the T_q value is verified acceptable for 12 hours or until 95% RTP is reached.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

T_q must be calculated at 12 hour intervals. The 12 hour Frequency prevents significant xenon redistribution between Surveillance.

REFERENCES

~~1. FSAR, Chapter [15]~~

~~2. FSAR, Chapter [6]~~

~~3. 10 CFR 50, Appendix A~~

~~4. 10 CFR 50~~

None

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.~~5~~4 Axial Shape Index (ASI) ~~(Analog)~~

BASES

BACKGROUND

~~The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analysis. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow accident, ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.~~

~~Methods of controlling the power distribution include:~~

~~a. Using CEAs to alter the axial power distribution;~~

~~b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and~~

~~c. Correcting off optimum conditions (e.g., a CEA drop or mis-operation of the unit) that cause margin degradations.~~

~~The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.~~

~~Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.~~

~~Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on linear heat rate (LHR) and departure from nucleate boiling (DNB).~~

~~The limits on LHR, Total Planar Radial Peaking Factor (F^T_{xy}), Total Integrated Radial Peaking Factor (F^T_r), T_q, and ASI represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis.~~

~~Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the LHR does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the ASI with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the ASI is maintained within the allowable limits specified in the COLR.~~

~~In conjunction with the use of the Excore Detector Monitoring System and in establishing the ASI limits, the following conditions are assumed:~~

~~a. The CEA insertion limits of LCO 3.1.6, "Shutdown CEA Insertion Limits," and LCO 3.1.7, "Regulating CEA Insertion Limits," are satisfied;~~

~~b. The T_q restrictions of LCO 3.2.4 are satisfied; and~~

~~c. F^T_{xy} does not exceed the limits of LCO 3.2.2.~~

~~The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors, and the alarms that have been established for the individual incore detector segments ensure that the peak LHR is maintained within the limits specified in the COLR. The setpoints for these alarms include tolerances, set in conservative directions, as follows:~~

~~a. A measurement calculational uncertainty factor of 1.062;~~

~~b. An engineering uncertainty factor of 1.03;~~

~~c. An allowance of 1.002 for axial fuel densification and thermal expansion; and~~

~~d. A THERMAL POWER measurement uncertainty factor of 1.02.~~

A detailed BACKGROUND description for LCO 3.2.4, "Axial Shape Index (AXI)" is included in the description of LCO 3.2.1, "Linear Heat Rates (LHR)."

APPLICABLE SAFETY ANALYSES

~~The fuel cladding must not sustain damage as a result of normal operation (Condition 1) or AOOs (Condition 2) (Ref. 3, GDC 10). The power distribution and CEA insertion and alignment LCOs prevent core power distributions from reaching levels that violate the following fuel design criteria:~~

~~a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 4);~~

~~b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 3, GDC 10);~~

~~c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 4); and~~

~~d. The 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, GDC 26).~~

~~The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. 4). This limitation is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations among measured quantities, the power distribution, and uncertainties in the determination of 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 Reactor Coolant System ensure that these criteria are met as long as the core is operated within the ASI, F_{xy}^T, and F_r^T limits specified in the COLR, and within the 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.~~

BASES

~~Fuel cladding damage does not occur while the reactor is operating at conditions outside these LCOs during normal operation. Fuel cladding damage results, however, when an accident occurs from initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.~~

~~The ASI satisfies Criterion 2 of the NRC Policy Statement.~~
A detailed APPLICABLE SAFETY ANALYSES description for LCO 3.2.4, "Axial Shape Index (ASI)" is included in the description of LCO 3.2.1, "Linear Heat Rates (LHR)."

LCO

~~The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and Departure from Nucleate Boiling Ratio (DNBR) operating limits. These power distribution LCO limits, except T_q, are provided in the COLR. The limitation on LHR ensures that in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F.~~

The limitation on ASI, along with the limitations of LCO 3.3.1, "Reactor Protection System Instrumentation," represents a conservative envelope of operating conditions consistent with the assumptions that have been analytically demonstrated adequate for maintaining an acceptable minimum DNBR throughout all AOOs. Of these, the loss of flow transient is the most limiting. Operation of the core with conditions within the specified limits ensures that an acceptable minimum margin from DNB conditions is maintained in the event of any AOO, including a loss of flow transient.

APPLICABILITY

In MODE 1 with THERMAL POWER > ~~20~~25% RTP, power distribution must be maintained within the limits assumed in the accident analyses to ensure that fuel damage does not result following an AOO. In other MODES, this LCO does not apply because THERMAL POWER is not sufficient to require a limit on the core power distribution. Below ~~20~~25% RTP the incore detector accuracy is not reliable.

BASES

ACTIONS

A.1

Operating the core within ASI limits specified in the COLR and within the limits of LCO 3.3.1 ensures an acceptable margin for DNB and for maintaining local power density in the event of an AOO. Maintaining ASI within limits also ensures that the limits of 10 CFR 50.46 are not exceeded during accidents. The Required Actions to restore ASI must be completed within 2 hours to limit the duration the plant is operated outside the initial conditions assumed in the accident analyses. In addition, this Completion Time is sufficiently short that the xenon distribution in the core cannot change significantly.

B.1

If the ASI cannot be restored to within its specified limits, or ASI cannot be determined because of Excore ~~Detector~~ Monitoring System inoperability, core power must be reduced. Reducing THERMAL POWER to ≤ ~~20~~25% RTP ensures that the core is operating further from thermal limits and places the core in a conservative condition. Four hours is a reasonable amount of time, based on operating experience, to reduce THERMAL POWER to ≤ ~~20~~25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.~~5~~4.1

Verifying that the ASI is within the specified limits ensures that the core is not approaching DNB conditions. A ~~Frequency of 12 hours~~ continuous frequency while using the incore system to monitor LHR is adequate for the operator to identify trends in conditions that result in an approach to the ASI limits, because the mechanisms that affect the ASI, such as xenon redistribution or ~~CEA~~ control rod drive mechanism malfunctions, cause the ASI to change slowly and should be discovered before the limits are exceeded.

REFERENCES

~~1. FSAR, Chapter [15]~~

~~2. FSAR, Chapter [6]~~

~~3. 10 CFR 50, Appendix A~~

~~4. 10 CFR 50.46~~

None

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 Excore Power Distribution Monitoring

BASES

BACKGROUND The excore power distribution monitoring system consists of Power Range Detector 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 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 upper and lower sections have a total active length of 12 feet. The DC current signal from each of the ion chambers is fed directly to the control room drawer assembly without preamplification. Each excore detector supplies data to a Thermal Margin Monitoring System (TMM) unit. Each TMM uses these excore signals to calculate Axial Shape Index (ASI) on a continuous basis.

ASI can be defined as the compensated ratio of power developed in the upper and lower sections of the core. The TMM takes the excore detector signals and develops a power ratio (YE) that describes the distribution of neutron flux developed in the core by the formula:

$$YE = (L - U)/(L + U)$$

Where;

L = Lower Excore segment flux

U = Upper Excore segment flux

The Excore detectors which are located within the concrete biological shield of the reactor must be compensated for the phenomenon of shape annealing. Shape Annealing factors are developed to correct the excore readings for neutron attenuation from the core periphery to the excore detector locations. This accounts for any material that would cause neutron attenuation within the detector path such as: concrete, structural steel and so forth. This allows the excore detectors to represent an accurate measurement of the core power distribution. Shape annealing has been found to be a linear relationship which can be correlated to the Axial Offset (AO) as determined by an Incore Detector System to the raw readings seen by the excore detectors.

BASES

Reactor Engineering has developed shape annealing factors for each individual Excore detector. The TMM uses the above calculated power ratio and the appropriate shape annealing factor to determine the ASI value for an individual excore detector channel. The equation below is a typical representation of the formula used to calculate ASI for a particular excore detector channel.
YI = ASI = 2.3YE

APPLICABLE SAFETY ANALYSES

The purpose of LCO 3.2.5, "Excore Power Distribution Monitoring" ensures that fuel design conditions and safety analysis assumptions are maintained. This is performed by ensuring ASI will be maintained within limits by an alarming function supplied by the excore detectors.

The applicable safety analysis entails the ability ensure that when ASI is deviating outside of setpoints the operator will be signaled to correct the ASI by adjusting the axial power distribution. The main reason for ASI is to identify the flux shape which would cause DNB for variations in power level.

The LCOs governing LHR, ASI and the Primary Coolant System ensure that the core is operated within assumptions made in the safety analysis. These criterion are met when the core is operated within ASI, F_r^A, and F_r^T limits specified in the COLR. These are process variables that characterize the reactor core three dimensional power distributions. Fuel cladding damage does not occur while the plant is operating at conditions outside the limits of these LCOs during normal operation and Abnormal Operational Occurrences (AOOs).

LCO

This LCO ensures that ASI calculated by the Thermal Margin Monitor (TMM) is operable with detection input signals from at least two channels. ASI being calculated from two input signals ensures that adequate ASI monitoring is occurring. Two channels of detection is an accurate representation of core conditions since in the absence on any significant flux tilt one channel could give an accurate picture of ASI for the given core conditions. The presence of two input signals from two separate core quadrants will see communication of significant flux tilts occurring in the other two undetected quadrants and the excore deviation alarm will also flag any flux tilts that may occur by comparing individual values to a calculated mean value. This allows confidence that the core ASI is within assumptions made in the safety analysis. This ensures a flux tilt will be seen and the ASI values calculated are accurate.

BASES

APPLICABILITY This LCO is only valid when the excore detectors are being used for LHR or Quadrant Power Tilt (T_q) monitoring. While using the encores for LHR or T_q monitoring the applicability range for this LCO is MODE 1 with RTP ≥ 25%. Below 25% RTP ample thermal margin exists to ensure LHR and T_q are within limits therefore, below 25% RTP the LCO does not apply.

ACTIONS A.1

In the event the excore detector deviation alarm is inoperable T_q must be calculated every 12 hours. The excore deviation alarm channel consistently looks at the deviation between the four detectors from the average which signals a potential flux tilt while this alarm channel is operating. This ensures that quadrant tilt is with limits while continuous monitoring of T_q may not be occurring.

B.1 and B.2

In the event the T_q measured from the excores is deviating from the T_q measured from the incores by ≥ 2% then T_q must be calculated using the Incore detectors within 12 hours. Under these conditions, Surveillance Requirements 3.2.5.2." Excore ASI CALIBRATION" must also be performed within 7 days. The incore monitoring system is generally more reliable therefore, when an absolute deviation of 2% occurs the incore system shall be used to verify T_q. The specified Completion Times are adequate to perform the stated actions in a safe and efficient manner.

C.1

When the condition arises that 3 or 4 ASI alarm channels are inoperable, LHR monitoring with the incore alarm system should be initiated within 1 hour. This action ensures that ASI is being monitored accurately in the event the excore system can no longer be used. This ensures that assumptions made in the safety analysis are being maintained within acceptable limits. The Completion Time of 1 hour is sufficient to safely complete this action and reflects the importance of monitoring ASI accurately at all times.

BASES

D.1

In the event that ASI is deviating from AO by > .02 under steady state operating conditions. Surveillance Requirement 3.2.5.2 should be performed within 12 hours. This allows for the recalibration of AO and T_q between the excore and incore monitoring system. This action ensures that the incore core averaged Axial Offset (AO) and the individual excore ASI are within acceptable limits. This yields further confidence that ASI is being monitored accurately and still supports the safety analysis. The Completion Time of 12 hours is adequate to safely complete this required action.

D.2.1 and D.2.2

When the excore ASI is deviating from incore AO by > .02 under steady state conditions, the ASI alarm setpoint and TM.LP trip function must be compensated for this deviation within 12 hours. This ensures that fuel design parameters can accurately be monitored and not exceeded, when the incore/excore alignment is not within normal tolerances. This may occur when the calibration cannot be performed or the alignment problem exists after the calibration. The Completion Time of 12 hours is sufficient to perform these adjustments, accurately and efficiently.

D.3.1 and D.3.2

In the event that excore ASI is deviating from incore AO > .02 under steady state conditions the ASI alarm channel and TM/LP RPS trip units must be declared inoperable within 12 hours. This ensures that operators are not relying on a potentially inaccurate excore monitoring system. By placing these items inoperable it forces the operators to take appropriate action to correct the problem or place the plant in a conservative condition. 12 hours is a sufficient time period to expedite these actions.

E.1

If the Required Action and associated Completion Times are not met the plant must be < 25% RTP within 6 hours. This action ensures that the plant is not operated in an applicability range that relies on accurate power monitoring from the excore system. Below 25% RTP ample thermal margin exists to ensure the plant conditions meet the safety analysis.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1

This surveillance calls for the verification of measured individual excore channel AO when compared to the total core AO measured by the incores in ≤ 0.02 on a 31 day frequency. This SR is performed in conjunction with incore alarm updates on a monthly basis. This ensures that the excore system is functional for power monitoring if needed during normal plant operations. The frequency of 31 days is adequate to detect any detector drifting or long term abnormalities that would affect the functionality of the excore system for power monitoring.

SR 3.2.5.2

The calibration of T_q and AO from excore detectors with T_q and AO measured from the incores is performed at least every 184 days (6 months) or when warranted by required actions. This calibration ensures that incore averaged axial offset corresponds to excore ASI within 2% for each individual excore detector. It also ensures that the incore/excore calculated T_q are within alignment tolerances of 2%. The SR supports a high degree of confidence that the excores can accuretly be used for power monitoring. The frequency of 184 days ensures that system is recalibrated regardless whether the 2% limit has been reached within 184 days.

SR 3.2.5.3

This surveillance requires the performance of a CHANNEL CALIBRATION of the Excore Detector Deviation Alarm Channel. The excore detector deviation alarm channel compares the reported flux from each individual excore to the computed average and determines if all detectors are within a 2% deviation. This supports flux tilt detection to ensure valid ASI values are being reported. This is essential especially when only two ASI alarm channels are operable to ensure that significant flux tilts are detected that would corrupt the ASI values generated from the functional two excore channels. A frequency of 18 months is sufficient to ensure that the excore deviation alarm is adequately calibrated for flux tilt detection.

BASES

SR 3.2.5.4

This surveillance requires that a CHANNEL CALIBRATION of the excore ASI circuitry is performed each 18 months. This SR ensures that the appropriate ASI value is generated by the TMM for a particular excore reading and that the ASI alarming circuit would actuate when the calculated ASI value falls outside of specified upper and lower setpoints. This instills a high level of confidence that ASI is calculated accuretly and out of tolerance condition would be annunciated. A frequency of 18 months is adequate to ensure a high degree of system performance during plant operations.

REFERENCES None

ATTACHMENT 6

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

3.2 POWER DISTRIBUTION LIMITS

Comparison of Revised and Standard Technical Specifications

Palisades Revised Tech Spec Requirement List. (03/28/96)

A listing of the proposed Palisades Revised Tech Specs (RTS) correlated to the CE Standard Tech Specs (STS).

First Column; Proposed Palisades Revised Tech Spec (RTS) number
Each RTS item is listed in the left-most column.
If a STS item has been omitted from RTS, the word 'Omitted' is used.

Second Column; CE Standard Tech Spec (STS) number
The corresponding STS item is listed in the second column.
If a RTS item does not appear in STS, it is noted as 'Added'.

Third Column; Existing Palisades Tech Spec (TS) number
The closest TS item is listed in the third column.
If a RTS item does not appear in TS, it is noted as 'New'.

Fourth Column; RTS Item Description
An abbreviation of the RTS item appears in the third column.
Each item is identified as: LCO, ACTION, SR, ADMIN, Exception, etc.
In cases where a STS item was omitted from RTS, the description is of the STS item.

Description Key: RTS requirement type:	Column 4 syntax:
Safety Limit	SL: Safety limit; Applicable conditions
Limiting Condition for Operation	LCO: LCO Description; Applicable conditions
Condition	COND: Description of non-conforming condition
Action	ACTN: Required action; Completion time
Surveillance Requirement	SR: Test description; Frequency
Table	TABL: Title
Administrative Requirement	ADMN: Administrative requirement
Defined Term	DEF: Name of defined term

Fifth Column; Comments and Explanations of Differences between RTS and STS.
A brief explanation of differences between RTS and STS is provided in the fifth column.

Other abbreviations used in the listing are:

NA:	Not Applicable
CFT:	Channel Functional Test
CHNL:	Channel

RTS Number	STS Number	TS Number	RTS (STS) requirement Description	Explanation of Differences

Global differences between the proposed Palisades Technical Specifications and the Standard Technical Specifications for CE plants, Nureg 1432:

The following changes are not discussed in the explanation of differences for each TS requirement.

1) Bracketed values have been replaced with appropriate values for Palisades. Typically, the basis for these values is provided in the bases document.

2) Each required action of the form "Perform SR X.X.X.X . . ." has been altered by a parenthetical summary of the SR requirements. This change allows a reader to understand the required actions without constantly turning pages to locate the referenced SR.

3) Terminology has been changed to reflect Palisades usage:

"RWT"	becomes	"SIRWT"	Safety Injection Refueling Water Tank
"CEA"	becomes	"Control Rod" or "Rod"	Palisades uses cruciform control rods rather than the multifingered "Control Element Assemblies" of later CE plants.
"RCS"	becomes	"PCS"	Palisades terminology is "Primary Coolant System" rather than "Reactor Coolant System"
"SIAS"	becomes	"SIS"	Palisades terminology is "Safety Injection Signal" rather than "Safety Injection Actuation Signal"
"AC Vital bus"	becomes	"Preferred AC bus"	Palisades terminology.
"PAMI"	becomes	"AMI"	Accident Monitoring Instrumentation, Palisades terminology
"ESFAS"	becomes	"ESF Instrumentation"	There is no stand-alone ESFAS system or cabinet at Palisades; ESF instruments actuate the ESF functions
"DG LOVS"	becomes	"DG UV Start"	Palisades Terminology
"Remote Shutdown System"	becomes	"Alternate Shutdown System"	Palisades Terminology
"Power Rate of Change-High"	becomes	"High Startup Rate"	Palisades Terminology

RTS Number	STS Number	TS Number	RTS (STS) requirement Description	Explanation of Differences
3.2	3.2	3.23	POWER DISTRIBUTION	Several major differences exist between Palisades and the "Standard" CE plant which affect this section: Palisades is the oldest CE PWR and has different hardware and analyses from the newer CE plants; Palisades also uses ANF fuel rather than CE fuel. Therefore several of the LCOs, actions, etc is this section differ from RSTS. Actions, Completion times, SRs and frequencies were kept as close to the RSTS as possible while implementing a different set of limitations and requirements. The conditions and actions specified reflect current tech specs and operating practice.
3.2.1	3.2.1	3.23.1	LCO: LHR < limit and Incore Alms OPERABLE; ≥50%	The LCO statement is essentially unchanged with the exception of changing the applicability to MODE 1 ≥ 50%. The 50% RTP applicability is retained from the current license. Palisades has a great deal of thermal margin below this power range.
3.2.1 A	3.2.1 A	3.23.1	COND: LHR not within limits incores	Essentially unchanged, reworded for clarity.
Omitted	3.2.1 A	3.23.1	COND: LHR not within limits excores	See note for section 3.2.
3.2.1 A.1	3.2.1 A.1	3.23.1	ACTN: Restore LHR; 1 hour	Unchanged.
3.2.1 B	Added	3.23.1	COND: Incore alarm system inoperable	Condition added to support Incore alm system in LCO.
3.2.1 B.1.1	Added	3.23.1	ACTN: Verify Excores avail for LHR monitoring; 2 hrs	Added action to support added LCO requirement
3.2.1 B.1.2	Added	3.23.1	ACTN: Restrict power to APL; 2 hrs	Added action to support added LCO requirement
3.2.1 B.1.3	Added	3.23.1	ACTN: Verify T_q; pwr & ASI; each 4 hrs	Added action to support added LCO requirement
3.2.1 B.2.1	Added	3.23.1	ACTN: Restrict power to < 85%; 2 hrs	Added action to support added LCO requirement
3.2.1 B.2.2	Added	3.23.1	ACTN: Verify LHR w. manual incore readings; each 4 hrs	Added action to support added LCO requirement
3.2.1 C	3.2.1 B	3.0.3	COND: Required Action not met	Unchanged.
3.2.1 C.1	3.2.1 B.1	3.23.1	ACTN: Be < 50% RTP; 2 hrs	Changed action taking plant out of applicability range < 50% RTP. This value is retained from current license and allows the core to be placed in a conservative state with ample thermal margin.
3.2.1.1	Added	4.19.1.2.d	SR: Verify AO is w/in 0.05 of target; 15 minutes	This is retained from Palisades current license and has no comparable function in STS
3.2.1.2	Added	New	SR: Verify Incore alm system function for LHR monitor; 12 hrs	This SR ensures that operators are informed by the Plant process computer that the incore system is not functional for LHR monitoring.

RTS Number	STS Number	TS Number	RTS (STS) requirement Description	Explanation of Differences
3.2.1.3	Added	New	SR: Verify LHR w/in limit; 4 hours	LHR is continuously monitored by the Incore Monitoring System however this surveillance has been modified to a frequency of 4 hours to allow for the manual reading of Incores if the condition warrants it necessary.
3.1.2.4	3.2.1.3	4.19.2.1	SR: Update Incore Alarm Set points; 31 days	Removed unnecessary notes and reworded to apply to reflect Palisades current Power Distribution monitoring and incore alarm updates.
3.2.1.5	Added	4.18.2.1.a	SR: Verify APL, T_q, Target AO; 31 days	The note modifies this SR to illustrate that this is only applicable when using the excore monitoring system for LHR monitoring. This SR ensures that the excore monitoring system can be used for LHR monitoring.
3.2.1.6	Added	4.17	SR: Perform incore alarm CHANNEL CALIBRATION	An incore alarm CHANNEL CALIBRATION is performed to ensure alarm actuation upon signal receipt within limits.
Omitted	3.2.1.1	NA	SR: (Verify ASI alm setpoints)	NA to Palisades LHR LCO; see Palisades RTS SR 3.2.5.2
Omitted	3.2.1.2	NA	SR: (Demonstrate Local power density alms)	NA Palisades
3.2.2	3.2.3	3.23.2	LCO: Radial peaking factors: ≥ 25%	Changed LCO and applicability to reflect Palisades requirements.
3.2.2 A	3.2.3 A	3.23.2	COND: Radial peaking not w/in limit	Reworded to reflect Palisades limits.
3.2.2 A.1	3.2.3 A.1	3.23.2	ACTN: Restore radial peaking factors; 6 hrs	Reworded to reflect Palisades limits.
Omitted	3.2.3 A.2	NA	ACTN: (Withdraw CEAs above long term limit; 6 hrs)	NA to Palisades; we have no comparable limit or requirement.
Omitted	3.2.3 A.3	NA	ACTN: (Establish a revised upper THERMAL POWER limit)	NA Palisades has no comparable limit or requirement.
3.2.2 B	3.2.3 B	3.0.3	COND: Required action not met	Changed applicability to take plant below 25% RTP placing the plant outside of the LCO requirements and placing the plant in a conservative state.
3.2.2.1	3.2.2.1	4.19.2.1.b	SR: Verify Radial Peaking and T_q	Changed SR to reflect Palisades limits and requirements. Standard SR 's 3.2.3.1,3.2.3.2, and 3.2.3.3 have been combined into the Palisades specific SR 3.2.2.1 since when radial peaking is verified quadrant power tilt is also verified by default.
Omitted	3.2.3.2	NA	SR: (Verify F_r)	Palisades does not use F_{xy}.
Omitted	3.2.3.3	NA	SR: Verify (T_q)	Palisades uses a different parameter labeled T_q, which is the subject of RTS LCO 3.2.3.

RTS Number	STS Number	TS Number	RTS (STS) requirement Description	Explanation of Differences
Omitted	3.2.3	NA	LCO: (Total Planar Radial Peaking Factor)	Palisades does not use this parameter. Entire LCO with actions and SRs omitted.
3.2.3	3.2.4	3.23.3	LCO: T_q < 5%; ≥ 25%	The parameters represented by T_q are similar, but not identical. The limits and specified actions were retained from current Palisades Tech Specs.
3.2.3 A	3.2.4 A	3.23.3.1	COND: 5% < T_q < 10%	Changed quadrant power tilt to be expressed in a percentage and changed the lower limit to a Palisades specific value from current T.S. of 5%.
Omitted	3.2.4 A.1	3.23.3.1.a	ACTN: (Restore Tq; 2 hrs)	Unnecessary; restoration is always an option, whether before or after 2 hours.
3.2.3 A.1	3.2.4 A.2	3.23.3.1.b	ACTN: Verify Radial Peaking Factors; 2 hrs & every 8 hrs	Reworded with intent and action remaining unchanged.
3.2.3 B	Added	3.23.3.2	COND: T_q > 10%	Unchanged. Retained from current T.S.
3.2.3 B.1	3.2.4 B.1	3.23.3.2.b	ACTN: Restrict power; 4 hrs	Unchanged. Retained from current T.S.
3.2.3 B.2	3.2.4 A.2	3.23.3.2.b	ACTN: Verify Radial Peaking Factors; 4 hrs & each 8 hrs	This step moved to after power reduction action since power reduction required w/o respect to radial peaking factor results, and the power decrease required by action B.1 is what LCO 3.2.2 would require for radial peaking out of spec.
3.2.3 C	Added	3.23.3.3	COND: Required action not met, or Tq > 15%	Unchanged. Retained from current T.S.
3.2.3 C.1	3.2.4 C.2	3.23.3.3	ACTN: be out of applicability (25%; 12 hrs	Unchanged. Retained from current T.S. A 12 hr completion time for power reduction is necessary to ensure a safe and orderly shutdown from this condition. Power reduction to < 25 % RTP ensures adequate thermal margin and places the plant in a conservative condition.
Omitted	3.2.4 C.1	NA	ACTN: (Verify Radial Peaking Factors; 1 hr)	Once in this condition action states to place the plant < 25% RTP which allows for adequate thermal margin and places the plant in a conservative condition. Peaking factor verification is not warranted.
Omitted	3.2.4 C.3	NA	ACTN: (Restore T_q ≤ 3%; Prior to increase)	Restore is always an option
3.2.3.1	3.2.4.1	4.18.2.1	SR: Verify T_q; 12 hrs	Unchanged.
3.2.4	3.2.5	New	LCO: ASI w/in limit; ≥ 25%	Used Palisades applicability. Deleted reference to specific figure in the COLR.
3.2.4 A	3.2.5 A	New	COND: ASI not w/in limit	Unchanged.

Palisades RTS Cross Reference to STS. (03/28/96).

RTS Number	STS Number	TS Number	RTS (STS) requirement Description	Explanation of Differences
3.2.4 A.1	3.2.5 A.1	New	ACTN: Restore ASI; 2 hrs	Unchanged.
3.2.4 B	3.2.5 B	New	COND: Required action not met	Changed to reflect different applicability.
3.2.4.1	3.2.5.1	New	SR: Verify ASI each 12 hrs	Unchanged
3.2.5	Added	4.18 & 4.19	LCO: Three ASI channels and deviation alarm operable	Added LCO and subsequent conditions required actions and surveillances to assure accurate LHR monitoring.

ATTACHMENT 2

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

3.2 POWER DISTRIBUTION LIMITS

Bases for the Revised Technical Specifications

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Linear Heat Rate (LHR)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the primary coolant in the event of a Loss Of Coolant Accident (LOCA), loss of flow accident, ejected control rod accident, or other postulated accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using control rods to alter the axial power distribution;

b. Decreasing control rod insertion by boration, thereby improving the radial power distribution; and

c. Correcting off optimum conditions (e.g., a control rod drop or misoperation of the plant) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., control rod insertion and alignment limits), the power distribution satisfies this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on Linear Heat Rate (LHR) and Departure from Nucleate Boiling Ratio (DNBR).

The limits on LHR, Assembly Radial Peaking F_r^T, Total Radial Peaking F_r^A, T_q, and ASI represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis.

BACKGROUND (continued)

Either of the two core power distribution monitoring systems, the Excore Monitoring System or the Incore Monitoring System, provides adequate monitoring of the core power distribution and is capable of verifying that the LHR is within its limits. The Excore Monitoring System performs this function by continuously monitoring ASI with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the ASI is maintained within the allowable limits specified in the COLR.

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

a. The control rod insertion limits of LCO 3.1.6, "Shutdown and Part Length Rod Insertion Limits," and LCO 3.1.7, "Regulating Rod Insertion Limits," are satisfied;

b. The T_q restrictions of LCO 3.2.3 are satisfied; and

c. Radial Peaking factors F_r^A and F_r^T are within limits of LCO 3.2.2

The Incore Monitoring System continuously provides a direct measure of the peaking factors and 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. Measurement uncertainty (See COLR Table 2.3);

b. Engineering uncertainty (3%);

c. THERMAL POWER uncertainty (2%).

The measurement uncertainties associated with LHR, F_r^A and F_r^T are based on a statistical analysis performed on Palisades Incore Detector Algorithm (PIDAL) power distribution benchmarking results. These values are included in the approved NRC submittal for Palisades use of PIDAL methodology. Table 2.3 of the COLR illustrates 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.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) and AOOs (Condition 2) (Ref. 3, GDC 10). The power distribution and control rod insertion and alignment LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 4);

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 3, GDC 10).

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 200 cal/gm (Ref.5≤); and

d. The 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, GDC 26).

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 ensure that these criteria are met as long as the core is operated within the ASI, F_r^A, and F_r^T limits specified in the COLR. 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.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Fuel cladding damage does not 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.

LCO

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

APPLICABILITY

In MODE 1, with THERMAL POWER ≥ 50% RTP, the power distribution must be maintained within the limits assumed in the accident analysis to ensure that fuel damage does not result following an AOO. Below 50% RTP and all other MODES, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on the core power distribution.

ACTIONS

A.1

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.

ACTIONS (continued)

B.1.1 - B 1.3

If the Incore Monitoring System is inoperable, verification must be performed to ensure that the excore monitoring system is operable for LHR monitoring. The THERMAL POWER must also be restricted to the excore Allowable Power Level (APL), within 2 hours. This ensures the analysis inputs to the safety analysis are not violated.

Verifying the following parameters:

$T_q \leq 3\%$, THERMAL POWER $\leq$ APL and ASI within 5% of target AO further ensures that the excore monitoring system will give accurate results for LHR monitoring. Operating within these limits bounds inputs used in the safety analysis and ensures that the LHR is within the limits used in the safety analysis. A four hour time interval is adequate to ensure the LHR is always within limits.

B 2.1 and 2.2

If the incore alarm system is inoperable and the excore monitoring system is not being used to monitor LHR, operation at less than or equal to 85% RTP may continue, provided LHR is verified through manual incore readings. Operation at $\leq$ 85% RTP ensures that ample thermal margin is maintained. These readings shall be manually collected at the terminal blocks in the control room utilizing a suitable signal detector. 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) within 4 hours and at least every 2 hours thereafter. The time interval of 2 hours and the minimum of 10 detectors per quadrant are sufficient to maintain adequate surveillance of the core power distribution to detect significant changes until the monitoring systems are returned to service.

C.1

If the required action and associated completion time is not met the plant must be < 50% RTP within 2 hours. At this power level ample THERMAL MARGIN exists to ensure that the fuel integrity is not jeopardized and safety analysis assumptions used are still valid. A 2 hour time period is ample to place the plant in conservative state < 50% RTP.

SURVEILLANCE REQUIREMENTS

A Note was added to the SRs to require LHR to be determined by either the Excore Monitoring System or the Incore Monitoring System.

SR 3.2.1.1

This SR is only applicable while the excore system is being used for LHR monitoring. The verification of measured ASI being within 0.05 of the target ASI value for at least 3 of 4, 2 of 3, and 2 of 2 operable channels ensures that ASI is being monitored adequately. ASI remaining within stated limits ensures that radial peaking factors and LHR limits are not exceeded. Operating within these limits allows the power distribution to be monitored adequately and with a high level of confidence. ASI monitored by the stated operable number of channels ensures that the minimum number of operable channels are being used to determine accurate ASI readings. Two channels is the minimum number of operable channels permitted to achieve accurate ASI values. The frequency of 15 minutes is adequate to safely monitor ASI while using the excore system. This time period is short enough to catch any significant changes in ASI. Changes in operating conditions that could cause a flux redistribution, i.e., boration/dilution, rod motion, and Xenon redistribution are either delayed response or flagged by other alarming functions within the control room.

SR 3.2.1.2

This SR ensures that the incore monitoring system is available for LHR monitoring. The incore system compares incoming incore data to an alarm data-base to determine if LHR is within alarm setpoints. This data is also fed to the PIDAL program that calculates T_q, AO, F_r^A, F_r^T, and LHR. Both of these programs run within the plant process computer environment. In the event that the incore system has failed a message would be flagged on the control room PC terminal that says, "Host Communication lost to Reach-Mon." It is more than adequate for an operator to verify that this message does not exist on the PC terminal every 12 hours.

SR 3.2.1.3

During normal plant operations, the incore alarm system supplies continuous monitoring of LHR. As data flows in from the incore detectors LHR is calculated and compared to the alarm setpoints continuously and automatically. If the excore detectors or the incore manual readings are used to monitor LHR then the time period allotted to perform LHR verification is specified by each particular condition. Monitoring LHR ensures that the assumptions made in the Safety Analysis are maintained.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.4

The Incore Monitoring System provides a detailed picture of core operating parameters. The Palisades Incore Detector Algorithm (PIDAL) provides LHR, ASI, T_q, F_r^A and F_r^T information for the entire core. The PIDAL program is used to generate alarm setpoints that are correct for fuel depletion. The frequency of 31 days is consistent with the historical testing frequency of the reactor monitoring system. The intent of updating Incore alarm setpoints is to compensate for fuel depletion effects on the core power distribution. This time period is adequate to capture the long term effects of fuel depletion. After each refueling, the incore alarm setpoints should be updated prior to reaching 50% RTP. These values would be adjusted once again once the excore monitoring system is calibrated with the incore alarm system at RTP conditions.

SR 3.2.1.5

This surveillance requirements ensures that the excore monitoring system is operable for use in LHR monitoring. This surveillance requirement can be satisfied by verifying values obtained from a current SR 3.2.1.1 performance. This Surveillance requirement is only applicable when the excore monitoring system is used for LHR Monitoring. The 31 day frequency ensures that these parameters are consistent for use of the excore monitoring system for LHR monitoring.

SR 3.2.1.6

This surveillance ensures that the incore alarm channels are functioning as designed. The channel calibration ensures that the setpoint value will initiate an incore alarm. The 18 month frequency is consistent with refueling outage scheduling. This is an adequate time period to perform SR 3.1.1.3 with a high level of confidence.

REFERENCES

1. FSAR, Chapter 14
2. FSAR, Chapter 6
3. 10 CFR 50, Appendix A
4. 10 CFR 50.46
5. FSAR, Chapter 14, Section 14

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Radial Peaking Factors F_r^A and F_r^T

BASES

BACKGROUND	A detailed BACKGROUND description for LCO 3.2.2, "Radial Peaking Factors F_r^A and F_r^T" is included in the description of LCO 3.2.1, "Linear Heat Rates (LHR)."
APPLICABLE SAFETY ANALYSES	A detailed APPLICABLE SAFETY ANALYSES description for LCO 3.2.2, "Radial Peaking Factors F_r^A and F_r^T is included in the description of LCO 3.2.1, "Linear Heat Rates (LHR)."
LCO	A detailed LCO description for LCO 3.2.2, "Radial Peaking Factors F_r^A and F_r^T" is included in the description of LCO 3.2.1, "Linear Heat Rates (LHR)."
APPLICABILITY	In MODE 1 $\geq$ 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 other MODES, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on the core power distribution.

ACTIONS

A.1

In the event F_r^A and/or F_r^T are not within specified limits, THERMAL POWER must be reduced to ensure the core is operating within the assumptions made in the safety analysis. The acceptable power level is specified by equation #1 below:

EQ #1

$$P = [1-3.33 \left(\frac{Fr}{F_L} - 1\right) (RTP)]$$

Where;

F_r = Measured valve of either F_r^A or F_r^T

F_L = Corresponding limit stated in the COLR

BASES

ACTIONS (continued)

Operating at or below this power level ensures that the assumptions made in the safety analysis are preserved. The completion time of 6 hours is an adequate period of time to perform power level adjustments.

B.1

If F_r^A and/or F_r^T cannot be returned to within its limit, THERMAL POWER must be reduced. Operation at or below 25% RTP ensures ample THERMAL MARGIN exists to meet applicable fuel design limits. A completion time of 6 hours is sufficient to complete a safe and orderly power reduction.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The periodic Surveillance to determine the calculated F_r^T and F_r^A ensures that radial peaking remains within the range assumed in the safety analysis throughout the fuel cycle. Determining the measured radial peaking factors once after each fuel loading prior to exceeding 50% RTP ensures that the core is properly loaded.

Performance of the Surveillance every 31 days of accumulated operation in MODE 1 ensures that unacceptable changes in the radial power distribution are promptly detected.

The power distribution map can only be obtained after THERMAL POWER exceeds 25% RTP because the incore detectors are not reliable below 25% RTP.

REFERENCES

None

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 Quadrant Power Tilt (T_q)

BASES

BACKGROUND	A detailed BACKGROUND description for LCO 3.2.3, "Quadrant Power Tilt (T_q)" is included in the description of LCO 3.2.1, "Linear Heat Rates (LHR)."
APPLICABLE SAFETY ANALYSES	A detailed APPLICABLE SAFETY ANALYSES description for LCO 3.2.3, "Quadrant Power Tilt (T_q)" is included in the description of LCO 3.2.1, "Linear Heat Rates (LHR)."
LCO	A detailed LCO description for LCO 3.2.3, "Quadrant Power Tilt (T_q)" is included in the description of LCO 3.2.1, "Linear Heat Rates (LHR)."
APPLICABILITY	In MODE 1 > 25% RTP, power distribution must be maintained within the limits assumed in accident analysis to ensure that fuel damage does not result following an AOO. In other MODES, this LCO does not apply because THERMAL POWER is not sufficient to require a limit on core power distribution.

ACTIONS

A.1

If the measured quadrant power tilt is > 5% but ≤ 10%, SR 3.2.2.1 (Peaking factor verification) must be performed within 2 hours. Two hours is sufficient time to allow the operator to verify radial peaking is within limits, and significant radial Xenon redistribution cannot occur within this time. This ensures the core is operated within the limits assumed in the safety analysis.

B.1 and B.2

If Quadrant Power Tilt (T_q) exceeds 10% then THERMAL POWER must be reduced to ≤ 50% RTP within four hours. The radial peaking factors must also be verified by performing SR 3.2.2.1 within 4 hours and once per 8 hours thereafter. This requirement ensures that the core is operating within its thermal limits and places the core in a conservative condition. A completion time of 4 hours is a reasonable time to reach 50% RTP and verify radial peaking in an orderly manner without challenging plant systems.

ACTIONS (continued)

C.1

If Quadrant Power Tilt (T_q) is > 15% or required action and associated completion times are not met the plant shall be < 25% RTP within 12 hours. Except as a result of control rod misalignment a T_q > 15% is not expected; if it occurs, continued operation of the reactor may be necessary to discover the cause of the tilt. 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 radial peaking factors used in core power distribution calculation are based on an untitled power distribution.

If T_q is not restored to within limits, the reactor continues to operate with an axial power distribution mismatch. Continued operation in this configuration may induce an axial xenon oscillation that causes increased LHGRs when the xenon redistributes.

If T_q cannot be restored to within limits, reactor power must be reduced to < 25% RTP within 12 hours. This provides conservative protection from increased peaking due to potential xenon redistributions.

The Required Actions are modified by a Note that requires all subsequent actions to be performed once power reduction commences after entering the Condition if T_q is not restored to < 15%. This procedure ensures corrective action is taken before unrestricted power operation resumes. Following THERMAL POWER reduction to ≤ 25% RTP, T_q must be restored to ≤ 3% before THERMAL POWER is increased. This action prevents the operator from increasing THERMAL POWER above the conservative limit when the Condition, T_q outside its limits, has existed but allows the plant to continue operation for diagnostic purposes. The Completion Time of Required Action C.1 is modified with a Note to indicate that the cause of the out of limit condition must be corrected prior to increasing THERMAL POWER.

This Note also indicates that subsequent power operation above 25% RTP may proceed provided that the measured T_q is verified ≤ 3% at least once per hour for 12 hours, or until verified at 95% RTP. This ensures that the power distribution is responding as predicted. The Completion Time of 12 hours is a historical value that allows an acceptable exit from the LCO after the T_q value is verified acceptable for 12 hours or until 95% RTP is reached.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

T_q must be calculated at 12 hour intervals. The 12 hour Frequency prevents significant xenon redistribution between Surveillance.

REFERENCES None

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Axial Shape Index (ASI)

BASES

BACKGROUND	A detailed BACKGROUND description for LCO 3.2.4, "Axial Shape Index (AXI)" is included in the description of LCO 3.2.1, "Linear Heat Rates (LHR)."
APPLICABLE SAFETY ANALYSES	A detailed APPLICABLE SAFETY ANALYSES description for LCO 3.2.4, "Axial Shape Index (ASI)" is included in the description of LCO 3.2.1, "Linear Heat Rates (LHR)."
LCO	The limitation on ASI, along with the limitations of LCO 3.3.1, "Reactor Protection System Instrumentation," represents a conservative envelope of operating conditions consistent with the assumptions that have been analytically demonstrated adequate for maintaining an acceptable minimum DNBR throughout all AOOs. Of these, the loss of flow transient is the most limiting. Operation of the core with conditions within the specified limits ensures that an acceptable minimum margin from DNB conditions is maintained in the event of any AOO, including a loss of flow transient.
APPLICABILITY	In MODE 1 with THERMAL POWER > 25% RTP, power distribution must be maintained within the limits assumed in the accident analyses to ensure that fuel damage does not result following an AOO. In other MODES, this LCO does not apply because THERMAL POWER is not sufficient to require a limit on the core power distribution. Below 25% RTP the incore detector accuracy is not reliable.
ACTIONS	<u>A.1</u> Operating the core within ASI limits specified in the COLR and within the limits of LCO 3.3.1 ensures an acceptable margin for DNB and for maintaining local power density in the event of an AOO. Maintaining ASI within limits also ensures that the limits of 10 CFR 50.46 are not exceeded during accidents. The Required Actions to restore ASI must be completed within 2 hours to limit the duration the plant is operated outside the initial conditions assumed in the accident analyses. In addition, this Completion Time is sufficiently short that the xenon distribution in the core cannot change significantly.

BASES

ACTIONS (continued)

B.1

If the ASI cannot be restored to within its specified limits, or ASI cannot be determined because of Excore Monitoring System inoperability, core power must be reduced. Reducing THERMAL POWER to ≤ 25% RTP ensures that the core is operating further from thermal limits and places the core in a conservative condition. Four hours is a reasonable amount of time, based on operating experience, to reduce THERMAL POWER to ≤ 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

Verifying that the ASI is within the specified limits ensures that the core is not approaching DNB conditions. A continuous frequency while using the incore system to monitor LHR is adequate for the operator to identify trends in conditions that result in an approach to the ASI limits, because the mechanisms that affect the ASI, such as xenon redistribution or control rod drive mechanism malfunctions, cause the ASI to change slowly and should be discovered before the limits are exceeded.

REFERENCES

None

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 Excore Power Distribution Monitoring

BASES

BACKGROUND

The excore power distribution monitoring system consists of Power Range Detector 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 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 upper and lower sections have a total active length of 12 feet. The DC current signal from each of the ion chambers is fed directly to the control room drawer assembly without preamplification. Each excore detector supplies data to a Thermal Margin Monitoring System (TMM) unit. Each TMM uses these excore signals to calculate Axial Shape Index (ASI) on a continuous basis.

ASI can be defined as the compensated ratio of power developed in the upper and lower sections of the core. The TMM takes the excore detector signals and develops a power ratio (YE) that describes the distribution of neutron flux developed in the core by the formula:

$$YE = (L - U)/(L + U)$$

Where;

L - Lower Excore segment flux

U = Upper Excore segment flux

The Excore detectors which are located within the concrete biological shield of the reactor must be compensated for the phenomenon of shape annealing. Shape Annealing factors are developed to correct the excore readings for neutron attenuation from the core periphery to the excore detector locations. This accounts for any material that would cause neutron attenuation within the detector path such as: concrete, structural steel and so forth. This allows the excore detectors to represent an accurate measurement of the core power distribution. Shape annealing has been found to be a linear relationship which can be correlated to the Axial Offset (AO) as determined by an Incore Detector System to the raw readings seen by the excore detectors.

BASES

BACKGROUND (continued)

Reactor Engineering has developed shape annealing factors for each individual Excore detector. The TMM uses the above calculated power ratio and the appropriate shape annealing factor to determine the ASI value for an individual excore detector channel. The equation below is a typical representation of the formula used to calculate ASI for a particular excore detector channel.

$$YI = ASI = 2.3YE$$

APPLICABLE SAFETY ANALYSES

The purpose of LCO 3.2.5, "Excore Power Distribution Monitoring" ensures that fuel design conditions and safety analysis assumptions are maintained. This is performed by ensuring ASI will be maintained within limits by an alarming function supplied by the excore detectors.

The applicable safety analysis entails the ability ensure that when ASI is deviating outside of setpoints the operator will be signaled to correct the ASI by adjusting the axial power distribution. The main reason for ASI is to identify the flux shape which would cause DNB for variations in power level.

The LCOs governing LHR, ASI and the Primary Coolant System ensure that the core is operated within assumptions made in the safety analysis. These criterion are met when the core is operated within ASI, F_r^A, and F_r^T limits specified in the COLR. These are process variables that characterize the reactor core three dimensional power distributions. Fuel cladding damage does not occur while the plant is operating at conditions outside the limits of these LCOs during normal operation and Abnormal Operational Occurrences (AOOs).

LCO

This LCO ensures that ASI calculated by the Thermal Margin Monitor (TMM) is operable with detection input signals from at least two channels. ASI being calculated from two input signals ensures that adequate ASI monitoring is occurring. Two channels of detection is an accurate representation of core conditions since in the absence on any significant flux tilt one channel could give an accurate picture of ASI for the given core conditions. The presence of two input signals from two separate core quadrants will see communication of significant flux tilts occurring in the other two undetected quadrants and the excore deviation alarm will also flag any flux tilts that may occur by comparing individual values to a calculated mean value. This allows confidence that the core ASI is within assumptions made in the safety analysis. This ensures a flux tilt will be seen and the ASI values calculated are accurate.

BASES

APPLICABILITY This LCO is only valid when the excore detectors are being used for LHR or Quadrant Power Tilt (T_q) monitoring. While using the encores for LHR or T_q monitoring the applicability range for this LCO is MODE 1 with RTP $\geq$ 25%. Below 25% RTP ample thermal margin exists to ensure LHR and T_q are within limits therefore, below 25% RTP the LCO does not apply.

ACTIONS

A.1

In the event the excore detector deviation alarm is inoperable T_q must be calculated every 12 hours. The excore deviation alarm channel consistently looks at the deviation between the four detectors from the average which signals a potential flux tilt while this alarm channel is operating. This ensures that quadrant tilt is with limits while continuous monitoring of T_q may not be occurring.

B.1 and B.2

In the event the T_q measured from the excores is deviating from the T_q measured from the incores by $\geq$ 2% then T_q must be calculated using the Incore detectors within 12 hours. Under these conditions, Surveillance Requirements 3.2.5.2. "Excore ASI Calibration" must also be performed within 7 days. The incore monitoring system is generally more reliable therefore, when an absolute deviation of 2% occurs the incore system shall be used to verify T_q. The specified Completion Times are adequate to perform the stated actions in a safe and efficient manner.

C.1

When the condition arises that 3 or 4 ASI alarm channels are inoperable, LHR monitoring with the incore alarm system should be initiated within 1 hour. This action ensures that ASI is being monitored accurately in the event the excore system can no longer be used. This ensures that assumptions made in the safety analysis are being maintained within acceptable limits. The Completion Time of 1 hour is sufficient to safely complete this action and reflects the importance of monitoring ASI accurately at all times.

BASES

ACTIONS (continued)

D.1

In the event that ASI is deviating from AO by > 0.02 under steady state operating conditions. Surveillance Requirement 3.2.5.2 should be performed within 12 hours. This allows for the recalibration of AO and T_q between the excore and incore monitoring system. This action ensures that the incore core averaged Axial Offset (AO) and the individual excore ASI are within acceptable limits. This yields further confidence that ASI is being monitored accurately and still supports the safety analysis. The Completion Time of 12 hours is adequate to safely complete this required action.

D.2.1 and D.2.2

When the excore ASI is deviating from incore AO by > 0.02 under steady state conditions, the ASI alarm setpoint and TM/LP trip function must be compensated for this deviation within 12 hours. This ensures that fuel design parameters can accurately be monitored and not exceeded, when the incore/excore alignment is not within normal tolerances. This may occur when the calibration cannot be performed or the alignment problem exists after the calibration. The Completion Time of 12 hours is sufficient to perform these adjustments, accurately and efficiently.

D.3.1 and D.3.2

In the event that excore ASI is deviating from incore AO > 0.02 under steady state conditions the ASI alarm channel and TM/LP RPS trip units must be declared inoperable within 12 hours. This ensures that operators are not relying on a potentially inaccurate excore monitoring system. By placing these items inoperable it forces the operators to take appropriate action to correct the problem or place the plant in a conservative condition. 12 hours is a sufficient time period to expedite these actions.

E.1

If the Required Action and associated Completion Times are not met the plant must be < 25% RTP within 6 hours. This action ensures that the plant is not operated in an applicability range that relies on accurate power monitoring from the excore system. Below 25% RTP ample thermal margin exists to ensure the plant conditions meet the safety analysis.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1

This surveillance calls for the verification of measured individual excore channel AO when compared to the total core AO measured by the incores in ≤ 0.02 on a 31 day frequency. This SR is performed in conjunction with incore alarm updates on a monthly basis. This ensures that the excore system is functional for power monitoring if needed during any detector drifting or long term abnormalities that would affect the functionality of the excore system for power monitoring.

SR 3.2.5.2

The calibration of T_q and AO from excore detectors with T_q and AO measured from the incores is performed at least every 184 days (6 months) or when warranted by required actions. This calibration ensures that incore averaged axial offset corresponds to excore ASI within 2% for each individual excore detector. It also ensures that the incore/excore calculated T_q are within alignment tolerances of 2%. The SR supports a high degree of confidence that the excores can accuretly be used for power monitoring. The frequency of 184 days ensures that system is recalibrated regardless whether the 2% limit has been reached within 184 days.

SR 3.2.5.3

This surveillance requires the performance of a CHANNEL CALIBRATION of the Excore Detector Deviation Alarm Channel. The excore detector deviation alarm channel compares the reported flux from each individual excore to the computed average and determines if all detectors are within a 2% deviation. This supports flux tilt detection to ensure valid ASI values are being reported. This is essential especially when only two ASI alarm channels are operable to ensure that significant flux tilts are detected that would corrupt the ASI values generated from the functional two excore channels. A frequency of 18 months is sufficient to ensure that the excore deviation alarm is adequately calibrated for flux tilt detection.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.5.4

This surveillance requires that a CHANNEL CALIBRATION of the excore ASI circuitry is performed each 18 months. This SR ensures that the appropriate ASI value is generated by the TMM for a particular excore reading and that the ASI alarming circuit would actuate when the calculated ASI value falls outside of specified upper and lower setpoints. This instills a high level of confidence that ASI is calculated accuretly and out of tolerance condition would be annunciated. A frequency of 18 months is adequate to ensure a high degree of system performance during plant operations.

REFERENCES None

ATTACHMENT 3

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

3.2 POWER DISTRIBUTION LIMITS

Comparison of Existing and Revised Technical Specifications

Palisades Tech Spec Requirement List. Corrected through Amendment 170

A list of the existing Palisades Tech Specs (TS) correlated to Palisades Revised Technical Specifications (RTS).

First Column; Existing Palisades Tech Spec (TS) number

Each numbered TS item is listed in the left-most column. Items which contain more than one requirement are listed once for each requirement.

Second Column; Palisades Revised Tech Spec (RTS) number

The nearest corresponding numbered RTS item is listed in the second column. If the item does not appear in RTS, it is noted as 'Deleted' or 'Relocated.'

Deleted is used where an item has been eliminated as a tech spec, ie deleting, iaw GL 84-15, the requirement to test a D.G. when an ECCS pump in the opposite train becomes inoperable.

Relocated is used where an item has been moved to a controlled program or document because it does not meet the "Criteria" of 10 CFR 50.36(2)(c)(ii).

Where an item is relocated or deleted, the number of the associated RTS section has been added to allow sorting the list by section number. Relocated items, such as heavy load restrictions, which are not associated with any particular RTS section are arbitrarily assigned the number 5.0.

Third Column; TS Item Description

An abbreviation of the TS requirement appears in the third column. Each item is identified as: LCO, ACTION, SR, Admin, Exception, etc. Some items are implied, rather than explicit, ie a LCO is implied when an ACTION exists without a stated LCO.

Description Key; TS requirement type:	Column 3 syntax:
Safety Limit	SL: Safety limit; Applicable conditions
Surveillance Requirement	SR: Equipment to be tested; Test description; Frequency
Limiting Safety Setting	LSS: RPS Trip Channel & required setting
Limiting Condition for Operation	LCO: Equipment to be operable; Applicable conditions
Action	ACTN: Condition requiring action; Required action; Completion time
Administrative Requirement	ADMN: Administrative requirement
Permitted Instrument Bypass	Byps: Bypassable component; conditions when bypass permitted
Defined Term	DEF: Name of defined item
Exception to other Requirement	XCPT: Excepted spec or condition; Applicable conditions
Descriptive material	DESC: Subject matter
Table	TBL: Table

Forth Column; Classification of Changes:

Each change is identified as ADMINISTRATIVE, RELOCATED, MORE RESTRICTIVE, or LESS RESTRICTIVE.

Fifth Column; Discussion of Changes:

Each change is discussed briefly.

TS Number	RTS Number	TS requirement description	Classification and Description of Changes	
3.1.1.e	3.2.4	LCO: ASI maintained iaw COLR;	ADMINISTRATIVE:	Requirement unchanged.
3.1.1.e.(1)a	3.2.4 A.1	ACTN: ASI not w/in limit; initiate action; 15 min	ADMINISTRATIVE:	RTS does not specify action initiation time, but completion time remains unchanged.
3.1.1.e.(1)b	3.2.4 A.1	ACTN: ASI not w/in limit; Restore w/in 1 hr	ADMINISTRATIVE:	Requirement unchanged.
3.1.1.e.(1)c	3.2.4 B.1	ACTN: ASI >limit >1 hr; Be <70%; 2 hrs	MORE RESTRICTIVE:	Action simplified & single completion time stipulated. RTS requirement is more conservative than TS requirement, and reflects actual plant operating practice.
3.11	3.2.1	Power Distribution Instrumentation		
3.11.1	3.2 Relocated	Incore Detectors	RELOCATED:	The incore detectors do not meet the criterion of 10 CFR 50.36, and the associated requirements have been relocated to plant procedures. The Incore detectors are used for monitoring linear heat rate, so certain incore related requirements are retained.
3.11.1.a	3.2 Relocated	LCO: Min incores; Meas Quad pwr tilt (Tq)	RELOCATED:	
3.11.1.a	3.2 Relocated	LCO: Min incores; Meas Radial peaking (Fr)	RELOCATED:	
3.11.1.a	3.2 Relocated	LCO: Min incores; Meas LHR	RELOCATED:	
3.11.1.a	3.2 Relocated	LCO: Min incores; Determining AO	RELOCATED:	
3.11.1.a	3.2 Relocated	LCO: Min incores; Determining APL	RELOCATED:	
3.11.1.b	3.2 Relocated	LCO: Min incores; Monitoring LHR	RELOCATED:	
3.11.1.b	3.2.1	LCO: Incore alarm operable; Monitoring LHR w/alarms	ADMINISTRATIVE:	Requirement unchanged.
3.11.1 A1	3.2 Relocated	ACTN: <min incores; No incore Tq, Fr, LHR, AO, APL	RELOCATED	
3.11.1 A2	3.2.1 B.1.1	ACTN: W/O incore alm; Don't use for monitoring LHR	ADMINISTRATIVE:	Requirement unchanged.
3.11.1 A2	3.2.1 B	ACTN: W/O incore alm; Comply w/3.11.2 or 3.23.1	ADMINISTRATIVE:	Requirement unchanged.
3.11.2	3.2	Excore Power Distribution Monitoring System		
3.11.2.a	3.2.1.3	SR: Incore AO Target & APL determined w/in 31 days; (for monitoring LHR w/excores)	ADMINISTRATIVE:	Requirement unchanged.

TS Number	RTS Number	TS requirement description	Classification and Description of Changes	
3.11.2.a	3.2.4.1 3.2.1 B.1.3	LCO: Measure AO w/in .05 of target for last 24 hrs; (for monitoring LHR w/excores)	ADMINISTRATIVE:	This measurement is inherent in the determination of ASI as required by SR 3.2.4.1 (Verify ASI within limits - Continuously) and in the action when shifting LHR measurement instruments from the incores to the excores as required by 3.2.1 B.1.3.
3.11.2.b	3.2.5.2	LCO: Excore AO cal w/incore; Monitoring LHR w/excores	ADMINISTRATIVE:	Requirement unchanged.
3.11.2.b	3.2.5.2	LCO: Excore AO cal w/incore; Each TM/LP trip chnl	ADMINISTRATIVE:	Requirement unchanged.
3.11.2.b	3.2.5.2	LCO: Excore AO cal w/incore; ASI alm	ADMINISTRATIVE:	Requirement unchanged.
3.11.2.c	3.2.5.2	LCO: Excore Tq cal w/incore; Monitoring LHR w/excores	ADMINISTRATIVE:	Requirement unchanged.
3.11.2.c	3.2.5.2	LCO: Excore Tq cal w/incore; Monitoring Tq w/excores	ADMINISTRATIVE:	Requirement unchanged.
3.11.2 A1	3.2.1 B.1.1	ACTN: Excore monit sys inop; Don't use for LHR	ADMINISTRATIVE:	Requirement unchanged.
3.11.2 A2	3.2.5 B.1	ACTN: Meas Tq not cal w/incores; Do not use for Tq	ADMINISTRATIVE:	Requirement unchanged.
3.11.2 A3	3.2.5 C.2.1	ACTN: Incore/excore AO diff >0.02; Adjust ASI alm; 12 hrs	ADMINISTRATIVE:	Requirement unchanged.
3.11.2 A4	3.2.5 B.1	ACTN: Incore/excore T_q diff >0.02; calc T_q each 12 hrs	ADMINISTRATIVE:	Requirement unchanged.
3.17.1.3.b	3.2.5 D.1/E.1	ACTN: 2 Pwr range instruments inoperable; be 70%; 2 hrs	ADMINISTRATIVE:	This action is replaced by actions which assure the desired protection is retained, or require a power reduction to below 25% RTP. The existing Bases (2nd paragraph, page B 3.17-8) state that the reason for the required power reduction is the loss of the ability to detect flux tilts when only two power range NI channels are available. With the proposed RTS, if 1 (of 3) required ASI monitoring channels (which are fed by the power range NI channels) are inoperable, Action 3.2.5 D.1 directs using the Incore detectors for measurement of LHR. The incores provide the ability to detect flux tilts, and thereby provide the information which the inoperable NIs are unable to provide. If this action cannot be completed, (ie the desired flux tilt detection function is lost) Action 3.2.5 E.1 requires a power reduction to below 25%.
3.17.6.15	3.2.5 A.1	ACTN: Deviation Alm inop; Calculate T_q each 12 hrs	ADMINISTRATIVE:	Requirement unchanged.
3.17.6.16	3.2 Relocated	ACTN: ASI Alm inop; Restore prior to startup	RELOCATED:	This requirement does not meet the criterion of 10 CFR 50.36.

TS Number	RTS Number	TS requirement description	Classification and Description of Changes	
3.17.6T#15	3.2.5	LCO: 1 Excore Deviation Alm; >25% power	ADMINISTRATIVE:	Requirements unchanged.
3.17.6T#16	3.2 Relocated	LCO: 4 chnls ASI Alarm; >25% power	RELOCATED:	This requirement does not meet the criterion of 10 CFR 50.36.
3.23	3.2	Power Distribution Limits		
3.23.1	3.2.1	LCO: LHR <COLR limit; >50% RTP	MORE RESTRICTIVE:	Added Incore Alarm System Operable to ensure it can be used for LHR monitoring.
3.23.1 A1	3.2.1 A.1	ACTN: >4 incore alms; Reduce LHR in 1 hr or be <50%	ADMINISTRATIVE:	Requirement unchanged.
3.23.1 A2	3.2.1 B.1.3	ACTN: AO >Target; stop using excore for LHR	ADMINISTRATIVE:	Requirement unchanged.
3.23.1 A2	3.2.1 B.2.1/.2	ACTN: Alm inop; be <85% w/in 2 hrs & follow A3	ADMINISTRATIVE:	Reworded for clarity.
3.23.1 A3	3.2.1 B.2.2	ACTN: Incore alm inop for LHR; Manual readings req	ADMINISTRATIVE:	Requirement unchanged.
3.23.1 A3	3.2.1 C.1	ACTN: Manual readings >alm; Follow Action A1	ADMINISTRATIVE:	Requirement unchanged.
3.23.1 A1	3.2.1 C	COND: Req action & ass Comp time not met	ADMINISTRATIVE:	Requirement unchanged.
3.23.1 A1	3.2.1 C.1	ACTN: Be <50% RTP; 2 hrs	ADMINISTRATIVE:	Requirement unchanged.
3.23.2	3.2.2	LCO: Radial Peaking w/in COLR; >25% RTP	ADMINISTRATIVE:	Requirement unchanged.
3.23.2 A1	3.2.2 A/B	ACTN: <50% & Fr>limit; HSD w/in 6 hrs	LESS CONSERVATIVE:	The action has been changed to reduce power to below applicability range of 25% RTP. This places the reactor in a conservative state with ample thermal margin.
3.23.2 A2	3.2.2 A.1	ACTN: >50% & Fr>limit; Reduce pwr; 6 hrs	ADMINISTRATIVE:	The action has been changed to reflect the standard. The power restriction stated by the equation in 3.23.2 A2 is relocated to the COLR. Reducing thermal power to bring the combination of thermal power and peaking to within limits is outlined in the COLR and calls upon the equations stated in the COLR or radial peaking and power level.
3.23.3	3.2.3	LCO: Tq <5%; W/>25% RTP	ADMINISTRATIVE:	
3.23.3 A1	3.2.3 A	ACTN: 10% >Tq >5%; do Action a, b, or c	ADMINISTRATIVE:	
3.23.3 A1.a	3.2 Deleted	ACTN: 10% >Tq >5%; fix w/in 2 hrs	ADMINISTRATIVE:	This is omitted from RTS, correction to within limits is always an option.
3.23.3 A1.b	3.2.3 A.1	ACTN: 10% >Tq >5%; check Fr w/in 2 hrs	ADMINISTRATIVE:	Requirement unchanged.

TS Number	RTS Number	TS requirement description	Classification and Description of Changes	
3.23.3 A1.b	3.2.3 A.1	ACTN: 10% >Tq >5%; check Fr in limit; Each 8 hrs	ADMINISTRATIVE:	Requirement unchanged.
3.23.3 A1.c	3.2.3 A.1	ACTN: 10% >Tq >5%; be <85% & check Fr	ADMINISTRATIVE:	Requirement unchanged.
3.23.3 A1.c	3.2.3 A.1	ACTN: 10% >Tq >5%; check Fr in limit; Each 8 hrs	ADMINISTRATIVE:	Requirement unchanged.
3.23.3 A2	3.2.3 B	ACTN: Tq >10%; Do Action a or b	ADMINISTRATIVE:	Requirement unchanged.
3.23.3 A2.a	3.2 Deleted	ACTN: Tq >10%; Fix w/in 2 hrs	ADMINISTRATIVE:	This is excluded from the standard, correction to within limits is always an option.
3.23.3 A2.b	3.2.3 B.1	ACTN: Tq >10%; Be <50% in 2 hrs	ADMINISTRATIVE:	Requirement unchanged.
3.23.3 A2.b	3.2.3 B.2	ACTN: Tq >10%; check Fr in limit; Each 8 hrs	ADMINISTRATIVE:	Requirement unchanged.
3.23.3 A3	3.2.3 C.1	ACTN: Tq >15%; SD w/in 12 hrs	LESS RESTRICTIVE:	Changed action to be <25% RTP within 12 hrs. This places the plant in a conservative condition and takes the plant out of the applicability range.
4.17.6T#15-cft	3.2.5.2	SR: 1 Excore Deviation Alm; Chnl func test; 18 mo	ADMINISTRATIVE:	Requirement unchanged. CFT is part of Channel Calibration.
4.17.6T#15-cal	3.2.5.2	SR: 1 Excore Deviation Alm; Chnl cal; 18 mo	ADMINISTRATIVE:	Requirement unchanged.
4.17.6T#16-cft	3.2 Relocated	SR: 4 chnls ASI Alarm; Chnl func test; 18 mo	RELOCATED:	This requirement does not meet the criterion of 10 CFR 50.36.
4.17.6T#16-cal	3.2 Relocated	SR: 4 chnls ASI Alarm; Chnl cal; 18 mo	RELOCATED:	This requirement does not meet the criterion of 10 CFR 50.36.
4.18.1	3.2 Relocated	Incore Detection System		
4.18.1.1.a	3.2 Relocated	SR: Incore Detection Sys; chnl check; 7 days	RELOCATED:	The incore detectors do not meet the criterion of 10 CFR 50.36, and the associated requirements have been relocated to plant procedures. The Incore detectors are used for monitoring linear heat rate, so certain incore related requirements are retained.
4.18.1.1.b	3.2.1.4	SR: Incore Detection Sys; chnl cal; Refueling	ADMINISTRATIVE:	Requirement unchanged.
4.18.1.2	3.2.1.4	SR: Datalogger Seq error alm; chnl check; Refueling	ADMINISTRATIVE:	Requirement unchanged.

TS Number	RTS Number	TS requirement description	Classification and Description of Changes	
4.18.2	3.2.1	Excore Monitoring System		
4.18.2.1.a	3.2.1.3	SR: Target AO; determine using excores; 31 days	ADMINISTRATIVE:	Requirement unchanged.
4.18.2.1.a	3.2.1.3	SR: Target AO; determine using incores; 31 days	ADMINISTRATIVE:	Requirement unchanged.
4.18.2.1.a	3.2.1.3	SR: APL; determine using excores; 31 days	ADMINISTRATIVE:	Requirement unchanged.
4.18.2.1.a	3.2.1.3	SR: APL; determine using incores; 31 days	ADMINISTRATIVE:	Requirement unchanged.
4.18.2.1.b	3.2.1.3	SR: AO; compare excore w incore; 31 days	ADMINISTRATIVE:	Requirement unchanged.
4.18.2.1.b	3.2.5 D	ACTN: Excore/incore AO diff >2%; Cal excore sys	ADMINISTRATIVE:	Requirement unchanged.
4.18.2.1.c	3.2.5.2	SR: Tq; compare excore w incore; 31 days	ADMINISTRATIVE:	Requirement unchanged.
4.18.2.1.c	3.2.5 b.2	ACTN: Excore/incore Tq diff >2%; cal excore sys	ADMINISTRATIVE:	Requirement unchanged.
4.19.1	3.2.1	Linear Heat Rates		
4.19.1.1	3.2.1.2	SR: Incore alm sys; Set before 50% w excore LHR	ADMINISTRATIVE:	Requirement unchanged.
4.19.1.1	3.2.1.2	SR: Incore alm sys; set; 7 days w using for LHR		Frequency changed to each 31 days in MODE 1.
4.19.1.2.a	3.2.4.1 3.2.1 B.1.3	SR: Excore AO; last day AO OK; B4 using excore LHR	ADMINISTRATIVE:	This measurement is inherent in the determination of ASI as required by SR 3.2.4.1 (Verify ASI within limits - Continuously) and in the action when shifting LHR measurement instruments from the incores to the excores as required by 3.2.1 B.1.3.
4.19.1.2.b	3.2.1 B.1.3	LCO: Excore Tq <3%; W/using excores LHR	ADMINISTRATIVE:	Requirement unchanged.
4.19.1.2.b	3.2.1 B.1.3	SR: Excore Tq; check in limit; 1 day; W/excore LHR	ADMINISTRATIVE:	Requirement unchanged.
4.19.1.2.c	3.2.1 B.1.2	LCO: Thermal Power <APL w using excore LHR	ADMINISTRATIVE:	Requirement unchanged.
4.19.1.2.c	3.2.1 B.1.3	LCO: Thermal Power <10% above pwr w APL	RELOCATED:	This requirement is included within the calculation methodology for determining the Allowable Power Limit. Therefore, it is relocated to the associated engineering procedure.
4.19.1.2.c	3.2.1 B.1.3	SR: Thermal Power; verify limits; 1 hr	LESS RESTRICTIVE:	APL is verified once per 4 hrs and previously while in this condition APL was verified each hour. The slightly relaxed SR period still allows ample time to detect THERMAL POWER that would be greater than APL. Ample

TS Number	RTS Number	TS requirement description	Classification and Description of Changes	
				margin exists for LHR to ensure 4 hrs is adequate to monitor this parameter.
4.19.1.2.d	3.2.1 B.1.3	LCO: AO <5% from target w using excore LHR	ADMINISTRATIVE:	Requirement unchanged.
4.19.1.2.d	3.2.4.1	SR: AO; verify w/in limit; Continuously	LESS RESTRICTIVE:	Frequency reduced from "continuously" to 15 minutes. This is adequate time to safely monitor ASI since any effects of Xenon redistribution due to rod insertion, boron changes, etc have other immediate indications which would flag the possibility of an ASI change.
4.19.2	3.2.2	Radial Peaking Factors		
4.19.2.1.a	3.2.2.1	SR: Assembly Fr; verify; Refueling (B4 50%)	ADMINISTRATIVE:	LCO 3.23.2 is applicable >25%; SRs must be current prior to entering applicability.
4.19.2.1.a	3.2.2.1	SR: Int Rod Fr; verify; Refueling (B4 50%)	ADMINISTRATIVE:	LCO 3.23.2 is applicable >25%; SRs must be current prior to entering applicability.
4.19.2.1.b	3.2.2.1	SR: Assembly Fr; verify w/in limit; Weekly at power	ADMINISTRATIVE:	Requirement unchanged.
4.19.2.1.b	3.2.2.1	SR: Int Rod Fr; verify w/in limit; Weekly at power	ADMINISTRATIVE:	Requirement unchanged.

ATTACHMENT 4

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

3.2 POWER DISTRIBUTION LIMITS

STS Pages Marked to Show the Differences Between RTS and STS

3.2 POWER DISTRIBUTION LIMITS

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

LCO 3.2.1 LHR shall not exceed the limits specified in the COLR and the Incore Alarm System shall be operable.

APPLICABILITY: MODE 1, ≥ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. LHR, as determined by the Incore Detector Monitoring System, exceeds the limits of Figure 3.2.1 1 of the COLR, as indicated by four or more coincident incore channels.~~ ~~OR~~ ~~LHR, as determined by the Excore Detector Monitoring System, exceeds the limits as indicated by the ASI outside the power dependent control limits as specified in Figure 3.2.1 2 of the COLR.~~	~~A.1 Restore LHR to within limits.~~	~~1 hour~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 2.~~	~~6 hours~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LHR, as determined by the Incore Monitoring System not within limits.	A.1 Restore LHR to within limits.	1 hour

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Incore Alarm System Inoperable.	B.1.1 Verify excore system is OPERABLE for LHR monitoring.	2 hours
	AND	
	B.1.2 Restrict THERMAL POWER to the excore Allowable Power Level (APL).	2 hours
	AND	
	B.1.3 Verify the following parameters: $T_q \leq 3\%$ THERMAL POWER $\leq$ APL and ASI within $\pm$ .05 of target.	2 hours AND Once per 4 hours, thereafter
	OR	
	B.2.1 Restrict THERMAL POWER to < 85% RTP.	2 hours
	AND	
	B.2.2 Verify LHR within limits using manual incore readings.	2 hours AND Once per 4 hours, thereafter
C. Required Action and associated Completion Time not met.	C.1 Be < 50% RTP.	2 hours

SURVEILLANCE REQUIREMENTS

----------NOTE----------
Either the Excore ~~Detector~~ Monitoring System or the Incore ~~Detector~~ Monitoring System shall be used to determine LHR.

SURVEILLANCE		FREQUENCY
	----------NOTE---------- This SR is only applicable while using the excore system to monitor LHR. ----------	
SR 3.2.1.1	Verify measured ASI is within 0.05 of target ASI for at least 3 or 4, 2 of 3 or 2 of 2 operable channels.	15 minutes
SR 3.2.1.2	Verify Incore Alarm System is functioning to monitor LHR.	12 hours
SR 3.2.1.~~1~~3	~~NOTE~~ ~~Only applicable when the Excore Detector Monitoring System is being used to determine LHR.~~ ~~Verify ASI alarm setpoints are within the limits specified in Figure 3.2.2 2 (ASI Operating Limits) in the COLR.~~ Verify LHR is within limits.	~~31 days~~ 12 hours
SR 3.2.1.~~2~~4	~~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.~~ ~~Verify incore detector local power density alarms satisfy the requirements of the core power distribution map, which shall be updated at least once per 31 days of accumulated operation in MODE 1.~~ Update incore alarm setpoints.	~~31 days~~ Prior to 50% RTP following refueling AND 31 days thereafter

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.~~35~~	~~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.~~ ~~Verify incore detector local power density alarm setpoints are less than or equal to the limits specified in the COLR.~~ Determine APL, T_q and Target AO using the excore and incore system.	31 days
SR 3.2.1.6	Perform a CHANNEL CALIBRATION on each incore alarm signal.	18 months

~~3.2 POWER DISTRIBUTION LIMITS~~

~~3.2.2 Total Planar Radial Peaking Factor (F^T_{xy}) (Analog)~~

~~LCO 3.2.2 The calculated value of F^T_{xy} shall not exceed the limits specified in the COLR.~~

~~APPLICABILITY: MODE 1.~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. NOTE Required Actions shall be completed if this Condition is entered.~~ ~~F^T_{xy} not within limits.~~	~~A.1 Reduce THERMAL POWER to bring the combination of THERMAL POWER and F^T_{xy} to within the limits specified in the COLR.~~ ~~AND~~ ~~A.2 Withdraw the control element assemblies (CEAs) to or beyond the long term steady state insertion limits of LCO 3.1.7, "Regulating CEAs," as specified in the COLR.~~	~~6 hours~~ ~~6 hours~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 2.~~	~~6 hours~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~		~~FREQUENCY~~
~~SR 3.2.2.1~~	~~NOTE~~ ~~SR 3.2.2.2 and SR 3.2.2.3 shall be completed each time SR 3.2.2.1 is required.~~ ~~F^T_{xy} shall be determined by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the long term steady state insertion limit, as specified in the COLR.~~ ~~Verify the value of F^T_{xy}.~~	~~Once prior to operation above 70% RTP after each fuel loading~~ ~~AND~~ ~~Each 31 days of accumulated operation in MODE 1~~
~~SR 3.2.2.2~~	~~Verify the value of F_{xy}.~~	~~In accordance with the Frequency requirements of SR 3.2.2.1~~
~~SR 3.2.2.3~~	~~Verify the value of T_q.~~	~~In accordance with the Frequency requirements of SR 3.2.2.1~~

3.2 POWER DISTRIBUTION LIMITS

3.2.~~3 Total Integrated Radial Peaking Factor (F_r^T) (Analog)~~2 Radial Peaking Factors F_r^T and F_r^A

LCO 3.2.~~3~~2 The ~~calculated~~ value of ~~F_r^T~~ F_r^A and F_r^T shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1, ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ~~NOTE Required Actions shall be completed if this Condition is entered.~~ F_r^A or F_r^T not within limit.	A.1 Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T or F_r^A to within limits specified in the COLR.	6 hours
	~~AND~~	
	~~A.2 Withdraw the control element assemblies (CEAs) to or beyond the long term steady state insertion limits of LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits," as specified in the COLR.~~	~~6 hours~~
	~~AND~~	
	~~A.3 Establish a revised upper THERMAL POWER limit as specified in the COLR.~~	~~6 hours~~
B. Required Actions and associated Completion Times not met.	B.1 Be ~~in MODE 2.~~ < 25% RTP.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.~~3~~2.1	~~NOTE~~ ~~SR 3.2.3.2 and SR 3.2.3.3 shall be completed each time SR 3.2.3.1 is required. F_r^T shall be determined by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the long term steady state insertion limit.~~	
	Verify the value of F_r^T, F_r^A within limits.	Prior to operation > ~~70~~50% RTP after each ~~fuel loading~~ refueling AND Each 31 days of accumulated operation in MODE 1
~~SR 3.2.3.2~~	~~Verify the value of F_r.~~	~~In accordance with the Frequency requirements of SR 3.2.3.1~~

~~(continued)~~

~~SR 3.2.3.3~~	~~Verify the value of T_q.~~	~~In accordance with the Frequency requirements of SR 3.2.3.1~~

3.2 POWER DISTRIBUTION LIMITS

3.2.~~4~~3 ~~AZIMUTHAL POWER TILT (T_q) (Analog)~~ Quadrant Power Tilt

LCO 3.2.~~4~~3 T_q shall be ≤ ~~[0.03]~~ 5%

APPLICABILITY: MODE 1, ~~with THERMAL POWER >~~ ≥ ~~50~~25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. Indicated T_q > [0.03] and ≤ 0.10.~~ A. T_q > 5% but ≤ 10%.	~~A.1 Restore T_q to ≤ [0.03].~~ ~~OR~~ ~~A.2 Verify F^T_{xy} and F^T_z are within the limits of LCO 3.2.2, "Total Planar Radial Peaking Factor (F^T_{xy})," and LCO 3.2.3, "Total Integrated Radial Peaking Factor (F^T_z)," respectively.~~ A.1 Perform SR 3.2.2.1 (Peaking factor verification).	2 hours AND Once per 8 hours thereafter
B. ~~Required Action and associated Completion Time of Condition A not met.~~ T_q > 10%	B.1 Reduce THERMAL POWER to ≤ 50% RTP. AND B.2 Perform SR 3.2.2.1 (Peaking factor verification).	4 hours 4 hours AND Once per 8 hours thereafter

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ~~Indicated~~ T_q > ~~0.10~~15%. OR Required Action and associated completion time not met.	----------NOTE---------- All subsequent Required Actions stated in Condition B must be completed if power reduction commences prior to restoring T_q ≤ ~~10~~ 15%. ---------- C.1 ~~Verify F^I_{xy} and F^I_z are within the limits of LCO 3.2.2 and LCO 3.2.3, respectively.~~ ~~AND~~ ~~C.2 Reduce THERMAL POWER to < 50% RTP.~~ ~~AND~~ ~~C.3 Restore T_q to ≤ [0.03].~~ Be < 25% RTP	~~1 hour~~ ~~2 hours~~ ~~Prior to increasing THERMAL POWER~~ 12 hours -------NOTE----- Correct the cause of the out of limit condition prior to increasing THERMAL POWER. Subsequent power operation above 50% RTP may proceed provided that the measured T_q is verified ≤ ~~[0.03]~~ 3% at least once per hour for 12 hours~~, or until verified at 95% RTP~~. ----------

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.43.1 Verify T_q is ~~within~~ below limits.	12 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.~~5~~4 Axial Shape Index (ASI) ~~(Analog)~~

LCO 3.2.~~5~~4 The ASI shall be maintained within the limits specified in ~~Figure 3.2.5-1 of~~ the COLR.

APPLICABILITY: MODE 1 ~~with THERMAL POWER > 20% RTP~~, ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ASI not within limits.	A.1 Restore ASI to within limits.	~~2~~1 hour~~s~~
B. Required Action and associated Completion Time not met.	B.1 ~~Reduce THERMAL POWER to < 20% RTP.~~ Be < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.~~5~~4.1 Verify ASI is within limits ~~specified in the COLR~~.	~~12 hours~~ Continuously, while using the incore alarm system to monitor LHR.

3.2 POWER DISTRIBUTION LIMITS

3.2.5 Ex-Core Power Distribution Monitoring

LCO 3.2.5 Three Axial Shape Index (ASI) Monitoring Channels and the Excore Deviation Alarm Channel shall be OPERABLE.

APPLICABILITY: MODE 1 ≥ 25% RTP, when Excore Power Range Channels are used to monitor Linear Heat Rate (LHR) or Quadrant Power Tilt (T_q).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Excore Detector Deviation Alarm Channel inoperable.	A.1 Calculate T_q.	Once per 12 hours
B. Excore T_q deviating from Incore T_q by ≥ 2%.	B.1 Calculate T_q using Incore detectors AND B.2 Perform SR 3.2.5.2 (Excore ASI cal)	Once per 12 hours 7 days

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ASI deviating from AO by > 0.02 under steady state operating conditions.	C.1 Perform SR 3.2.5.1 (Incore/Excore calibration)	12 hours
	OR	
	C.2.1 Adjust the ASI alarm setpoint to compensate for the deviation.	12 hours
	AND	
	C.2.2 Adjust the TM/LP trip function to compensate for the deviation.	12 hours
	OR	
	C.3.1 Declare ASI monitoring channel inoperable.	12 hours
	AND	
	C.3.2 Declare affected TM/LP RPS Trip Units inoperable.	
D. 1 required ASI monitoring channel inoperable.	D.1 Initiate LHR monitoring with the Incore Alarm System.	1 hour
E. Required action associated Completion Time not met.	E.1 Be < 25% RTP	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.5.1	Verify individual excore channel measured AO when compared to total core AO measured by the incores is ≤ 0.02.	31 days
SR 3.2.5.2	Calibrate T_q and AO from Excore with T_q and AO measured from Incores for each channel of TM/LP trip and the ASI alarm.	184 days
SR 3.2.5.3	Perform a CHANNEL CALIBRATION of the Excore Detector Deviation Alarm Channel.	18 months
SR 3.2.5.4	Perform a CHANNEL CALIBRATION of the Excore ASI circuitry.	18 months

ENCLOSURE 1

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

TECHNICAL SPECIFICATION CHANGE REQUEST

PART 2 - SECTION 2.0

March 27, 1996

CONSUMERS POWER COMPANY
Docket 50-255
Request for Change to the Technical Specifications
License DPR-20

2.0 SAFETY LIMIT CHANGE REQUEST

It is requested that the Safety Limit requirements of the Technical Specifications contained in the Facility Operating License DPR-20, Docket 50-255, issued to Consumers Power Company on February 21, 1991, for the Palisades Plant be changed as described below:

I. ARRANGEMENT AND CONTENT OF THIS PART OF THE CHANGE REQUEST:

This section of the Technical Specification Change Request (TSCR) proposes changes to those Palisades Technical Specification requirements addressing the reactor core safety limits and the PCS pressure SLs. These changes are intended to result in requirements which are appropriate for the Palisades plant, but closely emulate those of the Standard Technical Specifications, Combustion Engineering Plants, NUREG 1432, Revision 1.

This discussion and its supporting information frequently refer to three sets of Technical Specifications; the following abbreviations are used for clarity and brevity:

TS - The existing Palisades Technical Specifications,
RTS - The revised Palisades Technical Specifications,
STS - NUREG 1432, Revision 1.

Six attachments are provided to assist the reviewer. The numbering and content of the attachments is consistent with other parts of the TSCR.

1. Proposed RTS pages
2. Bases for the RTS
3. A line by line comparison of the TS and RTS
4. STS pages marked to show the differences between RTS and STS
5. STS Bases pages marked to show differences between RTS and STS Bases.
6. A line by line comparison of RTS and STS.

Attachment 3, the line by line comparison of TS and RTS, is presented in a tabular format. The first page contains an explanation of the syntax and abbreviations used. The table is arranged numerically by TS item number. Each requirement in Sections 1 through 4 of TS is listed individually. In some cases, where a single numbered TS requirement contains more than one requirement, each requirement is listed individually under the same number.

In each section of the proposed RTS, new requirements taken from STS have been proposed. Since there is no equivalent requirement in TS, these changes do not appear in Attachment 3. These changes are considered MORE RESTRICTIVE because they add requirements and operating restrictions which do not exist in the current Palisades TS. The new requirements do appear in Attachment 6 where they are identified by an entry of "New" in the third column.

Attachment 3 Provides the Following Information for Each TS Requirement:

Identifying number of TS item,
Identifying number of closest equivalent RTS item,
Identification of TS item as LCO, Action, SR, etc.,
A short paraphrase of requirement,
A description of each proposed change from STS to RTS.

Classification of change as one of the following categories:

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

RELOCATED - A change which only moves requirements, not meeting the 10 CFR 50.36(c)(2)(ii) criteria, from the TS to the FSAR, to the Operating Requirements Manual, or to other documents controlled under 10 CFR 50.59.

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

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

Attachment 6, the line by line comparison of RTS and STS, is also presented in a tabular format. The first page contains an explanation of the syntax and abbreviations used; the second page contains a list of Palisades terminology used in place of the generic STS terminology. The table is arranged numerically by RTS item number. Each requirement in Sections 1 through 3 of RTS or STS is listed individually. Requirements which appear in TS, but not in RTS or STS, do not appear in the Attachment 6 listing.

Attachment 6 Provides the Following Information for Each RTS Requirement:

Identifying number of RTS requirement,
Identifying number of equivalent STS requirement,
Identification of each requirement as LCO, Action, SR, etc.,
Short paraphrase of each requirement,
A description of each difference between RTS and STS.

II. TECHNICAL SPECIFICATION CHANGES PROPOSED:

The TS LCOs and action statements for the reactor core Safety Limits (SLs) and PCS pressure SLs appear in Section 2.0. All RTS requirements for the SLs appear in proposed Section 2.0. Each proposed change from TS to RTS is discussed in the attachments to this section.

Each change from TS to the proposed RTS is described in Attachment 3.

The Major Changes From TS to RTS Proposed in This Section Are:

1. The reactor core SLs of Section 2.1.1 include Departure from Nucleate Boiling Ratio (DNBR) limits and Linear Heat Rate Limits (LHR) limits. Palisades does not currently have a LHR limit for fuel centerline melt. This SL was adopted from RTS. It is prudent to incorporate this SL since a great deal of the Palisades safety analysis uses this value as a bounding LHR value corresponding to fuel centerline melt. Siemens Power Corporation (SPC) has stated this is the value used in the Palisades safety analysis for LHR and fuel centerline melt.

2. In each section of the proposed RTS, new requirements taken from STS have been proposed. Since there is no equivalent requirement in TS, these changes do not appear in Attachment 3. The new requirements do appear in Attachment 6 where they are identified by an entry of "New" in the third column.

   The changes identified as "New" are considered More Restrictive because they add requirements and operating restrictions which do not exist in the current Palisades TS.

The Major Difference Between RTS and STS in This Part of the TSCR are:

1. The main difference that occurs in Section 2.0 is the incorporation of three values for DNBR as opposed to a single limit stated in the standard. The different values for DNBR corresponding to various DNB correlations are retained from the current license. Palisades has three different DNBR limits corresponding to different mechanical fuel designs and core flow conditions. Palisades only intends to use High Thermal Performance (HTP) fuel in future reload designs; however, the DNB values for older fuel design fuel will remain in TSs in case any of these fuel assemblies would be needed for fluence reduction in the cores periphery.

III. NO SIGNIFICANT HAZARDS ANALYSIS:

Each change proposed is classified in Attachment 3 as either ADMINISTRATIVE, RELOCATED, MORE RESTRICTIVE, or LESS RESTRICTIVE. Each change proposed for Section 2.0 is classified as ADMINISTRATIVE, RELOCATED, or MORE RESTRICTIVE.

ADMINISTRATIVE changes and RELOCATED changes move requirements, either within the TS or to documents controlled under 10 CFR 50.59, or clarify existing TS requirements, without affecting their technical content. Since ADMINISTRATIVE and RELOCATED changes do not alter the technical content of any requirements, they cannot involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any previously evaluated, or involve a significant reduction in a margin of safety.

MORE RESTRICTIVE changes only add new requirements, or revise existing requirements to result in additional operational restrictions. Since the TS, with all MORE RESTRICTIVE changes incorporated, will still contain all of the requirements which existed prior to the changes; MORE RESTRICTIVE changes cannot involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any previously evaluated, or involve a significant reduction in a margin of safety.

IV. CONCLUSION

The Palisades Plant Review Committee has reviewed this part of the STS conversion Technical Specifications Change Request and has determined that proposing this change does not involve an unreviewed safety question. Further, the change involves no significant hazards consideration. This change has been reviewed by the Nuclear Performance Assessment Department.

ATTACHMENT 1

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

2.0 SAFETY LIMITS (SLs)

Proposed Technical Specifications Pages

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

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

Correlation	Safety Limits
XNB	1.170
ANFP	1.154
HTP	1.141

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

2.1.2 Primary Coolant System (PCS) Pressure SL

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

2.2.1 If SL 2.1.1.1 or SL 2.1.1.2 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

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

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

ATTACHMENT 2

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

2.0 SAFETY LIMITS (SLs)

Bases for the Revised Technical Specifications

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

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

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

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

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

BACKGROUND (continued)

The Reactor Protective System (RPS), in combination with its LCOs, is designed to prevent any anticipated combination of transient conditions for Primary Coolant System (PCS) temperature, pressure, and THERMAL POWER level that would result in a violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

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

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

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

LCO 3.2.1, "Linear Heat Rate (LHR)," and LCO 3.2.2, " Radial Peaking Factors," or the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

SL 2.1.1.1

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

Palisades uses three DNB correlations: the XNB, ANFP, and HTP detailed in References 5 through 10. The XNB correlation is used for non-HTP fuel assemblies (assemblies loaded prior to cycle 9), when the non-HTP assemblies could have been limiting. The non-HTP fuel assemblies are used for vessel fluence reduction and reside on the core periphery. The core periphery locations operate at relatively low relative power fractions; therefore, they are not DNB limiting assemblies. The XNB correlation provides administrative justification for using the non-HTP assemblies in Palisades low leakage core design. The ANFP and HTP correlations are used for Palisades High Thermal Performance (HTP) fuel assemblies (assemblies loaded in cycle 9 and later).

SAFETY LIMITS (continued)

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

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

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

The DNB correlations are used solely as analytical tools to ensure that plant conditions will not degrade to the point where DNB could be challenged. The limiting axial shapes are used in the XCOBRA-IIIC model to ensure that the minimum DNBR ratio, for conditions allowed by the previously mentioned protection mechanisms, is greater than the correlation 95/95 safety limit.

The Reactor Protection System (RPS), in combination with the LCOs, is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and THERMAL POWER level that would result in a violation of the reactor core SLs.

SAFETY LIMITS (continued)

SL 2.1.1.2

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

APPLICABILITY

SL 2.1.1.1 and 2.1.1.2 only apply in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. Set points for the reactor trip functions are specified in LCO 3.3.1.

The primary safety valves or RPS trip functions serve to prevent PCS heatup to the reactor core SL conditions or to initiate a reactor trip function which forces the plant into MODE 3.

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

SAFETY LIMIT VIOLATIONS

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

2.2.1

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

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

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, 1988
2. FSAR, Section 3.3
3. 10 CFR 50.72
4. 10 CFR 50.73
5. XN-NF-621(P)(A), Rev 1
6. XN-NF-709
7. ANF-1224(P)(A), May 1989
8. ANF-89-192(P), January 1990
9. XN-NF-82-21(A), Rev 1
10. EMF-92-153(P)(A) and Supplement 1, March 1994

B 2.0 **SAFETY LIMITS (SLs)**

B 2.1.2 Primary Coolant System (PCS) Pressure SL

BASES

BACKGROUND

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, the PCS pressure is kept from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2) and by the piping, valve and fitting limit (USAS, Section B31.a) of 120% of design pressure. The initial hydrostatic test was conducted at 125% of design pressure (3125 psia) to verify the integrity of the primary coolant system (Ref. 3).

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

APPLICABLE SAFETY ANALYSES

The PCS primary safety valves, the Main Stream Safety Valves (MSSVs), and the High Pressurizer Pressure trip have settings established to ensure that the PCS pressure SL will not be exceeded. The PCS primary safety valves are sized to prevent system pressure from exceeding the design by more than 10%, in accordance with Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence the valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained. The Reactor Protective System (RPS) trip setpoints (LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation"), together with the settings of the MSSVs (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") and the pressurizer safety valves, provide pressure protection for normal operation and AOOs. In particular, the High Pressurizer Pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). Safety analyses for the PCS primary safety valves are performed, using conservative assumptions relative to pressure control devices.

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 (Ref. 6), is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is established at 2750 psia.

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

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

2.2.2.1

If the PCS 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.

SAFETY LIMIT VIOLATIONS (continued)

2.2.2.2

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

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

REFERENCES

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

ATTACHMENT 3

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

2.0 SAFETY LIMITS (SLs)

Comparison of Existing and Revised Technical Specifications

Palisades Tech Spec Requirement List. Corrected through Amendment 170

A list of the existing Palisades Tech Specs (TS) correlated to Palisades Revised Technical Specifications (RTS).

First Column; Existing Palisades Tech Spec (TS) number

Each numbered TS item is listed in the left-most column. Items which contain more than one requirement are listed once for each requirement.

Second Column; Palisades Revised Tech Spec (RTS) number

The nearest corresponding numbered RTS item is listed in the second column. If the item does not appear in RTS, it is noted as 'Deleted' or 'Relocated.'

Deleted is used where an item has been eliminated as a tech spec, ie deleting, iaw GL 84-15, the requirement to test a D.G. when an ECCS pump in the opposite train becomes inoperable.

Relocated is used where an item has been moved to a controlled program or document because it does not meet the "Criteria" of 10 CFR 50.36(2)(c)(ii).

Where an item is relocated or deleted, the number of the associated RTS section has been added to allow sorting the list by section number. Relocated items, such as heavy load restrictions, which are not associated with any particular RTS section are arbitrarily assigned the number 5.0.

Third Column; TS Item Description

An abbreviation of the TS requirement appears in the third column. Each item is identified as: LCO, ACTION, SR, Admin, Exception, etc. Some items are implied, rather than explicit, ie a LCO is implied when an ACTION exists without a stated LCO.

Description Key; TS requirement type:	Column 3 syntax:
Safety Limit	SL: Safety limit; Applicable conditions
Surveillance Requirement	SR: Equipment to be tested; Test description; Frequency
Limiting Safety Setting	LSS: RPS Trip Channel & required setting
Limiting Condition for Operation	LCO: Equipment to be operable; Applicable conditions
Action	ACTN: Condition requiring action; Required action; Completion time
Administrative Requirement	ADMN: Administrative requirement
Permitted Instrument Bypass	Byps: Bypassable component; conditions when bypass permitted
Defined Term	DEF: Name of defined item
Exception to other Requirement	XCPT: Excepted spec or condition; Applicable conditions
Descriptive material	DESC: Subject matter
Table	TBL: Table

Forth Column; Classification of Changes:

Each change is identified as ADMINISTRATIVE, RELOCATED, MORE RESTRICTIVE, or LESS RESTRICTIVE.

Fifth Column; Discussion of Changes:

Each change is discussed briefly.

Comparison of existing Palisades Tech Specs and Proposed Palisades Tech Specs. (03/28/96)

TS Number	RTS Number	TS requirement description	Classification and Description of Changes	
2.0	2.0, 3.3	Safety Limits and Limiting Safety Settings		
2.1	2.1.1	SL: DNBR >1.17/1.154/1.141; Hot Standby & Power ops	ADMINISTRATIVE:	Requirement unchanged.
2.1.1	2.0 Deleted	ACTN: SL exceeded; SD & no restart w/o NRC, etc	ADMINISTRATIVE:	Requirement redundant to 10 CFR 50.36(c)(1)(i)(A).
2.2	2.1.3	SL: PCS Press <2750; W/ fuel in Rx	ADMINISTRATIVE:	Requirement unchanged.
2.2.1	2.0 Deleted	ACTN: SL exceeded; SD & no restart w/o NRC, etc	ADMINISTRATIVE:	Requirement redundant to 10 CFR 50.36(c)(1)(i)(A).
2.3	3.3	Limiting Safety Settings - RPS		
2.3	3.3.1	LSS: RPS settings iaw Tbl 2.3.1; When RPS req by 3.17.1	ADMINISTRATIVE:	Unchanged in intent. The explicit statement is eliminated in STS format. Implicit in LCO 3.0.1 definition/stated in Bases.
2.3.1	3.3.1	ACTN: Setting not w/in limits; declare inop; Immediately	ADMINISTRATIVE:	Unchanged in intent. The explicit statement is eliminated in STS format. The intent is satisfied by LCO 3.0.2 wording.
2.3.1	3.3.1	TBL: RPS Trip Settings	MORE RESTRICTIVE:	Unchanged for 4 PCP operation. Only 4 PCP values listed, as 2 or 3 PCP operation is no longer permitted.
2.3.1.1	3.3.1	LSS: Variable Hi power Trip settings	ADMINISTRATIVE:	Requirement Unchanged.
2.3.1.2	3.3.1	LSS: PCS Flow trip settings	ADMINISTRATIVE:	Requirement Unchanged.
2.3.1.3	3.3.1	LSS: Hi Pressurizer Press trip setting	ADMINISTRATIVE:	Requirement Unchanged.
2.3.1.4	3.3.1	LSS: TM/LP Trip settings	ADMINISTRATIVE:	Requirement Unchanged.
2.3.1.5	3.3.1	LSS: SG Lo level trip setting	ADMINISTRATIVE:	Requirement Unchanged.
2.3.1.6	3.3.1	LSS: SG Lo Press trip setting	ADMINISTRATIVE:	Requirement Unchanged.
2.3.1.7	3.3.1	LSS: CHP Trip setting	ADMINISTRATIVE:	Requirement Unchanged.

ATTACHMENT 4

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

2.0 SAFETY LIMITS (SLs)

STS Pages Marked to Show the Differences Between RTS and STS

2.0 SAFETY LIMITS (SLs) ~~(Digital)~~

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 ~~In MODES 1 and 2, departure from nucleate boiling ratio (DNBR) shall be maintained at ≥ [1.19].~~

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

Correlation	Safety Limits
XNB	1.170
ANFP	1.154
HTP	1.141

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

2.1.2 ~~Reactor~~ Primary Coolant System ~~(RCS)~~ (PCS) Pressure SL

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

2.2.1 If SL 2.1.1.1 or SL 2.1.1.2 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

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

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

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

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

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

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

ATTACHMENT 5

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

2.0 SAFETY LIMITS (SLs)

STS Bases Pages Marked to Show the Differences Between RTS and STS

B 2.0 **SAFETY LIMITS (SLs)**

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

BASES

BACKGROUND

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

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

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

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

BASES

The Reactor Protective System (RPS), in combination with ~~the~~ its LCOs, is designed to prevent any anticipated combination of transient conditions for ~~Reactor~~ Primary Coolant System (~~RCS~~PCS) temperature, pressure, and THERMAL POWER level that would result in a violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

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

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

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

~~The RPS setpoints, LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation," in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for RCS temperature, pressure, and THERMAL POWER level that would result in a Departure from Nucleate Boiling Ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.~~

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

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

~~b. Pressurizer Pressure - Low trip;~~

~~c. Linear Power Level - High trip;~~

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

~~e. Local Power Density - High trip;~~

~~f. DNBR - Low trip;~~

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

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

BASES

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

~~j. Steam Generator Safety Valves.~~

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

~~The SL represents a design requirement for establishing the protection system trip setpoints identified previously.~~ LCO 3.2.1, "Linear Heat Rate (LHR)," and LCO 3.2.4~~2~~, "~~Departure From Nucleate Boiling Ratio (DNBR)~~ Radial Peaking Factors," or the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

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

~~The minimum value of the DNBR during normal operation and design basis AOOs is limited to 1.19, based on a statistical combination of CE-1 CHF correlation and engineering factor uncertainties, and is established as an SL. Additional factors such as rod bow and spacer grid size and placement will determine the limiting safety system settings required to ensure that the SL is maintained. Maintaining the dynamically adjusted peak LHR to ≤ 21 kW/ft ensures that fuel centerline melt will not occur during normal operating conditions or design AOOs.~~

SL 2.1.1.1

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

Palisades uses three DNB correlations: the XNB, ANFP, and HTP detailed in References 5 through 10. The XNB correlation is used for non-HTP fuel assemblies (assemblies loaded prior to cycle 9), when the non-HTP assemblies could have been limiting. The non-HTP fuel assemblies are used for vessel fluence reduction and reside on the core periphery. The core periphery locations operate at relatively low relative power fractions; therefore, they are not DNB limiting assemblies. The XNB correlation provides administrative justification for using the non-HTP assemblies in Palisades low leakage core design. The ANFP and HTP correlations are used for Palisades High Thermal Performance (HTP) fuel assemblies (assemblies loaded in cycle 9 and later). The HTP correlation can be used when the calculated reactor coolant conditions fall within the correlation's applicable coolant condition ranges. Outside of the applicable range of the HTP correlation, the ANFP correlation can be used. The ANFP correlation may be used over a broader range of coolant conditions than the HTP correlation. The HTP correlation is an extension of the ANFP correlation and incorporates the results of test sections designed to represent HTP fuel designs for CE plants.

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

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

The DNB correlations are used solely as analytical tools to ensure that plant conditions will not degrade to the point where DNB could be challenged. The limiting axial shapes are used in the XCOBRA-IIIC model to ensure that the minimum DNBR ratio, for conditions allowed by the previously mentioned protection mechanisms, is greater than the correlation 95/95 safety limit.

BASES

The Reactor Protection System (RPS), in combination with the LCOs, is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and THERMAL POWER level that would result in a violation of the reactor core SLs.

SL 2.1.1.2

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

APPLICABILITY

SL 2.1.1.1 and 2.1.1.2 only apply in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. ~~The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3.~~ Set points for the reactor trip functions are specified in LCO 3.3.1.

The primary safety valves or RPS trip functions serve to prevent PCS heatup to the reactor core SL conditions or to initiate a reactor trip function which forces the plant into MODE 3.

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

SAFETY LIMIT VIOLATIONS

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

2.2.1

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

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

~~2.2.3~~

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

~~2.2.4~~

~~If SL 2.1.1.1 or SL 2.1.1.2 is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.~~

~~2.2.5~~

~~If SL 2.1.1.1 or SL 2.1.1.2 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 4). A copy of the report shall also be provided to the senior management of the nuclear plant, and the utility Vice President - Nuclear Operations.~~

~~2.2.6~~

~~If SL 2.1.1.1 or SL 2.1.1.2 is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, 1988
2. FSAR, Section 3.3
3. 10 CFR 50.72
4. 10 CFR 50.73

5. XN-NF-621(P)(A), Rev 1

6. XN-NF-709

7. ANF-1224(P)(A), May 1989

8. ANF-89-192(P), January 1990

9. XN-NF-82-21(A), Rev 1

10. EMF-92-153(P)(A) and Supplement 1, March 1994

B 2.0 **SAFETY LIMITS (SLs)**

B 2.1.2 ~~Reactor Coolant System (RCS) Pressure SL (Digital)~~ Primary Coolant System (PCS) Pressure SL

BASES

BACKGROUND

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

The design pressure of the ~~RCS~~ PCS is 2500 psia. During normal operation and AOOs, the ~~RCS~~ PCS pressure is kept from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2) and by the piping, valve and fitting limit (USAS, Section B31.a) of 120% of design pressure. The initial hydrostatic test was conducted at 125% of design pressure (3125 psia) to verify the integrity of the primary coolant system (Ref. 3). ~~To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation, when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).~~

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

BASES

APPLICABLE SAFETY ANALYSES

The ~~RCS~~ PCS ~~pressurizer~~ primary safety valves, the Main Steam Safety Valves (MSSVs), and the ~~Reactor Pressure — High~~ High Pressurizer Pressure trip have settings established to ensure that the ~~RCS~~ PCS pressure SL will not be exceeded. The ~~RCS~~ PCS ~~pressurizer~~ primary safety valves are sized to prevent system pressure from exceeding the design by more than 10%, in accordance with Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence the valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained. The Reactor Protective System (RPS) trip setpoints (LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation"), together with the settings of the MSSVs (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") and the pressurizer safety valves, provide pressure protection for normal operation and AOOs. In particular, the ~~Pressurizer Pressure — High Trip setpoint~~ High Pressurizer Pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). Safety analyses for ~~both the Pressure — High Trip and~~ the ~~RCS~~ PCS ~~pressurizer~~ primary safety valves are performed, using conservative assumptions relative to pressure control devices.

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

~~a. Pressurizer Power Operated Relief Valves (PORVs);~~

~~b. Steam Bypass Control System;~~

~~c. Pressurizer Level Control System; or~~

~~d. Pressurizer Pressure Control System.~~

BASES

SAFETY LIMITS

The maximum transient pressure allowable in the ~~RCS~~ PCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the ~~RCS~~ PCS piping, valves, and fittings under ~~USAS, Section B31.1 (~~Ref. 6), is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is established at 2750 psia.

APPLICABILITY

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

SAFETY LIMIT VIOLATIONS

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

2.2.2.1

If the ~~RCS~~ PCS 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 ~~RCS~~ 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 ~~that~~ than the value specified in SL 2.1.2 exceeds 110% of the ~~RCS~~ 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 ~~RCS~~ PCS pressure by terminating the cause of the pressure increase, removing mass or energy from the ~~RCS~~ PCS, or a combination of these actions, and to establish MODE 3 conditions.

2.2.2.2

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

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

~~2.2.3~~

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

~~2.2.4~~

~~If the RCS pressure SL is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and to assess the condition of the unit before reporting to the senior management.~~

~~2.2.5~~

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

~~2.2.6~~

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

BASES

REFERENCES

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

ATTACHMENT 6

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

2.0 SAFETY LIMITS (SLs)

Comparison of Revised and Standard Technical Specifications

Palisades Revised Tech Spec Requirement List. (03/28/96)

A listing of the proposed Palisades Revised Tech Specs (RTS) correlated to the CE Standard Tech Specs (STS).

First Column; Proposed Palisades Revised Tech Spec (RTS) number
Each RTS item is listed in the left-most column.
If a STS item has been omitted from RTS, the word 'Omitted' is used.

Second Column; CE Standard Tech Spec (STS) number
The corresponding STS item is listed in the second column.
If a RTS item does not appear in STS, it is noted as 'Added'.

Third Column; Existing Palisades Tech Spec (TS) number
The closest TS item is listed in the third column.
If a RTS item does not appear in TS, it is noted as 'New'.

Fourth Column; RTS Item Description
An abbreviation of the RTS item appears in the third column.
Each item is identified as: LCO, ACTION, SR, ADMIN, Exception, etc.
In cases where a STS item was omitted from RTS, the description is of the STS item.

Description Key: RTS requirement type:	Column 4 syntax:
Safety Limit	SL: Safety limit; Applicable conditions
Limiting Condition for Operation	LCO: LCO Description; Applicable conditions
Condition	COND: Description of non-conforming condition
Action	ACTN: Required action; Completion time
Surveillance Requirement	SR: Test description; Frequency
Table	TABL: Title
Administrative Requirement	ADMN: Administrative requirement
Defined Term	DEF: Name of defined term

Fifth Column; Comments and Explanations of Differences between RTS and STS.
A brief explanation of differences between RTS and STS is provided in the fifth column.

Other abbreviations used in the listing are:

NA:	Not Applicable
CFT:	Channel Functional Test
CHNL:	Channel

RTS Number	STS Number	TS Number	RTS (STS) requirement Description	Explanation of Differences

Global differences between the proposed Palisades Technical Specifications and the Standard Technical Specifications for CE plants, Nureg 1432:

The following changes are not discussed in the explanation of differences for each TS requirement.

1) Bracketed values have been replaced with appropriate values for Palisades. Typically, the basis for these values is provided in the bases document.

2) Each required action of the form "Perform SR X.X.X.X . . ." has been altered by a parenthetical summary of the SR requirements. This change allows a reader to understand the required actions without constantly turning pages to locate the referenced SR.

3) Terminology has been changed to reflect Palisades usage:

"RWT"	becomes	"SIRWT"	Safety Injection Refueling Water Tank
"CEA"	becomes	"Control Rod" or "Rod"	Palisades uses cruciform control rods rather than the multifingered "Control Element Assemblies" of later CE plants.
"RCS"	becomes	"PCS"	Palisades terminology is "Primary Coolant System" rather than "Reactor Coolant System"
"SIAS"	becomes	"SIS"	Palisades terminology is "Safety Injection Signal" rather than "Safety Injection Actuation Signal"
"AC Vital bus"	becomes	"Preferred AC bus"	Palisades terminology.
"PAMI"	becomes	"AMI"	Accident Monitoring Instrumentation, Palisades terminology
"ESFAS"	becomes	"ESF Instrumentation"	There is no stand-alone ESFAS system or cabinet at Palisades; ESF instruments actuate the ESF functions
"DG LOVS"	becomes	"DG UV Start"	Palisades Terminology
"Remote Shutdown System"	becomes	"Alternate Shutdown System"	Palisades Terminology
"Power Rate of Change-High"	becomes	"High Startup Rate"	Palisades Terminology

Palisades RTS Cross Reference to STS. (03/28/96).

RTS Number	STS Number	TS Number	RTS (STS) requirement Description	Explanation of Differences
2.0	2.0		SAFETY LIMITS	
2.1.1.1	2.1.1.1	2.1	SL: DNBR ≥ 1.17/1.154/1.141; MODES 1 & 2	The existing Safety Limits were retained. Palisades is currently licensed with three DNBR Safety Limits. These Safety Limits were approved by Amendment 168 on June 13, 1995.
2.1.1.2	2.1.1.2	New	SL: Linear heat rate ≤ 21 kw/ft; MODES 1 & 2	Unchanged.
2.1.2	2.1.2	2.2	SL: PCS pressure ≤ 2750 psia; MODES 1-5	Unchanged.
2.2.1	2.2.1	6.7	ACTN: If SL 2.1.1.1 or SL 2.1.1.2 is violated;	Unchanged.
2.2.2	2.2.2	6.7	ACTN: If SL 2.1.2 is violated;	Unchanged.
2.2.2.1	2.2.2.1	6.7	ACTN: SL 2.1.2 violated in MODES 1 or 2;	Unchanged.
2.2.2.2	2.2.2.2	6.7	ACTN: SL 2.1.2 violated in MODES 3, 4, 5, or 6.	Unchanged.

ENCLOSURE 1

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

TECHNICAL SPECIFICATION CHANGE REQUEST

PART 3 - SECTION 3.0

March 27, 1996

CONSUMERS POWER COMPANY
Docket 50-255
Request for Change to the Technical Specifications
License DPR-20

3.0 APPLICABILITY CHANGE REQUEST

It is requested that Section 3.0, Applicability, and Section 4.0, Surveillance Requirements, of the Technical Specifications contained in the Facility Operating License DPR-20, Docket 50-255, issued to Consumers Power Company on February 21, 1991, for the Palisades Plant be changed as described below:

I. ARRANGEMENT AND CONTENT OF THIS PART OF THE CHANGE REQUEST:

This part of the Technical Specification Change Request (TSCR) proposes changes to those Palisades Technical Specifications addressing Section 3.0, Applicability, and 4.0, Surveillance Requirements. These changes are intended to result in requirements which are appropriate for the Palisades plant, but closely emulate those of the Standard Technical Specifications, Combustion Engineering Plants, NUREG 1432, Revision 1.

This discussion and its supporting information frequently refer to three sets of Technical Specifications; the following abbreviations are used for clarity and brevity:

TS - The existing Palisades Technical Specifications,
RTS - The revised Palisades Technical Specifications,
STS - NUREG 1432, Revision 1.

Six attachments are provided to assist the reviewer. The numbering and content of the attachments is consistent with other parts of the TSCR.

1. Proposed RTS pages
2. Bases for the RTS
3. A line by line comparison of the TS and RTS
4. STS pages marked to show the differences between RTS and STS
5. STS Bases pages marked to show differences between RTS and STS Bases.
6. A line by line comparison of RTS and STS.

Attachment 3, the line by line comparison of TS and RTS, is presented in a tabular format. The first page contains an explanation of the syntax and abbreviations used. The table is arranged numerically by TS item number. Each requirement in Sections 1 through 4 of TS is listed individually. In some cases, where a single numbered TS requirement contains more than one requirement, each requirement is listed individually under the same number. Requirements which appear in RTS or STS, but not in TS, do not appear in the Attachment 3 listing; they are addressed in Attachment 6.

Attachment 3 Provides the Following Information for Each TS Requirement:

Identifying number of TS item,
Identifying number of closest equivalent RTS item,
Identification of TS item as LCO, Action, SR, etc.,
A short paraphrase of requirement,
A description of each proposed change from TS to RTS.

Classification of Change as One of the Following Categories:

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

RELOCATED - A change which only moves requirements, not meeting the 10 CFR 50.36(c)(2)(ii) criteria, from the TS to the FSAR, to the Operating Requirements Manual, or to other documents controlled under 10 CFR 50.59.

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

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

Attachment 6, the line by line comparison of RTS and STS, is also presented in a tabular format. The first page contains an explanation of the syntax and abbreviations used; the second page contains a list of Palisades terminology used in place of the generic STS terminology. The table is arranged numerically by RTS item number. Each requirement in Sections 1 through 3 of RTS or STS is listed individually. Requirements which appear in TS, but not in RTS or STS, do not appear in the Attachment 6 listing.

Attachment 6 Provides the Following Information for Each RTS Requirement:

Identifying number of RTS requirement,
Identifying number of equivalent STS requirement,
Identification of each requirement as LCO, Action, SR, etc.,
Short paraphrase of each requirement,
A description of each difference between RTS and STS.

II. TECHNICAL SPECIFICATION CHANGES PROPOSED:

TS Section 3.0 addresses the applicability and general usage rules for Limiting Conditions for Operation; TS Section 4.0 addresses the applicability and general usage rules for Surveillance Requirements. These sections are combined in RTS and STS as Section 3.0. Each proposed change from TS to RTS is discussed in the attachments to this part of the TSCR.

Each proposed change to a requirement in TS is described in Attachment 3.

Those proposed RTS requirements which have no counterpart in TS are described in Attachment 6. These new requirements are identified by the word "New" in the third column of Attachment 6.

The Major Changes From TS to RTS Proposed in This Part of the TSCR are:

RTS and STS both include LCOs 3.0.6 and 3.0.7 which do not appear in TS. LCO 3.0.6 addresses actions to be taken when a required support system becomes inoperable; LCO 3.0.7 addresses Test Exception LCOs.

There are no major differences between RTS and STS in this part of the TSCR.

III. NO SIGNIFICANT HAZARDS ANALYSIS:

Each change proposed is classified in Attachment 3 as either ADMINISTRATIVE, RELOCATED, MORE RESTRICTIVE, or LESS RESTRICTIVE. Each change proposed for Section 3.0 is classified as ADMINISTRATIVE or MORE RESTRICTIVE.

ADMINISTRATIVE changes move requirements, either within the TS or to documents controlled under 10 CFR 50.59, or clarify existing TS requirements, without affecting their technical content. Since ADMINISTRATIVE changes do not alter the technical content of any requirements, they cannot involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any previously evaluated, or involve a significant reduction in a margin of safety.

MORE RESTRICTIVE changes only add new requirements, or revise existing requirements to result in additional operational restrictions. Since the TS, with all MORE RESTRICTIVE changes incorporated, will still contain all of the requirements which existed prior to the changes, MORE RESTRICTIVE changes cannot involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any previously evaluated, or involve a significant reduction in a margin of safety.

IV. CONCLUSION

The Palisades Plant Review Committee has reviewed this part of the STS conversion Technical Specifications Change Request and has determined that proposing this change does not involve an unreviewed safety question. Further, the change involves no significant hazards consideration. This change has been reviewed by the Nuclear Performance Assessment Department.

ATTACHMENT 1

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

3.0 APPLICABILITY SECTION

Proposed Technical Specifications Pages

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the plant shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the plant, as applicable, in:

a. MODE 3 within 7 hours;

b. MODE 4 within 31 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the plant.

Exceptions to this Specification are stated in the individual Specifications. These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered allow plant operation in the MODE or other specified condition in the Applicability only for a limited period of time.

LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

LCO 3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification 5.5.13, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

LCO 3.0.7 Special Test Exception (STE) LCOs allow specified Technical Specifications (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with STE LCOs is optional. When an STE LCO is desired to be met but is not met, the ACTIONS of the STE LCO shall be met. When an STE LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1 SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

SR 3.0.4 Entry into a MODE or other specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the plant.

LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

ATTACHMENT 2

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

3.0 APPLICABILITY SECTION

Bases for the Revised Technical Specifications

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 through LCO 3.0.7 establish the general requirements applicable to all Specifications and apply at all times unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the plant is in the MODES or other specified conditions of the Applicability statement of each Specification).
LCO 3.0.2	LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that: a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified. There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the plant in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the plant that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

LCO (continued)

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "PCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Alternatives that would not result in redundant equipment being inoperable should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time other conditions exist which result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the plant may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable and the ACTIONS Condition(s) are entered.

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

a. An associated Required Action and Completion Time is not met and no other Condition applies; or

b. The condition of the plant is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the plant. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

This Specification delineates the time limits for placing the plant in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in plant operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the plant, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Primary Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, "Completion Times."

LCO (continued)

A plant shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met.

b. A Condition exists for which the Required Actions have now been performed.

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of Specification 3.0.3 allow 37 hours for the plant to be in MODE 5 when a shutdown is required during MODE 1 operation. If the plant is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the plant is already in the most restrictive Condition required by LCO 3.0.3.

LCO (continued)

The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken. Exceptions to LCO 3.0.3 are provided in instances where requiring a plant shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the plant. An example of this is in LCO 3.7.16, "Fuel Storage Pool Water Level." LCO 3.7.16 has an Applicability of "During movement of irradiated fuel assemblies in the fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.16 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the plant in a shutdown condition. The Required Action of LCO 3.7.16 of "Suspend movement of irradiated fuel assemblies in fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the plant in a MODE or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Plant conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the plant being required to exit the Applicability desired to be entered to comply with the Required Actions.

LCO (continued)

Compliance with Required Actions that permit continued operation of the plant for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the plant before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any plant shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4 or 5, MODE 2 from MODE 3 or 4 or 5, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, 3, or 4. The requirements of LCO 3.0.4 do not apply in MODES 4 and 5 or in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken. [In some cases (e.g., ..) these ACTIONS provide a Note that states "While this LCO is not met, entry into a MODE or other specified condition in the Applicability is not permitted, unless required to comply with ACTIONS." This Note is a requirement explicitly precluding entry into a MODE or other specified condition of the Applicability.

LCO (continued)

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed SRs. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the SRs.

BASES

LCO (continued)

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of an SR on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of an SR on another channel in the same trip system.

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the plant is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the plant is maintained in a safe condition in the support system's Required Actions.

LCO (continued)

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.13, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.7

Special tests and operations are required at various times over the plant's life to demonstrate performance characteristics, to perform maintenance activities, and to perform special evaluations. Because TS normally preclude these tests and operations, Special Test Exceptions (STEs) allow specified requirements to be changed or suspended under controlled conditions. STEs are included in applicable sections of the Specifications. Unless otherwise specified, all other TS requirements remain unchanged and in effect as applicable. This will ensure that all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed or suspended to perform the special test or operation will remain in effect.

LCO (continued)

The Applicability of an STE LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with STE LCOs is optional.

A special test may be performed under either the provisions of the appropriate STE LCO or the other applicable TS requirements. If it is desired to perform the special test under the provisions of the STE LCO, the requirements of the STE LCO shall be followed. This includes the SRs specified in the STE LCO.

Some of the STE LCOs require that one or more of the LCOs for normal operation be met (i.e., meeting the STE LCO requires meeting the specified normal LCOs). The Applicability, ACTIONS, and SRs of the specified normal LCOs, however, are not required to be met in order to meet the STE LCO when it is in effect. This means that, upon failure to meet a specified normal LCO, the associated ACTIONS of the STE LCO apply, in lieu of the ACTIONS of the normal LCO. Exceptions to the above do exist. There are instances when the Applicability of the specified normal LCO must be met, where its ACTIONS must be taken, where certain of its Surveillances must be performed, or where all of these requirements must be met concurrently with the requirements of the STE LCO.

Unless the SRs of the specified normal LCOs are suspended or changed by the special test, those SRs that are necessary to meet the specified normal LCOs must be met prior to performing the special test. During the conduct of the special test, those Surveillances need not be performed unless specified by the ACTIONS or SRs of the STE LCO.

ACTIONS for STE LCOs provide appropriate remedial measures upon failure to meet the STE LCO. Upon failure to meet these ACTIONS, suspend the performance of the special test and enter the ACTIONS for all LCOs that are then not met. Entry into LCO 3.0.3 may possibly be required, but this determination should not be made by considering only the failure to meet the ACTIONS of the STE LCO.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the plant is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a Special Test Exception (STE) are only applicable when the STE is used as an allowable exception to the requirements of a Specification.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary plant parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

Some examples of this process are:

a. Auxiliary Feed Water (AFW) pump turbine maintenance during refueling that requires testing at steam pressures > 800 psi. However, if other appropriate testing is satisfactorily completed, the AFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the testing.

b. High Pressure Safety Injection (HPSI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPSI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

SR 3.0.2 (continued)

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is a Surveillance with a Frequency of "in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions." The requirements of regulations take precedence over the TS. The TS cannot in and of themselves extend a test interval specified in the regulations. Therefore, there is a Note in the Frequency stating, "SR 3.0.2 is not applicable."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

SR 3.0.3 (continued)

This delay period provides an adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of plant conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified plant conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified Condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the plant.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any plant shutdown.

SR 3.0.4 (continued)

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SR 3.0.4 is only applicable when entering MODE 3 from MODE 4, Mode 2 from MODE 3 or 4, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, or 3. The requirements of SR 3.0.4 do not apply in MODES 4 and 5, or in other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

ATTACHMENT 3

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

3.0 APPLICABILITY SECTION

Comparison of Existing and Revised Technical Specifications

Palisades Tech Spec Requirement List. Corrected through Amendment 170

A list of the existing Palisades Tech Specs (TS) correlated to Palisades Revised Technical Specifications (RTS).

First Column; Existing Palisades Tech Spec (TS) number

Each numbered TS item is listed in the left-most column. Items which contain more than one requirement are listed once for each requirement.

Second Column; Palisades Revised Tech Spec (RTS) number

The nearest corresponding numbered RTS item is listed in the second column. If the item does not appear in RTS, it is noted as 'Deleted' or 'Relocated.'

Deleted is used where an item has been eliminated as a tech spec, ie deleting, iaw GL 84-15, the requirement to test a D.G. when an ECCS pump in the opposite train becomes inoperable.

Relocated is used where an item has been moved to a controlled program or document because it does not meet the "Criteria" of 10 CFR 50.36(2)(c)(ii).

Where an item is relocated or deleted, the number of the associated RTS section has been added to allow sorting the list by section number. Relocated items, such as heavy load restrictions, which are not associated with any particular RTS section are arbitrarily assigned the number 5.0.

Third Column; TS Item Description

An abbreviation of the TS requirement appears in the third column. Each item is identified as: LCO, ACTION, SR, Admin, Exception, etc. Some items are implied, rather than explicit, ie a LCO is implied when an ACTION exists without a stated LCO.

Description Key; TS requirement type:	Column 3 syntax:
Safety Limit	SL: Safety limit; Applicable conditions
Surveillance Requirement	SR: Equipment to be tested; Test description; Frequency
Limiting Safety Setting	LSS: RPS Trip Channel & required setting
Limiting Condition for Operation	LCO: Equipment to be operable; Applicable conditions
Action	ACTN: Condition requiring action; Required action; Completion time
Administrative Requirement	ADMN: Administrative requirement
Permitted Instrument Bypass	Byps: Bypassable component; conditions when bypass permitted
Defined Term	DEF: Name of defined item
Exception to other Requirement	XCPT: Excepted spec or condition; Applicable conditions
Descriptive material	DESC: Subject matter
Table	TBL: Table

Forth Column; Classification of Changes:

Each change is identified as ADMINISTRATIVE, RELOCATED, MORE RESTRICTIVE, or LESS RESTRICTIVE.

Fifth Column; Discussion of Changes:

Each change is discussed briefly.

Comparison of existing Palisades Tech Specs and Proposed Palisades Tech Specs. (03/28/96)

TS Number	RTS Number	TS requirement description	Classification and Description of Changes	
3.0	3.0 (LCOs)	Limiting Conditions for Operation		
3.0.1	3.0.1(LCO)	LCO: Compliance with LCOs required	ADMINISTRATIVE:	Changed wording to agree with STS. Very similar; used STS wording.
3.0.1	3.0.2(LCO)	LCO: Follow Action when not meeting LCO	ADMINISTRATIVE:	Changed wording to agree with STS. Very similar; used STS wording.
3.0.2	3.0.2(LCO)	LCO: Exit Action when LCO restored	ADMINISTRATIVE:	Changed wording to agree with STS. Very similar; used STS wording.
3.0.3	3.0.3(LCO)	ACTN: Required when beyond LCO & Actions	ADMINISTRATIVE:	Changed wording to agree with STS. Very similar; used STS wording; Note, however, that the changed definitions for operating conditions affects the actual requirements. Below the requirements of TS 3.0.3 and RTS 3.0.3 are compared step by step below. The requirements are essentially unchanged. Actions TS-3.0.3.2 and RTS-3.0.3.b are not alike due to the difference in the definitions involved. The total time to MODE 5 (formerly Cold Shutdown) is unchanged.
3.0.3	3.0.3(LCO)	ACTN: Initiate action to SD in 1 hr	ADMINISTRATIVE:	Requirement unchanged.
3.0.3.1	3.0.3.a(LCO)	ACTN: Be in Hot Standby (<2%) in next 6 (7 total) hrs	MORE RESTRICTIVE:	Changed to: Be in MODE 3 (Subcritical) in 7 hrs (total).
3.0.3.2	3.0.3.b(LCO)	ACTN: Be in HSD (subcritical) in next 6 (13 total) hrs	MORE RESTRICTIVE:	Changed to: Be in MODE 4 (<300°F) in 31 hrs (total).
3.0.3.3	3.0.3.c(LCO)	ACTN: Be in CSD (<210°F) in next 24 (37 total) hrs	MORE RESTRICTIVE:	Changed to: Be in MODE 5 (<200°F) in 37 hrs (total).
3.0.4	3.0.4(LCO)	LCO: Limits mode entry unless LCOs met	ADMINISTRATIVE:	Very similar; used STS wording.

Comparison of existing Palisades Tech Specs and Proposed Palisades Tech Specs. (03/28/96)

TS Number	RTS Number	TS requirement description	Classification and Description of Changes	
4.0	3.0 - (SRs)	Surveillance Requirements		
4.0.1	3.0.1(SR)	SR: Surv applicability same as LCO	ADMINISTRATIVE:	Very similar; used STS words.
4.0.2	3.0.2(SR)	SR: Surv req frequency	ADMINISTRATIVE:	Very similar; used STS words.
4.0.2	3.0.2(SR)	SR: Max freq extension, 1.25x	ADMINISTRATIVE:	Very similar; used STS words.
4.0.3	3.0.3(SR)	SR: Failing SR implies noncompliance w/LCO	ADMINISTRATIVE:	Very similar; used STS words.
4.0.4	3.0.4(SR)	SR: SRs must be current to enter condition in LCO	ADMINISTRATIVE:	Very similar; used STS words.
4.0.5	5.5.7	SR: Surv Req for ASME testing	ADMINISTRATIVE:	Moved to program 5.5.7 iaw STS.

ATTACHMENT 4

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

3.0 APPLICABILITY SECTION

STS Pages Marked to Show the Differences Between RTS and STS

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1

LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7.

LCO 3.0.2

Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

LCO 3.0.3

When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the ~~unit~~ plant shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the ~~unit~~ plant, as applicable, in:

a. MODE 3 within 7 hours;

b. ~~[~~MODE 4 within ~~13~~31 hours~~]~~; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

3.0 LCO APPLICABILITY

LCO 3.0.4

When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the ~~unit~~ plant.

Exceptions to this Specification are stated in the individual Specifications. These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered allow ~~unit~~ plant operation in the MODE or other specified condition in the Applicability only for a limited period of time.

LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

~~Reviewers's Note: LCO 3.0.4 has been revised so that changes in MODES or other specified conditions in the Applicability that are part of a shutdown of the unit shall not be prevented. In addition, LCO 3.0.4 has been revised so that it is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, and 3. The MODE change restrictions in LCO 3.0.4 were previously applicable in all MODES. Before this version of LCO 3.0.4 can be implemented on a plant specific basis, the licensee must review the existing technical specifications to determine where specific restrictions on MODE changes or Required Actions should be included in individual LCOs to justify this change; such an evaluation should be summarized in a matrix of all existing LCOs to facilitate NRC staff review of a conversion to the STS.~~

LCO 3.0.5

Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

3.0 LCO APPLICABILITY

LCO 3.0.6

When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification 5.5.~~15~~13, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

LCO 3.0.7

Special ~~t~~Test ~~e~~Exception (STE) LCOs ~~[in each applicable LCO section]~~ allow specified Technical Specifications (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with STE LCOs is optional. When an STE LCO is desired to be met but is not met, the ACTIONS of the STE LCO shall be met. When an STE LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

3.0 SR APPLICABILITY

SR 3.0.4

Entry into a MODE or other specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the ~~unit~~ plant.

LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

~~Reviewer's Note: SR 3.0.4 has been revised so that changes in MODES or other specified conditions in the Applicability that are part of a shutdown of the unit shall not be prevented. In addition, SR 3.0.4 has been revised so that it is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, and 3. The MODE change restrictions in SR 3.0.4 were previously applicable in all MODES. Before this version of SR 3.0.4 can be implemented on a plant specific basis, the licensee must review the existing technical specifications to determine where specific restrictions on MODE changes or Required Actions should be included in individual LCOs to justify this change; such an evaluation should be summarized in a matrix of all existing LCOs to facilitate NRC staff review of a conversion to the STS.~~

ATTACHMENT 5

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

3.0 APPLICABILITY SECTION

STS Bases Pages Marked to Show the Differences Between RTS and STS

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 through LCO 3.0.7 establish the general requirements applicable to all Specifications and apply at all times unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the ~~unit~~ plant is in the MODES or other specified conditions of the Applicability statement of each Specification).
LCO 3.0.2	LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that: a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified. There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the ~~unit~~ plant in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering

(continued)

BASES

LCO 3.0.2 (continued)

ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the ~~unit~~ plant that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "~~RCS~~ PCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Alternatives that would not result in redundant equipment being inoperable should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time other conditions exist which result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the ~~unit~~ plant may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable and the ACTIONS Condition(s) are entered.

(continued)

BASES (continued)

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

a. An associated Required Action and Completion Time is not met and no other Condition applies; or

b. The condition of the ~~unit~~ plant is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the ~~unit~~ plant. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

This Specification delineates the time limits for placing the ~~unit~~ plant in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in ~~unit~~ plant operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the ~~unit~~ plant, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the ~~Reactor~~ Primary Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

(continued)

BASES

LCO 3.0.3 (continued)

A ~~unit~~ plant shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met.

b. A Condition exists for which the Required Actions have now been performed.

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of Specification 3.0.3 allow 37 hours for the ~~unit~~ plant to be in MODE 5 when a shutdown is required during MODE 1 operation. If the ~~unit~~ plant is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the ~~unit~~ plant is already in the most restrictive Condition required by LCO 3.0.3.

The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken. Exceptions to LCO 3.0.3 are provided in instances where requiring a ~~unit~~ plant shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the ~~unit~~ plant. An example of this is in LCO 3.7.16, "Fuel Storage Pool Water Level." LCO 3.7.16 has an Applicability of "During movement of irradiated fuel

(continued)

BASES

LCO 3.0.3 (continued)

assemblies in the fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.16 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the ~~unit~~ plant in a shutdown condition. The Required Action of LCO 3.7.16 of "Suspend movement of irradiated fuel assemblies in fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

~~The requirement to be in MODE 4 in 13 hours is plant specific and depends on the ability to cool the pressurizer and degas.~~

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the ~~unit~~ plant in a MODE or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. ~~unit~~ Plant conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the ~~unit~~ plant being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the ~~unit~~ plant for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the ~~unit~~ plant before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

(continued)

BASES

LCO 3.0.4 (continued)

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any ~~unit~~ plant shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4 or 5, MODE 2 from MODE 3 or 4 or 5, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, 3, or 4 . The requirements of LCO 3.0.4 do not apply in MODES 4 and 5 or in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken. [In some cases (e.g., ..) these ACTIONS provide a Note that states "While this LCO is not met, entry into a MODE or other specified condition in the Applicability is not permitted, unless required to comply with ACTIONS." This Note is a requirement explicitly precluding entry into a MODE or other specified condition of the Applicability.]

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with

(continued)

BASES

LCO 3.0.5 (continued)

ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed SRs. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the SRs.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of an SR on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of an SR on another channel in the same trip system.

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the ~~unit~~ plant is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the

(continued)

BASES

LCO 3.0.6 (continued)

supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the ~~unit~~ plant is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.~~15~~13, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is

(continued)

BASES

LCO 3.0.6 (continued)

retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.7

Special tests and operations are required at various times over the ~~unit~~ plant's life to demonstrate performance characteristics, to perform maintenance activities, and to perform special evaluations. Because TS normally preclude these tests and operations, ~~s~~Special ~~t~~Test ~~e~~Exceptions (STEs) allow specified requirements to be changed or suspended under controlled conditions. STEs are included in applicable sections of the Specifications. Unless otherwise specified, all other TS requirements remain unchanged and in effect as applicable. This will ensure that all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed or suspended to perform the special test or operation will remain in effect.

The Applicability of an STE LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with STE LCOs is optional.

A special test may be performed under either the provisions of the appropriate STE LCO or the other applicable TS requirements. If it is desired to perform the special test under the provisions of the STE LCO, the requirements of the STE LCO shall be followed. This includes the SRs specified in the STE LCO.

Some of the STE LCOs require that one or more of the LCOs for normal operation be met (i.e., meeting the STE LCO requires meeting the specified normal LCOs). The Applicability, ACTIONS, and SRs of the specified normal LCOs, however, are not required to be met in order to meet the STE LCO when it is in effect. This means that, upon failure to meet a specified normal LCO, the associated ACTIONS of the STE LCO apply, in lieu of the ACTIONS of the normal LCO. Exceptions to the above do exist. There are instances when the Applicability of the specified normal LCO must be met, where its ACTIONS must be taken, where certain of its Surveillances must be performed, or where all of

(continued)

BASES

~~SR~~ LCO 3.0.7 (continued)

these requirements must be met concurrently with the requirements of the STE LCO.

Unless the SRs of the specified normal LCOs are suspended or changed by the special test, those SRs that are necessary to meet the specified normal LCOs must be met prior to performing the special test. During the conduct of the special test, those Surveillances need not be performed unless specified by the ACTIONS or SRs of the STE LCO.

ACTIONS for STE LCOs provide appropriate remedial measures upon failure to meet the STE LCO. Upon failure to meet these ACTIONS, suspend the performance of the special test and enter the ACTIONS for all LCOs that are then not met. Entry into LCO 3.0.3 may possibly be required, but this determination should not be made by considering only the failure to meet the ACTIONS of the STE LCO.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the ~~unit~~ plant is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a ~~s~~Special ~~t~~Test ~~e~~Exception (STE) are only applicable when the STE is used as an allowable exception to the requirements of a Specification.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

(continued)

BASES

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary ~~unit~~ plant parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

Some examples of this process are:

a. Auxiliary ~~f~~Feedwater (AFW) pump turbine maintenance during refueling that requires testing at steam pressures > 800 psi. However, if other appropriate testing is satisfactorily completed, the AFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the testing.

b. High ~~p~~Pressure ~~s~~Safety ~~i~~Injection (HPSI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPSI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may

(continued)

BASES

SR 3.0.2 (continued)

not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is a Surveillance with a Frequency of "in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions." The requirements of regulations take precedence over the TS. The TS cannot in and of themselves extend a test interval specified in the regulations. Therefore, there is a Note in the Frequency stating, "SR 3.0.2 is not applicable."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not

(continued)

BASES

SR 3.0.3 (continued)

been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides an adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of ~~unit~~ plant conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified ~~unit~~ plant conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay

(continued)

BASES

SR 3.0.3 (continued)

period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified Condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the ~~unit~~ plant.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability

(continued)

BASES

SR 3.0.4 (continued)

that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any ~~unit~~ plant shutdown.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SR 3.0.4 is only applicable when entering MODE 3 from MODE 4, Mode 2 from MODE 3 or 4, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, or 3. The requirements of SR 3.0.4 do not apply in MODES 4 and 5, or in other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

ATTACHMENT 6

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

3.0 APPLICABILITY SECTION

Comparison of Revised and Standard Technical Specifications

A listing of the proposed Palisades Revised Tech Specs (RTS) correlated to the CE Standard Tech Specs (STS).

First Column; Proposed Palisades Revised Tech Spec (RTS) number
Each RTS item is listed in the left-most column.
If a STS item has been omitted from RTS, the word 'Omitted' is used.

Second Column; CE Standard Tech Spec (STS) number
The corresponding STS item is listed in the second column.
If a RTS item does not appear in STS, it is noted as 'Added'.

Third Column; Existing Palisades Tech Spec (TS) number
The closest TS item is listed in the third column.
If a RTS item does not appear in TS, it is noted as 'New'.

Fourth Column; RTS Item Description
An abbreviation of the RTS item appears in the third column.
Each item is identified as: LCO, ACTION, SR, ADMIN, Exception, etc.
In cases where a STS item was omitted from RTS, the description is of the STS item.

Description Key: RTS requirement type:	Column 4 syntax:
Safety Limit	SL: Safety limit; Applicable conditions
Limiting Condition for Operation	LCO: LCO Description; Applicable conditions
Condition	COND: Description of non-conforming condition
Action	ACTN: Required action; Completion time
Surveillance Requirement	SR: Test description; Frequency
Table	TABL: Title
Administrative Requirement	ADMN: Administrative requirement
Defined Term	DEF: Name of defined term

Fifth Column; Comments and Explanations of Differences between RTS and STS.
A brief explanation of differences between RTS and STS is provided in the fifth column.

Other abbreviations used in the listing are:

NA:	Not Applicable
CFT:	Channel Functional Test
CHNL:	Channel

RTS Number	STS Number	TS Number	RTS (STS) requirement Description	Explanation of Differences

Global differences between the proposed Palisades Technical Specifications and the Standard Technical Specifications for CE plants, Nureg 1432:

The following changes are not discussed in the explanation of differences for each TS requirement.

1) Bracketed values have been replaced with appropriate values for Palisades. Typically, the basis for these values is provided in the bases document.

2) Each required action of the form "Perform SR X.X.X.X . . ." has been altered by a parenthetical summary of the SR requirements. This change allows a reader to understand the required actions without constantly turning pages to locate the referenced SR.

3) Terminology has been changed to reflect Palisades usage:

"RWT"	becomes	"SIRWT"	Safety Injection Refueling Water Tank
"CEA"	becomes	"Control Rod" or "Rod"	Palisades uses cruciform control rods rather than the multifingered "Control Element Assemblies" of later CE plants.
"RCS"	becomes	"PCS"	Palisades terminology is "Primary Coolant System" rather than "Reactor Coolant System"
"SIAS"	becomes	"SIS"	Palisades terminology is "Safety Injection Signal" rather than "Safety Injection Actuation Signal"
"AC Vital bus"	becomes	"Preferred AC bus"	Palisades terminology.
"PAMI"	becomes	"AMI"	Accident Monitoring Instrumentation, Palisades terminology
"ESFAS"	becomes	"ESF Instrumentation"	There is no stand-alone ESFAS system or cabinet at Palisades; ESF instruments actuate the ESF functions
"DG LOVS"	becomes	"DG UV Start"	Palisades Terminology
"Remote Shutdown System"	becomes	"Alternate Shutdown System"	Palisades Terminology
"Power Rate of Change-High"	becomes	"High Startup Rate"	Palisades Terminology

RTS Number	STS Number	TS Number	RTS (STS) requirement Description	Explanation of Differences
3.0	3.0		APPLICABILITY SECTION	
3.0.1	3.0.1	3.0.1	LCO: LCOs shall be met except as provided in LCO 3.0.2.	Unchanged.
3.0.2	3.0.2	3.0.1	LCO: Upon failing to meet a LCO the ACTIONS shall be met.	Unchanged.
3.0.3	3.0.3	3.0.3	LCO: SD Required when beyond LCO & ACTIONS; MODES 1,2,3,4	Changed time to MODE 4 from 13 to 31 hours. Palisades cannot degas primary coolant system sufficiently to allow opening system for maintenance within 13 hour period, and must have elevated pressure/temperature to accomplish degas flow. Time to MODE 5 is unchanged. In addition, the word "unit" was changed to "plant" as plant specific usage.
3.0.4	3.0.4	3.0.4	LCO: Limits MODE entry unless LCOs met	Unchanged.
3.0.5	3.0.5	New	LCO: Equip declared inop may be operated to show OPERABILITY	Unchanged.
3.0.6	3.0.6	New	LCO: Equip may be made inoperable for SRs and . . .	Unchanged.
3.0.7	3.0.7	New	LCO: Special Test Exceptions	Unchanged.
3.0.1	3.0.1	4.0.3	SR: Failure to meet SR or Frequency is failure to meet LCO.	Unchanged.
3.0.2	3.0.2	4.0.2	SR: The Frequency is met if within 1.25X interval specified.	Unchanged.
3.0.3	3.0.3	4.0.3	SR: When failed to do SR; may delay for up to 24 Hrs.	Unchanged.
3.0.4	3.0.4	4.0.4	SR: Shall not entry applicability unless the SRs met.	Unchanged.

ENCLOSURE 1

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

TECHNICAL SPECIFICATION CHANGE REQUEST

PART 4 - SECTION 3.1

March 28, 1996

CONSUMERS POWER COMPANY
Docket 50-255
Request for Change to the Technical Specifications
License DPR-20

3.1 REACTIVITY CONTROL SYSTEMS CHANGE REQUEST

It is requested that the Control Rod System requirements of the Technical Specifications contained in the Facility Operating License DPR-20, Docket 50-255, issued to Consumers Power Company on February 21, 1991, for the Palisades Plant be changed as described below:

I. ARRANGEMENT AND CONTENT OF THIS PART OF THE CHANGE REQUEST:

This section of the Technical Specification Change Request (TSCR) proposes changes to those Palisades Technical Specification requirements addressing the Control Rod Limits. These changes are intended to result in requirements which are appropriate for the Palisades plant, but closely emulate those of the Standard Technical Specifications, Combustion Engineering Plants, NUREG 1432, Revision 1.

This discussion and its supporting information frequently refer to three sets of Technical Specifications; the following abbreviations are used for clarity and brevity:

TS - The existing Palisades Technical Specifications,
RTS - The revised Palisades Technical Specifications,
STS - NUREG 1432, Revision 1.

Six attachments are provided to assist the reviewer. The numbering and content of the attachments is consistent with other parts of the TSCR.

1. Proposed RTS pages
2. Bases for the RTS
3. A line by line comparison of the TS and RTS
4. STS pages marked to show the differences between RTS and STS
5. STS Bases pages marked to show differences between RTS and STS Bases.
6. A line by line comparison of RTS and STS.

Attachment 3, the line by line comparison of TS and RTS, is presented in a tabular format. The first page contains an explanation of the syntax and abbreviations used. The table is arranged numerically by TS item number. Each requirement in Sections 1 through 4 of TS is listed individually. In some cases, where a single numbered TS requirement contains more than one requirement, each requirement is listed individually under the same number. Requirements which appear in RTS or STS, but not in TS, do not appear in the Attachment 3 listing.

Attachment 3 Provides the Following Information for Each TS Requirement:

Identifying number of TS item,
Identifying number of closest equivalent RTS item,
Identification of TS item as LCO, Action, SR, etc.,
A short paraphrase of requirement,
A description of each proposed change from TS to RTS.

Classification of Change as One of the Following Categories:

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

RELOCATED - A change which only moves requirements, not meeting the 10 CFR 50.36(c)(2)(ii) criteria, from the TS to the FSAR, to the Operating Requirements Manual, or to other documents controlled under 10 CFR 50.59.

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

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

Attachment 6, the line by line comparison of RTS and STS, is also presented in a tabular format. The first page contains an explanation of the syntax and abbreviations used; the second page contains a list of Palisades terminology used in place of the generic STS terminology. The table is arranged numerically by RTS item number. Each requirement in Sections 1 through 3 of RTS or STS is listed individually. Requirements which appear in TS, but not in RTS or STS, do not appear in the Attachment 6 listing.

Attachment 6 Provides the Following Information for Each RTS Requirement:

Identifying number of RTS requirement,
Identifying number of equivalent STS requirement,
Identification of each requirement as LCO, Action, SR, etc.,
Short paraphrase of each requirement,
A description of each difference between RTS and STS.

II. TECHNICAL SPECIFICATION CHANGES PROPOSED:

The TS LCOs and action statements for Control Rod Limits appear in Section 3.10; those for Moderator Temperature Coefficient (MTC) and Reactivity Balance requirements appear in Section 3.12 and 4.10. The TS surveillance requirements appear in TS Section 4. All RTS requirements for these LCOs appear in proposed Section 3.1 "Reactivity Control Systems." Each proposed change from TS to RTS is discussed in the attachments to this part of the TSCR.

Each proposed change to a requirement in TS is described in Attachment 3.

Those proposed RTS requirements which have no counterpart in TS are described in Attachment 6. These new requirements are identified by the word "New" in the third column of Attachment 6.

The Major Changes From TS to RTS Proposed in This Part of the TSCR are:

1. Shutdown Margin Requirements (SDM)

The SDM values stated in the current TS have not changed; however, a great deal of extraneous information existed in LCO 3.10.1 "Shutdown Margin Requirements" which has been moved, eliminated or condensed to make the RTS LCOs for SDM much more concise. This warrants some discussion on these details.

The SDM requirement has been split into two LCOs bounded by the different applicability ranges. LCO 3.1.1 covers SDM requirements for 3 and 4 pump operation while in MODE 3 with $T_{AVE} \geq 525$ °F and LCO 3.1.2 covers SDM requirements for MODE 3 < 525 °F, MODE 4 and MODE 5. The SDM values for these different applicability ranges have been retained from current TS. Items relating to PCS/Shutdown cooling pump flow requirements and electrical disabling of the charging pumps and charging pump operational surveillance have been moved to Section 3.4.4 of the RTS. This information is much more prudent to be contained in this area. These changes brought Palisades version of these LCOs almost identical to the Standard with the exception of Palisades retained SDM values and 3 vs 4 pump operation.

3.10.1 d in current TS states that boron concentration must be increased to account for an untrippable control rod; however, in RTS the definition of SDM a stuck or untrippable rod does not have to be accounted for if two independent means of rod position indication are available (i.e.the PIP and the SPI). The requirement for control rod drop times has been moved to 3.1.5.6 in the RTS. RTS LCOs 3.1.1 and 3.1.2 meet the same intent of current TS and conform to standard to allow for a very clear interpretation of SDM requirements under all operation conditions.

RTS 3.1.1 and 3.1.2 have added requirements to verify SDM on a 24 hour frequency that is the same as that in the STS. Palisades does not currently have a requirement to perform a SDM verification on a routine basis. It has been noted in the basis that this can be satisfied by a full SDM calculation or verified by data supplied by the fuel vendor in the technical data book that already considers the necessary component in the SDM calculation. A requirement to verify SDM within two hours following a reactor trip or shutdown has been added to 3.1.1 and 3.1.2 to account for the case where the SR has been performed and a very short time later, the plant is tripped or taken off line, and a long time period may exist before a SDM verification is required to be performed.

2. LCO 3.10.3 has been combined with RTS LCO 3.1.6 "Shutdown and Part Length Rod Insertion Limits." The Shutdown rod insertion limit has been added from STS along with its associated conditions, required actions and completion times. It is prudent to include this LCO since it reflects current operational practice at Palisades.

3. Control Rod Operability and Alignment

TS Section 3.10.4 definitions of Misaligned Rod, Inoperable control rod, and Inoperable Part Length rod have not been retained from current TS. These definitions are not defined terms in the RTS. The RTS conditions and required actions provide a path for the intent of these definitions to be retained without defining the terms explicitly.

TS Section 3.10.4 "Misaligned or inoperable CONTROL ROD or Part-Length Rod" has been significantly changed to conform with the STS, while the intent and actions have been retained. All of the current requirements of 3.10.4 have been retained with the exception of the ejected rod worth verification which has been deleted since it is incorporated into the SDM verification of 3.10.4 c.

Two separate conditions of rod misalignment with associated actions have been added to reflect the STS; they include rod misalignment $\geq$ 8 inches $\leq$ 20 inches, and rod misalignment > 20 inches. These conditions are supported by subsequent required actions that are a composite of current TS requirements and STS actions that were prudent to include to support these operational conditions.

RTS 3.1.5 condition D Rod position deviation alarm inoperable has been added from STS. This is a prudent addition since it allows for rod position verification within 15 minutes following any rod motion while the deviation alarm is inoperable. This assures that the rods remain within 8 inches of other rods in the same group and operation with an undetected misaligned rod would not occur.

4. Current TS Section 4.10 "Reactivity Anomalies" has undergone significant change, adding the conditions, required actions and completion times reflected in STS. The RTS allows for re-evaluation of the core design and safety analysis to ensure operation outside of the reactivity anomaly limit is bounded by design. These additional changes are more restrictive since they require more action to be taken than NRC notification stated in current TS.

5. Regulating Rod Insertion Limits

Current TS 3.10.5 covers withdrawal sequence, overlap and rod insertion limits which have all been retained in RTS. The RTS Section 3.1.7 "Regulating Rod Insertion Limits" supplies much more detail regarding stated conditions, required actions and associated completion times that have been adopted by the STS. This allows for a much more concise path for operators to follow when one of the above stated limits is outside of the COLR specifications.

The reference to overlap has been excluded since operating within sequence implies overlap shall be met. The RTS requires the PDIL alarm circuit to be operable and calls out the required action of verifying group positions within 15 minutes following rod motion. This ensures operators are aware of PDIL violations while performing rod manipulations. Since Palisades only has manual rod control as opposed to automatic rod control seen in other plants, the completion time of 15 minutes following any rod motion with no subsequent follow up is adequate to ensure compliance.

6. RTS 3.1.8 "Special Test Exception (STE) Shutdown Margin and Rod Limits" retains all of the exemptions necessary to perform PHYSICS TESTS. Current TS 3.10 also uses this exception for rod exercising; however, RTS modifies any affected LCO with a note to allow such surveillance. RTS 3.1.8 is solely dedicated to PHYSICS TESTS. RTS 3.1.8 supplies more restrictive actions that reflect current PHYSICS TESTS procedural performance at Palisades and conforms to STS.

7. In each section of the proposed RTS, new requirements taken from STS have been proposed. Since there is no equivalent requirement in TS, these changes do not appear in Attachment 3. The new requirements do appear in Attachment 6 where they are identified by an entry of "New" or "3.0.3" in the third column.

   The changes identified as "New" are considered MORE RESTRICTIVE because they add requirements and operating restrictions which do not exist in the current Palisades TS.

   The changes identified as "3.0.3" are considered LESS RESTRICTIVE because they extend the time available to restore compliance to the LCO (the Allowed Outage Time) beyond that allowed by LCO 3.0.3. In these cases, the proposed RTS contain a specific Action where the existing TS do not contain any Action for the associated LCO. These instances do not involve a loss of safety function, but occur due to the lack of structure of Technical Specifications circa 1970. There was not necessarily an intent that failure to meet these LCOs would force a plant shutdown or an entry into LCO 3.0.3 (the original TS contained no equivalent of LCO 3.0.3).

<u>The Major Differences Between RTS and STS in This Part of the TSCR are</u>:

1. STS SDM LCO 3.1.1 does not account for different SDM limits for 3 and 4 Primary Coolant Pump operation. Palisades safety analysis supports two different SDM values for applicability stated in RTS LCO 3.1.1 for less than 4 pump operation.

2. The applicability for 3.1.1 and 3.1.2 has been changed from STS to incorporate current operating practice and support assumptions made in the safety analysis. The temperature break point between LCOs 3.1.1 and 3.1.2 was changed to MODE 3 $\geq$ 525°F as opposed to the standard value of 200°F. The value used by Palisades is the old Hot shutdown break point of 525°F which allows for the two SDM LCOs to be molded around current licensed SDM values.

3. SR 3.1.1.1 and SR 3.1.2.1 have had a frequency added to the RTS of 2 hours following a reactor trip or shutdown. This was added to assure that if the SR was performed very shortly before the plant tripped or shutdown that the SR would be performed much sooner than the required 24 hours. This frequency did not appear in the STS.

4. STS LCO 3.1.4 states MTC shall be within limits stated in the COLR. RTS has been changed to state the single upper limit for Palisades within the LCO. This has been retained from the current license. Palisades has only a single MTC positive limit while other plants have an MTC curve with a bounding negative endpoint. Palisades has no comparable curve to place in the COLR.

5. STS SR 3.1.4.2 has been deleted from the RTS version. This SR calls for mid-cycle MTC testing, and Palisades has retained from its current license performing this test during PHYSICS TESTING. This SR is unnecessary to ensure MTC is bounded for the duration of the fuel cycle. Palisades MTC testing frequency meets ANSI 19.6.1 for PHYSICS TESTING. The Beginning of Core (BOC) MTC value is adequate verification of nuclear methods for predicting MTC. Determining MTC at BOC, yields the greatest challenge to nuclear methods predication due to excessively high boron concentration at BOC. The accurate predication of other nuclear parameters during the fuel cycle: boron rundown, radial peaking, and quadrant tilt ensure nuclear methods are adequate to predict MTC at different points in cycle. MTC is a global effect that could not be predicted accurately if the localized parameters such as radial peaking were not being predicted accurately. Historical data would show missed Estimated Critical Positions, missed radial peaks and etc if SR 3.1.4.2 would be warranted at Palisades.

6. STS 3.1.5 A3.1 and A3.2 have been omitted from RTS. By administrative definition, restore to compliance is always an implied option and does not need to be stated explicitly in an LCO action statement.

7. Condition 3.1.5 B and associated action 3.1.5 B.1 has been added to RTS and does not appear in the STS. This is due to Palisades cruciform rod configuration and Palisades ability to operate with one inoperable control rod. Adding the statement pertaining to declaring a rod inoperable for a specified condition has allowed Palisades to mold the various rod configuration related LCOs to conform as close as possible to the STS.

8. Condition 3.1.5 C from the STS has been omitted from Palisades RTS. This is due to the fact Palisades has no rod motion inhibit function. This is not an aspect of Palisades CRDM system. Primarily this is seen in plants using finger type control rods and associated drives.

9. Condition 3.1.5 D of Palisades RTS states 1 channel of rod position indication inoperable has been added from current licensing base and does not appear in the STS. Palisades is unique compared to many other plants since it has two independent means of rod position indication (i.e. PIP and SPI) while many other plants only have one means of rod position indication.

10. SR 3.1.5.1 rod position verification continuous frequency interval has been omitted from Palisades RTS. It states 15 minutes following any rod motion with the follow up frequency being deleted from the STS. The follow up frequency from the STS was based on plants that have automatic rod control systems while Palisades only has manual control; therefore, follow up rod position verification for this condition is unnecessary for Palisades.

11. Action 3.1.5 E.1 completion time of Palisades RTS has been changed from 6 hours in STS to the current TS value of 12 hours. In the event of a mispositioned rod, the local peaking factors could be substantially elevated in that locality. The much larger reactivity worth and peaking influence from a misaligned cruciform blade as opposed to a CEA would warrant a slower derate ensuring radial peaking remains within design limits and not initiating any core instabilities.

12. SR 3.1.5.3 CEA motion inhibit verification from the STS has been omitted. Palisades has no comparable equipment.

13. Action 3.1.6 B.2 completion time has been changed from 6 hours to 12 hours in the STS; see explanation 11.

14. Condition 3.1.7 A, Regulating rod beyond limit, has been omitted the reference to a transient insertion limit stated in STS. Palisades has no comparable limit. Palisades has only one PDIL curve located in the COLR.

15. Action 3.1.7 A.2.1 has been deleted from STS; see note 6 for explanation.

16. STS condition 3.1.7 B and 3.1.7 C have been omitted from RTS. These conditions reference two different insertion limits, and Palisades only has one insertion limit.

17. STS SR 3.1.7.2 has been omitted from RTS. It calls for the verification of the time period with a rod configuration between the transient and steady state insertion limits. Palisades does not have two insertion limits; therefore, this SR is not applicable to Palisades.

18. STS figure 3.1.7-1, Rod Insertion Limit, has been deleted from RTS. Palisades rod insertion limit figure is contained within the COLR.

19. RTS STE 3.1.8 has added additional LCOs to be suspended in accordance with existing STE 3.10.7. LCOs 3.1.2, 3.1.5 and 3.1.6 have been added to support Palisades PHYSICS TESTING. The current TS counterpart LCOs are included in TS 3.10.7.

20. STS 3.1.9 has been omitted from RTS. This LCO has been omitted since Palisades PHYSICS TESTING program is performed after each refueling and covers both STE 3.1.8 and 3.1.9. STE 3.1.8 has been modified to reflect this. STE 3.1.9 is geared toward mid-cycle MTC testing which Palisades does not perform per current license.

III. NO SIGNIFICANT HAZARDS ANALYSIS:

Each change proposed is classified in Attachment 3 as either ADMINISTRATIVE, RELOCATED, MORE RESTRICTIVE, or LESS RESTRICTIVE.

Analysis of ADMINISTRATIVE, RELOCATED, and MORE RESTRICTIVE Changes:

ADMINISTRATIVE changes and RELOCATED changes move requirements, either within the TS or to documents controlled under 10 CFR 50.59, or clarify existing TS requirements, without affecting their technical content. Since ADMINISTRATIVE and RELOCATED changes do not alter the technical content of any requirements, they cannot involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any previously evaluated, or involve a significant reduction in a margin of safety.

MORE RESTRICTIVE changes only add new requirements, or revise existing requirements to result in additional operational restrictions. Since the TS, with all MORE RESTRICTIVE changes incorporated, will still contain all of the requirements which existed prior to the changes; MORE RESTRICTIVE changes cannot involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any previously evaluated, or involve a significant reduction in a margin of safety.

Analysis of LESS RESTRICTIVE Changes:

The LESS RESTRICTIVE Changes Proposed in This Part of the TSCR are:

1. TS Section 4.10 requires in the event of a reactivity anomaly notify AEC within 24 hours. This requirement has been deleted since STS allow for alternative actions to correct the reactivity anomaly without NRC notification.

2. TS Section 4.10 requires an evaluation to be sent to the AEC within 30 days. This requirement has not been included in RTS since STS allow for alternative actions to correct the anomaly without NRC notification.

3. The proposed RTS add specific Action for failure to meet an LCO where no loss of function occurs, but the existing TS do not contain any. With the exsiting TS an entry into LCO 3.0.3 is required. Each of these changes is identified by an entry of "3.0.3" in the third column of Attachment 6. These changes are considered LESS RESTRICTIVE because they extend the time available to restore compliance to the LCO (the Allowed Outage Time) beyond that allowed by LCO 3.0.3.

Do these LESS RESTRICTIVE changes involve a significant increase in the probability or consequences of an accident previously evaluated?

Changes 1 and 2:

These changes are LESS RESTRICTIVE only in that they delete administrative reporting requirements to inform the NRC of plant condition. This does not impact any plant system design or operating conditions, or any operator action or plant response to any accident scenario currently evaluated. Therefore, these changes do not involve an increase in the probability or consequences of an accident previously evaluated.

Change 3:

These changes are LESS RESTRICTIVE only in their allowance of a longer Allowed Outage Time (AOT) for inoperable equipment. The proposed times are those stipulated in the STS. Changing an AOT alone, does not alter any plant design, operating conditions, operating practices, equipment settings, or equipment capabilities. Since these items are unchanged, changing an AOT would not increase the probability of any accident previously evaluated.

During the evaluation of potential accidents, the safety analyses assume the occurrence of the most limiting single failure. Typically, this single failure is assumed to disable one of the two trains of the equipment installed to mitigate an accident. In accordance with this assumption, the Technical Specifications allow continued operation with required equipment inoperable for limited periods of time (AOTs) only if the assumed level of equipment remains operable. Extending an AOT does not change level of safety equipment required to be available, and does not allow that level to drop below the level assumed to be available in the safety analyses. Therefore, changing an AOT cannot increase the consequences of an accident previously evaluated.

Do these LESS RESTRICTIVE changes create the possibility of a new or different kind of accident from any previously evaluated?

Changes 1 and 2:

These changes are LESS RESTRICTIVE only in that they delete administrative reporting requirements to inform the NRC of plant condition. This does not impact any plant system design or operating conditions, or any operator action or plant response to any accident scenario currently evaluated. Therefore, these changes do not create the possibility of a new or different kind of accident from any previously evaluated.

Change 3:

These changes are LESS RESTRICTIVE only in their allowance of a longer Allowed Outage Time (AOT) for inoperable equipment. The proposed times are those stipulated in the STS. Changing an AOT alone, cannot alter any plant operating conditions, operating practices, equipment settings, or equipment capabilities. Therefore, changing an AOT cannot create the possibility of a new or different kind of accident from any previously evaluated.

Do these LESS RESTRICTIVE changes involve a significant reduction in a margin of safety?

Changes 1 and 2:

These changes are LESS RESTRICTIVE only in that they delete administrative reporting requirements to inform the NRC of plant condition. This does not impact any plant system design or operating conditions, or any operator action or plant response to any accident scenario currently evaluated. Therefore, these changes do not change the margin of safety in any way.

Change 3:

These changes are LESS RESTRICTIVE only in their allowance of an extension to an Allowed Outage Time (AOT) for inoperable equipment. Extending an AOT alone, cannot alter any plant operating conditions, operating practices, equipment settings, or equipment capabilities.

An excessive AOT extension could reduce the margin of safety by allowing operation for an excessive period with less capability to mitigate an accident, or with parameters outside those assumed in the safety analysis. An overly restrictive AOT could also reduce the margin of safety by imposing unnecessary transients on the plant for minor deviations from the requirements of the LCOs.

The existing AOTs were based on the operating experience available when they were added to the TS. Typically this was done during the initial plant licensing, circa 1970. In each of these changes where it is proposed that an AOT be extended, the time proposed is that stipulated in the STS. The AOTs stipulated in the STS are based on a much larger accumulation of operating experience and have been judged by the NRC and by the industry to be appropriate for typical situations. There are no special features of the Palisades plant which would invalidate those judgements for these changes. Therefore, operation of the facility in accordance with the requirements proposed by these changes does not involve a significant reduction in a margin of safety.

IV. CONCLUSION

The Palisades Plant Review Committee has reviewed this part of the STS conversion Technical Specifications Change Request and has determined that proposing this change does not involve an unreviewed safety question. Further, the change involves no significant hazards consideration. This change has been reviewed by the Nuclear Performance Assessment Department.

ATTACHMENT 1

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

3.1 REACTIVITY CONTROL SYSTEMS

Proposed Technical Specifications Pages

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 Shutdown Margin (SDM)- T_{ave} ≥ 525°F

LCO 3.1.1 SDM shall be ≥ 2% Δ ρ with 4 Primary Coolant Pumps operating and 3.75% Δ ρ SDM with < 4 Primary Coolant Pumps operating.

APPLICABILITY: MODE 3, with T_{ave} ≥ 525°F.

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 is within limit.	Within 2 hours following a reactor shutdown or trip. <u>AND</u> 24 hours thereafter.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Shutdown Margin (SDM) - T_{ave}< 525°F

LCO 3.1.2 SDM shall be ≥ 3.5% Δ ρ.

APPLICABILITY: MODE 3, with T_{ave} < 525°F, MODE 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 3.1.2.1 Verify SDM is within limit.	Within 2 hours following a reactor trip or shutdown. AND 24 hours thereafter.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Reactivity Balance

LCO 3.1.3 The core reactivity balance shall be within ± 1% $\Delta \rho$ of predicted values.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Core reactivity balance not within limit.	A.1 Reevaluate core design and safety analysis and determine that the reactor core is acceptable for continued operation.	72 hours
	AND	
	A.2 Establish appropriate operating restrictions and Surveillance Requirements.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	----------NOTES---------- 1. The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 Effective Full Power Days (EFPD) after each refueling. 2. This Surveillance is not required to be performed prior to entry into MODE 2. ----------	
	Verify overall core reactivity balance is within ± 1% $\Delta \rho$ of predicted values.	Prior to entering MODE 1 after each refueling. <u>AND</u> ------NOTE------ Only required after initial 60 EFPD and; ---------- 31 EFPD thereafter.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Moderator Temperature Coefficient (MTC)

LCO 3.1.4 The Moderator Temperature Coefficient (MTC) shall be less positive than 0.5×10^{-4} $\Delta \rho/°F$.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MTC not within limit.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	----------NOTE---------- This Surveillance is not required to be performed prior to entry into MODE 2. ---------- Verify MTC is within the specified limit.	Once prior to exceeding 2% RTP after each refueling.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Rod Alignment

LCO 3.1.5 Each rod shall be OPERABLE and positioned within 8 inches of all other rods in the same group. The rod position deviation alarm shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

---NOTE---
This LCO is not applicable while performing SR 3.1.5.3 (verify rod position deviation alarm operating).

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One control rod inoperable.	A.1	Verify SDM ≥ 2% Δ ρ.	1 hour
	OR		
	A.2	Initiate boration to restore SDM to ≥ 2% Δ ρ.	1 hour
B. One rod misaligned from other rods in the same group by ≥ 8 inches, but ≤ 20 inches.	B.1	Declare the misaligned rods inoperable.	1 hour
	AND		
	B.2.1	Perform SR 3.2.2.1. (Peaking factor verification)	2 hours
	OR		
	B.2.2	Restrict thermal power ≤ 75% RTP.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One control rod misaligned from other rods in the same group by ≥ 20 inches.	C.1 Declare the rod inoperable. AND C.2 Verify SDM ≥ 2% Δ ρ. AND C.3.1 Perform SR 3.2.2.1 (Peaking Factor Verification). OR C.3.2 Restrict THERMAL POWER to ≤ 50% RTP.	1 hour 1 hour 4 hours 4 hours
D. Rod position deviation alarm is inoperable. OR 1 channel of rod position indication inoperable.	D.1 Perform SR 3.1.5.1. (rod position verification)	Once within 15 minutes following any rod motion.
E. Two or more control rods inoperable. OR Required Action and associated Completion Time not met.	E.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify the indicated position of each rod ≤ 8 inches of all other rods in the same group.	12 hours
SR 3.1.5.2	Verify that, each rod's operable position indication channel indicates within 5 inches of each other.	12 hours
SR 3.1.5.3	Demonstrate the rod position deviation alarm is OPERABLE.	92 days
SR 3.1.5.4	Verify rod freedom of movement and tripability by moving each full length rod that is not fully inserted into the reactor core ≥ 5 inches in either direction.	92 days
SR 3.1.5.5	Perform a CHANNEL CALIBRATION on each rod position indicator channel.	18 months
SR 3.1.5.6	Verify each rod full lenght drop time is ≤ 2.5 seconds.	18 months

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Shutdown and Part Length Rod Insertion Limits

LCO 3.1.6 All shutdown and part length rods shall be withdrawn to ≥ 128 inches.

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

----------NOTE----------
This LCO is not applicable while performing SR 3.1.5.4 (Rod Exercising).

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One shutdown or part length rod not within limits.	A.1	Verify SDM ≥ 2% Δ ρ.	1 hour
	OR		
	A.2	Initiate boration to restore SDM to ≥ 2% Δ ρ.	1 hour
	OR		
	A.3	Declare the rod inoperable.	1 hour
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify all shutdown and part length rods are ≥ 128 inches.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Regulating Rod Insertion Limits

LCO 3.1.7 The Power Dependent Insertion Limit (PDIL) alarm circuit shall be OPERABLE. The regulating rod groups shall be limited to the withdrawal sequence and to the insertion limits specified in the COLR.

APPLICABILITY: MODES 1 and 2.

----------NOTE----------
This LCO is not applicable while performing SR 3.1.5.4 (Rod Exercising).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Regulating groups inserted beyond the insertion limit.	A.1.1 Verify SDM $\geq 2\% \Delta \rho$.	2 hours
	OR	
	A.1.2 Initiate boration to restore rods above insertion limit.	2 hours
	AND	
	A.2 Reduce THERMAL POWER to less than or equal to the fraction of RTP allowed by the regulating group position and insertion limits specified in the COLR.	2 hours

(continued)

ACTION (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Regulating groups withdrawn out of sequence.	B.1 Verify SDM $\geq 2\% \Delta \rho$.	1 hour
	OR	
	B.2 Restore rods to within appropriate sequence.	1 hour
C. PDIL alarm circuit inoperable.	C.1 Perform SR 3.1.7.1. (verify group position)	Once within 15 minutes following any rod motion.
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	----------NOTE---------- This Surveillance is not required to be performed prior to entry into MODE 2. ---------- Verify each regulating group position is within its insertion limits.	12 hours
SR 3.1.7.2	Verify PDIL alarm circuit is OPERABLE. (Setpoint verification only)	31 days

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Special Test Exception (STE) - Shutdown Margin (SDM) and Rod Limits

LCO 3.1.8 The requirements of LCOs 3.1.1 and 3.1.2 (SHUTDOWN MARGIN) and LCOs 3.1.5, 3.1.6, and 3.1.7 (Rod Limits) may be suspended for the determination of rod worth, MTC, and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated rod worth is available for trip in OPERABLE control rods.

APPLICABILITY: MODES 2 and 3 during PHYSICS TESTS.

----------NOTE----------
Operation in MODE 3 shall be limited to 6 consecutive hours.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any full length rod not fully inserted and less than the above shutdown reactivity equivalent available for trip insertion. OR All full length rods inserted and the reactor subcritical by less than the above shutdown reactivity equivalent.	A.1 Initiate boration to restore required shutdown reactivity.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Verify that the position of each rod not fully inserted has sufficient negative reactivity addition, to provide adequate SDM.	2 hours
SR 3.1.8.2	Verify that each rod not fully inserted is capable of full insertion when tripped from at least the 50% withdrawn position.	Within 7 days prior to suspending LCOs 3.1.1 or 3.1.2.

ATTACHMENT 2

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

3.1 REACTIVITY CONTROL SYSTEMS

Bases for the Revised Technical Specifications

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 Shutdown Margin (SDM) $T_{ave} \geq 525°F$

BASES

BACKGROUND

The reactivity control systems must be redundant and capable of holding the reactor core subcritical when shutdown under cold conditions, in accordance with GDC 26 (Ref. 1). Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and Anticipated Operational Occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all rods, assuming the single rod of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable rods and soluble boric acid in the Primary Coolant System (PCS). The control rod system can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the control rods, together with the boration system, provide the SDM during power operation and are capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel design limits, assuming that the control rod of highest reactivity worth remains fully withdrawn.

The soluble boron system can compensate for fuel depletion during operation and all Xenon burnout reactivity changes, and maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown rods fully withdrawn and the regulating rods within the limits of LCO 3.1.7, "Regulating Control Rod Insertion Limits." When the plant is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the PCS boron concentration.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes a SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod 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 ≤ 200 cal/gm energy deposition for the 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 reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative Moderator Temperature Coefficient (MTC) this cooldown causes an increase in core reactivity. As PCS temperature decreases, the severity of a MSLB decreases until the MODE 5 conditions are 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, no fuel damage occurs as a result of the post trip return to power, and 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 must also protect against:

a. Inadvertent boron dilution; (Ref. 3)

b. An uncontrolled rod withdrawal from a subcritical or low power condition; (Ref. 5)

c. Startup of an inactive loop; and (Ref. 6)

d. Control rod ejection. (Ref. 7)

These events are described in detail in the respective FSAR section referenced above.

LCO

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, the DNBR limit is exceeded and 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4) may also be exceeded. 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.

The LCO statement contains two different SDM values which account for operation with 4 or less than 4 primary coolant pumps operating. A 2% ρ SDM value is applicable for 4 pumps with $T_{ave} \geq$ 525°F while operation with less than 4 pumps requires 3.75% ρ SDM for this same temperature range. The safety analysis uses these SDM values to support accident scenarios that are initiated under these conditions.

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

BASES

APPLICABILITY The applicability of this LCO is restricted to MODE 3 with T_{ave} > 525°F. LCO 3.1.1, SDM requirements of 2% $\Delta\rho$ pcm is sufficient to support the safety analyses.

The SDM requirements for all MODES of operation are sufficient to meet the assumptions made in the safety analyses discussed above. In MODES 1 and 2, the SDM requirement is satisfied by control rod positioning (regulating and shutdown) as stipulated by LCO 3.1.6, "Control Rod Insertion Limits." It should be noted that if the insertion limits stated in LCOs 3.1.6 and 3.1.7 are violated, SDM is not automatically compromised.

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.

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 PCS 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 SIRWT. The operator should borate with the best source available for the plant conditions.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

SDM is verified by performing a current reactivity balance calculation or by using technical data generated by Reactor Engineering which considers the following 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).

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 Palisades reactivity balance due to the fact that Palisades fuel vendor does not account for Samarium in fuel design calculations performed. 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 performed by the Palisades fuel vendor. Therefore, including Samarium into the SDM calculation would not be correct.

The frequency of SDM verification within 2 hours following a reactor trip or shutdown circumvents the case where the plant trips 1 minute after the 24 hour check was performed. This ensures that following a trip or shutdown the SDM assumed in the safety analysis is verified promptly.

The frequency of 24 hours 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 complete the calculation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26
2. FSAR, Section 14.14
3. FSAR, Section 14.3
4. 10 CFR 100
5. FSAR, Section 14.2
6. FSAR, Section 14.8
7. FSAR, Section 14.16

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Shutdown Margin (SDM)< 525°F

BASES

BACKGROUND

A detailed BACKGROUND description for B 3.1.2, "Shutdown Margin (SDM) T_{ave} < 525°F" is included in the description of B 3.1.1 Bases.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes a SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs with the assumption of the highest worth rod stuck out following a reactor trip. Specifically, for LCO 3.1.2, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

The acceptance criteria for the SDM requirements are that the 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, fuel centerline temperature limits for AOOs, and ≤ 200 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.

An inadvertent boron dilution is a moderate frequency incident as defined in Reference 2. The core is initially subcritical with all rods inserted. A Chemical and Volume Control System malfunction occurs, which causes unborated water to be pumped to the PCS via three charging pumps.

The reactivity change rate associated with boron concentration changes due to inadvertent dilution is within the capabilities of operator recognition and control.

BASES

LCO

The accident analysis has shown that the required SDM is sufficient to avoid unacceptable consequences to the fuel or PCS as a result of the events addressed above.

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.

By definition of SHUTDOWN MARGIN the stuck rod worth can be relaxed from the LCO if both the synchro and reed switch position indication systems can verify All Rods In (ARI) condition.

APPLICABILITY

In MODE 3 with T_{ave}, < 525°F, and MODES 4, and 5 the SDM requirements of LCO 3.1.2 are sufficient to meet the assumptions used in the safety analyses.

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.

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 PCS 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 SIRWT. The operator should borate with the best source available for the plant conditions.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

A detailed SURVEILLANCE REQUIREMENTS description for B 3.1.2, "Shutdown Margin (SDM) T_{ave} < 525°F" is included in the description of B 3.1.1 Bases.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26

2. FSAR, Section 14.3

B 3.1 **REACTIVITY CONTROL SYSTEMS**

B 3.1.3 Reactivity Balance

BASES

BACKGROUND

According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that, subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM) $T_{ave} \geq 525°F$") in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the critical boron curve, which provides an indication of the soluble boron concentration in the Primary Coolant System (PCS) versus cycle burnup. Periodic measurement of the PCS boron concentration for comparison with the predicted value with other variables fixed (such as control rod height, temperature, pressure, and power) provides a convenient method of ensuring that core reactivity is within design expectations, and that the calculational models used to generate the safety analysis are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the PCS boron concentration.

BASES

BACKGROUND (continued)

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the PCS boron concentration is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The critical boron curve is based on steady state operation at RTP. Therefore, deviations from the predicted critical boron curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

APPLICABLE SAFETY ANALYSES

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or control rod ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the PCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for calculational models used to predict core reactivity. If the measured and predicted PCS boron concentrations for identical core conditions at Beginning Of Cycle (BOC) are not within design tolerances, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted critical boron curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The normalization of predicted PCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

LCO

The reactivity balance limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the nuclear design methodology are larger than expected. A limit on the reactivity balance of ± 1% $\Delta\rho$ has been established, based on engineering judgment. A ± 1% $\Delta\rho$ deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

When measured core reactivity is within ±1% $\Delta\rho$ of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limits are normally detected by comparing predicted and measured steady state PCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the PCS boron concentration is unlikely.

APPLICABILITY

The limits on core reactivity must be maintained during MODES 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODES 3, 4, and 5 because the reactor is shutdown and the reactivity balance is not changing.

BASES

APPLICABILITY (continued) In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. A SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, or control rod replacement, or shuffling).

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 72 hours is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of PCS boron concentration sampling, then a recalculation of the PCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized, and power operation may continue. If operational restrictions or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 72 hours is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

BASES

ACTIONS (continued)

B.1

If the core reactivity cannot be restored to within the 1% $\Delta\rho$ limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at lease MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then boration required by Action 3.1.1 A.1 would occur. The allowed Completion Time 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 3.1.3.1

Core reactivity is verified by periodic comparisons of measured and predicted PCS boron concentrations. The comparison is made considering that other core conditions are fixed or stable including control rod position, moderator temperature, fuel temperature, fuel depletion, and xenon concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. The SR is modified by three Notes. Note 1 in the Surveillance column indicates that the normalization of predicted core reactivity to the measured value must take place within the first 60 Effective Full Power Days (EFPD) after each refueling. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. The required subsequent Frequency of 31 EFPD following the initial 60 EFPD after entering MODE 1, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (e.g., QPTR, etc.) for prompt indication of an anomaly. A second Note, "only required after 60 EFPD," is added to the Frequency column to allow this. Note 2 in the Surveillance column indicates that the performance of SR 3.1.3.1 is not required prior to entering MODE 2.

This Note is required to allow a MODE 2 entry to verify core reactivity, because LCO Applicability is for MODES 1 and 2.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29

2. FSAR, Section 3.3

B 3.1 **REACTIVITY CONTROL SYSTEMS**

B 3.1.4 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to GDC 11 (Ref. 1), the reactor core and its interaction with the Primary Coolant System (PCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended or rapid reactivity increases.

The MTC relates a change in core reactivity to a change in primary coolant temperature. A positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature. The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result. The same characteristic is true when the MTC is positive and coolant temperature decreases occur.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the Beginning Of Cycle (BOC) MTC is less positive than that allowed by the LCO. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons (lumped burnable poison assemblies) to yield an MTC at the BOC within the range analyzed in the plant accident analysis. The End Of Cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles that are designed to achieve high burnups or that have changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOC limit.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and

b. The MTC must be such that inherently stable power operations result during normal operation and during accidents, such as overheating and overcooling events.

Reference 2 contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions, such as very large soluble boron concentrations, to ensure the accident results are bounding (Ref. 3).

Accidents that cause core overheating, either by decreased heat removal or increased power production, must be evaluated for results when the MTC is positive. Reactivity accidents that cause increased power production include the control rod withdrawal transient from either zero or full THERMAL POWER. The limiting overheating event relative to plant response is based on the maximum difference between core power and steam generator heat removal during a transient. The most limiting event with respect to a positive MTC is a control rod withdrawal accident from zero power, also referred to as a startup accident (Ref. 4).

Accidents that cause core overcooling must be evaluated for results when the MTC is most negative. The event that produces the most rapid cooldown of the PCS, and is therefore the most limiting event with respect to the negative MTC, is a Main Steam Line Break (MLSB) event. Following the reactor trip for the postulated EOC SLB event, the large moderator temperature reduction combined with the large negative MTC may produce reactivity increases that are as much as the shutdown reactivity. When this occurs, a substantial fraction of core power is produced with all control rods inserted, except the most reactive one, which is assumed withdrawn. Even if the reactivity increase produces slightly subcritical conditions, a large fraction of core power may be produced through the effects of subcritical neutron multiplication.

MTC values are bounded in reload safety evaluations assuming steady state conditions at BOC.

BASES

LCO

LCO 3.1.4 requires the MTC to be less positive than the specified limit of 0.50×10^{-4} $\Delta\rho$°F to ensure the core safety evaluation. MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation. The limit on a positive MTC ensures that core overheating accidents will not violate the accident analysis assumptions.

MTC is a core physics parameter determined by the fuel and fuel cycle design and cannot be easily controlled once the core design is fixed.

APPLICABILITY

In MODE 1, the limits on the MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2, the limits must also be maintained to ensure startup and subcritical accidents, such as the uncontrolled control rod withdrawal, will not violate the assumptions of the accident analysis. In MODES 3, 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents (DBAs) using the MTC as an analysis assumption are initiated from these MODES. However, the variation of the MTC, with temperature in MODES 3, 4, and 5, for DBAs initiated in MODES 1 and 2, is accounted for in the subject accident analysis.

The variation of the MTC, with temperature assumed in the safety analysis, is accepted as valid once the BOC measurement is used for normalization.

ACTIONS

A.1

MTC is a function of the fuel and fuel cycle designs, and cannot be controlled directly once the designs have been implemented in the core. If MTC exceeds its limits, the reactor must be placed in MODE 3. This eliminates the potential for violation of the accident analysis bounds. The associated Completion Time of 6 hours is reasonable, considering the probability of an accident occurring during the time period that would require an MTC value within the LCO limits, and the time for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

The SR for MTC validation at the beginning of each fuel cycle provides adequate values used in the safety analysis. This confirms the nuclear methods used for the prediction of both BOC and EOC MTC values. This SR is performed in accordance with startup physics testing following each refueling outage. The MTC changes smoothing from the most positive (least negative) to the most negative value during fuel cycle operation as the PCS boron concentration is reduced to compensate for fuel depletion. MTC values may be extrapolated and compensated to permit direct comparison to the specified MTC limits.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11
2. FSAR, Section 14.1 and 14.14
3. FSAR, Section 3.3
4. FSAR, Section 14.2

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Control Rod Alignment

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and regulating rod is an initial assumption in all safety analyses that assume 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 applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10 and GDC 26 (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

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.

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 46 inches per minute.

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 control rods provide the required reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating control 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 the Plant Process Computer (PPC) synchro based rod position indication system and the PPC reed switch based position indication system.

BASES

BACKGROUND (continued)

The PPC synchro based position indication system measures the phase angle of a synchro connected to the CRDM rack. Full control rod travel is less than 1 turn of the synchro. Each control rod has its own synchro which is monitored continuously. Accuracy of this system is highly precise (± .1 inch).

The reed switch position indication system provides highly accurate rod position, but at a lower precision than the synchros. This system is based on a voltage dividing network consisting of a series of magnetic reed switches and resistors. The reed switches are spaced along a tube with a center to center spacing distance of 1.5 inches. The resolution of the SPI is ± 2 inches for conservatism.

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). 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 system, or by operator error. A stuck rod may be caused by mechanical jamming.

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

a. There shall be no violations of:

   1. Specified acceptable fuel design limits, or

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

b. The core must remain subcritical after accident transients.

Three types of misalignment are distinguished in the safety analysis (Ref. 1). 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 misalignment 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 misalignment occurs when one rod drops partially or fully into the reactor core.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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).

Misalignment of a control rod between 8 and 20 inches is bounded in the safety analysis by the dropped rod event.

Another type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth control rod also fully withdrawn (Ref. 4). Since the control rod drop incidents result in the most rapid approach to Specified Acceptable Fuel Design Limits (SAFDLs) caused by a rod misoperation, the accident analysis analyzed a single full length rod drop. The most rapid approach to the DNBR SAFDL may be caused by a single full length rod drop.

In the case of the full length control rod drop, a prompt decrease in core average power and a distortion in radial power are initially produced, which when conservatively coupled, results 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.

LCO

The limits on shutdown and regulating control 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 shutdown the reactor. The OPERABILITY requirements also ensure that the control rod banks maintain the correct power distribution and control rod alignment.

The requirement is to maintain the control rod alignment to within 8 inches between any control rod and its group. This assumes a total misalignment from fully withdrawn to fully inserted. This case bounds the safety analysis for a single rod in any intermediate position.

BASES

LCO (continued) Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMS, all 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 control rods are fully inserted at or below the Lower Electrical Limit (LEL), and the reactor is shutdown 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) — $T_{ave} \geq 525°F$," for SDM in MODES 3, 4, and 5, and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

ACTIONS

A.1

A control rod is considered inoperable if it cannot be moved by its operator or if it cannot be tripped. In the event a control rod is declared inoperable, a SDM verification shall be performed in accordance with ST 3.1.1.1. This will ensure that the core is operated within the assumptions made in the safety analysis. A time period of 1 hour is adequate to perform this calculation.

A.2

In the event a control rod is declared inoperable, SHUTDOWN MARGIN requirements shall be maintained by increasing the boron concentration by an amount equivalent in reactivity to the worth of the inoperable control rod. The time period of 1 hour is adequate to initiate boration to maintain SHUTDOWN MARGIN.

BASES

ACTIONS (continued)

B.1, B.2.1, and B.2.2

A control rod misaligned by more than 8 inches, but less than 15 inches shall be declared inoperable within 1 hour if the rod cannot be restored within limits. In addition, if the misaligned rod is not restored within 1 hour SR 3.2.2.1 shall be performed to verify hot channel factors are within design limits. In the event, the above actions cannot be met, THERMAL POWER shall be reduced to ≤ 75% RTP within 2 hours. This will ensure that hot channel factors are not exceeded. Continued operation with the effected rod fully inserted will only be permitted if the hot channel factors, SDM and ejected rod worths are satisfied.

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

A control rod misaligned more than 20 inches from other rods in the same group shall be declared inoperable within 1 hour if it cannot be restored within limits. In the event, the effected rod is declared inoperable, SDM should be verified to within limits. In addition, THERMAL POWER shall be restricted to ≤ 50% RTP within 4 hours in the event peaking factors are being violated when SR 3.2.2.1 is performed. This will ensure that the hot channel factors will not be violated due to a skewed power distribution and subsequent power peaking. At ≤ 50% RTP ample thermal margin exists to ensure hot channel factors are within design limits and meet the assumptions stated in the safety analysis. The time restraints specified are adequate to achieve the required actions in a safe manner.

D.1 and E.1

In the event, the rod position deviation alarm is inoperable or 1 channel of position indication in inoperable SR 3.1.5.1 shall be performed within 15 minutes following rod motion. Thin is adequate assurance that the rod configuration will not change since Palisades only has manual rod motion as opposed to automatic rod control systems used in other plants. The rod position verification will ensure control rods are within limits. This ensures the assumptions in the safety analysis have been met.

If two or more control rods are inoperable or the required action and associated completion times are not met, the plant shall be in MODE 3 within 12 hours. The time period specified allows the plant to derate in a safe manner. The time period to bring the plant to MODE 3 is needed to mitigate any potential power peaks due to a skewed power distribution brought on by a mispositioned cruciform blade and inherent to the plants low leakage core design.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

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 not in 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, and protection can be provided by the control rod deviation circuits.

SR 3.1.5.2

OPERABILITY of at least 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. The control rod "full in" and "full out" corresponds to the Upper electrical limits and the lower electrical limit which provide an additional independent 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 circuits.

SR 3.1.5.3

Demonstrating the control rod deviation circuit is OPERABLE verifies the circuit 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, and protection can be provided to ensure nuclear design limits are not violated.

SR 3.1.5.4

Verifying each 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 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 ≥ 5 inches in either direction 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.5.5

Performance of a CHANNEL CALIBRATION of each 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. Since the Surveillance must be performed when the reactor is shutdown, 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 Indication System.

SR 3.1.5.6

Verification of full length control rod drop times determined that the maximum control rod drop time permitted is consistent with the assumed drop time used in that safety analysis (Ref. 6). Measuring drop times prior to reactor criticality, after reactor vessel head installation removal, ensures that reactor internals and CRDMs will not interfere with control rod motion or drop time and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. Individual 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. The drop time includes the time span from trip signal actuation to 90% rod insertion.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26
2. 10 CFR 50.46
3. FSAR, Section 14.4
4. FSAR, Section 14.4.8
5. FSAR, Section 14.4.9

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown control 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 control rod worth, 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" (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 control rod insertion have been established, and all control rod positions are monitored and controlled during power operation to ensure that the reactivity limits, ejected control rod worth, and SDM limits are preserved.

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

The design calculations are performed with the assumption that the shutdown rods are withdrawn prior to the regulating rods. The shutdown control 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 control rods are controlled manually or automatically by the control room operator. During normal plant operation, the shutdown rods are fully withdrawn. The shutdown rods must be completely withdrawn from the core prior to withdrawing any regulating rods during an approach to criticality. The shutdown rods are then left in this position until the reactor is shutdown. They affect core power, burnup distribution, and add negative reactivity to shutdown the reactor upon receipt of a reactor trip signal.

BASES

APPLICABLE SAFETY ANALYSES

Accident analysis assumes that the shutdown control rods 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 a very 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 control rod, are assumed to insert into the core. The shutdown and regulating control rods shall be at their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The regulating rods may be partially inserted in the core as allowed by LCO 3.1.7, "Regulating Control Rod Insertion Limits." The shutdown rod insertion limit is established to ensure that a sufficient amount of negative reactivity is available to shutdown the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)— $T_{ave} \geq 525°F$") following a reactor trip from full power. The combination of regulating rods and shutdown rods (less the most reactive control rod, which is assumed to be 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). The shutdown rod insertion limit also limits the reactivity worth of an ejected shutdown rod.

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 rod insertion limits affect the safety analyses involving core reactivity, ejected rod worth, and SDM (Ref. 3).

BASES

LCO

The shutdown rods must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shutdown the reactor and maintain the required SDM following a reactor trip.

APPLICABILITY

The shutdown rods must be within their insertion limits, with the reactor in MODES 1 and 2. The Applicability in MODE 2 begins anytime any regulating rod is withdrawn above 5 inches. This ensures that a sufficient amount of negative reactivity is available to shutdown the reactor and maintain the required SDM following a reactor trip. In MODES 1 and 2, if shutdown rods are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation (considering the listed reactivity effects in Bases Section SR 3.1.1.1). In MODE 3, 4, 5, or 6, the shutdown rods are fully inserted in the core and contribute to the SDM. Refer to LCOs 3.1.1 and 3.1.2, "SHUTDOWN MARGIN (SDM)— $T_{ave} \leq 525°$ F," for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

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 control rods to move, and requires the shutdown rods to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1, A.2 and A.3

Prior to entering this condition, the shutdown rods were fully withdrawn. If a shutdown rod is then inserted into the core, its potential negative reactivity is added to the 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 control rod(s) is not restored to within limits prior to this time.

In the event one shutdown or part length rod is not within limits and other actions have not been performed, the rod shall be declared inoperable within 1 hour. This places the condition into LCO 3.1.5, "Rod Operability and Alignment" which allows for SDM verification of boration. This ensures that the rod is no longer credited and safety analysis assumptions are still preserved.

BASES

ACTIONS (continued)

When a shutdown or part length rod is not within limits, the rod can be declared inoperable within 1 hour time period. This action is performed if other options have not been elected. This reflects Palisades ability to run with an inoperable control rod. The 1 hour time period reflects the importance of addressing this condition when entered.

B.1 and B.2

When Required Action cannot be met or completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 12 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems. Due to Palisades low leakage core design the 12 hour Completion Time is required to not initiate any instability in the core while deescalating. A misaligned rod could cause a very skewed power distribution; therefore, a slower derate would be prudent.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Since the shutdown rods are positioned manually by the control room operator, verification of shutdown rod position at a Frequency of 12 hours is adequate to ensure that the shutdown rods 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 rods.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26
2. 10 CFR 50.46
3. FSAR, Section 14.2

B 3.1 **REACTIVITY CONTROL SYSTEMS**

B 3.1.7 Regulating Rod Insertion Limits

BASES

BACKGROUND

The insertion limits of the regulating rods are initial assumptions in all safety analyses that assume 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" (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 insertion 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, ejected control 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 control rod worth and control rod position (integral control 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 changing reactivity very quickly (compared to borating or diluting).

BASES

BACKGROUND (continued) 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, "Regulating Rod Insertion Limits"; LCO 3.2.4, " Quadrant POWER TILT (T_q)"; and LCO 3.2.5, "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)"); assembly radial peaking factor F_r^A LCO 3.2.2, "Assembly Radial Peaking; and total integrated radial peaking factor (F^T_r) (LCO 3.2.3, "Total Integrated Radial Peaking Factor (F^T_r)") limits in the COLR. Operation within the LHR limits given in the COLR prevents power peaks that would exceed the Loss Of Coolant Accident (LOCA) limits derived by the Emergency Core Cooling System analysis. Operation within the F^{rA} and F^T_r limits given in the COLR prevents Departure from Nucleate Boiling (DNB) during a loss of forced reactor coolant flow accident. In addition to the LHR, F^A_r, and F^T_r limits, certain reactivity limits are preserved by regulating rod insertion limits. The regulating rod insertion limits also restrict the ejected control rod worth to the values assumed in the safety analysis and preserve the minimum required SDM in MODES 1 and 2. The ejected rod case is limited to the reactivity worth for the highest worth rod ejected from the PDIL limit, thus limiting the maximum possible reactivity excursion.

The establishment of limiting safety system settings and LCOs requires that the expected long and short term behavior of the radial peaking factors be determined. The long term behavior relates to the variation of the steady state radial peaking factors with core burnup and is affected by the amount of control rod insertion assumed, the portion of a burnup cycle over which such assertion is assumed and the expected power level variation throughout the cycle.

The long term behavior relates to steady state full power operation. This condition of operation and resultant radial peaking factors are ensured by LCO 3.1.6 and SR 3.2.2.1 radial peaking factor verification. This ensures that the core is operated within nuclear design limits and verifies assumptions assumed in the safety analysis.

BASES

BACKGROUND (continued)

The short term behavior relates to transient perturbations to the steady state radial peaking factors. The magnitudes of such perturbations depend upon the expected use of control rods during transient mitigation. The PDIL curve stated on the COLR dictates the acceptable control rod positioning for anticipated power maneuvers and transient mitigation within the limits. The PDIL limitations stated in the COLR reflect the assumptions made in the safety analyses. This ensures that radial peaking is not violated during power level maneuvering or transient mitigation.

The regulating rod insertion and alignment limits are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the regulating bank insertion limits control the reactivity that could be added in the event of a control rod ejection accident, and the shutdown and regulating bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the primary coolant in the event of a LOCA, loss of flow, ejected control rod, or other accident requiring termination by a Reactor Protection System trip function.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition I) and anticipated operational occurrences (Condition II). The acceptance criteria for the regulating rod insertion, ASI, and T_q LCOs are such as to preclude core power distributions from occurring that would violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed a limit of 2200°F, 10 CFR 50.46 (Ref. 2);

b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot channel control rod in the core does not experience a DNB condition.

c. During an ejected control rod accident, the fission energy input to the fuel must not exceed 200 cal/gm (Ref. 3); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM, with the highest worth control rod stuck fully withdrawn, GDC 26 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES (continued)

Regulating rod position, ASI, and T_q are process variables that together characterize and control the three dimensional power distribution of the reactor core.

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 insertion limits, so that the allowable inserted worth of the control rods is such that sufficient reactivity is available to shutdown the reactor to hot zero power. SDM assumes the maximum worth control rod remains fully withdrawn upon trip (Ref. 4).

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 control rod worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected control rod worth.

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

LCO

The limits on regulating rods 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 control rod worth is maintained, and ensuring adequate negative reactivity insertion on trip. The overlap between regulating banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during regulating rod motion.

The Power Dependent Insertion Limit (PDIL) alarm circuit is required to be OPERABLE for notification that the rods are outside the required insertion limits. When the PDIL alarm circuit is inoperable, the verification of control rod positions is increased to ensure improper control rod alignment is identified before unacceptable flux distribution occurs.

BASES

APPLICABILITY The regulating rod 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 control 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 control 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.

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 control rods to move, and requires the regulating rods to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1.1, A.1.2, and A.2

Operation beyond the insertion limit may result in a loss of SDM and excessive peaking factors. If the regulating rod 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 insertion limit should not be violated during normal operation; this violation, however, may occur during transients when the operator is manually controlling the control rod 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 within the limits or to reduce THERMAL POWER to less than or equal to that allowed for the actual control rod insertion limit. A two hour limit provides a reasonable time to accomplish this, allowing the operator to deal with current plant conditions while limiting peaking factors to acceptable levels.

B.1

Verifying SHUTDOWN MARGIN within 1 hour allows for the ensurance that the assumptions made in the safety analysis are maintained. A 1 hour completion time is an adequate period of time to verify SDM is within limits.

In addition, the out of sequence rods must be restored within appropriate sequence within one hour. This places the plant back into the configuration assumed by the Palisades safety analysis.

BASES

ACTIONS (continued)

C.1

The required Completion Time of 2 hours from initial discovery of a regulating control rod group outside the limits until its restoration to within PDIL, shown on the figures in the COLR, allows sufficient time for borated water to enter the Primary Coolant System from the chemical addition and makeup systems, and to cause the regulating control rods to withdraw to the acceptable region. It is reasonable to continue operation for 2 hours after it is discovered. This Completion Time is based on limiting the potential xenon redistribution, the low probability of an accident, and the steps required to complete the action.

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 3.1.7.1

With the PDIL alarm circuit OPERABLE, verification of each regulating rod group position every 12 hours is sufficient to detect control 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 control rod group positions available to the operator in the control room.

SR 3.1.7.1 is modified by a Note indicating that entry is allowed into MODE 2 without having performed the SR. This is necessary, since the unit must be in the applicable MODES in order to perform Surveillances that demonstrate the LCO limits are met.

SR 3.1.7.2

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.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26
2. 10 CFR 50.46
3. FSAR, Section 14.4.2
4. FSAR, Section 14.4.8

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Special Test Exception Start-Up Physics Testing (SPT)

BASES

BACKGROUND

The primary purpose of the SDM STE is to permit relaxation of existing LCOs to allow the performance of certain PHYSICS TESTS. These tests are constructed to determine the control rod worth and SDM.

Section XI of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants" (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59, "Changes, Tests, and Experiments" (Ref. 2).

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in design and analysis;

c. Verify assumptions used for predicting plant response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that operating and emergency procedures are adequate.

To accomplish these objectives, testing is required prior to initial criticality, after each refueling shutdown, and during startup, low power operation, power ascension, and at power operation. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions, and that the core can be operated as designed (Ref. 4).

BACKGROUND (continued)

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long term power operation. Examples of PHYSICS TESTS include determination of critical boron concentration, control rod group worths, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because adequate limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

Reference 5 defines the requirements for initial testing of the facility, including PHYSICS TESTS. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits of all LCO, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. As long as the Linear Heat Rate (LHR) remains within its limit, fuel design criteria are preserved.

a. LCO 3.1.1, "SHUTDOWN MARGIN (SDM)— $T_{ave} \geq 525°F$"; and

b. LCO 3.1.2, "SHUTDOWN MARGIN (SDM)—T_{ave} 525°F"; and

c. LCO 3.1.5, "Control Rod Alignment";

d. LCO 3.1.6, "SHUTDOWN Rod Insertion Limits";

e. LCO 3.1.7, "Regulating Rod Insertion Limits."

Therefore, this LCO places limits on the minimum amount of control rod worth required to be available for reactivity control when control rod worth measurements are performed.

APPLICABLE SAFETY ANALYSES (continued)

The individual LCOs cited above govern SDM control rod group height, insertion, and alignment. Additionally, the LCOs governing Primary Coolant System (PCS) flow, reactor inlet temperature, and pressurizer pressure contribute to maintaining Departure from Nucleate Boiling (DNB) parameter limits. The initial condition criteria for accidents sensitive to core power distribution are preserved by the LHR and DNB parameter limits. The criteria for the Loss Of Coolant Accident (LOCA) are specified in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 6). The criteria for the loss of forced reactor coolant flow accident are specified in Reference 7. Operation within the LHR limit preserves the LOCA criteria; operation within the DNB parameter limits preserves the loss of flow criteria.

SRs are conducted as necessary to ensure that LHR and DNB parameters remain within limits during PHYSICS TESTS. Performance of these SRs allows PHYSICS TESTS to be conducted without decreasing the margin of safety.

Requiring that shutdown reactivity equivalent to at least the highest estimated control rod worth (of those control rods actually withdrawn) be available for trip insertion from the OPERABLE control rod provides a high degree of assurance that shutdown capability is maintained for the most challenging postulated accident, a stuck control rod. Since LCO 3.1.1 is suspended, however, there is not the same degree of assurance during this test that the reactor would always be shut down if the highest worth control rod was stuck out and calculational uncertainties or the estimated highest control rod worth was not as expected (the single failure criterion is not met). This situation is judged acceptable, however, because specified acceptable fuel damage limits are still met. The risk of experiencing a stuck control rod and subsequent criticality is reduced during this PHYSICS TEST exception by the requirements to determine control rod positions every 2 hours; by the trip of each control rod to be withdrawn 7 days prior to suspending the SDM; and by ensuring that shutdown reactivity is available, equivalent to the reactivity worth of the estimated highest worth withdrawn control rod (Ref. 5).

PHYSICS TESTS include measurement of core parameters or exercise of control components that affect process variables. Among the process variables involved are total assembly radial peaking total pin radial peaking, T_q and ASI, which represent initial condition input (power peaking) to the accident analysis. Also involved are the shutdown and regulating rods, which affect power peaking and are required for shutdown of the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

BASES

APPLICABLE SAFETY ANALYSES (continued)

PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Policy Statement.

LCO

This LCO provides that a minimum amount of control rod worth is immediately available for reactivity control when control rod worth measurement tests are performed. The STE is required to permit the periodic verification of the actual versus predicted core reactivity conditions. The SDM requirements of LCOs 3.1.1 , 3.1.2, 3.1.5, 3.1.6, and 3.1.7 may be suspended.

APPLICABILITY

This LCO is applicable in MODES 2 and 3. Although control rod worth testing is conducted in MODE 2, sufficient negative reactivity is inserted during the performance of these tests to result in temporary entry into MODE 3. Because the intent is to immediately return to MODE 2 to continue control rod worth measurements, the STE allows limited operation to 6 consecutive hours in MODE 3, as indicated by the Note, without having to borate to meet the SDM requirements of LCO 3.1.1.

ACTIONS

A.1

With any control rod not fully inserted and less than the minimum required reactivity equivalent available for insertion, or with all control rods inserted and the reactor subcritical by less than the reactivity equivalent of the highest worth control rod, restoration of the minimum SDM requirements must be accomplished by increasing the PCS boron concentration. The required Completion Time of 15 minutes for initiating boration allows the operator sufficient time to align the valves and start the boric acid pumps and is consistent with the Completion Time of LCO 3.1.1.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification of the position of each partially or fully withdrawn full length or part length control rod is necessary to ensure that the minimum negative reactivity requirements for insertion on a trip are preserved. A 2 hour Frequency is sufficient for the operator to verify that each control rod position is within the acceptance criteria.

SR 3.1.8.2

Prior demonstration that each control rod to be withdrawn from the core during PHYSICS TESTS is capable of full insertion, when tripped from at least a 50% withdrawn position, ensures that the control rod will insert on a trip signal. The 7 day Frequency ensures that the control rods are OPERABLE prior to reducing SDM to less than the limits of LCO 3.1.1.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI
2. 10 CFR 50.59
3. Regulatory Guide 1.68, Revision 2, August 1978
4. ANSI/ANS-19.6.1-1985, December 13, 1985
5. FSAR, Chapter 14.4.8
6. 10 CFR 50.46

ATTACHMENT 3

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

3.1 REACTIVITY CONTROL SYSTEMS

Comparison of Existing and Revised Technical Specifications

Palisades Tech Spec Requirement List. Corrected through Amendment 170

A list of the existing Palisades Tech Specs (TS) correlated to Palisades Revised Technical Specifications (RTS).

First Column; Existing Palisades Tech Spec (TS) number

Each numbered TS item is listed in the left-most column. Items which contain more than one requirement are listed once for each requirement.

Second Column; Palisades Revised Tech Spec (RTS) number

The nearest corresponding numbered RTS item is listed in the second column. If the item does not appear in RTS, it is noted as 'Deleted' or 'Relocated.'

Deleted is used where an item has been eliminated as a tech spec, ie deleting, iaw GL 84-15, the requirement to test a D.G. when an ECCS pump in the opposite train becomes inoperable.

Relocated is used where an item has been moved to a controlled program or document because it does not meet the "Criteria" of 10 CFR 50.36(2)(c)(ii).

Where an item is relocated or deleted, the number of the associated RTS section has been added to allow sorting the list by section number. Relocated items, such as heavy load restrictions, which are not associated with any particular RTS section are arbitrarily assigned the number 5.0.

Third Column; TS Item Description

An abbreviation of the TS requirement appears in the third column. Each item is identified as: LCO, ACTION, SR, Admin, Exception, etc. Some items are implied, rather than explicit, ie a LCO is implied when an ACTION exists without a stated LCO.

Description Key; TS requirement type:	Column 3 syntax:
Safety Limit	SL: Safety limit; Applicable conditions
Surveillance Requirement	SR: Equipment to be tested; Test description; Frequency
Limiting Safety Setting	LSS: RPS Trip Channel & required setting
Limiting Condition for Operation	LCO: Equipment to be operable; Applicable conditions
Action	ACTN: Condition requiring action; Required action; Completion time
Administrative Requirement	ADMN: Administrative requirement
Permitted Instrument Bypass	Byps: Bypassable component; conditions when bypass permitted
Defined Term	DEF: Name of defined item
Exception to other Requirement	XCPT: Excepted spec or condition; Applicable conditions
Descriptive material	DESC: Subject matter
Table	TBL: Table

Forth Column; Classification of Changes:

Each change is identified as ADMINISTRATIVE, RELOCATED, MORE RESTRICTIVE, or LESS RESTRICTIVE.

Fifth Column; Discussion of Changes:

Each change is discussed briefly.

TS Number	RTS Number	TS requirement description	Classification and Description of Changes	
3.1.3.c	3.1.2	LCO: Rx subcritical by required amount; <525°F	LESS RESTRICTIVE:	Using STS requirements.
3.1.3.d	3.1 Deleted	LCO: Only 1 rod out w/o bubble & normal lvl	LESS RESTRICTIVE:	Protection provided by new SDM definition and LCO 3.3.1.
3.1.3.e	3.1 Deleted	LCO: No dilution w/o bubble & normal lvl	LESS RESTRICTIVE:	SDM requirements are given in LCO 3.1.1 and LCO 3.1.2.
3.6.1.c	3.1.2, 3.3.1 3.6.1, 3.9.1	LCO: No reactivity addn; w/o integrity	ADMINISTRATIVE:	Proposed RTS do not explicitly prohibit dilution or multiple rod withdrawal without containment integrity; they do, however, contain requirements which accomplish the same thing: LCO 3.3.1 effectively prohibits withdrawal capability of more than one control rod while in conditions where containment Operability (integrity is not required. LCO 3.3.1 requires the Low Flow trip to be operable whenever more than one rod is capable of being withdrawn; with less than four pumps operating a low flow trip would block all rod withdrawal. Although not a RTS requirement, four pumps cannot be operated simultaneously until PCS temperature is well above 200°F, where containment integrity is required. LCO 3.9.1 required sufficient boron concentration to maintain K_{eff} <.95 and prohibits dilution if this LCO is not met; LCO 3.1.2 requires a SDM ≥3.75% while in MODE 5, and requires immediate boration if this LCO is not met. LCO 3.6.1 requires the containment to be Operable above MODE 5.
3.10.1	3.1.1 & .2	Shutdown Margin Requirements		
3.10.1.a	3.1.1	LCO: SDM = 2% >525°F; W/4 PCPs	ADMINISTRATIVE:	The SDM value is unchanged with the exception of the reference to 4 pump operation is not mentioned since the applicability range bounds the SDM value for 4 pump operation.
3.10.1.b	3.1.1	LCO: SDM >3.75%; W/<4 PCPs >525°F	ADMINISTRATIVE:	The SDM value is unchanged <4 pump operation is bounded by applicability range specified. This is contained within LCO 3.1.2 SDM Tave <525 F.
3.10.1.b	3.1.1	ACTN: <4 PCPs & >525°F; Borate to req SDM	ADMINISTRATIVE:	Reworded to conform with standard with technical content remaining the same.
3.10.1.c	3.1.2	LCO: Boron >CSD boron; <525°F w/≥2810 gpm	ADMINISTRATIVE:	The condition of MODE 3 <525°F applies to LCO 3.1.2 RTS. This applicability range bounds the SDM value required. For this condition the SDM value is 3750 pcm SDM. The requirement of being >CSD boron when Tave <525°F is incorporated in the additional SDM required in this

TS Number	RTS Number	TS requirement description	Classification and Description of Changes	
				applicability range. Therefore, the boron and flow requirements stated are not applicable.
3.10.1.d	3.1 Deleted	ACTN: Untrippable rod; Verify SDM	ADMINISTRATIVE:	Reactivity consideration pertaining to an inoperable rod is accounted for the in RTS definition of SHUTDOWN MARGIN Section 1.1. Therefore this statement has been deleted since it is no longer applicable.
3.10.1.e	3.1.5.6	LCO: Each rod drop time <2.5 sec	ADMINISTRATIVE:	This statement remains completely intact from the current license in SR 3.1.5.5.
3.10.2		Deleted		
3.10.3	3.1.6	LCO: PL rods withdrawn	ADMINISTRATIVE:	This LCO is unchanged, it has been incorporated in RTS 3.1.6 "Shutdown and Part Length Rod Insertion Limits."
3.10.3	3.1.6	XCPT: 3.10.3 (PL rods out LCO) N/A; Rod exercising	ADMINISTRATIVE:	Changed wording to reflect standard.
3.10.4	3.1.5	Misaligned or Inop Control Rods		
3.10.4.a	3.1 Deleted	DEF: Misaligned rod (>8in out)	ADMINISTRATIVE:	Not a defined term in STS
3.10.4.b	3.1 Deleted	DEF: Inop control rod	ADMINISTRATIVE:	Not a defined term in STS
3.10.4.b	3.1 Deleted	DEF: Inop PL rod	ADMINISTRATIVE:	Not a defined term in STS
3.10.4.b	3.1.5	LCO: <2 misaligned or inop rods; When >HSD	ADMINISTRATIVE:	Technical content unchanged, slight change in wording for this action.
3.10.4.b	3.1.5 E.1	ACTN: >1 misaligned or inop rod; HSD in 12 hrs	ADMINISTRATIVE:	Technical content unchanged, slight change in wording for this action.
3.10.4.c	3.1.5 B.2.1/.2	ACTN: Misaligned rod; Hot chnl OK or be <75%; 2 hrs	ADMINISTRATIVE:	Changed wording align with RTS.
3.10.4.c	3.1.5 A.1	ACTN: Misaligned rod; Verify SDM	ADMINISTRATIVE:	Changed wording to align with RTS.
3.10.4.c	3.1 Deleted	ACTN: Misaligned rod; Verify ejected rod worths	ADMINISTRATIVE:	This requirement has been deleted this worth of the misaligned rod is incorporated in the SDM verification of 3.1.5 A.1.
3.10.5	3.1.9	Regulating Group Insertion Limits		
3.10.5.a	3.1.7	LCO: Reg rod sequence, overlap, & insertion w/in COLR	ADMINISTRATIVE:	Changed wording to reflect the RTS. Overlap is not discussed explicitly, but it is an inherent part of the rod sequencing.

TS Number	RTS Number	TS requirement description	Classification and Description of Changes	
3.10.5.b	3.1.7 B.2	ACTN: Reg rod not w/in limit; Restore; 2 hrs	ADMINISTRATIVE:	Changed wording to reflect standard.
3.10.6	3.1.8	Shutdown Rod Limits		
3.10.6.a	3.1.6	LCO: All SD Rods out before any regulating rods	ADMINISTRATIVE:	Requirement unchanged.
3.10.6.b	3.1 Relocated	LCO: SD rods not withdrawn w/o bubble	RELOCATED:	No similar requirement appears in STS. The intent of this requirement was to assure that the reactor was not taken critical without a bubble in the pressurizer. The initial criticality and initial low power physics testing were performed at 260°F T_{ave}. At that temperature it would be possible to be in a solid water condition. Proposed RTS do not allow criticality below 500°F for physics testing or below 525°F for normal operation. These newer restrictions eliminate the need for this Tech Spec requirement.
3.10.6.c	3.1.6	LCO: SD rods not below exercise limit until reg rods in	ADMINISTRATIVE:	Requirement unchanged.
3.10.7	3.1.8	XCPT: 3.10.1.a (4 PCP SDM) N/A; Phy Test	ADMINISTRATIVE:	Changed wording to reflect standard.
3.10.7	3.1.7	XCPT: 3.10.1.a (4 PCP SDM) N/A; Rod exercise	ADMINISTRATIVE:	The exception for rod exercises is stated in a note in RTS for each applicable LCO that would require such a SR.
3.10.7	3.1.8	XCPT: 3.10.1.b (<4 PCP SDM) N/A; Phy Test	ADMINISTRATIVE:	Changed wording to reflect standard.
3.10.7	3.1 Deleted	XCPT: 3.10.1.b (<4 PCP SDM) N/A; Rod exercise	MORE RESTRICTIVE:	Exception was not included in RTS; it is not needed.
3.10.7	3.1.8	XCPT: 3.10.3 (PL rods out) N/A; Phy Test	ADMINISTRATIVE:	Changed wording to reflect standard.
3.10.7	3.1.6	XCPT: 3.10.3 (PL rods out) N/A; Rod exercise	ADMINISTRATIVE:	The exception for rod exercises is stated in a note in RTS for each applicable LCO that would require such a SR.
3.10.7	3.1.8	XCPT: 3.10.4.b (Misaligned or inop rod) N/A; Phy Test	ADMINISTRATIVE:	Changed wording to reflect standard.
3.10.7	3.1.5/3.1.6	XCPT: 3.10.4.b (Misaligned or inop rod) N/A; Rod exercise	ADMINISTRATIVE:	The exception for rod exercises is stated in a note in RTS for each applicable LCO that would require such a SR.
3.10.7	3.1.8	XCPT: 3.10.5 (rod insrtn, seq, overlap) N/A; Phy Test	ADMINISTRATIVE:	Changed wording to reflect standard.
3.10.7	3.1.5/3.1.6	XCPT: 3.10.5 (rod insrtn, seq, overlap) N/A; Rod ex	ADMINISTRATIVE:	The exception for rod exercises is stated in a note in RTS for each applicable LCO that would require such a SR.
3.10.7	3.1.8	XCPT: 3.10.6 (SD rod limits) N/A; Phy Test	ADMINISTRATIVE:	Changed wording to reflect standard.

TS Number	RTS Number	TS requirement description	Classification and Description of Changes	
3.10.7	3.1.6	XCPT: 3.10.6 (SD rod limits) N/A; Rod exercise	ADMINISTRATIVE:	The exception for rod exercises is stated in a note in RTS for each applicable LCO that would require such a SR.
3.12	3.1.4	Moderator Temperature Coefficient		
3.12	3.1.4	LCO: MTC <+0.5E-4; <2% RTP	ADMINISTRATIVE:	Requirement unchanged.
3.17.6.2	3.1.5 D.1	ACTN: 1 Rod Pos chnl inop; check rods 15; Min after motion	ADMINISTRATIVE:	Requirement unchanged.
3.17.6.18	3.1.7 C.1	ACTN: PDIL Alm inop; check rods; 15; Min after motion	ADMINISTRATIVE:	Requirement unchanged.
3.17.6T#2	3.1.5	LCO: 2 chnls Rod Pos; >1 rod capable of withdrawal	ADMINISTRATIVE:	Rod position indication is addressed as a functional part of Rod Operability in LCO 3.1.5.
3.17.6T#18	3.1.7	LCO: 2 chnls PDIL Alm; ≥Hot Standby	ADMINISTRATIVE:	Requirements unchanged.
4.2.1.2	3.1.1.1/3.1.2.1	SR: PCS Boron; 2/wk	MORE RESTRICTIVE:	Frequency changed from twice per week to once per day.
4.2.2.1	3.1.5.5	SR: FL Rod Drop; verify drop times; Refueling	ADMINISTRATIVE:	Requirement unchanged.
4.2.2.2	3.1.5.4	SR: All Rods; exercise; 92 days	ADMINISTRATIVE:	Requirement unchanged.
4.10	3.1.3	Reactivity Anomalies		
4.10	3.1.3	LCO: Critical Boron; Actual w/in 1% of Predicted	ADMINISTRATIVE:	Requirement Unchanged.
4.10	3.1 Deleted	ADMN: Crit/Predicted B δk >1%; notify AEC, 24 hrs	LESS RESTRICTIVE:	This reporting requirement was issued as part of the original Palisades Tech Specs, circa 1971. Since that time 10 CFR 50.72 and 50.73 have been issued to replace "Reporting Requirements" of this type.
4.10	3.1 Deleted	ADMN: Crit/Predicted B δk >1%; Eval to AEC, 30 days	LESS RESTRICTIVE:	This reporting requirement was issued as part of the original Palisades Tech Specs, circa 1971. Since that time 10 CFR 50.72 and 50.73 have been issued to replace "Reporting Requirements" of this type.
4.10	3.1.3.1	SR: Crit Boron; compare w predicted; periodically	MORE RESTRICTIVE:	RTS SR frequency is "31 days".
4.17.6T#2-cc	3.1.5.2	SR: 2 chnls Rod Pos; Chnl Check; 12 hrs	ADMINISTRATIVE:	Requirement unchanged.
4.17.6T#13-cft	3.1 Relocated	SR: 2 chnls Rod seq. cont/Alarm; Chnl fnc tst; 18 mo	RELOCATED:	This requirement does not meet the criterion of 10 CFR 50.36.
4.17.6T#13-cal	3.1 Relocated	SR: 2 chnls Rod sequence control/Alarm; Chnl cal; 18 mo	RELOCATED:	This requirement does not meet the criterion of 10 CFR 50.36.

TS Number	RTS Number	TS requirement description	Classification and Description of Changes	
4.17.6T#18-cft	3.1.7.2	SR: 2 chnls PDIL Alm; Chnl func test; 31 days	ADMINISTRATIVE:	Requirement unchanged.
4.17.6T#18-cal	3.1.7.2	SR: 2 chnls PDIL Alm; Chnl cal; 18 mo	ADMINISTRATIVE:	The PDIL is a computer generated alarm and is not subject to drift. A channel functional test is required by SR 3.1.7.2, which accomplishes the same function.
4.20	3.1.4	Moderator Temperature Coefficient		
4.20.1	3.1.4.1	SR: MTC; verify w/in limit; Refueling (B4 2%)	ADMINISTRATIVE:	Requirement unchanged.

ATTACHMENT 4

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

3.1 REACTIVITY CONTROL SYSTEMS

STS Pages Marked to Show the Differences Between RTS and STS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 Shutdown Margin (SDM)-~~T_{avg}~~ T_{ave} ≥ ~~200~~525°F ~~(Analog)~~

LCO 3.1.1 SDM shall be ≥ ~~[4.5]% Δ k/k.~~2% Δ ρ with 4 Primary Coolant Pumps operating and 3.75% Δ ρ SDM with < 4 Primary Coolant Pumps operating.

APPLICABILITY: MODE~~S~~ 3, with T_{ave} ≥ 525°F ~~and 4.~~

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 is ~~≥ [4.5]% Δk/k.~~ within limit.	Within 2 hours following a reactor shutdown or trip; AND 24 hours thereafter.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Shutdown Margin (SDM) - ~~T_{avg}~~ T_{ave} ~~≤ 200~~< 525°F ~~(Analog)~~

LCO 3.1.2 SDM shall be ≥ ~~[3.0]% Δk/k.~~3.5% Δ ρ.

APPLICABILITY: ~~MODE 5.~~MODE 3, with T_{ave} < 525°F, MODE 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 3.1.2.1 Verify SDM is ~~≥ [3.0]% Δk/k.~~within limit.	Within 2 hours following a reactor trip or shutdown. AND 24 hours thereafter.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Reactivity Balance ~~(Analog)~~

LCO 3.1.3 The core reactivity balance shall be within ± 1% ~~Δk/k~~ ρ of predicted values.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Core reactivity balance not within limit.	A.1	Reevaluate core design and safety analysis and determine that the reactor core is acceptable for continued operation.	72 hours
	AND		72 hours
	A.2	Establish appropriate operating restrictions and ~~SRs~~ Surveillances Requirements.	
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	----------NOTES---------- 1. The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 ~~e~~Effective ~~f~~Full ~~p~~Power ~~d~~Days (EFPD) after each ~~fuel loading~~ refueling. 2. This Surveillance is not required to be performed prior to entry into MODE 2. ----------	
	Verify overall core reactivity balance is within ± 1% Δ~~k/k~~ ρ of predicted values.	Prior to entering MODE 1 after each refueling. ~~loading~~ AND ------NOTE------ Only required after initial 60 EFPD and ---------- 31 EFPD thereafter

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Moderator Temperature Coefficient (MTC) ~~(Analog)~~

LCO 3.1.4 ~~The MTC shall be maintained within the limits specified in the COLR. The maximum positive limit shall be that specified in Figure 3.1.4-1.~~
The Moderator Temperature Coefficient (MTC) shall be less positive than $.5 \times 10^{-4}$ $\Delta \rho/°F$.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MTC not within limit~~s~~.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	----------NOTE---------- This Surveillance is not required to be performed prior to entry into MODE 2. ---------- Verify MTC is within the ~~upper~~ specified limit~~s specified in the COLR~~.	Once prior to ~~entering MODE 1~~ exceeding 2% RTP after each ~~fuel loading~~ refueling.

(continued)

SURVEILLANCE	FREQUENCY
~~SR 3.1.4.2~~ ~~NOTES~~ ~~1. This Surveillance is not required to be performed prior to entry into MODE 1 or 2.~~ ~~2. If the MTC is more negative than the COLR limit when extrapolated to the end of cycle, SR 3.1.4.2 may be repeated. Shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit.~~ ~~Verify MTC is within the lower limit specified in the COLR.~~	~~Each fuel cycle within 7 effective full power days (EFPD) of reaching 40 EFPD core burnup~~ ~~AND~~ ~~Each fuel cycle within 7 EFPD of reaching 2/3 of expected core burnup~~

~~Figure 3.1.4-1 (page 1 of 1)~~

~~Allowable Positive MTC Limit~~

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control ~~Element Assembly (CEA)~~ Rod Alignment ~~(Analog)~~

LCO 3.1.5 ~~All CEAs~~ Each rod shall be OPERABLE and ~~aligned to within [7] inches (indicated~~ positioned~~)~~ within 8 inches of all ~~their~~ other rods in ~~respective~~ the same group. ~~[, and the CEA motion inhibit and the CEA deviation circuit~~ The rod position deviation alarm shall be OPERABLE~~]~~.

APPLICABILITY: MODES 1 and 2.

---NOTE---
This LCO is not applicable while performing SR 3.1.5.3 (verify rod position deviation alarm operating).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ~~One or more regulating CEAs trippable and misaligned from its group by > [7 inches] and ≤ [15 inches].~~ One control rod inoperable.	~~A.1 Reduce THERMAL POWER to ≤ 70% RTP.~~	1 hour
	~~AND~~	
	~~A.2.1 Verify SDM is ≥ [4.5]% Δk/k.~~	~~1 hour~~
	~~OR~~	~~1 hour~~
~~OR~~	~~A.2.2 Initiate boration to restore SDM to within limit.~~	~~2 hours~~
~~One regulating CEA trippable and misaligned from its group by > [15 inches].~~	~~AND~~	
	~~A.3.1 Restore the misaligned CEA(s) to within [7 inches] (indicated position) of its group.~~	~~2 hours~~
	~~OR~~	
	~~A.3.2 Align the remainder of the CEAs in the group to within [7 inches] (indicated position) of the misaligned CEA(s) while maintaining the insertion limit of LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits."~~	1 hour
	A.1 Verify SDM ≥ 2% Δ ρ.	
	OR	
	A.2 Initiate boration to restore SDM to ≥ 2% Δ ρ.	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One ~~or more shutdown CEAs trippable and~~ rod misaligned from other rods in ~~its group~~ the same group by ~~> [7~~ ≥ 8 inches~~]~~, ~~and~~ but ≤ ~~[15~~20 inches~~]~~. ~~OR~~ ~~One shutdown CEA trippable and misaligned from its group by > [15 inches].~~	B.1 ~~Reduce THERMAL POWER to ≤ 70% RTP.~~ ~~AND~~ ~~B.2.1 Verify SDM is ≥ [4.5]% Δk/k.~~ ~~OR~~ ~~B.2.2 Initiate boration to restore SDM to within limit.~~ ~~AND~~ ~~B.3 Restore the misaligned CEA(s) to within [7 inches] (indicated position) of its group.~~Declare the misaligned rods inoperable. AND B.2.1 Perform SR 3.2.2.1. (Peaking factor verification) OR B.2.2 Restrict thermal power ≤ 75% RTP.	1 hour 2 hours 2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ~~CEA motion inhibit inoperable.~~One control rod misaligned from other rods in the same group by ≥ 20 inches.	C.1 ~~Perform SR 3.1.5.1.~~Declare the rod inoperable.	1 hour
	~~AND~~	
	C.2 Verify SDM ≥ 2% Δ ρ.	~~AND~~ ~~Every 4~~1 hours ~~thereafter~~
	~~AND~~	
	C.3.1 Perform SR 3.2.2.1 (Peaking Factor Verification).	~~6~~4 hours
	~~OR~~	~~6 hours~~
	C.3.2 Restrict THERMAL POWER to ≤ 50% RTP.	4 hours
	~~AND~~	
	~~C.2.1 Restore CEA motion inhibit to OPERABLE status.~~	
	~~OR~~	
	~~C.2.2~~	
	~~NOTE~~ ~~Performance of Required Action C.2.2 is allowed only when not in conflict with Required Action A.1, A.3.1, A.3.2, B.1, B.3, or D.1.~~ ~~Place and maintain the CEA drive switch in either the "off" or "manual" position [, and fully withdraw all CEAs in groups 3 and 4 and withdraw all CEAs in group 5 to < 5% insertion].~~	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. ~~CEA deviation circuit inoperable.~~ Rod position deviation alarm is inoperable. OR 1 channel of rod position indication inoperable.	D.1 Perform SR 3.1.5.1. (rod position verification)	~~1 hour~~ ~~AND~~ ~~Every 4 hours thereafter~~ Once within 15 minutes following any rod motion.

(continued)

E. ~~Required Action and associated Completion Time not met.~~ Two or more control rods inoperable. OR ~~One or more CEAs untrippable.~~ Required Action and associated Completion Time not met. ~~OR~~ ~~Two or more CEAs misaligned by >[15 inches.~~	E.1 Be in MODE 3.	~~6~~12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify the indicated position of each ~~CEA to be within [7 inches] of all other CEAs in its group~~ rod ≤ 8 inches of all other rods in the same group.	12 hours
SR 3.1.5.2	~~Verify that, for each CEA, the OPERABLE CEA position indicator channels, reed switch, and plant computer CEA position indication indicate within [5 inches] of each other.~~ Verify that, each rod's operable position indication channel indicates within 5 inches of each other.	12 hours
SR 3.1.5.3	~~Verify the CEA motion inhibit~~ Demonstrate the rod position deviation alarm is OPERABLE.	~~31~~92 days
SR 3.1.5.4	Verify ~~the CEA deviation circuit is OPERABLE~~ rod freedom of movement and tripability by moving each full length rod that is not fully inserted into the reactor core ≥ 5 inches in either direction.	~~31~~92 days
~~SR 3.1.5.5~~	~~Verify CEA freedom of movement (trippability) by moving each individual CEA that is not fully inserted into the reactor core [5 inches] in either direction.~~	~~92 days~~
SR 3.1.5.~~6~~5	Perform a CHANNEL ~~FUNCTIONAL TEST of the reed switch position transmitter channel~~ CALIBRATION on each rod position indicator channel.	18 months
SR 3.1.5.~~7~~6	Verify each ~~CEA~~ full length rod drop time is ≤ ~~[3.1]~~ 2.5 seconds.	~~Prior to reactor criticality, after each removal of the reactor head~~ 18 months

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Shutdown ~~Control Element Assembly (CEA) Insertion Limits (Analog)~~and Part Length Rod Insertion Limits

LCO 3.1.6 All shutdown ~~CEAs shall be withdrawn to ≥ [129] inches~~ and part length rods shall be withdrawn to ≥ 128 inches.

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

----------NOTE----------
This LCO is not applicable while performing SR 3.1.5.~~54~~ (Rod Exercising).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ~~or more shutdown CEAs not within limit~~ shutdown or part length rod not within limits.	~~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~~	
	~~A.2 Restore shutdown CEA(s) to within limit.~~	~~2 hours~~
	A.1 Verify SDM ≥ 2% Δ ρ.	1 hour
	OR	
	A.2 Initiate boration to restore SDM to ≥ 2% Δ ρ.	1 hour
	OR	
	A.3 Declare the rod inoperable.	1 hour

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	~~6~~12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	~~Verify each shutdown CEA is withdrawn ≥ [129] inches.~~ Verify all shutdown and part length rods are ≥ 128 inches.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Regulating ~~Control Element Assembly (CEA)~~ Rod Insertion Limits ~~(Analog)~~

LCO 3.1.7 The ~~p~~Power ~~d~~Dependent ~~i~~Insertion ~~I~~Limit (PDIL) alarm circuit shall be OPERABLE. The regulating ~~CEA~~ rod groups shall be limited to the withdrawal sequence 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 [or during reactor power cutback operation]~~ 3.1.5.4 (Rod Exercising).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Regulating ~~CEA~~ groups inserted beyond the ~~transient~~ insertion limit.	A.1.1 Verify SDM ≥ ~~[4.5]% Δk/k.~~2% Δρ	~~1~~2 hours
	~~OR~~	
	~~A.1.2 Initiate boration to restore SDM to within limit.~~	
	~~AND~~	
	~~A.2.1 Restore regulating CEA groups to within limits.~~	~~1~~2 hours
	OR	
	A.1.2 Initiate boration to restore rods above insertion limit.	2 hours
	AND	
	A.2 Reduce THERMAL POWER to less than or equal to the fraction of RTP allowed by the ~~CEA~~ regulating group position and insertion limits specified in the COLR.	

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. ~~Regulating CEA groups inserted between the long term steady state insertion limit and the transient insertion limit for > 4 hours per 24 hour interval.~~ Regulating groups withdrawn out of sequence.	B.1 ~~Verify short term steady state insertion limits are not exceeded.~~ ~~OR~~ ~~B.2 Restrict increases in THERMAL POWER to ≤ 5% RTP per hour.~~ Verify SDM ≥ 2% Δ ρ. ~~OR~~ B.2 Restore rods to within appropriate sequence	~~15 minutes~~ ~~15 minutes~~ 1 hour 1 hour
~~C. Regulating CEA groups inserted between the long term steady state insertion limit and the transient insertion limit for intervals > 5 effective full power days (EFPD) per 30 EFPD interval or > 14 EFPD per 365 EFPD.~~	~~C.1 Restore regulating CEA groups to within limits.~~	~~2 hours~~
~~D~~C. PDIL alarm circuit inoperable.	~~D~~C.1 Perform SR 3.1.7.1. (verify group position)	~~1 hour~~ ~~AND~~ ~~Once per 4 hours thereafter~~ Once within 15 minutes following any rod motion

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~E~~D. Required Action and associated Completion Time not met.	~~E~~D.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	----------NOTE---------- This Surveillance is not required to be performed prior to entry into MODE 2. ---------- Verify each regulating ~~CEA~~ group position is within its insertion limits.	12 hours
~~SR 3.1.7.2~~	~~Verify the accumulated times during which the regulating CEA groups are inserted beyond the steady state insertion limits but within the transient insertion limits.~~	~~24 hours~~
SR 3.1.7.~~3~~2	Verify PDIL alarm circuit is OPERABLE. (Setpoint verification only)	31 days

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Special Test Exception (STE) - Shutdown Margin (SDM) ~~(Analog)~~ and Rod Limits

LCO 3.1.8 The ~~SDM~~ requirements of LCOs 3.1.1 and 3.1.2 (SHUTDOWN MARGIN ~~(SDM) T_{avg} > 200°F," and the regulating control element assembly (CEA) insertion limits of LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits," may be suspended for measurement of CEA worth and the SDM, provided shutdown reactivity equivalent to at least the highest estimated CEA worth (of those CEAs actually withdrawn) is available for trip insertion~~ and LCOs 3.1.5, 3.1.6, and 3.1.7 (Rod Limits) may be suspended for the determination of rod worth, MTC, and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated rod worth is available for trip in OPERABLE control rods.

APPLICABILITY: MODES 2 and 3 during PHYSICS TESTS.

----------NOTE----------
Operation in MODE 3 shall be limited to 6 consecutive hours.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any ~~CEA~~ full length rod not fully inserted and less than the above shutdown reactivity equivalent available for trip insertion. OR All ~~CEAs~~ full length rods inserted and the reactor subcritical by less than the above shutdown reactivity equivalent.	A.1 Initiate boration to restore required shutdown reactivity.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Verify that the position of each ~~CEA~~ rod not fully inserted ~~is within the acceptance criteria for available~~ has sufficient negative reactivity addition, to provide adequate SDM.	2 hours
SR 3.1.8.2	Verify that each ~~CEA~~ rod not fully inserted is capable of full insertion when tripped from at least the 50% withdrawn position.	Within ~~[~~7 days~~]~~ prior to ~~reducing SDM to less than the limits of~~ suspending LCOs 3.1.1 or 3.1.2.

~~3.1 REACTIVITY CONTROL SYSTEMS~~

~~3.1.9 Special Test Exception (STE) MODES 1 and 2 (Analog)~~

~~LCO 3.1.9 During the performance of PHYSICS TESTS, the requirements of~~

~~LCO 3.1.4, "Moderator Temperature Coefficient (MTC)";~~
~~LCO 3.1.5, "Control Element Assembly (CEA) Alignment";~~
~~LCO 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits";~~
~~LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits";~~
~~LCO 3.2.2, "Total Planar Radial Peaking Factor (F^T_{xy})";~~
~~LCO 3.2.3, "Total Integrated Radial Peaking Factor (F^T_r)"; and~~
~~LCO 3.2.4, "AZIMUTHAL POWER TILT (T_q)"~~

~~may be suspended, provided:~~

~~a. THERMAL POWER is restricted to the test power plateau, which shall not exceed 85% RTP; and~~

~~b. SDM is ≥ [4.5]% Δk/k.~~

~~APPLICABILITY: MODES 1 and 2 during PHYSICS TESTS.~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. Test power plateau exceeded.~~	~~A.1 Reduce THERMAL POWER to less than or equal to test power plateau.~~	~~15 minutes~~

~~(continued)~~

ACTIONS (continued)

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~B. SDM not within limit.~~	~~B.1 Initiate boration to restore SDM to within limit.~~	~~15 minutes~~
	~~AND~~	
	~~B.2 Suspend PHYSICS TESTS.~~	~~1 hour~~
~~C. Required Action and associated Completion Time not met.~~	~~C.1 Suspend PHYSICS TESTS.~~	~~1 hour~~
	~~AND~~	
	~~C.2 Be in MODE 3.~~	~~6 hours~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~	~~FREQUENCY~~
~~SR 3.1.9.1 Verify THERMAL POWER is equal to or less than the test power plateau.~~	~~1 hour~~
~~SR 3.1.9.2 Verify SDM is ≥ [4.5]% Δk/k.~~	~~24 hours~~

ATTACHMENT 5

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

3.1 REACTIVITY CONTROL SYSTEMS

STS Bases Pages Marked to Show the Differences Between RTS and STS

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 Shutdown Margin (SDM) ~~T_{avg} > 200~~ T_{ave} ≥ 525°F ~~(Analog)~~

BASES

BACKGROUND

The reactivity control systems must be redundant and capable of holding the reactor core subcritical when shutdown under cold conditions, in accordance with GDC 26 (Ref. 1). Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and ~~a~~Anticipated ~~o~~Operational ~~O~~Occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all ~~control element assemblies (CEAs)~~ rods, assuming the single ~~CEA~~ rod of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable ~~CEAs~~ rods and soluble boric acid in the ~~Reactor Coolant System (RCS)~~ Primary Coolant System (PCS). The ~~CEA~~ control rod ~~S~~system can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the ~~CEAs~~ control rods, together with the boration system, provide the SDM during power operation and are capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel ~~damage~~ design limits, assuming that the ~~CEA~~ control rod of highest reactivity worth remains fully withdrawn.

The soluble boron system can compensate for fuel depletion during operation and all ~~x~~Xenon burnout reactivity changes, and maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown ~~CEAs~~ rods fully withdrawn and the regulating ~~CEAs~~ rods within the limits of LCO 3.1.7, "Regulating Control ~~Element Assembly (CEA)~~ Rod Insertion Limits." When the ~~unit~~ plant is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the ~~RCS~~ PCS boron concentration.

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 of the highest worth ~~CEA~~ rod 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 (~~d~~Departure from ~~n~~Nucleate ~~b~~Boiling ~~r~~Ratio ~~(~~(DNBR~~)~~), fuel centerline temperature limit AOOs, and ≤ ~~280~~200 cal/gm energy deposition for the ~~CEA~~ 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 ~~m~~Main ~~s~~Steam ~~l~~Line ~~b~~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 ~~s~~Steam ~~g~~Generator (SG), and consequently the ~~RCS~~ PCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative ~~m~~Moderator ~~t~~Temperature ~~c~~Coefficient (MTC) this cooldown causes an increase in core reactivity. As ~~RCS~~ PCS temperature decreases, the severity of a~~n~~ MSLB decreases until the MODE 5 ~~value is~~ conditions are 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~~ PCS 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 power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirement must also protect against:

a. Inadvertent boron dilution; (Ref. 3)

b. An uncontrolled ~~CEA~~ rod withdrawal from a subcritical or low power condition; (Ref. 5)

c. Startup of an inactive ~~reactor coolant pump (RCP); and~~ loop; and (Ref. 6)

d. ~~CEA~~ Control rod ejection. (Ref. 7)

BASES

These events are described in detail in the respective FSAR section referenced above.

~~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 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.~~

~~The withdrawal of 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 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, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.~~

~~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 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.~~

LCO

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, the DNBR limit is exceeded. ~~there is a potential to exceed the DNBR limit and to exceed~~ ~~and~~ 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 CEA positioning (regulating and shutdown CEA) and through the soluble boron concentration.~~

BASES

The LCO statement contains two different SDM values which account for operation with 4 or less than 4 primary coolant pumps operating. A 2% ρ SDM value is applicable for 4 pumps with $T_{ave} \geq$ 525°F while operation with less than4 pumps requires 3.75% ρ SDM for this same temperature range. The safety analysis uses these SDM values to support accident scenarios that are initiated under these conditions.

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

APPLICABILITY ~~In MODES 3 and 4, 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 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits," and LCO 3.1.7. 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."~~
The applicability of this LCO is restricted to MODE 3 with T_{ave} > 525°F. LCO 3.1.1, SDM requirements of 2% $\Delta\rho$ pcm is sufficient to support the safety analyses.

The SDM requirements for all MODES of operation are sufficient to meet the assumptions made in the safety analyses discussed above. In MODES 1 and 2, the SDM requirement is satisfied by control rod positioning (regulating and shutdown) as stipulated by LCO 3.1.6, "Control Rod Insertion Limits." It should be noted that if the insertion limits stated in LCOs 3.1.6 and 3.1.7 are violated, SDM is not automatically compromised.

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.

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~~ PCS 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~~ SIRWT. The operator should borate with the best source available for the plant conditions.

~~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 [ ] gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 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 [ ] gpm and [ ] ppm represent typical values and are provided for the purpose of offering a specific example.~~

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

~~SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects:~~SDM is verified by performing a current reactivity balance calculation or by using technical data generated by Reactor Engineering which considers the following reactivity effects:

a. ~~RCS~~ PCS boron concentration;

b. ~~CEA~~ Control rod positions;

c. ~~RCS~~ PCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration; and

~~f. Samarium concentration; and~~

~~g~~f. Isothermal ~~t~~Temperature ~~c~~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 ~~RCS~~ PCS.

Samarium is not considered in the Palisades reactivity balance due to the fact that Palisades fuel vendor does not account for Samarium in fuel design calculations performed. 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 performed by the Palisades fuel vendor. Therefore, including Samarium into the SDM calculation would not be correct.

The frequency of SDM verification within 2 hours following a reactor trip or shutdown circumvents the case where the plant trips 1 minute after the 24 hour check was performed. This ensures that following a trip or shutdown the SDM assumed in the safety analysis is verified promptly.

The frequency of 24 hours 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 complete the calculation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26
2. FSAR, Section ~~[ ]~~14.14
3. FSAR, Section ~~[ ]~~14.3
4. 10 CFR 100
5. FSAR, Section 14.2
6. FSAR, Section 14.8
7. FSAR, Section 14.16

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Shutdown Margin (SDM) ~~$T_{avg} \leq 200$~~ < 525°F ~~(Analog)~~

BASES

BACKGROUND

~~The reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions, in accordance with GDC 26 (Ref. 1). Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all control element assemblies (CEAs), assuming the single CEA of highest reactivity worth is fully withdrawn.~~

~~The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable CEAs and soluble boric acid in the Reactor Coolant System (RCS). The CEA System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the CEAs, together with the boration system, provide the SDM during power operation and are capable of making the core subcritical rapidly enough to prevent exceeding the acceptable fuel damage limits, assuming that the CEA of highest reactivity worth remains fully withdrawn.~~

~~The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes, as well as maintain the reactor subcritical under cold conditions.~~

~~During power operation, SDM control is ensured by operating with the shutdown CEAs fully withdrawn and the regulating CEAs within the limits of LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.~~

A detailed BACKGROUND description for B 3.1.2, "Shutdown Margin (SDM) T_{avg} < 525°F" is included in the description of B 3.1.1 Bases.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes a~~n~~ SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs with the assumption of the highest worth ~~CEA~~ rod stuck out following a reactor trip. Specifically, for ~~MODE 5~~ LCO 3.1.2, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

The acceptance criteria for the SDM requirements are that the 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, fuel centerline temperature limits for AOOs, and ≤ ~~280~~ 200 cal/gm energy deposition for the ~~CEA~~ control rod ejection accident); and

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

An inadvertent boron dilution is a moderate frequency incident as defined in Reference 2. The core is initially subcritical with all ~~CEAs~~ rods inserted. A Chemical and Volume Control System malfunction occurs, which causes unborated water to be pumped to the ~~RCS~~ PCS via three charging pumps.

The reactivity change rate associated with boron concentration changes due to inadvertent dilution is within the capabilities of operator recognition and control.

~~The high neutron flux alarm on the startup channel instrumentation will alert the operator to the boron dilution with a minimum of 15 minutes remaining before the core becomes critical.~~

~~SDM satisfies Criterion 2 of the NRC Policy Statement.~~

LCO

The accident analysis has shown that the required SDM is sufficient to avoid unacceptable consequences to the fuel or ~~RCS~~ PCS as a result of the events addressed above.

BASES

~~The boron dilution (Ref. 2) accident initiated in MODE 5 is the most limiting analysis that establishes the SDM value of the LCO. 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.~~
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 CEA positioning (regulating and shutdown CEAs) and through soluble boron concentration.~~

By definition of SHUTDOWN MARGIN the stuck rod worth can be relaxed from the LCO if both the synchro and reed switch position indication systems can verify All Rods In (ARI) condition.

APPLICABILITY ~~In MODE 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 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits," and LCO 3.1.7. 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 3.1.2.1). In MODES 3 and 4, the SDM requirements are given in LCO 3.1.1, "SHUTDOWN MARGIN T_{avg} > 200°F." In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."~~
In MODE 3 with T_{ave}, < 525°F, and MODES 4, and 5 the SDM requirements of LCO 3.1.2 are sufficient to meet the assumptions used in the safety analyses.

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.~~

~~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.~~

~~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 [ ] gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 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 [ ] gpm and [ ] ppm represent typical values and are provided for the purpose of offering a specific example.~~
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.

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 PCS 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 SIRWT. The operator should borate with the best source available for the plant conditions.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

~~In MODE 5, the SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects:~~

~~a. RCS boron concentration;~~

~~b. CEA positions;~~

~~c. RCS average temperature;~~

~~d. Fuel burnup based on gross thermal energy generation;~~

BASES

~~e. Xenon concentration;~~

~~f. Samarium concentration; and~~

~~g. 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 that of the RCS.~~

~~The Frequency of 24 hours 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 complete the calculation.~~
A detailed SURVEILLANCE REQUIREMENTS description for B 3.1.2, "Shutdown Margin (SDM) T_{ave} < 525°F" is included in the description of B 3.1.1 Bases.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26
2. FSAR, Section ~~[ ]~~14.3

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Reactivity Balance ~~(Analog)~~

BASES

BACKGROUND

According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that, subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, ~~control element assembly (CEA)~~ rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM) ~~T_{avg} > 200~~T_{ave} ≥ 525°F") in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the critical boron curve, which provides an indication of the soluble boron concentration in the ~~Reactor Coolant System (RCS)~~ Primary Coolant System (PCS) versus cycle burnup. Periodic measurement of the ~~RCS~~ PCS boron concentration for comparison with the predicted value with other variables fixed (such as ~~CEA~~ control rod height, temperature, pressure, and power) provides a convenient method of ensuring that core reactivity is within design expectations, and that the calculational models used to generate the safety analysis are adequate.

BASES

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), ~~CEAs~~ control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the ~~RCS~~ PCS boron concentration.

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the ~~RCS~~ PCS boron concentration is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The critical boron curve is based on steady state operation at RTP. Therefore, deviations from the predicted critical boron curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

APPLICABLE SAFETY ANALYSES

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as ~~CEA~~ control rod withdrawal accidents or ~~CEA~~ control rod ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the ~~RCS~~ PCS boron concentration requirements for reactivity control during fuel depletion.

BASES

The comparison between measured and predicted initial core reactivity provides a normalization for calculational models used to predict core reactivity. If the measured and predicted ~~RCS~~ PCS boron concentrations for identical core conditions at ~~b~~Beginning ~~o~~Of ~~c~~Cycle (BOC) ~~do not agree~~ are not within design tolerances, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted critical boron curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred.

The normalization of predicted ~~RCS~~ PCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the ~~CEAs~~ control rods in their normal positions for power operation. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

~~The reactivity balance satisfies Criterion 2 of the NRC Policy Statement.~~

LCO

The reactivity balance limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the nuclear design methodology are larger than expected. A limit on the reactivity balance of ± 1% $\Delta k/k$ $\Delta\rho$ has been established, based on engineering judgment. A ± 1% $\Delta\rho$ deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

BASES

When measured core reactivity is within ±1% ~~Δk/k~~ ρ of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limits are normally detected by comparing predicted and measured steady state ~~RCS~~ PCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the ~~RCS~~ PCS boron concentration ~~are~~ is unlikely.

APPLICABILITY The limits on core reactivity must be maintained during MODES 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODES 3, 4, and 5 because the reactor is shutdown and the reactivity balance is not changing.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. A~~n~~ SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, or ~~CEA~~ control rod replacement, or shuffling).

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 72 hours is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of ~~RCS~~ PCS boron concentration sampling, then a recalculation of the ~~RCS~~ PCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized, and power operation may continue. If operational restrictions or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 72 hours is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity cannot be restored to within the 1% $\Delta\rho$ limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at lease MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then boration required by ~~SR 3.1.1.1~~ Action 3.1.1 A.1 would occur. The allowed Completion Time 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 3.1.3.1

Core reactivity is verified by periodic comparisons of measured and predicted ~~RCS~~ PCS boron concentrations. The comparison is made considering that other core conditions are fixed or stable including ~~CEA~~ control rod position, moderator temperature, fuel temperature, fuel depletion, and xenon concentration~~, and samarium concentration~~. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. The SR is modified by three Notes. Note 1 in the Surveillance column indicates that the normalization of predicted core reactivity to the measured value must take place within the first 60 ~~e~~Effective ~~f~~Full ~~p~~Power ~~d~~Days (EFPD) after each ~~fuel loading~~ refueling. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. The required subsequent Frequency of 31 EFPD following the initial 60 EFPD after entering MODE 1, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (e.g., QPTR, etc.) for prompt indication of an anomaly. A second Note, "only required after 60 EFPD," is added to the Frequency column to allow this. Note 2 in the Surveillance column indicates that the performance of SR 3.1.3.1 is not required prior to entering MODE 2.

This Note is required to allow a MODE 2 entry to verify core reactivity, because LCO Applicability is for MODES 1 and 2.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29
2. FSAR, Section ~~[ ]~~ 3.3

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Moderator Temperature Coefficient (MTC) ~~(Analog)~~

BASES

BACKGROUND

According to GDC 11 (Ref. 1), the reactor core and its interaction with the ~~Reactor Coolant System (RCS)~~ Primary Coolant System (PCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended or rapid reactivity increases.

The MTC relates a change in core reactivity to a change in ~~reactor~~ primary coolant temperature. A positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature. The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result. The same characteristic is true when the MTC is positive and coolant temperature decreases occur.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the ~~b~~Beginning ~~o~~Of ~~c~~Cycle (BOC) MTC is less positive than that allowed by the LCO. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons (lumped burnable poison assemblies) to yield an MTC at the BOC within the range analyzed in the plant accident analysis. The ~~e~~End ~~o~~Of ~~c~~Cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles that are designed to achieve high burnups or that have changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOC limit.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and

b. The MTC must be such that inherently stable power operations result during normal operation and during accidents, such as overheating and overcooling events.

Reference 2 contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions, such as very large soluble boron concentrations, to ensure the accident results are bounding (Ref. 3).

Accidents that cause core overheating, either by decreased heat removal or increased power production, must be evaluated for results when the MTC is positive. Reactivity accidents that cause increased power production include the ~~control element assembly (CEA)~~ control rod withdrawal transient from either zero or full THERMAL POWER. The limiting overheating event relative to plant response is based on the maximum difference between core power and steam generator heat removal during a transient. The most limiting event with respect to a positive MTC is a ~~CEA~~ control rod withdrawal accident from zero power, also referred to as a startup accident (Ref. 4).

Accidents that cause core overcooling must be evaluated for results when the MTC is most negative. The event that produces the most rapid cooldown of the ~~RCS~~ PCS, and is therefore the most limiting event with respect to the negative MTC, is a Main ~~s~~Steam ~~l~~Line ~~b~~Break (~~SLB~~MLSB) event. Following the reactor trip for the postulated EOC SLB event, the large moderator temperature reduction combined with the large negative MTC may produce reactivity increases that are as much as the shutdown reactivity. When this occurs, a substantial fraction of core power is produced with all ~~CEAs~~ control rods inserted, except the most reactive one, which is assumed withdrawn. Even if the reactivity increase produces slightly subcritical conditions, a large fraction of core power may be produced through the effects of subcritical neutron multiplication.

MTC values are bounded in reload safety evaluations assuming steady state conditions at BOC. ~~and EOC. A middle of cycle (MOC) measurement is conducted at conditions when the RCS boron concentration reaches approximately 300 ppm. The measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions.~~

~~The MTC satisfies Criterion 2 of the NRC Policy Statement.~~

BASES

LCO

LCO 3.1.4 requires the MTC to be ~~within specified limits of the COLR to ensure the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation. The limit on a positive MTC ensures that core overheating accidents will not violate the accident analysis assumptions. The negative MTC limit for EOC specified in the COLR ensures that core overcooling~~ less positive than the specified limit of 0.50×10^4 $\Delta\rho$°F to ensure the core safety evaluation. MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation. The limit on a positive MTC ensures that core overheating accidents will not violate the accident analysis assumptions.

MTC is a core physics parameter determined by the fuel and fuel cycle design and cannot be easily controlled once the core design is fixed. ~~During operation, therefore, the LCO can only be ensured through measurement. The surveillance checks at BOC and MOC on an MTC provide confirmation that the MTC is behaving as anticipated, so that the acceptance criteria are met.~~

APPLICABILITY

In MODE 1, the limits on the MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2, the limits must also be maintained to ensure startup and subcritical accidents, such as the uncontrolled ~~CEA or group~~ control rod withdrawal, will not violate the assumptions of the accident analysis. In MODES 3, 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents (DBAs) using the MTC as an analysis assumption are initiated from these MODES. However, the variation of the MTC, with temperature in MODES 3, 4, and 5, for DBAs initiated in MODES 1 and 2, is accounted for in the subject accident analysis. ~~The variation of the MTC, with temperature assumed in the safety analysis, is accepted as valid once the BOC and MOC measurements are used for normalization.~~

The variation of the MTC, with temperature assumed in the safety analysis, is accepted as valid once the BOC measurement is used for normalization.

BASES

ACTIONS

A.1

MTC is a function of the fuel and fuel cycle designs, and cannot be controlled directly once the designs have been implemented in the core. If MTC exceeds its limits, the reactor must be placed in MODE 3. This eliminates the potential for violation of the accident analysis bounds. The associated Completion Time of 6 hours is reasonable, considering the probability of an accident occurring during the time period that would require an MTC value within the LCO limits, and the time for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1 ~~and SR 3.1.4.2~~

~~The SRs for measurement of the MTC at the beginning and middle of each fuel cycle provide for confirmation of the limiting MTC values. The MTC changes smoothly from most positive (least negative) to most negative value during fuel cycle operation, as the RCS boron concentration is reduced to compensate for fuel depletion. The requirement for measurement prior to operation > 5% RTP satisfies the confirmatory check on the most positive (least negative) MTC value. The requirement for measurement, within 7 days after reaching 40 effective full power days and ⅔ core burnup, satisfies the confirmatory check of the most negative MTC value. The measurement is performed at any THERMAL POWER, so that the projected EOC MTC may be evaluated before the reactor actually reaches the EOC condition. MTC values may be extrapolated and compensated to permit direct comparison to the specified MTC limits.~~

The SR for MTC validation at the beginning of each fuel cycle provides adequate values used in the safety analysis. This confirms the nuclear methods used for the prediction of both BOC and EOC MTC values. This SR is performed in accordance with startup physics testing following each refueling outage. The MTC changes smoothing from the most positive (least negative) to the most negative value during fuel cycle operation as the PCS boron concentration is reduced to compensate for fuel depletion. MTC values may be extrapolated and compensated to permit direct comparison to the specified MTC limits.

~~SR 3.1.4.2 is modified by a Note that indicates performance is not required prior to entering MODE 1 or 2. Although this Surveillance is applicable in MODES 1 and 2, the reactor must be critical before the Surveillance can be completed. Therefore, entry into the applicable MODE prior to accomplishing the Surveillance is necessary.~~

BASES

~~SR 3.1.4.2 is modified by a second Note, which indicates that if the extrapolated MTC is more negative than the EOC COLR limit, the Surveillance may be repeated, and that shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. An engineering evaluation is performed if the extrapolated value of MTC exceeds the Specification limits.~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11
2. FSAR, Section ~~[ ].~~14.1 and 14.14
3. FSAR, Section ~~[ ].~~ 3.3
4. FSAR, Section ~~[ ].~~14.2

B 3.1 **REACTIVITY CONTROL SYSTEMS**

B 3.1.5 Control ~~Element Assembly (CEA)~~ Control Rod Alignment ~~(Analog)~~

BASES

BACKGROUND

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

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10 and GDC 26 (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a ~~CEA~~ control rod to become inoperable or to become misaligned from its group. ~~CEA~~ Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available ~~CEA~~ control rod worth for reactor shutdown. Therefore, ~~CEA~~ 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 ~~CEA~~ control rod alignment and OPERABILITY have been established, and all ~~CEA~~ 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.

~~CEAs~~ Control rods are moved by their ~~c~~Control ~~element~~ Rod ~~d~~Drive ~~m~~Mechanisms (~~CEDM~~CRDMs). Each ~~CEDM~~ CRDM moves its ~~CEA one step (approximately ¾ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Control Element Drive Mechanism Control System (CEDMCS).~~rod 46 inches per minute.

~~The CEAs are arranged into groups that are radially symmetric. Therefore, movement of the CEAs does not introduce radial asymmetries in the core power distribution. The shutdown and regulating CEAs provide the required reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating CEAs also provide reactivity (power level) control during normal operation and transients. Their movement may be automatically controlled by the Reactor Regulating System.~~

~~The axial position of shutdown and regulating CEAs is indicated by two separate and independent systems, which are the Plant Computer CEA Position Indication System and the Reed Switch Position Indication System.~~

~~The Plant Computer CEA Position Indication System counts the commands sent to the CEA gripper coils from the CEDM Control System that moves the CEAs. There is a one step counter for each group of CEAs. Individual CEAs in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. Plant Computer CEA Position Indication System is considered highly precise (± 1 step or ± ¾ inch). If a CEA does not move one step for each command signal, the step counter will still count the command and incorrectly reflect the position of the CEA.~~

~~The Reed Switch Position Indication System provides a highly accurate indication of actual CEA position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of reed switches spaced along a tube with a center to center distance of 1.5 inches, which is two steps. To increase the reliability of the system, there are redundant reed switches at each position.~~

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 control rods provide the required reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating control 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 the Plant Process Computer (PPC) synchro based rod position indication system and the PPC reed switch based position indication system.

The PPC synchro based position indication system measures the phase angle of a synchro connected to the CRDM rack. Full control rod travel is less than 1 turn of the synchro. Each control rod has its own synchro which is monitored continuously. Accuracy of this system is highly precise (± .1 inch).

The reed switch position indication system provides highly accurate rod position, but at a lower precision than the synchros. This system is based on a voltage dividing network consisting of a series of magnetic reed switches and resistors. The reed switches are spaced along a tube with a center to center spacing distance of 1.5 inches. The resolution of the SPI is ± 2 inches for conservatism.

APPLICABLE SAFETY ANALYSES

~~CEA~~ Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The accident analysis defines ~~CEA~~ 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, ~~CEA~~ control rod misalignment may be caused by a malfunction of the ~~CEDM, CEDMCS~~ system, or by operator error. A stuck ~~CEA~~ rod may be caused by mechanical jamming. ~~of the CEA fingers or of the gripper. Inadvertent withdrawal of a single CEA may be caused by the opening of the electrical circuit of the CEDM holding coil for a full length or part length CEA. A dropped CEA could be caused by an electrical failure in the CEA coil power programmers.~~

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

a. There shall be no violations of:

   1. Specified acceptable fuel design limits, or

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

b. The core must remain subcritical after accident transients.

Three types of misalignment are distinguished in the safety analysis (Ref. 1). During movement of a group, one ~~CEA~~ control rod may stop moving while the other ~~CEAs~~ control rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one ~~CEA~~ 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 ~~CEAs~~ control rods to meet the SDM requirement with the maximum worth ~~CEA~~ rod stuck fully withdrawn. If a ~~CEA~~ 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 ~~CEAs~~ rods into account. The third type of misalignment occurs when one ~~CEA~~ 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 ~~l~~Linear ~~h~~Heat ~~r~~Rates (LHRs).

~~Two types of analyses are performed in regard to static CEA misalignment (Ref. 4). With CEA banks at their insertion limits, one type of analysis considers the case when any one CEA is inserted [ ] inches into the core. The second type of analysis considers the case of a single CEA withdrawn [ ] inches from a bank inserted into its insertion limit. Satisfying limits on departure from nucleate boiling ratio (DNBR) in both of these cases bounds the situation when a CEA is misaligned from its group by [7 inches].~~
Misalignment of a control rod between 8 and 20 inches is bounded in the safety analysis by the dropped rod event.

Another type of misalignment occurs if one ~~CEA~~ rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth ~~CEA~~ control rod also fully withdrawn (Ref. ~~5~~4). Since the ~~CEA~~ control rod drop incidents result in the most rapid approach to ~~s~~Specified ~~a~~Acceptable ~~f~~Fuel ~~d~~Design ~~l~~Limits (SAFDLs) caused by a ~~CEA~~ rod misoperation, the accident analysis analyzed a single full length ~~CEA~~ rod drop. The most rapid approach to the DNBR SAFDL may be caused by a single full length ~~CEA~~ rod drop. ~~or a CEA subgroup drop, depending upon initial conditions.~~

~~All of the above CEA misoperations will result in an automatic reactor trip. In the case of the full length CEA 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.~~

In the case of the full length control rod drop, a prompt decrease in core average power and a distortion in radial power are initially produced, which when conservatively coupled, results in a local power and heat flux increase, and a decrease in DNBR parameters.

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

~~CEA alignment satisfies Criteria 2 and 3 of the NRC Policy Statement.~~

BASES

LCO

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

The requirement is to maintain the ~~CEA~~ control rod alignment to within ~~[7~~ 8 inches~~]~~ between any ~~CEA~~ control rod and its group. ~~The minimum misalignment assumed in safety analysis is [15 inches], and in some cases,~~ This assumes a total misalignment from fully withdrawn to fully inserted ~~is assumed~~. This case bounds the safety analysis for a single rod in any intermediate position.

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

APPLICABILITY

The requirements on ~~CEA~~ 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 ~~CEAs~~ 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 ~~CEAs~~ control rods are fully inserted at or below the Lower Electrical Limit (LEL), ~~bottomed,~~ and the reactor is shutdown and not producing fission power. In the shutdown ~~Modes~~ MODES, the OPERABILITY of the shutdown and regulating ~~CEAs~~ 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 ~~RCS~~ PCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM) — ~~$T_{avg} > 200$~~ $T_{ave} \geq 525$°F," for SDM in MODES 3, 4, and 5, and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

BASES

ACTIONS

~~A.1, A.2.1, A.2.2, A.3.1, and A.3.2~~

~~A CEA may become misaligned, yet remain trippable. In this condition, the CEA can still perform its required function of adding negative reactivity should a reactor trip be necessary.~~

~~If one or more regulating CEAs are misaligned by > [7 inches] and ≤ [15 inches] but trippable, or one regulating CEA is misaligned by > [15 inches] but trippable, continued operation in MODES 1 and 2 may continue, provided, within 1 hour, the power is reduced to ≤ 70% RTP and SDM is ≥ [5.0]% Δk/k, and within 2 hours the misaligned CEA(s) is aligned within [7 inches] of its group, or the misaligned CEA's group is aligned within [7 inches] of the misaligned CEA(s).~~

~~Xenon redistribution in the core starts to occur as soon as a CEA becomes misaligned. Reducing THERMAL POWER in accordance with Figure 3.1.5-1 (in the associated LCO) ensures acceptable power distributions are maintained (Ref. 6). For small misalignments (< [15 inches]) of the CEAs, there is:~~

~~a. A small effect on the time dependent long term power distributions relative to those used in generating LCOs and Limiting Safety System Settings (LSSS) setpoints;~~

~~b. A small effect on the available SDM; and~~

~~c. A small effect on the ejected CEA worth used in the accident analysis.~~

~~With a large CEA misalignment (≥ [15 inches]), however, this misalignment would cause distortion of the core power distribution. This distortion may, in turn, have a significant effect on:~~

~~a. The available SDM;~~

~~b. The time dependent, long term power distributions relative to those used in generating LCOs and LSSS setpoints; and~~

~~c. The ejected CEA worth used in the accident analysis.~~

~~Therefore, this condition is limited to a single CEA misalignment, while still allowing 2 hours for recovery.~~

~~In both cases, a 2-hour time period is sufficient to:~~

~~a. Identify cause of a misaligned CEA;~~

~~b. Take appropriate corrective action to realign the CEAs; and~~

~~c. Minimize the effects of xenon redistribution.~~

~~If a CEA is untrippable, it is not available for reactivity insertion during a reactor trip. With an untrippable CEA, meeting the insertion limits of LCO 3.1.6 and LCO 3.1.7 does not ensure that adequate SDM exists. In this condition, an additional allowance must be made for the worth of the affected CEA when calculating the available SDM. This is necessary, since the OPERABLE CEAs must still meet the single failure criterion. If additional negative reactivity is required to provide the necessary SDM, it must be provided by increasing the RCS boron concentration. One hour allows sufficient time to perform the SDM calculation and initiate any required boron adjustment to the RCS.~~

A.1

A control rod is considered inoperable if it cannot be moved by its operator or if it cannot be tripped. In the event a control rod is declared inoperable, a SDM verification shall be performed in accordance with ST 3.1.1.1. This will ensure that the core is operated within the assumptions made in the safety analysis. A time period of 1 hour is adequate to perform this calculation.

A.2

In the event a control rod is declared inoperable, SHUTDOWN MARGIN requirements shall be maintained by increasing the boron concentration by an amount equivalent in reactivity to the worth of the inoperable control rod. The time period of 1 hour is adequate to initiate boration to maintain SHUTDOWN MARGIN.

~~B.1, B.2.1, B.2.2., and B.3~~

~~If one or more shutdown CEAs are misaligned by > [7 inches] and ≤ [15 inches] but trippable, or one shutdown CEA is misaligned by > [15 inches] but trippable, continued operation in MODES 1 and 2 may continue, provided, within 1 hour, the power is reduced to ≤ 70% RTP and SDM is ≥ [5.0]% Δk/k, and within 2 hours the misaligned CEA(s) is aligned within [7 inches] of its group.~~

B.1, B.2.1, and B.2.2

A control rod misaligned by more than 8 inches, but less than 15 inches shall be declared inoperable within 1 hour if the rod cannot be restored within limits. In addition, if the misaligned rod is not restored within 1 hour SR 3.2.2.1 shall be performed to verify hot channel factors are within design limits. In the event, the above actions cannot be met, THERMAL POWER shall be reduced to ≤ 75% RTP within 2 hours. This will ensure that hot channel factors are not exceeded. Continued operation with the effected rod fully inserted will only be permitted if the hot channel factors, SDM and ejected rod worths are satisfied.

~~C.1, C.2.1, and C.2.2~~

~~The CEA motion inhibit permits CEA motion within the requirements of LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits," and prevents regulating CEAs from being misaligned from other CEAs in the group.~~

~~Performing SR 3.1.5.1 within 1 hour and every 4 hours thereafter, is considered acceptable in view of other information continuously available to the operator in the control room.~~

~~With the CEA motion inhibit inoperable, a Completion Time of 6 hours is allowed for restoring the CEA motion inhibit to OPERABLE status, or placing and maintaining the CEA drive switch in either the "off" or "manual" position, fully withdrawing the CEAs in groups 3 and 4, and withdrawing all CEAs in group 5 to < 5% insertion.~~

~~Placing the CEA drive switch in the "off" or "manual" position ensures the CEAs will not move in response to Reactor Regulating System automatic motion commands. Withdrawal of the CEAs to the positions required in the Required Action C.2.2 ensures that core perturbations in local burnup, perking factors, and SDM will not be more adverse than the Conditions assumed in the safety analyses and LCO setpoint determination (Ref. 6).~~

~~The 6 hour Completion Time takes into account Required Action C.1, the protection afforded by the CEA deviation circuits, and other information continuously available to the operator in the control room, so that during actual CEA motion, deviations can be detected.~~

~~Required Action C.2.2 is modified by a Note indicating that performing this Required Action is not required when in conflict with Required Actions A.1, A.3.1, A.3.2, B.1, B.3, or D.1.~~

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

A control rod misaligned more than 20 inches from other rods in the same group shall be declared inoperable within 1 hour if it cannot be restored within limits. In the event, the effected rod is declared inoperable, SDM should be verified to within limits. In addition, THERMAL POWER shall be restricted to ≤ 50% RTP within 4 hours in the event peaking factors are being violated when SR 3.2.2.1 is performed. This will ensure that the hot channel factors will not be violated due to a skewed power distribution and subsequent power peaking. At ≤ 50% RTP ample thermal margin exists to ensure hot channel factors are within design limits and meet the assumptions stated in the safety analysis. The time restraints specified are adequate to achieve the required actions in a safe manner.

~~D.1~~

~~When the CEA deviation circuit is inoperable, performing SR 3.1.5.1, within 1 hour and every 4 hours thereafter, ensures improper CEA alignments are identified before unacceptable flux distributions occur. The specified Completion Times take into account other information continuously available to the operator in the control room, so that during CEA movement, deviations can be detected, and the protection provided by the CEA inhibit and deviation circuit is not required.~~

~~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 regulating or shutdown CEAs are untrippable, or two or more CEAs are misaligned by > [15 inches], the unit is required to be brought to MODE 3. By being brought to MODE 3, the unit is brought outside its MODE of applicability. Continued operation is not allowed in the case of more than one CEA misaligned from any other CEA in its group by > [15 inches], or one or more CEAs untrippable. This is because these cases are indicative of a loss of SDM and power distribution, and a loss of safety function, respectively.~~

~~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.~~

D.1 and E.1

In the event, the rod position deviation alarm is inoperable or 1 channel of position indication in inoperable SR 3.1.5.1 shall be performed within 15 minutes following rod motion. Thin is adequate assurance that the rod configuration will not change since Palisades only has manual rod motion as opposed to automatic rod control systems used in other plants. The rod position verification will ensure control rods are within limits. This ensures the assumptions in the safety analysis have been met.

BASES

If two or more control rods are inoperable or the required action and associated completion times are not met, the plant shall be in MODE 3 within 12 hours. The time period specified allows the plant to derate in a safe manner. The time period to bring the plant to MODE 3 is needed to mitigate any potential power peaks due to a skewed power distribution brought on by a mispositioned cruciform blade and inherent to the plants low leakage core design.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that individual ~~CEA~~ control rod positions are within ~~[7~~ 8 inches~~]~~ ~~(indicated reed switch positions)~~ of all other ~~CEAs~~ control rods in the group at a 12 hour Frequency allows the operator to detect a ~~CEA~~ control rod that is not in its expected position. ~~beginning to deviate from its expected position.~~ The specified Frequency takes into account other ~~CEA~~ control rod position information that is 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~~ control rod ~~motion inhibit and~~ deviation circuits.

SR 3.1.5.2

OPERABILITY of at least two ~~CEA~~ control rod position indicator channels is required to determine ~~CEA~~ control rod positions, and thereby ensure compliance with the ~~CEA~~ control rod alignment and insertion limits. The ~~CEA~~ control rod "full in" and "full out" corresponds to the Upper electrical limits and the lower electrical limit~~s~~ which provide an additional independent means for determining the ~~CEA~~ control rod positions when the ~~CEAs~~ 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 ~~CEA~~ control rod movement, deviations can be detected, and protection can be provided by the ~~CEA~~ control rod ~~motion inhibit and~~ deviation circuits.

SR 3.1.5.3

~~Demonstrating the CEA motion inhibit OPERABLE verifies that the CEA motion inhibit is functional, even if it is not regularly operated. The 31 day Frequency takes into account other information continuously available to the operator in the control room, so that during CEA movement, deviations can be detected, and protection can be provided by the CEA deviation circuits.~~

~~SURVEILLANCE REQUIREMENTS~~ ~~SR 3.1.5.4~~

Demonstrating the ~~CEA~~ control rod deviation circuit is OPERABLE verifies the circuit is functional. The ~~31~~ 92 day ~~F~~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~~ to ensure nuclear design limits are not violated.

SR 3.1.5.~~5~~4

Verifying each ~~CEA~~ 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~~ 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 ~~[5 inches]~~ ≥ 5 inches in either direction 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~~ control rods. ~~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. At any time, if a CEA(s) is immovable, a determination of the trippability (OPERABILITY) of the CEA(s) must be made, and appropriate action taken.~~

SR 3.1.5.~~6~~5

Performance of a CHANNEL ~~FUNCTIONAL TEST of each reed switch position transmitter channel ensures the channel is OPERABLE and capable of indicating CEA position over the entire length of the CEA'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 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 CEA Reed Switch Indication System.~~

BASES

CALIBRATION of each 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. Since the Surveillance must be performed when the reactor is shutdown, 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 Indication System.

SR 3.1.5.~~7~~6

Verification of ~~CEA~~ full length control rod drop times determined that the maximum ~~CEA~~ control rod drop time permitted is consistent with the assumed drop time used in that safety analysis (Ref. ~~7~~6). Measuring drop times prior to reactor criticality, after reactor vessel head installation removal, ensures that reactor internals and ~~CEDM~~ CRDMs will not interfere with ~~CEA~~ control rod motion or drop time and that no degradation in these systems has occurred that would adversely affect ~~CEA~~ control rod motion or drop time. Individual ~~CEAs~~ 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 ~~unit~~ plant outage and because of the potential for an unplanned ~~unit~~ plant transient if the Surveillance were performed with the reactor at power. The drop time includes the time span from trip signal actuation to 90% rod insertion.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26

2. 10 CFR 50.46

3. FSAR, Section ~~[ ]~~14.4

~~4. FSAR, Section [ ]~~

~~5~~4. FSAR, Section 14.4.8

~~6~~5. FSAR, Section 14.4.9

~~7. FSAR, Section [ ]~~

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 ~~Shutdown Control Element Assembly (CEA) Insertion Limits (Analog)~~ Rod Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown ~~CEAs~~ control rods are initial assumptions in all safety analyses that assume ~~CEA~~ control rod insertion upon reactor trip. The insertion limits directly affect core power distributions and assumptions of available SDM, ejected ~~CEA~~ control rod worth, 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" (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~~ control rod insertion have been established, and all ~~CEA~~ control rod positions are monitored and controlled during power operation to ensure that the reactivity limits, ejected ~~CEA~~ control rod worth, and SDM limits are preserved.

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

The design calculations are performed with the assumption that the shutdown ~~CEAs~~ rods are withdrawn prior to the regulating ~~CEAs~~ rods. The shutdown ~~CEAs~~ control 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 ~~CEAs~~ control rods are controlled manually or automatically by the control room operator. During normal ~~unit~~ plant operation, the shutdown ~~CEAs~~ rods are fully withdrawn. The shutdown ~~CEAs~~ rods must be completely withdrawn from the core prior to withdrawing any regulating ~~CEAs~~ rods during an approach to criticality. The shutdown ~~CEAs~~ rods are then left in this position until the reactor is shutdown. They affect core power, burnup distribution, and add negative reactivity to shutdown the reactor upon receipt of a reactor trip signal.

APPLICABLE SAFETY ANALYSES

Accident analysis assumes that the shutdown ~~CEAs~~ control rods 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 ~~CEA~~ control rod ejection accident are limited to acceptable limits.

~~CEAs~~ Control rods are considered fully withdrawn at ~~129~~128 inches, since this position places them ~~outside the active region of the core~~ in a very insignificant reactivity worth region of the integral worth curve for each bank.

On a reactor trip, all ~~CEAs~~ full length control rods (shutdown and regulating), except the most reactive ~~CEA~~ control rod, are assumed to insert into the core. The shutdown and regulating ~~CEAs~~ control rods shall be at their insertion limits and available to insert the maximum 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, "Regulating Control ~~Element Assembly (CEA)~~ Rod Insertion Limits." The shutdown ~~CEA~~ rod insertion limit is established to ensure that a sufficient amount of negative reactivity is available to shutdown the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM) ~~T_{avg}~~ T_{avg} ~~≥~~ > ~~200~~525°F") following a reactor trip from full power. The combination of regulating ~~CEAs~~ rods and shutdown ~~CEAs~~ rods (less the most reactive ~~CEA~~ control rod, which is assumed to be 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). The shutdown ~~CEA~~ rod insertion limit also limits the reactivity worth of an ejected shutdown ~~CEA~~ rod.

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

a. There be no violation of:

   1. Specified acceptable fuel design limits, or

   2. ~~Reactor~~ Primary Coolant System pressure boundary damage; and

b. The core remains subcritical after accident transients.

As such, the shutdown ~~CEA~~ rod insertion limits affect the safety analyses involving core reactivity, ejected ~~CEA~~ rod worth, and SDM (Ref. 3).

~~The shutdown CEA insertion limits satisfy Criterion 2 of the NRC Policy Statement.~~

BASES

LCO

The shutdown ~~CEAs~~ rods must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shutdown the reactor and maintain the required SDM following a reactor trip.

APPLICABILITY

The shutdown ~~CEAs~~ rods must be within their insertion limits, with the reactor in MODES 1 and 2. The Applicability in MODE 2 begins anytime any regulating ~~CEA~~ rod is ~~not fully inserted~~ withdrawn above 5 inches. This ensures that a sufficient amount of negative reactivity is available to shutdown the reactor and maintain the required SDM following a reactor trip. In MODES 1 and 2, if shutdown ~~CEAs~~ rods are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation (considering the listed reactivity effects in Bases Section SR 3.1.1.1). In MODE 3, 4, 5, or 6, the shutdown ~~CEAs~~ rods are fully inserted in the core and contribute to the SDM. Refer to LCOs 3.1.1 and 3.1.2, "SHUTDOWN MARGIN (SDM) ~~$T_{avg} \leq 200$~~ $T_{ave} \leq 525$° F," for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

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~~ control rods to move, and requires the shutdown ~~CEAs~~ rods to move below the LCO limits, which would normally violate the LCO.

ACTIONS

~~A.1.1, A.1.2, and A.2~~
A.1, A.2 and A.3

Prior to entering this condition, the shutdown ~~CEAs~~ rods were fully withdrawn. If a shutdown ~~CEA~~ rod~~(s)~~ is then inserted into the core, its potential negative reactivity is added to the 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~~ control rod(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."~~

BASES

In the event one shutdown or part length rod is not within limits and other actions have not been performed, the rod shall be declared inoperable within 1 hour. This places the condition into LCO 3.1.5, "Rod Operability and Alignment" which allows for SDM verification of boration. This ensures that the rod is no longer credited and safety analysis assumptions are still preserved.

When a shutdown or part length rod is not within limits, the rod can be declared inoperable within 1 hour time period. This action is performed if other options have not been elected. This reflects Palisades ability to run with an inoperable control rod. The 1 hour time period reflects the importance of addressing this condition when entered.

B.1 and B.2

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~~12 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems. Due to Palisades low leakage core design the 12 hour Completion Time is required to not initiate any instability in the core while deescalating. A misaligned rod could cause a very skewed power distribution; therefore, a slower derate would be prudent.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

~~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 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 are withdrawn during a unit startup.~~

Since the shutdown ~~CEAs~~ rods are positioned manually by the control room operator, verification of shutdown ~~CEA~~ rod position at a Frequency of 12 hours is adequate to ensure that the shutdown ~~CEAs~~ rods 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~~ rods.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26
2. 10 CFR 50.46
3. FSAR, Section ~~[ ]~~14.2

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Regulating ~~Control Element Assembly (CEA)~~ Rod Insertion Limits ~~(Analog)~~

BASES

BACKGROUND

The insertion limits of the regulating ~~CEAs~~ rods 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" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling ~~s~~Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Limits on regulating ~~CEA~~ rod insertion have been established, and all ~~CEA~~ 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, ejected ~~CEA~~ control rod worth, reactivity insertion rate, and SDM limits are preserved.

The regulating ~~CEA~~ rod groups operate with a predetermined amount of position overlap, in order to approximate a linear relation between ~~CEA~~ control rod worth and ~~CEA~~ control rod position (integral ~~CEA~~ control 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.

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~~ changing 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 3.1.7, "Regulating ~~Control Element Assembly (CEA)~~ Rod Insertion Limits"; LCO 3.2.4, "~~AZIMUTHAL~~ Quadrant POWER TILT (T_q)"; and LCO 3.2.5, "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 radial peaking factor (F^T_{xy}) (LCO 3.2.2, "Total Planar Radial Peaking Factor F^T_{xy})~~ assembly radial peaking factor F^A_r LCO 3.2.2, "Assembly Radial Peaking; and total integrated radial peaking factor (F^T_r) (LCO 3.2.3, "Total Integrated Radial Peaking Factor (F^T_r)") limits in the COLR. Operation within the LHR limits given in the COLR prevents power peaks that would exceed the ~~1~~Loss ~~o~~Of ~~c~~Coolant ~~a~~Accident (LOCA) limits derived by the Emergency Core Cooling System analysis. Operation within the F'^{xy} and F^T_r limits given in the COLR prevents ~~d~~Departure from ~~n~~Nucleate ~~b~~Boiling (DNB) during a loss of forced reactor coolant flow accident. In addition to the LHR, F^T_{xy}, F^A_r, and F^T_r limits, certain reactivity limits are preserved by regulating ~~CEA~~ rod insertion limits. The regulating ~~CEA~~ rod insertion limits also restrict the ejected ~~CEA~~ control rod worth to the values assumed in the safety analysis and preserve the minimum required SDM in MODES 1 and 2. The ejected rod case is limited to the reactivity worth for the highest worth rod ejected from the PDIL limit, thus limiting the maximum possible reactivity excursion.

The establishment of limiting safety system settings and LCOs requires that the expected long and short term behavior of the radial peaking factors be determined. The long term behavior relates to the variation of the steady state radial peaking factors with core burnup and is affected by the amount of ~~CEA~~ control rod insertion assumed, the portion of a burnup cycle over which such ~~insertion~~ assertion is assumed and the expected power level variation throughout the cycle.

The long ~~short~~ term behavior relates to ~~transient perturbations to the~~ steady state ~~radial peaks~~ full power operation. ~~due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions and load maneuvering. Analyses are performed, based on the expected mode of operation of the Nuclear Steam Supply System (base loaded, maneuvering, etc.). From these analyses, CEA insertions are determined and a consistent set of radial peaking factors defined. The long term steady state and short term insertion limits are determined, based upon the assumed mode of operation used in the analyses, and provide a means of preserving the assumption on CEA insertions used. The long and short term insertion limits of LCO 3.1.7 are specified for the plant, which has been designed primarily for base loaded operation, but has the ability to accommodate a limited amount of load maneuvering.~~This condition of operation and resultant radial peaking factors are ensured by LCO 3.1.6 and SR 3.2.2.1 radial peaking factor verification. This ensures that the core is operated within nuclear design limits and verifies assumptions assumed in the safety analysis.

The short term behavior relates to transient perturbations to the steady state radial peaking factors. The magnitudes of such perturbations depend upon the expected use of control rods during transient mitigation. The PDIL curve stated on the COLR dictates the acceptable control rod positioning for anticipated power maneuvers and transient mitigation within the limits. The PDIL limitations stated in the COLR reflect the assumptions made in the safety analyses. This ensures that radial peaking is not violated during power level maneuvering or transient mitigation.

The regulating ~~CEA~~ rod insertion and alignment limits are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the regulating bank insertion limits control the reactivity that could be added in the event of a ~~CEA~~ control rod ejection accident, and the shutdown and regulating bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the ~~reactor~~ primary coolant in the event of a LOCA, loss of flow, ejected ~~CEA~~ control rod, or other accident requiring termination by a Reactor Protection System trip function.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition I) and anticipated operational occurrences (Condition II). The acceptance criteria for the regulating ~~CEA~~ rod insertion, ASI, and T_q LCOs are such as to preclude core power distributions from occurring that would violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed a limit of 2200°F, 10 CFR 50.46 (Ref. 2);

b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot ~~fuel CEA~~ channel control rod in the core does not experience a DNB condition.

c. During an ejected ~~CEA~~ control rod accident, the fission energy input to the fuel must not exceed ~~280~~200 cal/gm (Ref. 3); and

d. The ~~CEAs~~ control rods must be capable of shutting down the reactor with a minimum required SDM, with the highest worth ~~CEA~~ control rod stuck fully withdrawn, GDC 26 (Ref. 1).

Regulating ~~CEA~~ rod position, ASI, and T_q are process variables that together characterize and control the three dimensional power distribution of the reactor core.

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 ~~CEA~~ rod insertion limits, so that the allowable inserted worth of the ~~CEAs~~ control rods is such that sufficient reactivity is available to shutdown the reactor to hot zero power. SDM assumes the maximum worth ~~CEA~~ control rod remains fully withdrawn upon trip (Ref. 4).

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 ~~CEA~~ control rod worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected ~~CEA~~ control rod worth.

BASES

The regulating and shutdown ~~CEA~~ rod insertion limits ensure that safety analyses assumptions for reactivity insertion rate, SDM, ejected ~~CEA~~ control rod worth, and power distribution peaking factors are preserved ~~(Ref. 5)~~.

~~The regulating CEA insertion limits satisfy Criterion 2 of the NRC Policy Statement.~~

LCO

The limits on regulating ~~CEAs~~ rods 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 ~~CEA~~ control rod worth is maintained, and ensuring adequate negative reactivity insertion on trip. The overlap between regulating banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during regulating ~~CEA~~ rod motion.

The ~~p~~Power ~~d~~Dependent ~~i~~Insertion ~~l~~Limit (PDIL) alarm circuit is required to be OPERABLE for notification that the ~~CEAs~~ rods are outside the required insertion limits. When the PDIL alarm circuit is inoperable, the verification of ~~CEA~~ control rod positions is increased to ensure improper ~~CEA~~ control rod alignment is identified before unacceptable flux distribution occurs.

APPLICABILITY

The regulating ~~CEA~~ rod 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~~ ejected control 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 ~~CEA~~ control 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.

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~~ control rods to move, and requires the regulating ~~CEAs~~ rods 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.~~

ACTIONS

A.1.1, A.1.2, and 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~~ rod 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~~ control rod 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~~ within the limits or to reduce THERMAL POWER to less than or equal to that allowed for the actual ~~CEA~~ control rod insertion limit. ~~Two hours~~ A two hour limit provides a reasonable time to accomplish this, allowing the operator to deal with current plant conditions while limiting peaking factors to acceptable levels.

B.1~~ and B.2~~

~~If the CEAs are inserted between the long term steady state insertion limits and the transient insertion limits for intervals > 4 hours per 24 hour period, and the short term steady state insertions are exceeded, peaking factors can develop that are of immediate concern (Ref. 6).~~

~~Verifying the short term steady state insertion limits are not exceeded ensures that the peaking factors that do develop are within those allowed for continued operation. Fifteen minutes provides adequate time for the operator to verify if the short term steady state insertion limits are exceeded.~~

~~Experience has shown that rapid power increases in areas of the core, in which the flux has been depressed, can result in fuel damage, as the LHR in those areas rapidly increases. Restricting the rate of THERMAL POWER increases to ≤ 5% RTP per hour, following CEA insertion beyond the long term steady state insertion limits, ensures the power transients experienced by the fuel will not result in fuel failure (Ref. 7).~~
Verifying SHUTDOWN MARGIN within 1 hour allows for the ensurance that the assumptions made in the safety analysis are maintained. A 1 hour completion time is an adequate period of time to verify SDM is within limits.

In addition, the out of sequence rods must be restored within appropriate sequence within one hour. This places the plant back into the configuration assumed by the Palisades safety analysis.

C.1

~~With the regulating CEAs inserted between the long term steady state insertion limit and the transient insertion limit, and with the core approaching the 5 effective full power days (EFPD) per 30 EFPD or 14 EFPD per 365 EFPD limits, the core approaches the acceptable limits placed on operation with flux patterns outside those assumed in the long term burnup assumptions (Ref. 8). In this case, the CEAs must be returned to within the long term steady state insertion limits, or the core must be placed in a condition in which the abnormal fuel burnup cannot continue. A Completion Time of 2 hours is allotted to return the CEAs to within the long term steady state insertion limits.~~

The required Completion Time of 2 hours from initial discovery of a regulating ~~CEA~~ control rod group outside the limits until its restoration to within ~~the long term steady state limits~~ PDIL, shown on the figures in the COLR, allows sufficient time for borated water to enter the ~~Reactor~~ Primary Coolant System from the chemical addition and makeup systems, and to cause the regulating ~~CEAs~~ control rods to withdraw to the acceptable region. It is reasonable to continue operation for 2 hours after it is discovered ~~that the 5 day or 14 day EFPD limit has been exceeded~~. This Completion Time is based on limiting the potential xenon redistribution, the low probability of an accident, and the steps required to complete the action.

D.1

~~When the PDIL alarm circuit is inoperable, performing SR 3.1.7.1 within 1 hour and once per 4 hours thereafter ensures improper CEA alignments are identified before unacceptable flux distributions occur.~~

~~E.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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

With the PDIL alarm circuit OPERABLE, verification of each regulating ~~CEA~~ rod group position every 12 hours is sufficient to detect ~~CEA~~ control 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 ~~CEA~~ control rod group positions available to the operator in the control room.

SR 3.1.7.1 is modified by a Note indicating that entry is allowed into MODE 2 without having performed the SR. This is necessary, since the unit must be in the applicable MODES in order to perform Surveillances that demonstrate the LCO limits are met.

~~SR 3.1.7.2~~

~~Verification of the accumulated time of CEA group insertion between the long term steady state insertion limits and the transient insertion limits ensures the cumulative time limits are not exceeded. The 24 hour Frequency ensures the operator identifies a time limit that is being approached before it is reached.~~

SR 3.1.7.~~3~~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 ~~CEA~~ control rod alignments.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26
2. 10 CFR 50.46
3. FSAR, Section ~~[ ], Section [ ], and Section [ ]~~14.4.2
4. FSAR, Section 14.4.8
5. ~~FSAR, Section [ ]~~
6. ~~FSAR, Section [ ]~~
7. ~~FSAR, Section [ ]~~
8. ~~FSAR, Section [ ]~~

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Special Test Exception ~~(STE) SHUTDOWN MARGIN (SDM) (Analog)~~ Start-Up Physics Testing (SPT)

BASES

BACKGROUND

The primary purpose of the SDM STE is to permit relaxation of existing LCOs to allow the performance of certain PHYSICS TESTS. These tests are constructed to determine the control ~~element assembly (CEA)~~ rod worth and SDM.

Section XI of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants" (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59, "Changes, Tests, and Experiments" (Ref. 2).

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in design and analysis;

c. Verify assumptions used for predicting plant response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that operating and emergency procedures are adequate.

To accomplish these objectives, testing is required prior to initial criticality, after each refueling shutdown, and during startup, low power operation, power ascension, and at power operation. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions, and that the core can be operated as designed (Ref. 4).

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long term power operation. Examples of PHYSICS TESTS include determination of critical boron concentration, ~~CEA~~ control rod group worths, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because adequate limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

Reference 5 defines the requirements for initial testing of the facility, including PHYSICS TESTS. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits of all LCO~~s~~, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. As long as the ~~l~~Linear ~~h~~Heat ~~r~~Rate (LHR) remains within its limit, fuel design criteria are preserved.

~~In this test, the following LCOs are suspended:~~

a. LCO 3.1.1, "SHUTDOWN MARGIN (SDM) ~~T_{avg}~~ → $T_{ave} \geq$ ~~200~~525°F"; and

b. LCO 3.1.2, "SHUTDOWN MARGIN (SDM) ~~T_{ave}~~ 525°F"; and

c. LCO 3.1.5, "Control Rod Alignment";

d. LCO 3.1.6, "SHUTDOWN Rod Insertion Limits";

~~b~~e. LCO 3.1.7, "Regulating ~~Control Element Assembly (CEA)~~ Rod Insertion Limits."

Therefore, this LCO places limits on the minimum amount of ~~CEA~~ control rod worth required to be available for reactivity control when ~~CEA~~ control rod worth measurements are performed.

The individual LCOs cited above govern SDM ~~CEA~~ control rod group height, insertion, and alignment. Additionally, the LCOs governing ~~Reactor~~ Primary Coolant System (~~RCS~~PCS) flow, reactor inlet temperature, and pressurizer pressure contribute to maintaining ~~d~~Departure from ~~n~~Nucleate ~~b~~Boiling (DNB) parameter limits. The initial condition criteria for accidents sensitive to core power distribution are preserved by the LHR and DNB parameter limits. The criteria for the ~~l~~Loss ~~o~~Of ~~c~~Coolant ~~a~~Accident (LOCA) are specified in 10 CFR 50.46, "Acceptance Criteria for Emergency ~~c~~Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 6). The criteria for the loss of forced reactor coolant flow accident are specified in Reference 7. Operation within the LHR limit preserves the LOCA criteria; operation within the DNB parameter limits preserves the loss of flow criteria.

SRs are conducted as necessary to ensure that LHR and DNB parameters remain within limits during PHYSICS TESTS. Performance of these SRs allows PHYSICS TESTS to be conducted without decreasing the margin of safety.

Requiring that shutdown reactivity equivalent to at least the highest estimated ~~CEA~~ control rod worth (of those ~~CEAs~~ control rods actually withdrawn) be available for trip insertion from the OPERABLE ~~CEA~~ control rod provides a high degree of assurance that shutdown capability is maintained for the most challenging postulated accident, a stuck ~~CEA~~ control rod. Since LCO 3.1.1 is suspended, however, there is not the same degree of assurance during this test that the reactor would always be shut down if the highest worth ~~CEA~~ control rod was stuck out and calculational uncertainties or the estimated highest ~~CEA~~ control rod worth was not as expected (the single failure criterion is not met). This situation is judged acceptable, however, because specified acceptable fuel damage limits are still met. The risk of experiencing a stuck ~~CEA~~ control rod and subsequent criticality is reduced during this PHYSICS TEST exception by the requirements to determine ~~CEA~~ control rod positions every 2 hours; by the trip of each ~~CEA~~ control rod to be withdrawn ~~24 hours~~ 7 days prior to suspending the SDM; and by ensuring that shutdown reactivity is available, equivalent to the reactivity worth of the estimated highest worth withdrawn ~~CEA~~ control rod (Ref. 5).

BASES

PHYSICS TESTS include measurement of core parameters or exercise of control components that affect process variables. Among the process variables involved are total ~~planar radial peaking factor, total integrated radial peaking factor~~ assembly radial peaking total pin radial peaking, T_q and ASI, which represent initial condition input (power peaking) to the accident analysis. Also involved are the shutdown and regulating ~~CEAs~~ rods, which affect power peaking and are required for shutdown of the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Policy Statement.

LCO

This LCO provides that a minimum amount of ~~CEA~~ control rod worth is immediately available for reactivity control when ~~CEA~~ control rod worth measurement tests are performed. The STE is required to permit the periodic verification of the actual versus predicted core reactivity conditions. ~~occurring as a result of fuel burnup or fuel cycling operations.~~ The SDM requirements of LCOs 3.1.1 ~~and the regulating CEA insertion limits of LCO 3.1.7~~ , 3.1.2, 3.1.5, 3.1.6, and 3.1.7 may be suspended.

APPLICABILITY

This LCO is applicable in MODES 2 and 3. Although ~~CEA~~ control rod worth testing is conducted in MODE 2, sufficient negative reactivity is inserted during the performance of these tests to result in temporary entry into MODE 3. Because the intent is to immediately return to MODE 2 to continue ~~CEA~~ control rod worth measurements, the STE allows limited operation to 6 consecutive hours in MODE 3, as indicated by the Note, without having to borate to meet the SDM requirements of LCO 3.1.1.

BASES

ACTIONS

A.1

With any ~~CEA~~ control rod not fully inserted and less than the minimum required reactivity equivalent available for insertion, or with all ~~CEAs~~ control rods inserted and the reactor subcritical by less than the reactivity equivalent of the highest worth ~~CEA~~ control rod, restoration of the minimum SDM requirements must be accomplished by increasing the ~~RCS~~PCS boron concentration. The required Completion Time of 15 minutes for initiating boration allows the operator sufficient time to align the valves and start the boric acid pumps and is consistent with the Completion Time of LCO 3.1.1.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification of the position of each partially or fully withdrawn full length or part length ~~CEA~~ control rod is necessary to ensure that the minimum negative reactivity requirements for insertion on a trip are preserved. A 2 hour Frequency is sufficient for the operator to verify that each ~~CEA~~ control rod position is within the acceptance criteria.

SR 3.1.8.2

Prior demonstration that each ~~CEA~~ control rod to be withdrawn from the core during PHYSICS TESTS is capable of full insertion, when tripped from at least a 50% withdrawn position, ensures that the ~~CEA~~ control rod will insert on a trip signal. The ~~[~~7 day~~]~~ Frequency ensures that the ~~CEAs~~ control rods are OPERABLE prior to reducing SDM to less than the limits of LCO 3.1.1.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI

2. 10 CFR 50.59

3. Regulatory Guide 1.68, Revision 2, August 1978

4. ANSI/ANS-19.6.1-1985, December 13, 1985

5. FSAR, Chapter ~~[~~14~~]~~.4.8

6. 10 CFR 50.46

~~7. FSAR, Chapter [14]~~

~~B 3.1 REACTIVITY CONTROL SYSTEMS~~

~~B 3.1.9 Special Test Exceptions (STE) MODES 1 and 2 (Analog)~~

~~BASES~~

~~BACKGROUND~~ ~~The primary purpose of these MODES 1 and 2 STEs is to permit relaxation of existing LCOs to allow the performance of certain PHYSICS TESTS. These tests are conducted to determine specific reactor core characteristics.~~

~~Section XI of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants" (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59, "Changes, Tests, and Experiments" (Ref. 2).~~

~~The key objectives of a test program are to (Ref. 3):~~

~~a. Ensure that the facility has been adequately designed;~~

~~b. Validate the analytical models used in design and analysis;~~

~~c. Verify assumptions used for predicting plant response;~~

~~d. Ensure that installation of equipment in the facility has been accomplished in accordance with design; and~~

~~e. Verify that operating and emergency procedures are adequate.~~

~~To accomplish these objectives, testing is required prior to initial criticality, after each refueling shutdown, and during startup, low power operation, power ascension, and at power operation. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions, and that the core can be operated as designed (Ref. 4).~~

~~PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of testing required to ensure that design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.~~

~~BASES~~

~~Examples of PHYSICS TESTS include determination of critical boron concentration, control element assembly (CEA) group worths, reactivity coefficients, flux symmetry, and core power distribution.~~

~~APPLICABLE SAFETY ANALYSES~~

~~It is acceptable to suspend certain LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded. Even if an accident occurs during a PHYSICS TEST with one or more LCOs suspended, fuel damage criteria are preserved because the limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.~~

~~Reference 5 defines the requirements for initial testing of the facility, including PHYSICS TESTS. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS 19.6.1 1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. As long as the linear heat rate (LHR) remains within its limit, fuel design criteria are preserved.~~

~~In this test, the following LCOs are suspended:~~

~~LCO 3.1.4, "Moderator Temperature Coefficient (MTC)";~~
~~LCO 3.1.5, "Control Element Assembly (CEA) Alignment";~~
~~LCO 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits";~~
~~LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits";~~
~~LCO 3.2.2, "Total Planar Radial Peaking Factor (F^T_{xy})";~~
~~LCO 3.2.3, "Total Integrated Radial Peaking Factor (F^T_r)"; and~~
~~LCO 3.2.4, "AZIMUTHAL POWER TILT (T_q)."~~

~~The safety analysis (Ref. 6) places limits on allowable THERMAL POWER during PHYSICS TESTS and requires the LHR and the departure from nucleate boiling (DNB) parameter to be maintained within limits. The power plateau of < 85% RTP and the associated trip setpoints are required to ensure [explain]. SDM shall be maintained ≥ [4.5]% Δk/k.~~

~~The individual LCOs governing CEA group height, insertion and alignment, ASI, F^T_{xy}, F^T_r, and T_q preserve the LHR limits. Additionally, the LCOs governing Reactor Coolant System (RCS) flow, reactor inlet temperature (T_c), and pressurizer pressure contribute to maintaining DNB parameter limits. The initial condition criteria for accidents sensitive to core power distribution are preserved by the LHR and DNB parameter limits. The criteria for the loss of coolant accident (LOCA) are specified in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 7). The criteria for the loss of forced reactor coolant flow accident are specified in Reference 7. Operation within the LHR limit preserves the LOCA criteria; operation within the DNB parameter limits preserves the loss of flow criteria.~~

~~During PHYSICS TESTS, one or more of the LCOs that normally preserve the LHR and DNB parameter limits may be suspended. The results of the accident analysis are not adversely impacted, however, if LHR and DNB parameters are verified to be within their limits while the LCOs are suspended. Therefore, SRs are placed as necessary to ensure that LHR and DNB parameters remain within limits during PHYSICS TESTS. Performance of these Surveillances allows PHYSICS TESTS to be conducted without decreasing the margin of safety.~~

~~PHYSICS TESTS include measurement of core parameters or exercise of control components that affect process variables. Among the process variables involved are F^T_{xy}, F^T_r, T_q, and ASI, which represent initial condition input (power peaking) to the accident analysis. Also involved are the shutdown and regulating CEAs, which affect power peaking and are required for shutdown of the reactor. The limits for these variables are specified for each fuel cycle in the COLR.~~

~~PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Policy Statement.~~

~~LCO~~ ~~This LCO permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of PHYSICS TESTS such as those required to:~~

~~a. Measure CEA worth;~~

~~b. Determine the reactor stability index and damping factor under xenon oscillation conditions;~~

~~c. Determine power distributions for nonnormal CEA configurations;~~

~~BASES~~

~~d. Measure rod shadowing factors; and~~

~~e. Measure temperature and power coefficients.~~

~~Additionally, it permits the center CEA to be misaligned during PHYSICS TESTS required to determine the isothermal temperature coefficient (ITC), MTC, and power coefficient.~~

~~The requirements of LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.1.7, LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4 may be suspended during the performance of PHYSICS TESTS, provided:~~

~~a. THERMAL POWER is restricted to test power plateau, which shall not exceed 85% RTP;~~

~~b. SDM shall be ≥ [4.5]% Δk/k.~~

~~APPLICABILITY~~ ~~This LCO is applicable in MODES 1 and 2 because the reactor must be critical at various THERMAL POWER levels to perform the PHYSICS TESTS described in the LCO section. Limiting the test power plateau to < 85% RTP ensures that LHRs are maintained within acceptable limits.~~

~~ACTIONS~~ ~~A.1~~

~~If THERMAL POWER exceeds the test power plateau, THERMAL POWER must be reduced to restore the additional thermal margin provided by the reduction. The 15 minute Completion Time ensures that prompt action shall be taken to reduce THERMAL POWER to within acceptable limits.~~

~~B.1 and B.2~~

~~If the SDM requirement is 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. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until the SDM is within limit.~~

~~Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.~~

~~BASES~~

~~C.1 and C.2~~

~~If Required Action A.1 or B.1 cannot be completed within the required Completion Time, PHYSICS TESTS must be suspended within 1 hour, and the reactor must be brought to MODE 3. Allowing 1 hour for suspending PHYSICS TESTS allows the operator sufficient time to change any abnormal CEA configuration back to within the limits of LCO 3.1.5, LCO 3.1.6, and LCO 3.1.7. Bringing the reactor to MODE 3 within 6 hours increases thermal margin and is consistent with the Required Actions of the power distribution LCOs. The required Completion Time of 6 hours is adequate for performing a controlled shutdown from full power conditions in an orderly manner and without challenging plant systems, and is consistent with power distribution LCO Completion Times.~~

~~SURVEILLANCE REQUIREMENTS~~

~~SR 3.1.9.1~~

~~Verifying that THERMAL POWER is equal to or less than that allowed by the test power plateau, as specified in the PHYSICS TEST procedure and required by the safety analysis, ensures that adequate LHR and DNB parameter margins are maintained while LCOs are suspended. The 1 hour Frequency is sufficient, based on the slow rate of power change and increased operational controls in place during PHYSICS TESTS.~~

~~SR 3.1.9.2~~

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

~~a. RCS boron concentration;~~

~~b. Control bank position;~~

~~c. RCS average temperature;~~

~~d. Fuel burnup based on gross thermal energy generation;~~

~~e. Xenon concentration;~~

~~f. Samarium concentration; and~~

~~g. 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 RCS.~~

~~BASES~~

~~The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.~~

~~REFERENCES~~

~~1. 10 CFR 50, Appendix B, Section XI.~~

~~2. 10 CFR 50.59.~~

~~3. Regulatory Guide 1.68, Revision 2, August 1978.~~

~~4. ANSI/ANS 19.6.1 1985, December 13, 1985.~~

~~5. FSAR, Chapter [14].~~

~~6. FSAR, Section [15.3.2.1].~~

~~7. 10 CFR 50.46.~~

ATTACHMENT 6

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

3.1 REACTIVITY CONTROL SYSTEMS

Comparison of Revised and Standard Technical Specifications

Palisades Revised Tech Spec Requirement List. (03/28/96)

A listing of the proposed Palisades Revised Tech Specs (RTS) correlated to the CE Standard Tech Specs (STS).

First Column; Proposed Palisades Revised Tech Spec (RTS) number
Each RTS item is listed in the left-most column.
If a STS item has been omitted from RTS, the word 'Omitted' is used.

Second Column; CE Standard Tech Spec (STS) number
The corresponding STS item is listed in the second column.
If a RTS item does not appear in STS, it is noted as 'Added'.

Third Column; Existing Palisades Tech Spec (TS) number
The closest TS item is listed in the third column.
If a RTS item does not appear in TS, it is noted as 'New'.

Fourth Column; RTS Item Description
An abbreviation of the RTS item appears in the third column.
Each item is identified as: LCO, ACTION, SR, ADMIN, Exception, etc.
In cases where a STS item was omitted from RTS, the description is of the STS item.

Description Key: RTS requirement type:	Column 4 syntax:
Safety Limit	SL: Safety limit; Applicable conditions
Limiting Condition for Operation	LCO: LCO Description; Applicable conditions
Condition	COND: Description of non-conforming condition
Action	ACTN: Required action; Completion time
Surveillance Requirement	SR: Test description; Frequency
Table	TABL: Title
Administrative Requirement	ADMN: Administrative requirement
Defined Term	DEF: Name of defined term

Fifth Column; Comments and Explanations of Differences between RTS and STS.
A brief explanation of differences between RTS and STS is provided in the fifth column.

Other abbreviations used in the listing are:

NA:	Not Applicable
CFT:	Channel Functional Test
CHNL:	Channel

RTS Number	STS Number	TS Number	RTS (STS) requirement Description	Explanation of Differences

Global differences between the proposed Palisades Technical Specifications and the Standard Technical Specifications for CE plants, Nureg 1432:

The following changes are not discussed in the explanation of differences for each TS requirement.

1) Bracketed values have been replaced with appropriate values for Palisades. Typically, the basis for these values is provided in the bases document.

2) Each required action of the form "Perform SR X.X.X.X . . ." has been altered by a parenthetical summary of the SR requirements. This change allows a reader to understand the required actions without constantly turning pages to locate the referenced SR.

3) Terminology has been changed to reflect Palisades usage:

"RWT"	becomes	"SIRWT"	Safety Injection Refueling Water Tank
"CEA"	becomes	"Control Rod" or "Rod"	Palisades uses cruciform control rods rather than the multifingered "Control Element Assemblies" of later CE plants.
"RCS"	becomes	"PCS"	Palisades terminology is "Primary Coolant System" rather than "Reactor Coolant System"
"SIAS"	becomes	"SIS"	Palisades terminology is "Safety Injection Signal" rather than "Safety Injection Actuation Signal"
"AC Vital bus"	becomes	"Preferred AC bus"	Palisades terminology.
"PAMI"	becomes	"AMI"	Accident Monitoring Instrumentation, Palisades terminology
"ESFAS"	becomes	"ESF Instrumentation"	There is no stand-alone ESFAS system or cabinet at Palisades; ESF instruments actuate the ESF functions
"DG LOVS"	becomes	"DG UV Start"	Palisades Terminology
"Remote Shutdown System"	becomes	"Alternate Shutdown System"	Palisades Terminology
"Power Rate of Change-High"	becomes	"High Startup Rate"	Palisades Terminology

RTS Number	STS Number	TS Number	RTS (STS) requirement Description	Explanation of Differences
3.1	3.1	3.10	REACTIVITY CONTROL SECTION	Several major differences exist between Palisades and the "Standard" CE plant which affect this section: Palisades is the oldest CE PWR and has different hardware and analyses from the newer CE plants; Palisades also uses Siemens Power Corp.(SPC) fuel rather than CE fuel. Therefore several of the LCOs, Actions, etc in this section differ from RSTS. Actions, Completion times, SRs and frequencies were kept as close to the RSTS as possible while implementing a different set of limitations and requirements. The conditions and actions specified reflect current tech specs and operating practice.
3.1.1	3.1.1	3.10.1.a	LCO: Shutdown margin ≥ 2%; MODE 3 ≥ 525 F	Used Palisades value (2%) and applicability. Palisades Safety Analysis assumes 2% SDM as an initial condition therefore; Siemens Power Corp. analyzes to this value for this applicability range. MODES 1 and 2 SDM requirements are assured by LCO 3.1.6 and 3.1.7. Reactivity units changed to meet industry standard.
3.1.1 A	3.1.1 A	3.0.3	COND: SDM < limit	Unchanged.
3.1.1 A.1	3.1.1 A.1	3.0.3	ACTN: Borate to restore SDM; 15 Min.	Unchanged.
3.1.1.1	3.1.1.1	New	SR: Verify SDM; 24 hours	Reworded SR to agree with SR 3.1.2.1, which must support 2 different limits. Wording of 2 SRs requiring the same task should be worded alike.
3.1.2	3.1.2	3.10.b	LCO: SDM shall be ≥ 3.75% MODE 3 < 525, MODE 4, and 5	Used Palisades bounding value for former < 4 PCP operation. This value is an initial assumption to the Palisades safety analysis and allows for the maintenance of SL integrity in the event of a DBA for stated range of applicability. Reactivity units changed to meet industry standard.
3.1.2 A	3.1.1 A	3.0.3	COND: SDM < limit	Unchanged.
3.1.2 A.1	3.1.2 A.1	3.0.3	ACTN: Borate to restore SDM; 15 Min.	Unchanged.
3.1.2.1	3.1.2.1	New	SR: Verify SDM; Within 2 hours following a RX trip; 24 hours	Changed wording to remove SDM limit. Added frequency of verifying SDM within 2 hours following a reactor trip or shutdown. This added frequency allows SDM to be verified within 2 hours from a reactor trip. This circumvents the case were the reactor trips 1 minute after the 24 hour surveillance was performed and then would not be required to be performed for 23 hours and 59 minutes following a reactor trip.
3.1.3	3.1.3	4.10	LCO: Reactivity Balance; MODES 1 & 2	Unchanged.

RTS Number	STS Number	TS Number	RTS (STS) requirement Description	Explanation of Differences
3.1.3 A	3.1.3 A	4.10	COND: Core reactivity balance not w/in limit	Reworded to agree with LCO.
3.1.3 A.1	3.1.3 A.1	4.10	ACTN: Determine Rx OK; 72 hrs	Unchanged.
3.1.3 A.2	3.1.3.A.2	4.10	ACTN: Establish restrictions; 72 hrs	Unchanged.
3.1.3 B	3.1.3 B	4.10	COND: Required action & completion time not met	Unchanged.
3.1.3 B.1	3.1.3 B.1	4.10	ACTN: Be in MODE 3; 6 hours	Unchanged.
3.1.3.1	3.1.3.1	4.10	SR: Verify core reactivity balance OK; 31 days	Reworded to agree with LCO.
3.1.4	3.1.4	3.12	LCO: MTC w/in limits stated in the COLR; MODES 1, and 2	Added maximum positive value from current license.
3.1.4 A	3.1.4 A	3.0.3	COND: MTC not w/in limits	Unchanged.
3.1.4 A.1	3.1.4 A.1	3.0.3	ACTN: MODE 3; 6 hours	Unchanged.
3.1.4.1	3.1.4.1	4.20.1	SR: Verify MTC w/in limits in COLR.	Retained only 'Prior to MODE 1 operation after each refueling' frequency. This frequency meets ANSI standard 19.6.1 for startup physics testing. The Beginning Of Core (BOC) MTC value is an adequate verification of nuclear methods for predicting MTC. Determining the BOC MTC yields the greatest challenge to nuclear methods prediction due to excessively high boron concentrations at BOC. Therefore, the current SR is adequate to ensure that the MTC is within design limits throughout the fuel cycle.
Omitted	3.1.4.2	NA	SR: (Mid-Cycle MTC Test)	See note for SR 3.1.4.1.
3.1.5	3.1.5	3.10.5	LCO: Control Rod Operability and Alignment	Used Palisades terms and values; omitted CEA Motion Inhibit, which has no equivalent at Palisades since Palisades uses cruciform control blades. Conditions and required actions were changed to reflect different hardware, analyses, and operating practice.
3.1.5 A	New	3.10.4.b	COND: One rod inoperable	See note for LCO 3.1.5
3.1.5 A.1	3.1.5 A.2.1	3.10.1.d	ACTN: Verify SDM; 1 hr	Unchanged.
3.1.5 A.2	3.1.5 A.2.2	3.10.4.c	ACTN: Initiate boration to restore SDM; 1 hr	Unchanged.
Omitted	3.1.5 A.3.1	3.0.3	ACTN: (Restore misaligned rod; 2 hrs)	It is implied that restoration to a condition that places the plant within the limits of the LCO is an option.
Omitted	3.1.5 A.3.2	3.0.3	ACTN: (Re-Align; 2 hrs)	Re-Align is always an option.

Palisades RTS Cross Reference to STS. (03/28/96).

RTS Number	STS Number	TS Number	RTS (STS) requirement Description	Explanation of Differences
3.1.5 B	3.1.5 B	3.10.4.c	COND: One rod misaligned	See note for LCO 3.1.5
3.1.5 B.1	Added	3.10.4.a	ACTN: Declare rod inoperable; 1 hr	See note for LCO 3.1.5
3.1.5 B.2.1	Added	3.10.4.c	ACTN: Verify radial peaking factors; 2 hrs	Retained from current license base. See note for LCO 3.1.5
3.1.5 B.2.2	3.1.5 B.1	3.10.4.c	ACTN: Restrict power ≤ 75%; 2 hrs	Unchanged except 75% RTP retained from current license. See note for LCO 3.1.5
Omitted	3.1.5 C	NA	COND: (Motion Inhibit inoperable)	No comparable function at Palisades.
3.1.5 C	Added	3.10.4.c	COND: One rod misaligned by > 20"	See note for LCO 3.1.5
3.1.5.C.1	Added	3.10.4.a	ACTN: Declare the rod inoperable	See note for LCO 3.1.5
3.1.5 C.2	Added	New	ACTN: Verify SDM; 1 hr	Verifies SDM with a misaligned rod.
3.1.5 C.3	Added	New	ACTN: Restrict power ≤ 50%; 4 hrs	A rod misaligned greater than 20 inches may be considered a dropped rod. Reflects Palisades ability to pick up a dropped rod as long as power is restricted.
3.1.5 D	3.1.5 D	3.10.4	COND: Deviation alarm circuit inoperable	Unchanged
3.1.5 D.1	3.1.5 D.1	3.10.4	ACTN: Perform SR 3.1.5.1 (Rod Position Verification); 15 Min.	Changed time to 15 minutes following rod motion. This reflects Palisades rod movement is only a manual capability.
3.1.5 E	3.1.5 E	3.0.3	COND: Two or More control rods inoperable	Changed to agree with preceding wording.
Omitted	3.1.5 E	NA	COND: (One or more rods Untrippable)	Omitted since untrippable rods are declared inoperable and fall into condition stated previously.
3.1.5 E	3.1.5 E	3.0.3	COND: Required Action not met	Changed to agree with preceding conditions
3.1.5.E.1	3.1.5 E.1	3.10.4.b	ACTN: MODE 3; 12 hrs	The action time of 12 hours in retained. This differs from the CE standard due to Palisades inherent low leakage core design. In the event a rod is misaligned, the local peaking factors are substantially elevated in that locality. The much larger reactivity worth and peaking influence from a mispositioned cruciform blade as opposed to a CEA would warrant a slower derate ensuring radial peaking remains in design limits while not initiating any core instabilities.
3.1.5.1	3.1.5.1	New	SR: Check Rod position; 12 hours	Used Palisades value; no other changes
3.1.5.2	3.1.5.2	4.1.3.2.a	SR: Compare rod pos indicators; 12 hours	Changed wording to reflect different equipment and terminology; used Palisades value; no other changes.

Palisades RTS Cross Reference to STS. (03/28/96).

RTS Number	STS Number	TS Number	RTS (STS) requirement Description	Explanation of Differences
Omitted	3.1.5.3	NA	SR: (CEA Motion Inhibit; 31 days)	No comparable equipment at Palisades.
3.1.5.3	3.1.5.4	New	SR: Rod pos deviation Alm; 92 days	Used Palisades terminology and retained surveillance frequency.
3.1.5.4	3.1.5.5	4.2.2.2	SR: Exercise each rod; 92 days	Used Palisades terminology. Palisades has different type rod drive mechanisms.
3.1.5.5	3.1.5.6	4.2.2.1	SR: Rod drop times; 18 months	Used Palisades value; no other changes
3.1.5.6	3.1.5.7	4.1.3.2.c	SR: Primary rod pos channel cal test; 18 months	Used Palisades terms; no other changes
3.1.6	3.1.6	3.10.3/.6	LCO: SD & PL Rod withdrawal; MODES 1 & 2	Added Part Length rods to LCO, there is no counter part in RSTS; used Palisades value & terminology. No other changes.
3.1.6 A	3.1.6 A	3.0.3	COND: SD or PL Rod not within limits	Added PL rods.
3.1.6 A.1.1	3.1.6 A.1	3.0.3	ACTN: Verify SDM ≥ 2%; 1 hour	Unchanged, with the exception of updating the SDM value to match Palisades requirements for MODES 1 and 2.
3.1.6 A.1.2	3.1.6.A.1.2	3.0.3	ACTN: Initiate boration to restore SDM ≥ 2%; 1 hr	Unchanged.
3.1.6 B	3.1.6 B	3.0.3	COND: Required action not met	Unchanged.
3.1.6.B.1	Added	3.0.3	ACTN: Verify SDM ≥ 2%	Added to verify core reactivity configuration prior to power reduction.
3.1.6.B.2	3.1.6.B.1	3.0.3	ACTN: Be in MODE 3; 12 hours	Changed completion time from 6 to 12 hours to ensure safe shutdown with a skewed power distribution. Palisades low leakage core design and cruciform blades provide a very steep gradient on the radial power distribution. A longer time period would be warranted to shutdown from a mispositioned rod condition to ensure peaking factors are not violated.
3.1.6.1	3.1.6.1	New	SR: Verify SD & PL rod position; 12 hrs & etc	Unchanged.
3.1.7	3.1.7	3.10.5	LCO: Reg rod insertion limits; MODES 1 & 2	Unchanged.
3.1.7 A	3.1.7 A	3.10.5.a	COND: Reg rods beyond limit	Omitted reference to "transient limit" since Palisades has no such limit.
3.1.7 A.1.1	3.1.7 A.1.1	3.10.5.a	ACTN: Verify SDM ≥ 2%; 1 hr	Unchanged.
3.1.7 A.1.2	3.1.7 A.1.2	3.10.5.a	ACTN: Initiate boration to restore SDM ≥ 2%; 1 hr	Unchanged.
3.1.7 A.2	3.1.7 A.2	3.10.5.a	ACTN: Reduce Thermal Power to limits stated in the COLR; 2 hr	Unchanged.

Palisades RTS Cross Reference to STS. (03/28/96).

RTS Number	STS Number	TS Number	RTS (STS) requirement Description	Explanation of Differences
Omitted	3.1.7 A.2.1	3.10.5.a	ACTN: (Restore rod within limits; 2 hrs)	Omitted since this is implied that restoring the condition to within limits satisfies the LCO.
3.1.7 B	Added	3.0.3	COND: Seq or Overlap exceeds limit	RSTS provided no Condition or Action for part of LCO.
3.1.7 B.1	Added	3.0.3	ACTN: Verify SDM ≥ 2%; 1 hr	Specified action like that in 3.1.7 A.1
Omitted	3.1.7 B	NA	COND: (Rods between Transient & SS limits)	Omitted, Palisades has only a single limit
Omitted	3.1.7 C	NA	COND: (Rods between Transient & SS limits)	Omitted, Palisades has only a single limit
3.1.7 C	3.1.7 D	3.0.3	COND: PDIL Alm inoperable	Unchanged.
3.1.7 C.1	3.1.7 D.1	New	ACTN: Verify rod group pos; 15 minutes	SR changed to 15 minutes following any rod motion since Palisades only has manual rod manipulation not automatic.
3.1.7 D	3.1.7 E	3.0.3	COND: Required action not met	Reworded to agree with altered Actions.
3.1.7 D.1	Added	New	ACTN: Verify SDM; 1 hr	Added since if Condition 3.1.7 A is not corrected, SDM is not assured.
3.1.7 D.2	3.1.7 E.1	3.0.3	ACTN: Be in MODE 3; 6 hrs	Unchanged.
3.1.7.1	3.1.7.1	New	SR: Verify reg group position; 12 hrs	Unchanged.
Omitted	3.1.7.2	NA	SR: (Verify times between limits; 24 hrs)	Palisades does not have a transient limit.
3.1.7.2	3.1.7.3	New	SR: Demonstrate PDIL alm OPERABLE; 31 days	Unchanged.
Omitted	3.1.7-1	NA	Figr: (Rod insertion limit figure)	Palisades single insertion limit is contained within the COLR.
3.1.8	3.1.8	3.10.7/.8	LCO: Test exemption, SDM MTC & Rods; MODES 2 & 3	Added LCO 3.1.2, 3.1.5, 3.1.6 to be suspended to support current startup physics testing program used for Palisades.
3.1.8 A	3.1.8 A	New	COND: LCO not met	Unchanged.
3.1.8 A.1	3.1.8 A.1	New	ACTN: Initiate boration; 15 min	Unchanged.

Palisades RTS Cross Reference to STS. (03/28/96).

RTS Number	STS Number	TS Number	RTS (STS) requirement Description	Explanation of Differences
3.1.8.1	3.1.8.1	New	SR: Verify rod position; 2 hrs	Unchanged.
3.1.8.2	3.1.8.2	New	SR: Verify withdrawn rods trippable; w/in 7 days	Unchanged.
Omitted	3.1.9	3.10.7/.8	LCO: (Test exemption, Rods, SDM, Power; MODES 1 & 2)	This LCO has been omitted since Palisades startup physics testing is performed after each refueling and covers both STE's 3.1.8 and 3.1.9. STE 3.1.8 has been modified to incorporate this. STE 3.1.9 is geared toward Mid-cycle MTC testing which Palisades does not perform per current license.

ENCLOSURE 1

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

TECHNICAL SPECIFICATION CHANGE REQUEST

PART 5 - SECTION 3.2

March 27, 1996

CONSUMERS POWER COMPANY
Docket 50-255
Request for Change to the Technical Specifications
License DPR-20

3.2 POWER DISTRIBUTION LIMITS CHANGE REQUEST

It is requested that the Power Distribution Limits and the Excore Power Monitoring Limits of the Technical Specifications contained in the Facility Operating License DPR-20, Docket 50-255, issued to Consumers Power Company on February 21, 1991, for the Palisades Plant be changed as described below:

I. ARRANGEMENT AND CONTENT OF THIS PART OF THE CHANGE REQUEST:

This section of the Technical Specification Change Request (TSCR) proposes changes to those Palisades Technical Specification requirements addressing Power Distribution Limits and the Excore Power Monitoring Limits. These changes are intended to result in requirements which are appropriate for the Palisades plant, but closely emulate those of the Standard Technical Specifications, Combustion Engineering Plants, NUREG 1432, Revision 1.

This discussion and its supporting information frequently refer to three sets of Technical Specifications; the following abbreviations are used for clarity and brevity:

TS - The existing Palisades Technical Specifications,
RTS - The revised Palisades Technical Specifications,
STS - NUREG 1432, Revision 1.

Six attachments are provided to assist the reviewer. The numbering and content of the attachments is consistent with other parts of the TSCR.

1. Proposed RTS pages
2. Bases for the RTS
3. A line by line comparison of the TS and RTS
4. STS pages marked to show the differences between RTS and STS
5. STS Bases pages marked to show differences between RTS and STS Bases.
6. A line by line comparison of RTS and STS.

Attachment 3, the line by line comparison of TS and RTS, is presented in a tabular format. The first page contains an explanation of the syntax and abbreviations used. The table is arranged numerically by TS item number. Each requirement in Sections 1 through 4 of TS is listed individually. In some cases, where a single numbered TS requirement contains more than one requirement, each requirement is listed individually under the same number. Requirements which appear in RTS or STS, but not in TS, do not appear in the Attachment 3 listing.

Attachment 3 Provides the Following Information for Each TS Requirement:

Identifying number of TS item,
Identifying number of closest equivalent RTS item,
Identification of TS item as LCO, Action, SR, etc.,
A short paraphrase of requirement,
A description of each proposed change from TS to RTS.

Classification of Change as One of the Following Categories:

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

RELOCATED - A change which only moves requirements, not meeting the 10 CFR 50.36(c)(2)(ii) criteria, from the TS to the FSAR, to the Operating Requirements Manual, or to other documents controlled under 10 CFR 50.59.

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

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

Attachment 6, the line by line comparison of RTS and STS, is also presented in a tabular format. The first page contains an explanation of the syntax and abbreviations used; the second page contains a list of Palisades terminology used in place of the generic STS terminology. The table is arranged numerically by RTS item number. Each requirement in Sections 1 through 3 of RTS or STS is listed individually. Requirements which appear in TS, but not in RTS or STS, do not appear in the Attachment 6 listing.

Attachment 6 Provides the Following Information for Each RTS Requirement:

Identifying number of RTS requirement,
Identifying number of equivalent STS requirement,
Identification of each requirement as LCO, Action, SR, etc.,
Short paraphrase of each requirement,
A description of each difference between RTS and STS.

II. TECHNICAL SPECIFICATION CHANGES PROPOSED:

The TS LCOs and action statements for Power Distribution Limits and Excore Power Monitoring Limits appear in Sections 3.23 and 3.11. The TS surveillance requirements appear in TS Section 4. All RTS requirements for these LCOs appear in proposed Section 3.2, "Power Distribution Limits." Each proposed change from TS to RTS is discussed in the attachments to this part of the TSCR.

Each proposed change to a requirement in TS is described in Attachment 3.

Those proposed RTS requirements which have no counterpart in TS are described in Attachment 6. These new requirements are identified by the word "New" in the third column of Attachment 6.

The Major Changes From TS to RTS Proposed in This Part of the TSCR are:

1. RTS 3.2.1 LCO has required incore alarm system to be operable. This more restrictive change was added to RTS LCO 3.2.1 to ensure operability for LHR monitoring. This allowed for the structure of the LCO to be molded more closely to STS.

2. RTS added condition 3.2.1 B incore alarm system inoperable. This condition sets up a series of conditions that allow for LHR to be monitored without the incore system.

3. Added action 3.2.1 B.1.1 to RTS to verify the excore system is ready for LHR monitoring in the event the incores are not available. This action ensures that LHR is monitored at all times and does not exceed limits assumed the accident analysis.

4. Added 3.2.1 B.1.2 to RTS to restrict thermal power to the Allowed Power Level (APL). This requirement ensures thermal power is appropriate for monitoring LHR with the excore system. This ensures accurate predictions of LHR by the excore system.

5. TS 3.23.3 A1a and A2a have been deleted and are not included in RTS. These actions call for restoring quadrant power tilt (T_q) within limits. Restoration is always an option and does not have to be explicitly stated.

6. RTS LCO 3.2.4, "Axial Shape Index (ASI)" has been retained from current TS and appropriate conditions, required actions and completion times were added to support the LCO.

7. RTS LCO 3.2.5 has been added from current TS 3.11.2. This LCO states that three ASI alarm channels shall be operable and the excore deviation alarm channel shall be operable. The ASI alarm channels operable ensures the excores can adequately monitor LHR and the excore deviation alarm operable ensures that the excores can be used to monitor Quadrant power tilt adequately. The associated conditions, required actions, completion times and surveillance requirements have been added to support this LCO. It is prudent to add this LCO to ensure the excore system can safely monitor LHR and T_q.

8. In each section of the proposed RTS, new requirements taken from STS have been proposed. Since there is no equivalent requirement in TS, these changes do not appear in Attachment 3. The new requirements do appear in Attachment 6 where they are identified by an entry of "New" or "3.0.3" in the third column.

   The changes identified as "New" are considered MORE RESTRICTIVE because they add requirements and operating restrictions which do not exist in the current Palisades TS.

The changes identified as "3.0.3" are considered LESS RESTRICTIVE because they extend the time available to restore compliance to the LCO (the Allowed Outage Time) beyond that allowed by LCO 3.0.3. In these cases, the proposed RTS contain a specific Action where the existing TS do not contain any Action for the associated LCO. These instances do not involve a loss of safety function, but occur due to the lack of structure of Technical Specifications circa 1970. There was not necessarily an intent that failure to meet these LCOs would force a plant shutdown or an entry into LCO 3.0.3 (the original TS contained no equivalent of LCO 3.0.3).

The Major Differences Between RTS and STS in This Part of the TSCR are:

1. 3.2.1 B added to RTS incore alarm system inoperable was added to support the LHR LCO. This condition and associated, required actions, completion times and surveillance were added to ensure that the incore alarm system is operable for LHR monitoring.

2. Added SR 3.2.1.1 to RTS to verify incore alarm system is functioning for LHR monitoring. This SR ensures that the Incore system is operable on a 12 hour frequency and ensures continuous monitoring of LHR.

3. Added SR 3.2.1.2 to RTS to verify LHR is within limits. In reality LHR is monitored continuously and the moment LHR is outside of limits the incore system will alarm. However, a 12 hour frequency is added to support the LCO for normal LHR monitoring or monitoring by one of the other options stated in the LCO. This allows for the use of other LHR monitoring methods that are not continuous monitoring as an extra precaution.

4. Added SR 3.2.1.4 to RTS to verify APL, Quadrant power tilt, and target Axial Offset (AO) on a monthly basis. This SR was added to support the use of the excore system in the event it is needed for LHR monitoring.

5. Added SR 3.2.1.5 to RTS to perform a CHANNEL CALIBRATION on each incore alarm signal on an 18 month frequency. This ensures that the incore alarm system signal is translating into accurate flux reading for LHR monitoring.

6. Omitted STS SR 3.1.2 which demonstrate local power density alarms. This is not applicable to Palisades. Palisades does not have a comparable parameter to measure that is illustrated in this SR.

7. Omitted STS action 3.2.3 A.2 which says withdraw CEAs above long term insertion limit. Palisades has no comparable limit or requirement.

8. Omitted STS action 3.2.3 A.3 which states establish a revised upper thermal power limit. Palisades has no comparable limit or requirement.

9. Omitted LCO 3.2.3 and associated conditions, required actions, completion times and surveillance. This LCO pertains to planar radial peaking F_{xy} which is not a monitored parameter at Palisades.

10. STS 3.2.4 C.3 has been omitted from RTS. This action allows for restoring quadrant tilt prior to power increase. Restoration to compliance is always an option that does not have to be explicitly stated.

III. NO SIGNIFICANT HAZARDS ANALYSIS:

Each change proposed is classified in Attachment 3 as either ADMINISTRATIVE, RELOCATED, MORE RESTRICTIVE, or LESS RESTRICTIVE.

Analysis of ADMINISTRATIVE, RELOCATED, and MORE RESTRICTIVE Changes:

ADMINISTRATIVE changes and RELOCATED changes move requirements, either within the TS or to documents controlled under 10 CFR 50.59, or clarifying existing TS requirements, without affecting their technical content. Since ADMINISTRATIVE and RELOCATED changes do not alter the technical content of any requirements, they cannot involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any previously evaluated, or involve a significant reduction in a margin of safety.

MORE RESTRICTIVE changes only add new requirements, or revise existing requirements to result in additional operational restrictions. Since the TS, with all MORE RESTRICTIVE changes incorporated, will still contain all of the requirements which existed prior to the changes, MORE RESTRICTIVE changes cannot involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any previously evaluated, or involve a significant reduction in a margin of safety.

Analysis of LESS RESTRICTIVE Changes:

The LESS RESTRICTIVE Changes Proposed in This Part of the TSCR are:

1. While using the excores to monitor LHR, the Current TS states APL to be calculated every hour. RTS 3.2.1 B.1.3 allows for the Allowable Power Level (APL) to be calculated once per 4 hours.

2. RTS 3.2.2, "Radial Peaking Factors" surveillance period has been extended from 7 days to 31 days in the STS. This encompasses both assembly and total radial peaking factors.

3. RTS 3.2.1.3, "Update Incore Alarm Setpoints" surveillance frequency has been extended from TS 7 days to STS 31 days. This entails the incore alarms are adjusted to the measured power distribution.

4. RTS 3.2.5.1 requires the verification of AO within 5% of target every 15 minutes while the excore system is used to monitor LHR. This SR has been relaxed from TS 4.19.1.2d frequency of continuously.

5. The proposed RTS add specific Action for failure to meet an LCO where no loss of function occurs, but the existing TS do not contain any. With the exsiting TS an entry into LCO 3.0.3 is required. Each of these changes is identified by an entry of "3.0.3" in the third column of Attachment 6. These changes are considered LESS RESTRICTIVE because they extend the time available to restore compliance to the LCO (the Allowed Outage Time) beyond that allowed by LCO 3.0.3.

Do these LESS RESTRICTIVE changes involve a significant increase in the probability or consequences of an accident previously evaluated?

Changes 1, 2, 3, 4, and 5:

These changes are LESS RESTRICTIVE only in their allowance of a longer Allowed Outage Time (AOT) for inoperable equipment or a longer surveillance testing interval. The proposed times are those stipulated in the STS. Changing an AOT or a surveillance interval, alone, does not alter any plant design, operating conditions, operating practices, equipment settings, or equipment capabilities. Since these items are unchanged, changing an AOT or a surveillance interval would not increase the probability of any accident previously evaluated.

During the evaluation of potential accidents, the safety analyses assume the occurrence of the most limiting single failure. Typically, this single failure is assumed to disable one of the two trains of the equipment installed to mitigate an accident. In accordance with this assumption, the Technical Specifications allow continued operation with required equipment inoperable for limited periods of time (AOTs) only if the assumed level of equipment remains operable. Extending an AOT does not change level of safety equipment required to be available, and does not allow that level to drop below the level assumed to be available in the safety analyses. Therefore, changing an AOT cannot increase the consequences of an accident previously evaluated.

Excessively extending a surveillance interval could affect the probability that a piece of equipment will function properly upon demand. An overly restrictive surveillance interval could also affect the ability of the equipment to mitigate an accident by imposing unnecessary testing wear, equipment manipulations, and system transients on the plant, and thereby affect the consequences of an accident. The existing surveillance intervals were based on the operating experience available when they were added to the TS. Typically, this was done during the initial plant licensing, circa 1970. In each of these changes where it is proposed that a surveillance interval be extended, the time proposed is that stipulated in the STS. The surveillance intervals stipulated in the STS are based on a much larger accumulation of operating experience and have been judged by the NRC and by the industry to be appropriate for typical situations. There are no special features of the Palisades plant which would invalidate those judgements for these changes. Therefore, operation of the facility in accordance with the requirements proposed by these changes does not involve a significant increase in the probability of an accident previously evaluated.

Do these LESS RESTRICTIVE changes create the possibility of a new or different kind of accident from any previously evaluated?

Changes 1, 2, 3, 4, and 5:

These changes are LESS RESTRICTIVE only in their allowance of a longer Allowed Outage Time (AOT) for inoperable equipment or a longer surveillance testing interval. The proposed times are those stipulated in the STS. Changing an AOT or surveillance interval, alone, cannot alter any plant operating conditions, operating practices, equipment settings, or equipment capabilities. Therefore, changing an AOT or a surveillance interval cannot create the possibility of a new or different kind of accident from any previously evaluated.

Do these LESS RESTRICTIVE changes involve a significant reduction in a margin of safety?

Changes 1, 2, 3, 4, and 5:

These changes are LESS RESTRICTIVE only in their allowance of an extension to an Allowed Outage Time (AOT) for inoperable equipment or to a surveillance testing interval. Extending an AOT or a surveillance interval, alone, cannot alter any plant operating conditions, operating practices, equipment settings, or equipment capabilities.

An excessive AOT extension could reduce the margin of safety by allowing operation for an excessive period with less capability to mitigate an accident, or with parameters outside those assumed in the safety analysis. An overly restrictive AOT could also reduce the margin of safety by imposing unnecessary transients on the plant for minor deviations from the requirements of the LCOs. Similarly, an excessive surveillance interval extension could reduce the margin of safety by reducing assurance that required equipment will function as designed or that parameters are within the required limits. An overly restrictive surveillance interval could also reduce the margin of safety by imposing unnecessary testing wear, equipment manipulations, and system transients on the plant.

The existing AOTs and surveillance intervals were based on the operating experience available when they were added to the TS. Typically this was done during the initial plant licensing, circa 1970. In each of these changes where it is proposed that an AOT or surveillance interval be extended, the time proposed is that stipulated in the STS. The AOTs and surveillance intervals stipulated in the STS are based on a much larger accumulation of operating experience and have been judged by the NRC and by the industry to be appropriate for typical situations. There are no special features of the Palisades plant which would invalidate those judgements for these changes. Therefore, operation of the facility in accordance with the requirements proposed by these changes does not involve a significant reduction in a margin of safety.

IV. CONCLUSION

The Palisades Plant Review Committee has reviewed this part of the STS conversion Technical Specifications Change Request and has determined that proposing this change does not involve an unreviewed safety question. Further, the change involves no significant hazards consideration. This change has been reviewed by the Nuclear Performance Assessment Department.

ATTACHMENT 1

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

3.2 POWER DISTRIBUTION LIMITS

Proposed Technical Specifications Pages

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Linear Heat Rate (LHR)

LCO 3.2.1 LHR shall not exceed the limits specified in the COLR and the Incore Alarm System shall be operable.

APPLICABILITY: MODE 1, ≥ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LHR, as determined by the Incore Monitoring System not within limits.	A.1 Restore LHR to within limits.	1 hour

(continued)

ACTION (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Incore Alarm System Inoperable.	B.1.1 Verify excore system is OPERABLE for LHR monitoring.	2 hours
	AND	
	B.1.2 Restrict THERMAL POWER to the excore Allowable Power Level (APL).	2 hours
	AND	
	B.1.3 Verify the following parameters: $T_q \leq 3\%$ THERMAL POWER $\leq$ APL and ASI within $\pm$ .05 of target.	2 hours AND Once per 4 hours, thereafter
	OR	
	B.2.1 Restrict THERMAL POWER to < 85% RTP.	2 hours
	AND	
	B.2.2 Verify LHR within limits using manual incore readings.	2 hours AND Once per 4 hours, thereafter
C. Required Action and associated Completion Time not met.	C.1 Be < 50% RTP.	2 hours

SURVEILLANCE REQUIREMENTS

----------NOTE----------
Either the Excore Monitoring System or the Incore Monitoring System shall be used to determine LHR.

SURVEILLANCE		FREQUENCY
----------NOTE---------- This SR is only applicable while using the excore system to monitor LHR. ----------		
SR 3.2.1.1	Verify measured ASI is within 0.05 of target ASI for at least 3 of 4, 2 of 3, or 2 of 2 operable channels.	15 minutes
SR 3.2.1.2	Verify Incore Alarm System is functioning to monitor LHR.	12 hours
SR 3.2.1.3	Verify LHR is within limits.	12 hours
SR 3.2.1.4	Update incore alarm setpoints.	Prior to 50% RTP following refueling. AND 31 days thereafter
SR 3.2.1.5	Determine APL, T_q and Target AO using the excore and incore system.	31 days
SR 3.2.1.6	Perform a CHANNEL CALIBRATION on each incore alarm signal.	18 months

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Radial Peaking Factors F_r^T and F_r^A

LCO 3.2.2 The value of F_r^A and F_r^T shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1, ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. F_r^A or F_r^T not within limit.	A.1 Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T or F_r^A to within limits specified in the COLR.	6 hours
B. Required Actions and associated Completion Times not met.	B.1 Be < 25% RTP.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify the value of F_r^T, F_r^A within limits.	Prior to operation > 50% RTP after each refueling. AND Each 31 days of accumulated operation in MODE 1.

3.2 POWER DISTRIBUTION LIMITS

3.2.3 Quadrant Power Tilt

LCO 3.2.3 T_q shall be $\leq$ 5%

APPLICABILITY: MODE 1, $\geq$ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. T_q > 5% but $\leq$ 10%.	A.1	Perform SR 3.2.2.1 (Peaking factor verification).	2 hours AND Once per 8 hours thereafter
B. T_q > 10%.	B.1	Reduce THERMAL POWER to $\leq$ 50% RTP.	4 hours
	AND		
	B.2	Perform SR 3.2.2.1 (Peaking factor verification).	4 hours AND Once per 8 hours thereafter

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. T_q > 15%. OR Required Action and associated completion time not met.	----------NOTE---------- All subsequent Required Actions stated in Condition B must be completed if power reduction commences prior to restoring $T_q \leq$ 15%. ---------- C.1 Be < 25% RTP.	 12 hours ------NOTE------ Correct the cause of the out of limit condition prior to increasing THERMAL POWER. Subsequent power operation above 50% RTP may proceed provided that the measured T_q is verified $\leq$ 3% at least once per hour for 12 hours. ----------

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.3.1 Verify T_q is below limits.	12 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Axial Shape Index (ASI)

LCO 3.2.4 The ASI shall be maintained within the limits specified in the COLR.

APPLICABILITY: MODE 1, ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ASI not within limits.	A.1 Restore ASI to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.4.1 Verify ASI is within limits.	Continuously, while using the incore alarm system to monitor LHR.

3.2 POWER DISTRIBUTION LIMITS

3.2.5 Excore Power Distribution Monitoring

LCO 3.2.5 Three Axial Shape Index (ASI) Monitoring Channels and the Excore Deviation Alarm Channel shall be OPERABLE.

APPLICABILITY: MODE 1 $\geq$ 25% RTP, when Excore Power Range Channels are used to monitor Linear Heat Rate (LHR) or Quadrant Power Tilt (T_q).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Excore Detector Deviation Alarm Channel inoperable.	A.1 Calculate T_q.	Once per 12 hours
B. Excore T_q deviating from Incore T_q by $\geq$ 2%.	B.1 Calculate T_q using Incore detectors. AND B.2 Perform SR 3.2.5.2 (Excore AIS cal)	Once per 12 hours 7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ASI deviating from AO by > 0.02 under steady state operating conditions.	C.1 Perform SR 3.2.5.1 (Incore/Excore calibration).	12 hours
	OR	
	C.2.1 Adjust the ASI alarm setpoint to compensate for the deviation.	12 hours
	AND	
	C.2.2 Adjust the TM/LP trip function to compensate for the deviation.	12 hours
	OR	
	C.3.1 Declare ASI monitoring channel inoperable.	12 hours
	AND	
	C.3.2 Declare affected TM/LP RPS Trip Units inoperable.	12 hours
D. 1 required ASI monitoring channel inoperable.	D.1 Initiate LHR monitoring with the Incore Alarm System.	1 hour
E. Required action associated Completion Time not met.	E.1 Be < 25% RTP.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.5.1	Verify individual excore channel measured AO when compared to total core AO measured by the incores is $\leq$ 0.02.	31 days
SR 3.2.5.2	Calibrate T_q and AO from Excore with T_q and AO measured from Incores for each channel of TM/LP trip and the ASI alarm.	184 days
SR 3.2.5.3	Perform a CHANNEL CALIBRATION of the Excore Detector Deviation Alarm Channel.	18 months
SR 3.2.5.4	Perform a CHANNEL CALIBRATION of the Excore ASI circuitry.	18 months

ATTACHMENT 4

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

3.3 INSTRUMENTATION PART

STS Pages Marked to Show the Differences Between RTS and STS

3.3 INSTRUMENTATION

3.3.1 REACTOR PROTECTIVE SYSTEM (RPS) INSTRUMENTATION~~—Operating (Analog)~~

LCO 3.3.1 Four RPS trip units and associated instrument and bypass removal channels for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5, when more than one Control Rod is capable of being withdrawn and PCS boron concentration is less than that required by the COLR for LCO 3.9.1.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each RPS trip or bypass removal Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one RPS trip unit or associated instrument channel inoperable, except for Condition C (Hi Startup Rate or Loss of Load). ~~(excore channel not calibrated with incore detectors).~~	A.1 Place affected trip unit in ~~bypass or~~ trip.	7 Days ~~1 hour~~
	~~AND~~	
	~~A.2.1 Restore channel to OPERABLE status.~~	~~[48] hours~~
	~~OR~~	
	~~A.2.2 Place affected trip unit in trip.~~	~~48 hours~~

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more Functions with two RPS trip units or associated instrument channels inoperable except for Condition D (Hi Startup Rate or Loss of Load). ~~Condition C (excore channel not calibrated with incore detectors).~~	----------NOTE---------- LCO 3.0.4 is not applicable. ---------- B.1 Place one trip unit in ~~bypass and place the other trip unit in~~ trip. AND B.2 Restore one trip unit to OPERABLE status.	 1 hour 7 days ~~[48] hours~~
C. One Loss of Load or Hi Startup Rate trip unit or associated instrument channel inoperable. ~~C. One or more Functions with one or more power range excore channels not calibrated with the incore detectors.~~	----------NOTE---------- LCO 3.0.4 is not applicable. ---------- C.1 Restore trip unit to OPERABLE status. ~~Perform SR 3.3.1.3.~~ ~~OR~~ ~~C.2 Restrict THERMAL POWER to ≤ 90% of the maximum allowed THERMAL POWER level.~~	 Prior to entering MODE 2 following MODE 5 entry ~~24 hours~~ ~~24 hours~~

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two Loss of Load or two Hi Startup Rate trip units or associated instrument channels inoperable ~~D. One or more Functions with one automatic bypass removal channel inoperable.~~	----------NOTE---------- LCO 3.0.4 is not applicable. ---------- D.1 Place one affected trip unit in trip AND D.2 Restore one trip unit to OPERABLE status ~~D.1 Disable bypass channel.~~ ~~OR~~ ~~D.2.1 Place affected trip units in bypass or trip.~~ ~~AND~~ ~~D.2.2.1 Restore bypass removal channel and affected trip units to OPERABLE status.~~ ~~OR~~ ~~D.2.2.2 Place affected trip units in trip.~~	1 hour Prior to entering MODE 2 following MODE 5 entry ~~1 hour~~ ~~[48] hours~~ ~~48 hours~~

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One or more Functions with one ~~two~~ or two automatic bypass removal channels inoperable.	~~----------NOTE----------~~ ~~LCO 3.0.4 is not applicable.~~ ~~----------~~ E.1 ~~Disable bypass channels.~~ Remove the bypass function OR E.2 Declare the affected trip units inoperable, and enter the appropriate Condition. ~~E.2.1 Place one affected trip unit in bypass and place the other in trip for each affected trip Function.~~ ~~AND~~ ~~E.2.2 Restore one bypass channel and the associated trip unit to OPERABLE status for each affected trip Function.~~	1 hour 1 hour ~~1 hour~~ ~~[48] hours~~

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time not met.	F.1 Be in MODE 3.	6 hours
	AND	
	F.2.1 Ensure that no more than one Control Rod is capable of being withdrawn.	6 hours
	OR	
	F.2.2 Ensure PCS boron concentration is at least that required by the COLR for LCO 3.9.1.	6 hours
G. Control room temperature > 90°F.	G.1 Enter LCO 3.0.3	Immediately

SURVEILLANCE REQUIREMENTS

----------NOTE----------
Refer to Table 3.3.1-1 to determine which SR shall be performed for each RPS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform a CHANNEL CHECK of each RPS instrument channel except Loss of Load and Hi Containment Pressure.	12 hours
SR 3.3.1.2	----------NOTES---------- 1. Not required to be performed until 12 hours after THERMAL POWER is ≥ 15% ~~≥ [20]%~~ RTP. ~~2. The daily calibration may be suspended during PHYSICS TESTS, provided the calibration is performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.~~ --- Perform calibration (heat balance only) and adjust the excore power range and ΔT power channels to agree with calorimetric calculation if the absolute difference is ≥ [1.5]%.	24 hours
SR 3.3.1.3	~~----------NOTE----------~~ ~~Not required to be performed until 12 hours after THERMAL POWER is ≥ [20]% RTP.~~ ~~----------~~ Perform a CHANNEL FUNCTIONAL TEST of each RPS channel except Loss of Load and Hi Startup Rate ~~Calibrate the power range excore channels using the incore detectors.~~	92 days ~~31 days~~

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.4	Calibrate the Excore Power Range Channels with a test signal ~~Perform a CHANNEL FUNCTIONAL TEST of each RPS channel except Loss of Load and Power Rate of Change.~~	31 days ~~[92] days~~
SR 3.3.1.5	~~NOTE~~ ~~Neutron detectors are excluded from CHANNEL CALIBRATION.~~ Verify constants in each Thermal Margin Monitor ~~Perform a CHANNEL CALIBRATION on excore power range channels.~~	92 days
SR 3.3.1.6	Perform a CHANNEL FUNCTIONAL TEST of each Hi Startup Rate ~~Power Rate of Change Channel~~ and each Loss of Load Functional Unit.	Once within 7 days prior to each reactor startup
SR 3.3.1.7	Perform a CHANNEL CALIBRATION of each RPS instrument channel, including bypass removal functions. ~~Perform a CHANNEL FUNCTIONAL TEST on each automatic bypass removal function.~~	18 months ~~Once within 92 days prior to each reactor startup~~

(continued)

~~SR 3.3.1.8~~	~~Perform a CHANNEL CALIBRATION of each RPS instrument channel, including bypass removal functions.~~	~~[18] months~~
~~SR 3.3.1.9~~	~~Verify RPS RESPONSE TIME is within limits.~~	~~[18] months on a STAGGERED TEST BASIS~~
SR 3.3.1.8	Verify control room temperature is ≤ 90°F.	12 hours

Table 3.3.1-1 (page 1 of 2)
Reactor Protective System Instrumentation

FUNCTION	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Variable High Power Trip	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 ~~SR 3.3.1.5~~ SR 3.3.1.7 ~~SR 3.3.1.8~~ ~~SR 3.3.1.9~~	≤ 15% RTP above current THERMAL POWER but not < 30% RTP nor > 106.5% RTP ~~≤ [10]% RTP above current THERMAL POWER but not < [30]% RTP nor > [107]% RTP~~
2. Hi Startup Rate [a] ~~2. Power Rate of Change - High [a]~~	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.7 ~~SR 3.3.1.8~~	~~≤ 2.6 dpm~~ N/A
3. Low PCS Flow ~~3. Reactor Coolant Flow - Low [b]~~	SR 3.3.1.1 SR 3.3.1.3 ~~SR 3.3.1.4~~ SR 3.3.1.7 ~~SR 3.3.1.8~~ ~~SR 3.3.1.9~~	≥ 95%
4. Low SG-A Level ~~4. Pressurizer Pressure - High~~	SR 3.3.1.1 SR 3.3.1.3 ~~SR 3.3.1.4~~ SR 3.3.1.7 ~~SR 3.3.1.8~~ ~~SR 3.3.1.9~~	≥ 25.9% ~~≤ [2400] psia~~
5. Low SG-B Level ~~5. Containment Pressure - High~~	SR 3.3.1.1 SR 3.3.1.3 ~~SR 3.3.1.4~~ SR 3.3.1.7 ~~SR 3.3.1.8~~ ~~SR 3.3.1.9~~	≥ 25.9% ~~≤ [4.0] psig~~

(continued)

6. Low SG-A Pressure	SR 3.3.1.1	≥ 500 psia
~~6. Steam Generator Pressure - Low[c]~~	~~SR 3.3.1.3~~	~~≥ [685] psia~~
	~~SR 3.3.1.4~~	
	SR 3.3.1.7	
	~~SR 3.3.1.8~~	
	~~SR 3.3.1.9~~	
7. Low SG-B Pressure		
	SR 3.3.1.1	≥ 500 psia
	SR 3.3.1.3	
	SR 3.3.1.7	
~~7a. Steam Generator A Level — Low~~		
	~~SR 3.3.1.1~~	~~≥ [24.7]%~~
	~~SR 3.3.1.4~~	
	~~SR 3.3.1.8~~	
	~~SR 3.3.1.9~~	
~~7b. Steam Generator B Level — Low~~		
	~~SR 3.3.1.1~~	~~≥ [24.7]%~~
	~~SR 3.3.1.4~~	
	~~SR 3.3.1.8~~	
	~~SR 3.3.1.9~~	
8. High Pressurizer Pressure		
	SR 3.3.1.1	≤ 2255 psia
	SR 3.3.1.3	
	SR 3.3.1.7	
~~8. Axial Power Distribution — High[d]~~	~~SR 3.3.1.1~~	~~Figure 3.3.1-3~~
	~~SR 3.3.1.2~~	
	~~SR 3.3.1.3~~	
	~~SR 3.3.1.4~~	
	~~SR 3.3.1.5~~	
	~~SR 3.3.1.7~~	
	~~SR 3.3.1.8~~	
	~~SR 3.3.1.9~~	

(continued)

Table 3.3.1-1 (page 2 of 2)
Reactor Protective System Instrumentation

FUNCTION	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
9. Thermal Margin/Low Pressure (TM/LP)	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.5 SR 3.3.1.7	(c)
~~9a. Thermal Margin/Low Pressure (TM/LP)~~	~~SR 3.3.1.1~~ ~~SR 3.3.1.2~~ ~~SR 3.3.1.3~~ ~~SR 3.3.1.4~~ ~~SR 3.3.1.5~~ ~~SR 3.3.1.7~~ ~~[SR 3.3.1.8]~~ ~~SR 3.3.1.9~~	~~Figures 3.3.1-1 and 3.3.1-2~~
~~9b. Steam Generator Pressure Difference~~[b]	~~SR 3.3.1.1~~ ~~SR 3.3.1.4~~ ~~SR 3.3.1.8~~ ~~SR 3.3.1.9~~	~~≤ [135] psid~~
10. Loss of Load (~~turbine stop valve control~~auto stop oil pressure)[b]	SR 3.3.1.6 SR 3.3.1.7 ~~SR 3.3.1.8~~	N/A ~~≥ [800] psig~~
11. Containment High Pressure	SR 3.3.1.3 SR 3.3.1.7	≤ 3.70 psig

(a) Trip may be bypassed when THERMAL POWER IS $< 1X10^{-4}$% RTP or $>$ 13% RTP.
Bypass shall be automatically removed when THERMAL POWER is $\geq 1X10^{-4}$% RTP and $\geq$ 13% RTP.

(b) Trip may be bypassed and is not required to be OPERABLE when THERMAL POWER is $<$ 17% RTP.
Bypass shall be automatically removed when THERMAL POWER is $\geq$ 17% RTP.

N/A No specific allowable value required.

The pressure setpoint for the Thermal Margin/Low Pressure Trip, Ptrip, is the higher of two values, Pmin and Pvar, both in psia:

P_{min} = 1750
P_{var} = 2012(QA)(QR$_1$) + 17.0(T_{in}) - 9493

where:

QA = -0.720(ASI) + 1.028; when -0.628 $\leq$ ASI $<$ -0.100

QA = -0.333(ASI) + 1.087; when -0.100 $\leq$ ASI $<$ +0.200

QA = +0.375(ASI) + 0.925; when +0.200 $\leq$ ASI $<$ +0.565

ASI = Measured ASI when Q $\geq$ 0.0625
ASI = 0.0 when Q $<$ 0.0625

QR$_1$ = 0.412(Q) + 0.588; when Q $\leq$ 1.0
QR$_1$ = Q when Q $>$ 1.0

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

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

~~(b) Trips may be bypassed when THERMAL POWER is < [1E-4]%. Bypass shall be automatically removed when THERMAL POWER is ≥ [1E-4]% RTP. During testing pursuant to LCO 3.4.17, RCS Loops — Test Exceptions, trips may be bypassed below 5% RTP. Bypass shall be automatically removed when THERMAL POWER is ≥ 5% RTP.~~

~~(c) Trip may be bypassed when steam generator pressure is < [785] psig. Bypass shall be automatically removed when steam generator pressure is ≥ [785] psig.~~

~~(d) Trip may be bypassed when THERMAL POWER is < [15]% RTP. Bypass shall be automatically removed when THERMAL POWER is ≥ [15]% RTP.~~

3.3 INSTRUMENTATION

3.3.2 Reactor Protective System (RPS) Logic and Trip Initiation ~~(Analog)~~
~~3.3.3~~

LCO 3.3.2 Six channels of RPS Matrix Logic, four channels of RPS Initiation Logic, ~~four channels of reactor trip circuit breakers (RTCBs),~~ and two ~~four~~ channels of Manual Trip shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5, when more than one CONTROL ROD is capable of being withdrawn and PCS boron concentration is less than that required by LCO 3.9.1. ~~with any RTCBs closed and any control element assemblies capable of being withdrawn.~~

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ----------NOTE---------- This action also applies when three Matrix Logic channels are inoperable due to a common power source failure de-energizing three matrix power supplies. ---------- One Matrix Logic channel inoperable.	A.1 Restore channel to OPERABLE status.	48 hours
~~--------NOTE-------- RTCBs associated with one inoperable channel may be closed for up to 1 hour for the performance of an RPS CHANNEL FUNCTIONAL TEST. ----------~~		
B. One channel of ~~Manual Trip, RTCBs, or~~ Initiation Logic inoperable in MODE 1 or 2.	B.1 Deenergize affected clutch power supplies. ~~B.1 Open the affected RTCBs.~~	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One channel of Manual Trip inoperable.	C.1 Restore channel to OPERABLE status.	Prior to entering MODE 2 following MODE 3 entry.
~~C. --------NOTE-------- RTCBs associated with one inoperable channel may be closed for up to 1 hour for the performance of an RPS CHANNEL FUNCTIONAL TEST. ---------- One channel of Manual Trip, RTCBs, or Initiation Logic inoperable in MODE 3, 4, or 5.~~	~~C.1 Open all RTCBs.~~	~~48 hours~~
D. Required Action and associated Completion Times not met.	D.1 Be in MODE 3.	6 hours
	AND	
	D.2.1 Ensure that no more than one Control Rod is capable of being withdrawn.	6 hours
	OR	
~~D. Two channels of RTCBs or Initiation Logic affecting the same trip leg inoperable.~~	D.2.2 Ensure PCS boron concentration is at least that required by the COLR for LCO 3.9.1.	6 hours ~~Immediately~~
	~~D.1 Open the affected RTCBs.~~	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~E. Required Action and associated Completion Time of Condition A, B, or D not met.~~ ~~OR~~ ~~One or more Functions with two or more Manual Trip, Matrix Logic, Initiation Logic, or RTCB channels inoperable for reasons other than Condition A or D.~~	~~E.1 Be in MODE 3.~~ ~~AND~~ ~~E.2 Open all RTCBs.~~	~~6 hours~~ ~~6 hours~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform a CHANNEL FUNCTIONAL TEST on each RPS Logic channel.	92 days
~~SR 3.3.3.1~~	~~Perform a CHANNEL FUNCTIONAL TEST on each RPS Logic channel and RTCB channel.~~	
SR 3.3.~~3.~~2.2	Perform a CHANNEL FUNCTIONAL TEST on each RPS Manual Trip channel.	Once within 7 days prior to each reactor startup
~~SR 3.3.3.3~~	~~Perform a CHANNEL FUNCTIONAL TEST, including separate verification of the undervoltage and shunt trips, on each RTCB.~~	~~[18] months~~

SURVEILLANCE REQUIREMENTS (continued)

3.3 INSTRUMENTATION

3.3.3 Engineered Safety Features (ESF) Instrumentation
~~3.3.4 Engineered Safety Features Actuation System (ESFAS) Instrumentation (Analog)~~

LCO 3.3.3 ~~LCO 3.3.4~~ Four ESF ~~ESFAS~~ trip units and associated instrument and bypass removal channels
for each Function in Table 3.3.3-1 ~~3.3.4-~~1shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTIONS:

----------NOTE----------
Separate Condition entry is allowed for each ESF ~~ESFAS~~ trip or bypass removal Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more functions with one ESF trip unit or associated instrument channel inoperable, except SIRWT level. ~~A. One Containment Spray Actuation Signal (CSAS) trip unit or associated instrument inoperable.~~	A.1 Place affected bistable ~~trip unit~~ in trip ~~bypass~~.	7 days ~~1 hour~~

(continued)

SURVEILLANCE REQUIREMENTS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more Functions with two ~~one~~ ESF ~~ESFAS~~ trip units or associated instrument channels ~~(except CSAS)~~ inoperable, except SIRWT level.	----------NOTE---------- LCO 3.0.4 is not applicable. ----------	
	B.1 Place one ~~affected~~ bistable ~~trip unit~~ in ~~bypass or~~ trip.	8 hours ~~1 hour~~
	AND	
	B.2 Restore one trip unit to ~~B.2.1~~ ~~channel~~ OPERABLE status.	7 days ~~[48] hours~~
	~~AND~~	~~1 hour~~
	~~B.2.1 Restore channel to OPERABLE status.~~	~~[48] hours~~
	~~OR~~	
	~~B.2.2 Place affected trip unit in trip.~~	~~48 hours~~
C. One SIRWT Level switch inoperable.	----------NOTE---------- LCO 3.0.4 is not applicable. ----------	
	C.1 Bypass the SIRWT Level Switch.	8 hours ~~1 hour~~
~~C. One or more Functions with two ESFAS trip units or associated instrument channels (except CSAS) inoperable.~~	~~C.1 Place one trip unit in bypass and place the other trip unit in trip.~~	
	AND	
	C.2 Restore channel ~~one trip unit~~ to OPERABLE status.	7 days ~~[48] hours~~

(continued)

SURVEILLANCE REQUIREMENTS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more Functions with one or two automatic bypass removal channels inoperable.	D.1 ~~Disable bypass channels.~~ Remove the bypass function	8 hours ~~1 hour~~
	OR	
	D.2 Declare the affected Logic Channel inoperable and enter the appropriate Condition.	8 hours
	~~OR~~	
	~~D.2.1 Place affected trip units in bypass or trip.~~	~~1 hour~~
	~~AND~~	
	~~D.2.2.1 Restore bypass removal channel and affected trip units to OPERABLE status.~~	~~[48] hours~~
	~~OR~~	
	~~D.2.2.2 Place affected trip units in trip.~~	~~48 hours~~
E. ~~F.~~ Required Action and associated Completion Time not met.	E.1 ~~F.1~~ Be in MODE 3.	6 hours
	AND	
	E.2 ~~F.2~~ Be in MODE 4	30 hours ~~[12] hours~~

(continued)

SURVEILLANCE REQUIREMENTS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~E. One or more Functions with two automatic bypass removal channels inoperable.~~	~~NOTE~~ ~~LCO 3.0.4 is not applicable.~~	
	~~E.1 Disable bypass channels.~~	~~1 hour~~
	~~OR~~	
	~~E.2.1 Place one affected trip unit in bypass and place the other in trip for each affected ESFAS Function.~~	~~1 hour~~
	~~AND~~	
	~~E.2.2 Restore one bypass channel and the associated trip unit to OPERABLE status for each affected trip Function.~~	~~48 hours~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1 ~~SR 3.3.4.1~~	Perform a CHANNEL CHECK of each ESF ~~ESFAS~~ instrument channel except RAS SIRWT Level Switches and CHP Containment Pressure Switches.	12 hours
SR 3.3.3.2 ~~SR 3.3.4.2~~	Perform a CHANNEL FUNCTIONAL TEST of each ESF ~~ESFAS~~ instrument channel except SIRWT Level Switches.	92 days
~~SR 3.3.4.3~~	~~Perform a CHANNEL FUNCTIONAL TEST on each automatic bypass removal function.~~	~~Once within 92 days prior to each reactor startup~~
SR 3.3.3.3 ~~SR 3.3.4.4~~	Perform a CHANNEL CALIBRATION of each ESF ~~ESFAS~~ instrument channel, including bypass removal functions.	18 months
~~SR 3.3.4.5~~	~~Verify ESF RESPONSE TIMR is within limits.~~	~~[18] months on a STAGGERED TEST BASIS~~

Table 3.3.3-1
Engineered Safety Features Instrumentation

FUNCTION	~~MODES~~	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Safety Injection ~~Actuation~~ Signal (SIS)~~(SIAS)~~			
		SR 3.3.3.1	≥ 1593 psia
~~a. Containment Pressure - High~~		SR 3.3.3.2	~~≤ [19.0] psia~~
a.~~b.~~ Pressurizer Pressure - Low(a)		SR 3.3.3.3 ~~SR 3.3.3.4~~ ~~SR 3.3.4.5~~	
2. Containment High Pressure Signal(CHP) ~~Containment Spray Actuation Signal~~(b)			
a. Containment Pressure - Hi, left train		~~SR 3.3.4.1~~	≥ 3.7 and ≤ 4.4 psig
		SR 3.3.3.2	~~≤ [19.0] psia~~
		SR 3.3.3.3 ~~SR 3.3.3.4~~ ~~SR 3.3.4.5~~	
b. Containment Pressure - Hi, right train			≥ 3.7 and ≤ 4.4 psig
		SR 3.3.3.2	
		SR 3.3.3.33	
3. Containment High Radiation (CHR) ~~Isolation Actuation Signal~~			
a. Containment Area Radiation - Hi ~~Pressure - High~~			
		SR 3.3.3.1	≤ 20 R/hr
		SR 3.3.3.2	~~≤ [19.0] psia~~
		SR 3.3.3.3 ~~SR 3.3.3.4~~ ~~SR 3.3.4.5~~	
~~b. Containment Radiation - High~~			
		~~SR 3.3.4.1~~ ~~SR 3.3.4.2~~ ~~SR 3.3.4.4~~ ~~SR 3.3.4.5~~	~~≤[2x Background]~~
4. Steam Generator Low Pressure (SGLP) ~~4. Main Steam Isolation Signal~~			
a. "A" Steam Generator Pressure - Low (b)			
		SR 3.3.3.1	
		SR 3.3.3.2	≥ 500 psia
		SR 3.3.4.3	~~≥ [495] psia~~
		~~SR 3.3.3.4~~ ~~SR 3.3.4.5~~	
b. "B" Steam Generator Pressure - Low (b)			
		SR 3.3.3.1	≥ 500 psia
		SR 3.3.3.2	
		SR 3.3.3.3	

FUNCTION	~~MODES~~	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5.~~6.~~ Aux Feedwater Actuation Signal (AFAS)			
a. "A" Steam Generator Level - Low		SR 3.3.3.1 SR 3.3.3.2 SR 3.3.3.3 ~~SR 3.3.4.4~~ ~~SR 3.3.4.5~~	≥ 25.9 % ~~≥ [45.7] %~~
b. "B" Steam Generator Level - Low		SR 3.3.3.1 SR 3.3.3.2 SR 3.3.3.3~~SR3.3.4.4.4~~ ~~SR 3.3.4.5~~	≥ 25.9 % ~~≥ [45.7] %~~
~~c. Steam Generator Pressure Difference - High (A > B) or (B > A)~~		~~SR 3.3.4.1~~ ~~SR 3.3.4.2~~ ~~SR 3.3.4.4~~ ~~SR 3.3.4.5~~	~~≤ [48.3] psid~~
6.~~5.~~ Recirculation Actuation Signal (RAS)			
a. SIRWT Level Switches - Low ~~a. Refueling Water Tank Level - Low~~		SR 3.3.3.3 ~~[SR 3.3.4.1]~~ ~~SR 3.3.4.2~~ ~~SR 3.3.4.4~~ ~~SR 3.3.4.5~~	≥ 21 and ≤ 27 inches inches above SIRWT floor ~~[≥ 24 inches and ≤ 30] inches above tank bottom~~

(a) Pressurizer Pressure - Low may be manually bypassed when pressurizer pressure is < 1700 psia. The bypass shall be automatically removed whenever pressurizer pressure is ≥ 1700 psia.

(b) Steam Generator Pressure - Low may be manually bypassed when steam generator pressure is < 550 psia. The bypass shall be automatically removed whenever steam generator pressure is ≥ 550 psia.

~~(b) SIAS is also required as a permissive to initiate containment spray.]~~

~~(c) Steam Generator Pressure - Low may be manually bypassed when steam generator pressure is < [785] psia. The bypass shall be automatically removed whenever steam generator pressure is ≥ [785] psia.~~

~~(d) Only the Main Steam Isolation Signal (MSIS) Function and the Steam Generator Pressure - Low and Containment Pressure - High signals are not required to be OPERABLE when all associated valves isolated by the MSIS Function are closed and [de-activated].~~

3.3 INSTRUMENTATION

3.3.4 Engineered Safety Features (ESF) System Logic and Manual Initiation~~L~~

~~3.3.5 Engineered Safety Features Actuation System (ESFAS) Logic and Manual Trip (Analog)~~

LCO 3.3.4 Two ESF ~~ESFAS~~ Manual Initiation and two ESF ~~ESFAS~~ Actuation Logic channels shall be OPERABLE for each ESF ~~ESFAS~~ Function specified in Table 3.3.4-1. ~~3.3.5-1~~

APPLICABILITY: MODES 1, 2, and 3. ~~According to Table 3.3.5-1.~~

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one Manual Initiation ~~Auxiliary Feedwater Actuation Signal (AFAS) Manual Trip or~~ Actuation Logic channel inoperable.	A.1 Restore channel to OPERABLE status.	48 hours
B. Required Action and associated Completion Time ~~of Condition A~~ not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4	6 hours 30 hours ~~[12] hours~~
~~C. One or more Functions with one Manual Trip or Actuation Logic channel inoperable except AFAS.~~	~~C.1 Restore channel to OPERABLE status.~~	~~48 hours~~

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~D. Required Action and associated Completion Time of Condition C not met.~~	~~D.1 Be in MODE 3.~~ ~~AND~~ ~~D.2 Be in MODE 5.~~	~~6 hours~~ ~~36 hours~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
	~~NOTES~~ ~~1. Testing of Actuation Logic shall include verification of the proper operation of each initiation relay.~~ ~~2. Relays associated with plant equipment that cannot be operated during plant operation are only required to be tested during each MODE 5 entry exceeding 24 hours unless tested during the previous 6 months.~~	
~~SR 3.3.5.1~~ SR 3.3.4.1	Perform a CHANNEL FUNCTIONAL TEST on the AFAS ~~ESFAS~~ Logic channels	92 days
SR 3.3.4.2	Perform a CHANNEL FUNCTIONAL TEST on the SIS Logic normal and emergency power functions using simulated signal.	92 days
SR3.3.4.3 SR 3.3.5.2	Perform a CHANNEL FUNCTIONAL TEST on each ESF ~~ESFAS~~ Manual Initiation ~~Trip~~ channel.	18 months
SR 3.3.4.4	Perform a CHANNEL FUNCTIONAL TEST on each ESF logic channel to verify all automatic actuations and automatic resetting of Low Pressure Bypasses.	18 months

Table 3.3.4.1 ~~3.3.5-1~~
Engineered Safety Features ~~Actuation System~~ Logic and Manual Channel ~~Applicability~~

FUNCTION	~~Applicable Modes~~	SURVEILLANCE REQUIREMENTS
1. Safety Injection ~~Actuation~~ Signal (SIS)		
a. Manual Initiation	~~1,2,3,(4)~~	SR 3.3.4.3
b. SIS Logic Trains (Initiation, Actuation, and Low Pressure Bypass auto reset)		SR 3.3.4.2 SR 3.3.4.4
c. CHP Signal SIS Initiation (5P Relay Output)		SR 3.3.4.4
2. Containment High Pressure Signal (CHP) ~~Spray Actuation Signal~~	~~1,2,3,(4)~~	
a. Manual Initiation (a)		SR 3.3.4.3
a. CHP Logic Trains		SR 3.3.4.4
3. Containment Hi Radiation (CHR) ~~Isolation Actuation Signal~~	~~1,2,3,~~	
a. Manual Initiation		SR 3.3.4.3
b. CHR Logic Trains		SR 3.3.4.4
~~4. Steam Generator Low Pressure (SGLP) Main Steam Isolation Signal~~	~~1,2,3,4~~	
a. Manual Initiation (a)		SR 3.3.4.3
b. SGLP Logic Trains		SR 3.3.4.4
5. Aux Feedwater Actuation Signal (AFAS) ~~Recirculation Actuation Signal~~	~~1,2,3,4~~	
a. Manual Initiation (a)		SR 3.3.4.3
b. AFAS Logic Trains		SR 3.3.4.1
6. Recirculation Actuation Signal (RAS) ~~Auxiliary Feedwater Actuation Signal~~		
a. Manual Initiation (a)	~~1,2,34~~	SR 3.3.4.3
b. RAS Logic Trains		SR 3.3.4.4

(a) manual initiation achieved by individual component controls

~~3.3 INSTRUMENTATION~~

~~3.3.5 Diesel Generator (DG) — Undervoltage Start (UV Start)~~
~~3.3.6 Diesel Generator (DG) — Loss of Voltage Start (LOVS) (Analog)~~

LCO 3.3.5 Three Loss of Voltage and Three Degraded Voltage sensors and associated auto start logic channels shall be OPERABLE for each DG.
~~[Four] channels of Loss of Voltage Function and [four] channels of Degraded Voltage Function auto initiation instrumentation per DG shall be OPERABLE.~~

APPLICABILITY: ~~Modes 1, 2, 3, and 4~~
When associated DG is required to be OPERABLE. ~~by LCO 3.8.2, "AC Sources — Shutdown."~~

ACTIONS

~~NOTE~~
~~Separate Condition entry is allowed for each Function.~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more sensors or logic channels per DG inoperable.	A.1 Declare the affected DG Inoperable and enter the appropriate Condition. ~~Place channel in bypass or trip.~~ ~~AND~~ ~~A.2.1 Restore channel to OPERABLE status.~~ ~~OR~~ ~~A.2.2 Place the channel in trip.~~	Immediately ~~1 hour~~ ~~[48] hours~~ ~~48 hours~~

~~B. One or more Functions with two channels per DG inoperable.~~	~~B.1 Enter applicable Conditions and Required Actions for the associated DG made inoperable by DG – LOVS instrumentation.~~ ~~OR~~ ~~B.2.1 NOTE~~ ~~LCO 3.0.4 is not applicable.~~ ~~Place one channel in bypass and the other channel in trip.~~ ~~AND~~ ~~B.2.2 Restore one channel to OPERABLE status.~~	~~1 hour~~ ~~1 hour~~ ~~[48] hours~~
~~C. One or more Functions with more than two channels inoperable.~~	~~C.1 Restore all but two channels to OPERABLE status.~~	~~1 hour~~
~~D. Required Action and associated Completion Time not met.~~	~~D.1 Enter applicable Conditions and Required Actions for the associated DG made inoperable by DG – LOVS instrumentation.~~	~~Immediately~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	Perform CHANNEL CALIBRATION with setpoint Allowable Values as follows: a. Loss of Voltage Function ≥ 1860 V dropout; and time delays of: ≤ 8.15 seconds at 1400 Volts; and ≤ 4.2 seconds at 930 Volts. b. Degraded Voltage Function ≥ 2184 Volts and Time delay of: ≤ 0.8 seconds.	18 months
~~SR 3.3.6.1~~	~~Perform CHANNEL CHECK.~~	~~12 hours~~
~~SR 3.3.6.2~~	~~Perform CHANNEL FUNCTIONAL TEST.~~	~~[92] days~~
~~SR 3.3.6.3~~	~~Perform CHANNEL CALIBRATION with setpoint Allowable Values as follows:~~ ~~a. Degraded Voltage Function ≥ [3180] V and ≤ [3220] V~~ ~~Time delay: ≥ [ ] seconds and ≤ [ ] seconds at [ ] V; and~~ ~~b. Loss of Voltage Function ≥ [3180] V and ≤ [3220] V~~ ~~Time delay: ≥ [ ] seconds and ≤ [ ] seconds at [ ] V.~~	~~[18] months~~

3.3 INSTRUMENTATION

3.3.6 Refueling Containment High Radiation (CHR) Initiation ~~7 Containment Purge Isolation Signal (CPIS) (Analog)~~

LCO 3.3.6 Two Refueling CHR channels shall be OPERABLE.~~7 [Four] CPIS containment radiation monitor channels and one CPIS automatic Actuation Logic and one Manual Trip train shall be OPERABLE.~~

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or Two Refueling CHR channels inoperable. ~~One radiation monitor channel inoperable.~~	A. ~~A.1 Place the affected channel in trip.~~	Immediately
	~~OR~~	
	~~A.2.~~1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.~~2.~~2 Suspend movement of irradiated fuel assemblies within containment.	~~4 hours~~
		~~Immediately~~
		~~Immediately~~

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~B. One required Manual Trip or automatic Actuation Logic train inoperable.~~ ~~OR~~	~~B.1 Place and maintain containment purge and exhaust valves in closed position.~~ ~~AND~~	~~Immediately~~ ~~(continued)~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform a CHANNEL CHECK on each refueling CHR channel.	24 hours
~~7.1~~	~~Perform a CHANNEL CHECK on each containment radiation monitor channel.~~	~~12 hours~~
SR 3.3.6.2	Perform a CHANNEL FUNCTIONAL TEST on each refueling CHR channel. Verify Containment refueling Radiation Monitor Channel - high radiation setpoint is ≤ 15 mRem/hr above background.	31 days ~~[92] days~~
~~SR 3.3.7.2~~	~~Perform a CHANNEL FUNCTIONAL TEST on each containment radiation monitor channel.~~ ~~Verify CPIS high radiation setpoint Allowable Value is ≤ [220 mR/hr].~~	~~(continued)~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.6.3~~SR 3.3.7.4~~	Perform a CHANNEL CALIBRATION on each containment refueling radiation monitor channel.	18 months
~~SR 3.3.7.3~~	~~NOTE~~ ~~Testing of Actuation Logic shall include verification of the proper operation of each initiation relay.~~ ~~Perform a CHANNEL FUNCTIONAL TEST on each CPIS Actuation Logic channel.~~	~~[31] days~~

~~SR 3.3.7.5 Perform a CHANNEL FUNCTIONAL TEST on each CPIS Manual Trip channel.~~
~~[18] months~~

~~SR 3.3.7.6 Verify CPIS response time of each containment radiation channel is within limits.~~

~~[18] months on a STAGGERED TEST BASIS~~

3.3 INSTRUMENTATION

3.3.7 ~~3.3.11 Post~~ Accident Monitoring ~~(PAM)~~ Instrumentation ~~(Analog)~~

LCO 3.3.7 ~~3.3.11~~ The accident monitoring instrumentation for each Function in Table 3.3.7-1 shall be OPERABLE.
~~The PAM instrumentation for each Function in Table 3.3.11-1 shall be OPERABLE.~~

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTES----------

1. LCO 3.0.4 is not applicable.

2. Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	7 days ~~30 days~~
B. One or more Functions with two required channels inoperable. ~~Required Action and associated Completion Time of Condition A not met.~~	B.1 Restore one channel to OPERABLE status. ~~Initiate action in accordance with Specification 5.6.8.~~	48 hours ~~Immediatley~~

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~NOTE~~ ~~Not applicable to hydrogen monitor channels.~~		
C. Required Action and associated Completion Time of Condition A not met for Functions 1 through 15. ~~One or more Functions with two required channels inoperable.~~ OR Required Action and associated Completion Time of Condition B not met for Functions 1 through 19.	C.1 Be in MODE 3. ~~Restore one channel to OPERABLE status~~ AND C.2 Be in MODE 4.	6 hours ~~7 days~~ 30 hours
D. Required Action and associated Completion Time of Condition A not met for Functions 16 through 21. ~~Two hydrogen monitor channels inoperable.~~ OR Required Action and associated Completion Time of Condition B not met for Functions 20 or 21.	D.1 Initiate action in accordance with Specification 5.6.7. (Report to NRC) ~~Restore one hydrogen monitor channel to OPERABLE status.~~ AND D.2 Restore channel to OPERABLE status	Immediately 7~~2 hours~~ Prior to entering MODE 2 following the next MODE 6 entry
~~E. Required Action and associated Completion Time of Condition C or D not met.~~	~~E.1 Enter the Condition referenced in Table 3.3.11-1 for the channel.~~	~~Immediately~~

~~F. As required by Required Action E.1 and referenced in Table 3.3.11-1.~~	~~F.1 Be in MODE 3.~~ ~~AND~~ ~~F.2 Be in MODE 4.~~	~~6 hours~~ ~~12 hours~~
~~G. As required by Required Action E.1 and referenced in Table 3.3.11-1.~~	~~G.1 Initiate action in accordance with Specification 5.6.8.~~	~~Immediately~~

SURVEILLANCE REQUIREMENTS

~~NOTE~~
~~These SRs apply to each PAM instrumentation Function in Table 3.3.7-1~~

SURVEILLANCE		FREQUENCY
SR 3.3.7.1 ~~3.3.11.1~~	Perform CHANNEL CHECK for each required instrumentation channel except valve position ~~that is normally energized.~~	31 days
SR 3.3.7.2~~3.3.11.2~~	NOTE ~~1. Neutron detectors are excluded from CHANNEL CALIBRATION.~~ 1. Calibrate Core Exit Thermocouple circuitry by substituting a known voltage for thermocouple voltage. Perform CHANNEL CALIBRATION.	18 ~~[18]~~ months

Table 3.3.7-1
Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	~~CONDITIONS REFERENCED FROM REQUIRED ACTION~~ ~~D.1~~
1. PCS Wide Range Hot Leg Temperature	2	
2. PCS Wide Range Cold Leg Temperature	2	
3. Wide Range Flux	2	
4. Containment Floor Water Level	2	
5. Subcooled Margin Monitor	2	
6. Wide Range Pressurizer Level	2	
7. Containment H_2 Concentration	2	
8. Condensate Storage Tank Level	2	
9. Wide Range Pressurizer Pressure	2	
10. Wide Range Containment Pressure	2	
11. Wide Range "A" SG Level	2	
12. Wide Range "B" SG Level	2	
13. Narrow Range "A" SG Pressure	2	
14. Narrow Range "B" SG Pressure	2	
15. Position Indication for each Containment Isolation Valve [a]	1/valve	
16. Core Exit Thermocouples - Quadrant 1	4	
17. Core Exit Thermocouples - Quadrant 2	4	
18. Core Exit Thermocouples - Quadrant 3	4	
19. Core Exit Thermocouples - Quadrant 4	4	
20. Reactor Vessel Level Monitoring Sys.	2	
21. High Range Containment Radiation	2	

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and de-activated automatic valve, closed manual valve, blind flange or check valve, with flow through the valve secured.

~~Table 3.3.11-1 (page 1 of 1)~~
~~Post-Accident Monitoring Instrumentation~~

~~FUNCTION~~	~~REQUIRED CHANNELS~~	~~CONDITIONS REFERENCED FROM REQUIRED ACTION D.1~~
~~1. [Logarithmic] Neutron Flux~~	~~2~~	~~F~~
~~2. Reactor Coolant System Hot Leg Temperature~~	~~2 per loop~~	~~F~~
~~3. Reactor Coolant System Cold Leg Temperature~~	~~2 per loop~~	~~F~~
~~4. Reactor Coolant System Pressure (wide range)~~	~~2~~	~~F~~
~~5. Reactor Vessel Water Level~~	~~2~~	~~[G]~~
~~6. Containment Sump Water Level (wide range)~~	~~2~~	~~F~~
~~7. Containment Pressure (wide range)~~	~~2~~	~~F~~
~~8. Containment Isolation Valve Position~~	~~2 per penetration flow path[(a)(b)]~~	~~F~~
~~9. Containment Area Radiation (high range)~~	~~2~~	~~[G]~~
~~10. Containment Hydrogen Monitors~~	~~2~~	~~F~~
~~11. Pressurizer Level~~	~~2~~	~~F~~
~~12. Steam Generator Water Level (wide range)~~	~~2 per steam generator~~	~~F~~
~~13. Condensate Storage Tank Level~~	~~2[(c)]~~	~~F~~
~~14. Core Exit Temperature - Quadrant [1]~~	~~2[(c)]~~	~~F~~
~~15. Core Exit Temperature - Quadrant [2]~~	~~2[(c)]~~	~~F~~
~~16. Core Exit Temperature - Quadrant [3]~~	~~2[(c)]~~	~~F~~
~~17. Core Exit Temperature - Quadrant [4]~~	~~2[(c)]~~	~~F~~
~~18. Auxiliary Feedwater Flow~~	~~2~~	~~F~~

~~(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.~~

~~(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.~~

~~(c) A channel consists of two or more core exit thermocouples.~~

~~Note:Table 3.3.11-1 shall be amended for each unit as necessary to list:~~

~~(1) all Regulatory Guide 1.97, Type A instruments, and~~
~~(2) all Regulatory Guide 1.97, Category I, non-Type A instruments specified in the unit's Regulatory Guide 1.97, Safety Evaluation Report.~~

3.3 INSTRUMENTATION

3.3.8 Alternate Shutdown System
~~3.3.12 Remote Shutdown System (Analog)~~

LCO 3.3.8~~3.3.12~~ The Alternate Shutdown System Functions in Table 3.3.8-1 shall be OPERABLE.
~~The Remote Shutdown System Functions in Table 3.3.12-1 shall be OPERABLE.~~

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTES----------

1. LCOs 3.0.3 and 3.0.4 are ~~is~~ not applicable.

2. Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required Channels ~~Functions~~ inoperable.	A.1 Provide equivalent shutdown capability. Restore required Functions to OPERABLE status. AND A.2 Restore Channel to OPERABLE status.	7 days ~~30 days~~ 60 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4	6 hours 30 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.8.1~~3.3.12.1~~	Perform CHANNEL CHECK for each required instrumentation Function 2 through 12 in Table 3.3.8-1. ~~channel that is normally energized.~~	92 days ~~31 days~~
SR 3.3.8.2~~3.3.12.2~~	Perform a CHANNEL CHECK of Function 1 in Table 3.3.8-1 (Neutron Flux). ~~Verify each required control circuit and transfer switch is capable of performing the intended function.~~	Once within 7 days prior to each Reactor Startup ~~[18] months~~
SR 3.3.8.3~~3.3.12.3~~	Perform a CHANNEL CHECK of Functions 13 and 14 in Table 3.3.8-1. (P-8B flow to SGs) ~~NOTE~~ ~~Neutron detectors are excluded from the CHANNEL CALIBRATION.~~ ~~Perform CHANNEL CALIBRATION for each required instrumentation channel.~~	18 months
SR 3.3.8.4~~3.3.12.4~~	Perform a CHANNEL FUNCTIONAL TEST on Functions 13 through 18 in Table 3.3.8-1. ~~Perform CHANNEL FUNCTIONAL TEST of the reactor trip circuit breaker open/closed indication.~~	18 months
SR 3.3.8.5	Perform CHANNEL CALIBRATION for each required instrumentation channel of Functions 1 through 15 in Table 3.3.8-1.	18 months

Table 3.3.8-1 (page 1 of 1)
Alternate Shutdown System Instrumentation and Controls

FUNCTION/INSTRUMENT OR CONTROL PARAMETER	SURVEILLANCE REQUIREMENTS	REQUIRED CHANNELS
1. Neutron Flux	SR 3.3.8.2 SR 3.3.8.5	1
2. Pressurizer Pressure	SR 3.3.8.1 SR 3.3.8.5	1
3. Pressurizer Level	SR 3.3.8.1 SR 3.3.8.5	1
4. PCS #1 Hot Leg Temperature	SR 3.3.8.1 SR 3.3.8.5	1
5. PCS #2 Hot Leg Temperature	SR 3.3.8.1 SR 3.3.8.5	1
6. PCS #1 Cold Leg Temperature	SR 3.3.8.1 SR 3.3.8.5	1
7. PCS #2 Cold Leg Temperature	SR 3.3.8.1 SR 3.3.8.5	1
8. "A" SG Pressure	SR 3.3.8.1 SR 3.3.8.5	1
9. "B" SG Pressure	SR 3.3.8.1 SR 3.3.8.5	1
10. "A" SG Level	SR 3.3.8.1 SR 3.3.8.5	1
11. "B" SG Level	SR 3.3.8.1 SR 3.3.8.5	1
12. SIRW Tank Level	SR 3.3.8.1 SR 3.3.8.5	1
13. AFW Pump P-8B Flow to "A" SG	SR 3.3.8.3 SR 3.3.8.4 SR 3.3.8.5	1
14. AFW Pump P-8B Flow to "B" SG	SR 3.3.8.3 SR 3.3.8.4 SR 3.3.8.5	1
15. AFW Pump P-8B Suction Pressure Alarm	SR 3.3.8.4 SR 3.3.8.5	1
16. AFW Pump P-8B Steam Valve Control	SR 3.3.8.4	1
17. AFW Flow Control "A" SG	SR 3.3.8.4	1
18. AFW Flow Control "B" SG	SR 3.3.8.4	1

~~Table 3.3.12-1 (page 1 of 1)~~
~~Remote Shutdown System Instrumentation and Controls~~

~~NOTE~~

~~This table is for illustration purposes only. It does not attempt to encompass every Function used at every unit, but does contain the types of Functions commonly found.~~

~~FUNCTION/INSTRUMENT OR CONTROL PARAMETER~~	~~REQUIRED NUMBER OF DIVISIONS~~
~~1. Reactivity Control~~	
~~a. Log Power Neutron Flux~~	~~[1]~~
~~b. Source Range Neutron Flux~~	~~[1]~~
~~c. Reactor Trip Circuit Breaker Position~~	~~[1 per trip breaker]~~
~~d. Manual Reactor Trip~~	~~[2]~~
~~2. Reactor Coolant System Pressure Control~~	
~~a. Pressurizer Pressure or RCS Wide Range Pressure~~	~~[1]~~
~~b. Pressurizer Power Operated Relief Valve Control and Block Valve Control~~	~~[1, controls must be for power operated relief valve and block valves on same line]~~
~~3. Decay Heat Removal via Steam Generators~~	
~~a. Reactor Coolant Hot Leg Temperature~~	~~[1 per loop]~~
~~b. Reactor Coolant Cold Leg Temperature~~	~~[1 per loop]~~
~~c. Auxiliary Feedwater Controls~~	~~[1]~~
~~d. Steam Generator Pressure~~	~~[1 per steam generator]~~
~~e. Steam Generator Level or Auxiliary Feedwater Flow~~	~~[1 per steam generator]~~
~~f. Condensate Storage Tank Level~~	~~[1]~~
~~4. Reactor Coolant System Inventory Control~~	
~~a. Pressurizer Level~~	~~[1]~~
~~b. Reactor Coolant Charging Pump Controls~~	~~[1]~~

3.3 INSTRUMENTATION

3.3.9 Neutron Flux Monitoring Channels
~~3.3.13 [Logarithmic] Power Monitoring Channels (Analog)~~

LCO 3.3.9 Two channels of Neutron Flux monitoring instrumentation shall be OPERABLE.

~~LCO 3.3.13 Two channels of [logarithmic] power level monitoring instrumentation shall be OPERABLE.~~

APPLICABILITY: MODES 3, 4, and 5 with no more than one Control Rod capable of withdrawal.
~~MODES 3, 4, and 5, with the reactor trip circuit breakers open or Control Element Assembly (CEA) Drive System not capable of CEA withdrawal.~~

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Suspend all operations involving positive reactivity additions.	Immediately
	AND	
	A.2 Perform SDM verification in accordance with SR 3.1.1.1, if T_{avg} > 525°F, or SR 3.1.2.1, if T_{avg} ≤ 525°F ~~if T_{avg} > 200°F, or SR 3.1.2.1, if T_{avg} ≤ 200°F~~	4 hours AND Once per 12 hours thereafter

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.9.1 ~~SR 3.3.12.1~~ Perform CHANNEL CHECK on each required neutron flux channel.	12 hours
SR 3.3.9.2 ~~SR 3.3.12.2~~ Perform CHANNEL CALIBRATION on each required neutron flux channel. ~~Perform CHANNEL FUNCTIONAL TEST.~~	18 months ~~[92] days~~
~~SR 3.3.13.3 NOTE~~ ~~Neutron detectors are excluded from CHANNEL CALIBRATION.~~ ~~Perform CHANNEL CALIBRATION.~~	~~[18] months~~

3.3 INSTRUMENTATION

3.3.10 Spent Fuel Pool Radiation Monitor

LCO 3.3.10 Two Spent Fuel Pool (SFP) Radiation Monitors shall be OPERABLE.

APPLICABILITY: Whenever fuel is in the SFP area.

---NOTE---
LCOs 3.0.3 and 3.0.4 are not Applicable

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or Two Spent Fuel Pool Radiation Monitors inoperable.	A.1 Suspend fuel movement in the SFP area.	Immediately
	AND	
	A.2.1 Restore Monitors to Operable Status	72 hours
	OR	
	A.2.2 Provide equivalent monitoring capability	72 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.10.1	Perform a CHANNEL CHECK on each SFP Radiation Monitoring channel.	24 hours
SR 3.3.10.2	Perform a CHANNEL FUNCTIONAL TEST on each SFP Radiation Monitoring channel.	31 days
SR 3.3.10.3	Perform a CHANNEL CALIBRATION on each SFP Radiation Monitoring channel.	18 months

ENCLOSURE 1

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

TECHNICAL SPECIFICATION CHANGE REQUEST

PART 6 - SECTION 3.3

March 27, 1996

CONSUMERS POWER COMPANY
Docket 50-255
Request for Change to the Technical Specifications
License DPR-20

3.3 INSTRUMENTATION SECTION CHANGE REQUEST

It is requested that the Instrumentation Systems requirements of the Technical Specifications contained in the Facility Operating License DPR-20, Docket 50-255, issued to Consumers Power Company on February 21, 1991, for the Palisades Plant be changed as described below:

I. ARRANGEMENT AND CONTENT OF THIS PART OF THE CHANGE REQUEST:

This section of the Technical Specification Change Request (TSCR) proposes changes to those Palisades Technical Specification requirements addressing the Instrumentation Systems components. These changes are intended to result in requirements which are appropriate for the Palisades plant, but closely emulate those of the Standard Technical Specifications, Combustion Engineering Plants, NUREG 1432, Revision 1.

This discussion and its supporting information frequently refer to three sets of Technical Specifications; the following abbreviations are used for clarity and brevity:

TS - The existing Palisades Technical Specifications,
RTS - The revised Palisades Technical Specifications,
STS - NUREG 1432, Revision 1.

Six attachments are provided to assist the reviewer. The numbering and content of the attachments is consistent with other parts of the TSCR.

1. Proposed RTS pages
2. Bases for the RTS
3. A line by line comparison of the TS and RTS
4. STS pages marked to show the differences between RTS and STS
5. STS Bases pages marked to show differences between RTS and STS Bases.
6. A line by line comparison of RTS and STS.

Attachment 3, the line by line comparison of TS and RTS, is presented in a tabular format. The first page contains an explanation of the syntax and abbreviations used. The table is arranged numerically by TS item number. Each requirement in Sections 1 through 4 of TS is listed individually. In some cases, where a single numbered TS requirement contains more than one requirement, each requirement is listed individually under the same number. Requirements which appear in RTS or STS, but not in TS, do not appear in the Attachment 3 listing.

Attachment 3 Provides the Following Information for Each TS Requirement:

Identifying number of TS item,
Identifying number of closest equivalent RTS item,
Identification of TS item as LCO, Action, SR, etc.,
A short paraphrase of requirement,
A description of each proposed change from TS to RTS.

Classification of Change as One of the Following Categories:

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

RELOCATED - A change which only moves requirements, not meeting the 10 CFR 50.36(c)(2)(ii) criteria, from the TS to the FSAR, to the Operating Requirements Manual, or to other documents controlled under 10 CFR 50.59.

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

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

Attachment 6, the line by line comparison of RTS and STS, is also presented in a tabular format. The first page contains an explanation of the syntax and abbreviations used; the second page contains a list of Palisades terminology used in place of the generic STS terminology. The table is arranged numerically by RTS item number. Each requirement in Sections 1 through 3 of RTS or STS is listed individually. Requirements which appear in TS, but not in RTS or STS, do not appear in the Attachment 6 listing.

Attachment 6 Provides the Following Information for Each RTS Requirement:

Identifying number of RTS requirement,
Identifying number of equivalent STS requirement,
Identification of each requirement as LCO, Action, SR, etc.,
Short paraphrase of each requirement,
A description of each proposed change from RTS to STS.

II. TECHNICAL SPECIFICATION CHANGES PROPOSED:

The TS LCOs and action statements for the RPS Instrumentation settings (LSSS) appear in Section 2.3. Those for the ESF Instrumentation are in Section 3.16. The LCOs for the RPS, ESF Instrumentation, Accident Monitoring, Remote Shutdown, and a variety of miscellaneous instrumentation are in Section 3.17. The TS surveillance requirements appear in TS Section 4. All RTS requirements for the RPS Instrumentation, ESF Instrumentation, DG UV Start Instrumentation, Accident Monitoring Instrumentation, Remote Shutdown Instrumentation, and some miscellaneous instrumentation (Refueling CHR, Neutron Flux Monitoring, Spent Fuel Pool Monitoring) appear in proposed Section 3.3. Each proposed change from TS to RTS is discussed in the attachments to this part of the TSCR.

Each proposed change to a requirement in TS is described in Attachment 3.

Those proposed RTS requirements which have no counterpart in TS are described in Attachment 6. These new requirements are identified by the word "New" in the third column of Attachment 6.

The Major Changes From TS to RTS Proposed in This Part of the TSCR are:

1. Some of the requirements for the miscellaneous instrumentation of TS Section 3.17.6 have been relocated. Others have been retained in the Instrument chapter (RTS Section 3.3), or moved to other RTS sections. There are very few cases in which other instrumentation in TS 3.17 has been relocated.

2. Current TS 3.17.3 requires that the CHP, CHR, and SGLP functions be operable when above cold shutdown. In the RTS, the requirements for MODE 4 operability have been eliminated, and these functions are only required in MODES 1, 2, and 3, as in the other ESF functions required by TS Section 3.17.2. In the case of SGLP, the bypass permissive below 550 psi makes this function unnecessary in the lower modes. In the case of CHP and CHR, the STS bases statement that accidents are slow to develop in the lower modes is valid here. The ability to manually actuate in MODE 4 is retained, but only by virtue of individual component actuation per LCO 3.6.3, "Containment Isolation Valves."

3. The CHANNEL FUNCTIONAL TEST interval in the RPS and ESF Instrument chapters has been changed from 31 days to 92 days, in accordance with the justification provided in "NRC evaluation of CEOG Topical Report CEN-327, RPS/ESFAS Extended Test Interval Evaluation" dated November 6 1989. By the time of RTS implementation, Palisades will have reviewed instrument drift information for each instrument channel involved over a 2 year period. Records of this review, including as-found and as-left values and other supporting data covering the period will be available for audit.

4. In each section of the proposed RTS, new requirements taken from STS have been proposed. Since there is no equivalent requirement in TS, these changes do not appear in Attachment 3. The new requirements do appear in Attachment 6 where they are identified by an entry of "New" in the third column.

   The changes identified as "New" are considered MORE RESTRICTIVE because they add requirements and operating restrictions which do not exist in the current Palisades TS.

<u>The Major Differences Between RTS and STS in This Part of the TSCR are</u>:

1. RTS LCO 3.3.1 addresses RPS instrumentation in MODES 1, 2, 3, 4, and 5, with more than one Control Rod capable of being withdrawn, and requires all RPS functions to be OPERABLE. In STS, LCO 3.3.1 addresses MODE 1 and 2 operation, requiring all RPS functions. STS LCO 3.3.2 addresses the lower modes, requiring only Rate of change of power - High trip protection. The more restrictive RTS requirements are retained from existing TS, and RTS Section 3.3.1 roughly encompasses STS LCOs 3.3.1 and 3.3.2.

2. RTS LCO 3.3.2 (RPS Logic) is analogous to STS LCO 3.3.3

3. RTS LCO 3.3.3 and 3.3.4 (ESF Instrumentation/Logic) are analogous to STS LCOs 3.3.4 and 3.3.5, respectively.

4. RTS LCO 3.3.5 (DG UV Start) is analogous to RTS LCO 3.3.6 (DG LOVS).

5. RTS LCO 3.3.6 (Refueling CHR) has no STS counterpart, though CPIS (STS LCO 3.3.7) was the closest, and used as the model.

6. There is no Palisades equivalent of STS LCO 3.3.8, CRIS.

7. There is no Palisades equivalent of STS LCO 3.3.9, CVCS Isolation.

8. There is no Palisades equivalent of STS LCO 3.3.10, SBFAS.

9. RTS LCO 3.3.7, Accident Monitoring, is equivalent to STS LCO 3.3.11.

10. RTS LCO 3.3.8 Alternate Shutdown is equivalent to STS LCO 3.3.12.

11. RTS LCO 3.3.9 (Neutron Flux Monitoring) is equivalent to STS LCO 3.3.13.

12. RTS LCO 3.3.10, Spent Fuel Pool Monitoring, has no STS equivalent.

III. NO SIGNIFICANT HAZARDS ANALYSIS:

Each change proposed is classified in Attachment 3 as either ADMINISTRATIVE, RELOCATED, MORE RESTRICTIVE, or LESS RESTRICTIVE.

Analysis of ADMINISTRATIVE, RELOCATED, and MORE RESTRICTIVE Changes:

ADMINISTRATIVE changes and RELOCATED changes move requirements, either within the TS or to documents controlled under 10 CFR 50.59, or clarifying existing TS requirements, without affecting their technical content. Since ADMINISTRATIVE and RELOCATED changes do not alter the technical content of any requirements, they cannot involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any previously evaluated, or involve a significant reduction in a margin of safety.

MORE RESTRICTIVE changes only add new requirements, or revise existing requirements to result in additional operational restrictions. Since the TS, with all MORE RESTRICTIVE changes incorporated, will still contain all of the requirements which existed prior to the changes, MORE RESTRICTIVE changes cannot involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any previously evaluated, or involve a significant reduction in a margin of safety.

Analysis of LESS RESTRICTIVE Changes:

The LESS RESTRICTIVE Changes Proposed in This Part of the TSCR are:

1. TS LCO 3.17.3 (Instrumentation Requirements for Isolation Functions) requires both automatic and manual initiation capability for the Steam Generator Low Pressure (SGLP), Containment High Pressure (CHP), and Containment High Radiation (CHR) isolation functions be Operable when in the equivalent of MODES 1, 2, 3, and 4 (above Cold Shutdown). Proposed RTS LCOs 3.3.3 (Automatic Initiation) and 3.3.4 (Manual Initiation) require these functions to be operable in MODES 1, 2, and 3. SGLP closes the Main Steam Isolation Valves and Feedwater Regulating Valves to isolate the steam generators; CHP switches control room ventilation to the emergency mode, initiates Safety Injection, Containment Spray, and closure of containment isolation valves; CHR switches control room ventilation to the emergency mode and initiates closure of containment isolation valves.

   The change of applicability for the automatic initiation instrumentation is in agreement with the applicability in equivalent STS LCOs 3.3.4 and 3.3.5. In MODE 4, which was omitted from the applicability, PCS temperature is less than 300°F and saturation pressure is less than 67 psia. With these relatively low pressures and temperatures, the probability of an accident and the consequent energy release are both quite low. Automatic isolation of the containment is not necessary.

The change in applicability does not actually affect the operability requirements for manual initiation of SGLP or CHP. There is no single control (i.e., pushbutton or switch) which would manually initiate either SGLP or CHP. Manual initiation of these functions can only be achieved by individually actuating the manual control for each affected component. Manual control operability is considered part of the component operability, and is required, for isolation components, by LCO 3.6.3, "Containment Isolation Valves."

The proposed change in applicability does reduce the requirement for manual initiation of CHR. Manual initiation of CHR can be accomplished by actuation of a single push button in each channel, which was installed for the purposes of testing the circuitry. Since the low event probability and low system energy content allows time for isolating the containment, and the capability to manually actuate each containment isolation valve is required by LCO 3.6.3, the operability of the CHR initiation pushbuttons is not required in MODE 4.

2. Tables 4.17.1, 4.17.2, and 4.17.3, CHANNEL FUNCTIONAL TEST specified at 31 day intervals has been changed to 92 days in STS Sections 3.3.1, through 3.3.4, in accordance with "NRC evaluation of CEOG Topical Report CEN-327, RPS/ESFAS Extended Test Interval Evaluation," November 6, 1989.

Do these LESS RESTRICTIVE changes involve a significant increase in the probability or consequences of an accident previously evaluated?

Change 1:

Change 1 is LESS RESTRICTIVE in the sense that there is no required automatic CHP, CHR, or SGLP in MODE 4 in the RTS, whereas it is required in the TS. In the case of the SGLP function, it is permissible to manually bypass this function when below 550 psi, where it is not needed. This corresponds to a temperature above 300 °F, the distinction between MODE 3 and 4. In the case of CHP and CHR, manual actuation is still possible, since containment isolation valve operability is required by LCO 3.6.3, "Containment Isolation Valves."

Change 2:

Change 2 is LWSS RESTRICTIVE only in its allowance of a longer Allowed Outage Time (AOT) for inoperable equipment,or a longer surveillance testing interval. The proposed times are those stipulated in the STS. Changing an AOT or a surveillance interval, alone, does not alter any plant design, operating conditions, operating practices, equipment settings, or equipment capabilities. Since these items are unchanged, changing an AOT or a surveillance interval would not increase the probability of any accident previously evaluated.

During the evaluation of potential accidents, the safety analyses assume the occurrence of the most limiting single failure. Typically, this single failure is assumed to disable one of the two trains of the equipment installed to mitigate an accident. In accordance with this assumption, the Technical Specifications allow continued operation with required equipment inoperable for limited periods of time (AOTs) only if the assumed level of equipment remains operable. Extending an AOT does not change level of safety equipment required to be available, and does not allow that level to drop below the level assumed to be available in the safety analyses. Therefore, changing an AOT cannot increase the consequences of an accident previously evaluated.

Excessively extending a surveillance interval could affect the probability that a piece of equipment will function properly upon demand. An overly restrictive surveillance interval could also affect the ability of the equipment to mitigate an accident by imposing unnecessary testing wear, equipment manipulations, and system transients on the plant, and thereby affect the consequences of an accident. The existing surveillance intervals were based on the operating experience available when they were added to the TS. Typically, this was done during the initial plant licensing, circa 1970. In each of these changes where it is proposed that a surveillance interval be extended, the time proposed is that stipulated in the STS. The surveillance intervals stipulated in the STS are based on a much larger accumulation of operating experience and have been judged by the NRC and by the industry to be appropriate for typical situations. There are no special features of the Palisades plant which would invalidate those judgements for these changes. Therefore, operation of the facility in accordance with the requirements proposed by change 2 does not involve a significant increase in the probability of an accident previously evaluated.

Do these LESS RESTRICTIVE changes create the possibility of a new or different kind of accident from any previously evaluated?

Change 1:

Change 1 does not create the possibility of a new or different kind of accident from that previously evaluated because it only affects the conditions under which certain instrumentation, used to mitigate an accident, is required to be operable. Change 1 does not affect the design, operating conditions, or operating methods of any plant equipment whose failure could initiate an accident.

Change 2:

Change 2 is LESS RESTRICTIVE only in its allowance of a longer Allowed Outage Time (AOT) for inoperable equipment or a longer surveillance testing interval. The proposed times are those stipulated in the STS. Changing an AOT or surveillance interval, alone, cannot alter any plant operating conditions, operating practices, equipment settings, or equipment capabilities. Therefore, changing an AOT or a surveillance interval cannot create the possibility of a new or different kind of accident from any previously evaluated.

Do these LESS RESTRICTIVE changes involve a significant reduction in a margin of safety?

Change 1:

Change 1 does not involve a significant reduction in a margin of safety because the Anticipated Operation Occurrences, transients, or accidents which might require a SGLP, CHP, or CHR isolation function are unlikely to occur with the plant in MODE 4. In MODE 4 the steam pressure is less than 67 psia, and the SGLP signal, set at 500 psig, would be blocked. The low PCS and main steam pressures make occurrence of an event requiring containment isolation very improbable. Therefore, the proposed change in SGLP, CHP, and CHR operability would not result in a significant reduction in the margin of safety.

Change 2:

Change 2 is LESS RESTRICTIVE only in its allowance of an extension to an Allowed Outage Time (AOT) for inoperable equipment or to a surveillance testing interval. Extending an AOT or a surveillance interval, alone, cannot alter any plant operating conditions, operating practices, equipment settings, or equipment capabilities.

An excessive AOT extension could reduce the margin of safety by allowing operation for an excessive period with less capability to mitigate an accident, or with parameters outside those assumed in the safety analysis. An overly restrictive AOT could also reduce the margin of safety by imposing unnecessary transients on the plant for minor deviations from the requirements of the LCOs. Similarly, an excessive surveillance interval extension could reduce the margin of safety by reducing assurance that required equipment will function as designed, or that parameters are within the required limits. An overly restrictive surveillance interval could also reduce the margin of safety by imposing unnecessary testing wear, equipment manipulations, and system transients on the plant.

The existing AOTs and surveillance intervals were based on the operating experience available when they were added to the TS. Typically this was done during the initial plant licensing, circa 1970. In each of these changes where it is proposed that an AOT or surveillance interval be extended, the time proposed is that stipulated in the STS. The AOTs and surveillance intervals stipulated in the STS are based on a much larger accumulation of operating experience and have been judged by the NRC and by the industry to be appropriate for typical situations. There are no special features of the Palisades plant which would invalidate those judgements for these changes. Therefore, operation of the facility in accordance with the requirements proposed by Change 2 does not involve a significant reduction in a margin of safety.

IV. CONCLUSION

The Palisades Plant Review Committee has reviewed this part of the STS conversion Technical Specifications Change Request and has determined that proposing this change does not involve an unreviewed safety question. Further, the change involves no significant hazards consideration. This change has been reviewed by the Nuclear Performance Assessment Department.

ATTACHMENT 1

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-25

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

3.3 INSTRUMENTATION PART

Proposed Technical Specifications Pages

3.3 INSTRUMENTATION

3.3.1 Reactor Protective System (RPS) Instrumentation

LCO 3.3.1 Four RPS trip units and associated instrument and bypass removal channels for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5, when more than one Control Rod is capable of being withdrawn and PCS boron concentration is less than that required by the COLR for LCO 3.9.1.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each RPS trip or bypass removal Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one RPS trip unit or associated instrument channel inoperable, except for Condition C (Hi Startup Rate or Loss of Load).	A.1 Place affected trip unit in trip.	7 Days
B. One or more Functions with two RPS trip units or associated instrument channels inoperable except for Condition D (Hi Startup Rate or Loss of Load).	----------NOTE---------- LCO 3.0.4 is not applicable. ----------	
	B.1 Place one trip unit in trip.	1 hour
	AND	
	B.2 Restore one trip unit to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One Loss of Load or Hi Startup Rate trip unit or associated instrument channel inoperable.	----------NOTE---------- LCO 3.0.4 is not applicable. ----------	
	C.1 Restore trip unit to OPERABLE status.	Prior to entering MODE 2 following MODE 5 entry.
D. Two Loss of Load or two Hi Startup Rate trip units or associated instrument channels inoperable.	----------NOTE---------- LCO 3.0.4 is not applicable. ----------	
	D.1 Place one affected trip unit in trip.	1 hour
	AND	
	D.2 Restore one trip unit to OPERABLE status.	Prior to entering MODE 2 following MODE 5 entry.
E. One or more Functions with one or two automatic bypass removal channels inoperable.	E.1 Remove the bypass function.	1 hour
	OR	
	E.2 Declare the affected trip units inoperable, and enter the appropriate Condition.	1 hour

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. Required Action and associated Completion Time not met.	F.1	Be in MODE 3.	6 hours
	AND		
	F.2.1	Ensure that no more than one Control Rod is capable of being withdrawn.	6 hours
	OR		
	F.2.2	Ensure PCS boron concentration is at least that required by the COLR for LCO 3.9.1.	6 hours
G. Control room temperature > 90°F.	G.1	Enter LCO 3.0.3	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform a CHANNEL CHECK of each RPS instrument channel except Loss of Load and Hi Containment Pressure.	12 hours
SR 3.3.1.2	----------NOTES---------- Not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP. ---------- Perform calibration (heat balance only) and adjust the excore power range and ΔT power channels to agree with calorimetric calculation if the absolute difference is ≥ 2%.	24 hours
SR 3.3.1.3	Perform a CHANNEL FUNCTIONAL TEST of each RPS channel except Loss of Load and Hi Startup Rate.	92 days
SR 3.3.1.4	Calibrate the Excore Power Range Channels with a test signal.	31 days
SR 3.3.1.5	Verify constants in each Thermal Margin Monitor.	92 days
SR 3.3.1.6	Perform a CHANNEL FUNCTIONAL TEST of each Hi Startup Rate and each Loss of Load Functional Unit.	Once within 7 days prior to each reactor startup.
SR 3.3.1.7	Perform a CHANNEL CALIBRATION of each RPS instrument channel, including bypass removal functions.	18 months
SR 3.3.1.8	Verify control room temperature is ≤ 90°F.	12 hours

Table 3.3.1-1 (page 1 of 2)
Reactor Protective System Instrumentation

FUNCTION	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Variable High Power Trip	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.7	≤ 15% RTP above current THERMAL POWER with a minimum of ≤ 30% RTP and a maximum of ≤ 106.5% RTP
2. Hi Startup Rate[a]	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.7	N/A
3. Low PCS Flow	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.7	≥ 95%
4. Low SG-A Level	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.7	≥ 25.9%
5. Low SG-B Level	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.7	≥ 25.9%
6. Low SG-A Pressure	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.7	≥ 500 psia
7. Low SG-B Pressure	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.7	≥ 500 psia
8. High Pressurizer Pressure	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.7	≤ 2255 psia
9. Thermal Margin/Low Pressure (TM/LP)	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.5 SR 3.3.1.7	Setpoint equation on following page
10. Loss of Load (auto stop oil pressure)[b]	SR 3.3.1.6 SR 3.3.1.7	N/A
11. Containment High Pressure	SR 3.3.1.3 SR 3.3.1.7	≤ 3.70 psig

(a) Trip may be bypassed when THERMAL POWER is < 1 x 10^{-4}% RTP or > 13% RTP.
Bypass shall be automatically removed when THERMAL POWER is ≥ 1 x 10^{-4}% RTP and ≤ 13% RTP.

(b) Trip may be bypassed and is not required to be OPERABLE when THERMAL POWER is < 17% RTP.
Bypass shall be automatically removed when THERMAL POWER is ≥ 17% RTP.

N/A No specific allowable valve required.

Table 3.3.1-1 (page 2 of 2)
Reactor Protective System Instrumentation

The pressure setpoint for the Thermal Margin/Low Pressure Trip, Ptrip, is the higher of two values, Pmin and Pvar, both in psia:

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

where:

$QA = -0.720(ASI) + 1.028$; when $-0.628 \leq ASI < -0.100$

$QA = -0.333(ASI) + 1.067$; when $-0.100 \leq ASI < +0.200$

$QA = +0.375(ASI) + 0.925$; when $+0.200 \leq ASI < +0.565$

ASI = Measured ASI when $Q \geq 0.0625$
ASI = 0.0 when $Q < 0.0625$

$QR_1 = 0.412(Q) + 0.588$; when $Q \leq 1.0$
$QR_1 = Q$ when $Q > 1.0$

Q = (THERMAL POWER)/(RATED THERMAL POWER)
T_c = Maximum primary coolant inlet temperature, °F.

ASI, T_c, and Q are the existing values as measured by the associated instrument channel.

3.3 INSTRUMENTATION

3.3.2 Reactor Protective System (RPS) Logic and Trip Initiation

LCO 3.3.2 Six channels of RPS Matrix Logic, four channels of RPS Initiation Logic, and two channels of Manual Trip shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5, when more than one CONTROL ROD is capable of being withdrawn and PCS boron concentration is less than that required by LCO 3.9.1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE------- This action also applies when three Matrix Logic channels are inoperable due to a common power source failure de-energizing three matrix power supplies. ---------- One Matrix Logic channel inoperable.	A.1 Restore channel to OPERABLE status.	48 hours
B. One channel of Initiation Logic inoperable.	B.1 De-energize affected clutch power supplies.	1 hour
C. One channel of Manual Trip inoperable.	C.1 Restore channel to OPERABLE status.	Prior to entering MODE 2 following MODE 3 entry.

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Times not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2.1	Ensure that no more than one Control Rod is capable of being withdrawn.	6 hours
	OR		
	D.2.2	Ensure PCR boron concentration is at least that required by the COLR for LCO 3.9.1	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform a CHANNEL FUNCTIONAL TEST on each RPS Logic channel.	92 days
SR 3.3.2.2	Perform a CHANNEL FUNCTIONAL TEST on each RPS Manual Trip channel.	Once within 7 days prior to each reactor startup.

3.3 INSTRUMENTATION

3.3.3 Engineered Safety Features (ESF) Instrumentation

LCO 3.3.3 Four ESF bistables and associated instrument and bypass removal channels for each Function in Table 3.3.3-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTIONS:

----------NOTE----------
Separate Condition entry is allowed for each ESF bistable trip or bypass removal Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SIRWT Level switch inoperable.	----------NOTE---------- LCO 3.0.4 is not applicable. ----------	
	A.1 Bypass the SIRWT Level Switch.	8 hours
	AND	
	A.2 Restore channel to OPERABLE status.	7 days
B. One or more functions with one ESF bistable or associated instrument channel inoperable, except SIRWT level.	B.1 Place affected bistable in trip.	7 days

(continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more Functions with two ESF bistables or associated instrument channels inoperable, except SIRWT level.	----------NOTE---------- LCO 3.0.4 is not applicable. ----------		
	C.1	Place one bistable in trip.	8 hours
	AND		
	C.2	Restore one bistable to OPERABLE status.	7 days
D. One or more Functions with one or two automatic bypass removal channels inoperable.	D.1	Remove the bypass.	8 hours
	OR		
	D.2	Declare the affected Logic Channel inoperable and enter the appropriate Condition.	8 hours
E. Required Action and associated Completion Time not met.	E.1	Be in MODE 3.	6 hours
	AND		
	E.2	Be in MODE 4.	30 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Perform a CHANNEL CHECK of each ESF instrument channel except RAS SIRWT Level Switches and CHP Containment Pressure Switches.	12 hours
SR 3.3.3.2	Perform a CHANNEL FUNCTIONAL TEST of each ESF instrument channel except SIRWT Level Switches.	92 days
SR 3.3.3.3	Perform a CHANNEL CALIBRATION of each ESF instrument channel, including bypass removal functions.	18 months

Table 3.3.3-1
Engineered Safety Features Instrumentation

FUNCTION	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Safety Injection Signal (SIS)		
a. Pressurizer Pressure - Low[a]	SR 3.3.3.1 SR 3.3.3.2 SR 3.3.3.3	≥ 1593 psia
2. Containment High Pressure Signal(CHP)		
a. Containment Pressure - Hi, left train	SR 3.3.3.2 SR 3.3.3.3	≥ 3.7 and ≤ 4.4 psig
b. Containment Pressure - Hi, right train	SR 3.3.3.2 SR 3.3.3.3	≥ 3.7 and ≤ 4.4 psig
3. Containment High Radiation (CHR)		
a. Containment Area Radiation - Hi	SR 3.3.3.1 SR 3.3.3.2 SR 3.3.3.3	≤ 20 R/hr
4. Steam Generator Low Pressure (SGLP)		
a. "A" Steam Generator Pressure - Low [b]	SR 3.3.3.1 SR 3.3.3.2 SR 3.3.3.3	≥ 500 psia
b. "B" Steam Generator Pressure - Low [b]	SR 3.3.3.1 SR 3.3.3.2 SR 3.3.3.3	≥ 500 psia
5. Aux Feedwater Actuation Signal (AFAS)		
a. "A" Steam Generator Level - Low	SR 3.3.3.1 SR 3.3.3.2 SR 3.3.3.3	≥ 25.9 %
b. "B" Steam Generator Level - Low	SR 3.3.3.1 SR 3.3.3.2 SR 3.3.3.3	≥ 25.9 %
6. Recirculation Actuation Signal (RAS)		
a. SIRWT Level Switches - Low	SR 3.3.3.3	≥ 21 and ≤ 27 inches above SIRWT floor

(a) Pressurizer Pressure - Low may be manually bypassed when pressurizer pressure is ≤ 1700 psia. The bypass shall be automatically removed whenever pressurizer pressure is > 1700 psia.

(b) Steam Generator Pressure - Low may be manually bypassed when steam generator pressure is < 550 psia. The bypass shall be automatically removed whenever steam generator pressure is ≥ 550 psia.

3.3 INSTRUMENTATION

3.3.4 Engineered Safety Features (ESF) Logic and Manual Initiation

LCO 3.3.4 Two ESF Manual Initiation and two ESF Actuation Logic channels shall be OPERABLE for each ESF Function specified in Table 3.3.4-1.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one Manual Initiation or Actuation Logic channel inoperable.	A.1 Restore channel to OPERABLE status.	48 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 30 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Perform a CHANNEL FUNCTIONAL TEST on the AFAS Logic channels.	92 days
SR 3.3.4.2	Perform a CHANNEL FUNCTIONAL TEST on the SIS Logic normal and emergency power functions using simulated signal.	92 days
SR 3.3.4.3	Perform a CHANNEL FUNCTIONAL TEST on each ESF Manual Initiation channel.	18 months
SR 3.3.4.4	Perform a CHANNEL FUNCTIONAL TEST on each ESF logic channel to verify all automatic actuation and automatic resetting of Low Pressure Bypasses.	18 months

Table 3.3.4-1
Engineered Safety Features Logic and Manual Channel

FUNCTION	SURVEILLANCE REQUIREMENTS
1. Safety Injection Signal (SIS)	
a. Manual Initiation	SR 3.3.4.3
b. SIS Logic Trains (Initiation, Actuation, and Low Pressure Bypass auto reset)	SR 3.3.4.2 SR 3.3.4.4
c. CHP Signal SIS Initiation (5P Relay Output)	SR 3.3.4.4
2. Containment High Pressure Signal (CHP)	
a. Manual Initiation[a]	SR 3.3.4.3
b. CHP Logic Trains	SR 3.3.4.4
3. Containment Hi Radiation (CHR)	
a. Manual Initiation	SR 3.3.4.3
b. CHR Logic Trains	SR 3.3.4.4
4. Steam Generator Low Pressure (SGLP)	
a. Manual Initiation[a]	SR 3.3.4.3
b. SGLP Logic Trains	SR 3.3.4.4
5. Aux Feedwater Actuation Signal (AFAS)	
a. Manual Initiation[a]	SR 3.3.4.3
b. AFAS Logic Trains	SR 3.3.4.1
6. Recirculation Actuation Signal (RAS)	
a. Manual Initiation[a]	SR 3.3.4.3
b. RAS Logic Trains	SR 3.3.4.4

(a) Manual Initiation may be achieved by individual component controls.

3.3 INSTRUMENTATION

3.3.5 Diesel Generator (DG) - Undervoltage Start (UV START)

LCO 3.3.5 Three Loss of Voltage and Three Degraded Voltage sensors and associated auto start logic channels shall be OPERABLE for each DG.

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more sensors or logic channels per DG inoperable.	A.1 Declare the affected DG Inoperable and enter the appropriate Condition.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	Perform CHANNEL CALIBRATION with setpoint Allowable Values as follows: a. Loss of Voltage Function ≥ 1860 V dropout; and time delays of: ≤ 8.15 seconds at 1400 Volts; and ≤ 4.2 seconds at 930 Volts. b. Degraded Voltage Function ≥ 2184 Volts and time delay of: ≤ 0.8 seconds.	18 months

3.3 INSTRUMENTATION

3.3.6 Refueling Containment High Radiation (CHR) Initiation

LCO 3.3.6 Two Refueling CHR channels shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or Two Refueling CHR channels inoperable.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform a CHANNEL CHECK on each refueling CHR channel.	24 hours
SR 3.3.6.2	Perform a CHANNEL FUNCTIONAL TEST on each refueling CHR channel. Verify Containment refueling Radiation Monitor Channel - high radiation setpoint is ≤ 15 mRem/hr above background.	31 days
SR 3.3.6.3	Perform a CHANNEL CALIBRATION on each containment refueling radiation monitor channel.	18 months
SR 3.3.6.4	Perform a CHANNEL FUNCTIONAL TEST on each manual CHR action channel.	18 months

3.3 INSTRUMENTATION

3.3.7 Accident Monitoring Instrumentation

LCO 3.3.7 The accident monitoring instrumentation for each Function in Table 3.3.7-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTES----------

1. LCO 3.0.4 is not applicable.

2. Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1	Restore required channel to OPERABLE status.	7 days
B. One or more Functions with two required channels inoperable.	B.1	Restore one channel to OPERABLE status.	48 hours
C. Required Action and associated Completion Time of Condition A not met for Functions 1 through 15. OR Required Action and associated Completion Time of Condition B not met for Functions 1 through 19.	C.1 AND C.2	Be in MODE 3. Be in MODE 4.	6 hours 30 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A not met for Functions 16 through 21.	D.1 Initiate action in accordance with Specification 5.6.7 (Report to NRC).	Immediately
	AND	
OR		
Required Action and associated Completion Time of Condition B not met for Functions 20 or 21.	D.2 Restore channel to OPERABLE status.	Prior to entering MODE 2 following the next MODE 6 entry.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform CHANNEL CHECK for each required instrumentation channel.	31 days
SR 3.3.7.2	----------NOTE---------- Calibrate Core Exit Thermocouple circuitry by substituting a known voltage for thermocouple voltage. ----------	
	Perform CHANNEL CALIBRATION on each required instrumentation channel.	18 months

Table 3.3.7-1
Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS
1. PCS Wide Range Hot Leg Temperature	2
2. PCS Wide Range Cold Leg Temperature	2
3. Wide Range Flux	2
4. Containment Floor Water Level	2
5. Subcooled Margin Monitor	2
6. Wide Range Pressurizer Level	2
7. Containment H_2 Concentration	2
8. Condensate Storage Tank Level	2
9. Wide Range Pressurizer Pressure	2
10. Wide Range Containment Pressure	2
11. Wide Range "A" SG Level	2
12. Wide Range "B" SG Level	2
13. Narrow Range "A" SG Pressure	2
14. Narrow Range "B" SG Pressure	2
15. Position Indication for each Containment Isolation Valve	1/valve[a]
16. Core Exit Thermocouples - Quadrant 1	4
17. Core Exit Thermocouples - Quadrant 2	4
18. Core Exit Thermocouples - Quadrant 3	4
19. Core Exit Thermocouples - Quadrant 4	4
20. Reactor Vessel Level Monitoring Sys.	2
21. High Range Containment Radiation	2

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve, with flow through the valve secured.

3.3 INSTRUMENTATION

3.3.8 Alternate Shutdown System

LCO 3.3.8 The Alternate Shutdown System Functions in Table 3.3.8-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTES----------

1. LCOs 3.0.3 and 3.0.4 are not applicable.

2. Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required Channels inoperable.	A.1	Provide equivalent shutdown capability.	7 days
	AND		
	A.2	Restore Channel to OPERABLE status.	60 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 4.	30 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.8.1	Perform CHANNEL CHECK for each required instrumentation Function 2 through 12 in Table 3.3.8-1.	92 days
SR 3.3.8.2	Perform a CHANNEL CHECK of Function 1 in Table 3.3.8-1 (Neutron Flux).	Once within 7 days prior to each Reactor Startup
SR 3.3.8.3	Perform a CHANNEL CHECK of Functions 13 and 14 in Table 3.3.8-1. (P-8B flow to SGs)	18 months
SR 3.3.8.4	Perform a CHANNEL FUNCTIONAL TEST on Functions 13 through 18 in Table 3.3.8-1.	18 months
SR 3.3.8.5	Perform CHANNEL CALIBRATION for each required instrumentation channel of Functions 1 through 15 in Table 3.3.8-1.	18 months

Table 3.3.8-1
Alternate Shutdown System Instrumentation and Controls

FUNCTION/INSTRUMENT OR CONTROL PARAMETER	SURVEILLANCE REQUIREMENTS	REQUIRED CHANNELS
1. Neutron Flux	SR 3.3.8.2 SR 3.3.8.5	1
2. Pressurizer Pressure	SR 3.3.8.1 SR 3.3.8.5	1
3. Pressurizer Level	SR 3.3.8.1 SR 3.3.8.5	1
4. PCS #1 Hot Leg Temperature	SR 3.3.8.1 SR 3.3.8.5	1
5. PCS #2 Hot Leg Temperature	SR 3.3.8.1 SR 3.3.8.5	1
6. PCS #1 Cold Leg Temperature	SR 3.3.8.1 SR 3.3.8.5	1
7. PCS #2 Cold Leg Temperature	SR 3.3.8.1 SR 3.3.8.5	1
8. "A" SG Pressure	SR 3.3.8.1 SR 3.3.8.5	1
9. "B" SG Pressure	SR 3.3.8.1 SR 3.3.8.5	1
10. "A" SG Level	SR 3.3.8.1 SR 3.3.8.5	1
11. "B" SG Level	SR 3.3.8.1 SR 3.3.8.5	1
12. SIRW Tank Level	SR 3.3.8.1 SR 3.3.8.5	1
13. AFW Pump P-8B Flow to "A" SG	SR 3.3.8.3 SR 3.3.8.4 SR 3.3.8.5	1
14. AFW Pump P-8B Flow to "B" SG	SR 3.3.8.3 SR 3.3.8.4 SR 3.3.8.5	1
15. AFW Pump P-8B Suction Pressure Alarm	SR 3.3.8.4 SR 3.3.8.5	1
16. AFW Pump P-8B Steam Valve Control	SR 3.3.8.4	1
17. AFW Flow Control "A" SG	SR 3.3.8.4	1
18. AFW Flow Control "B" SG	SR 3.3.8.4	1

3.3 INSTRUMENTATION

3.3.9 Neutron Flux Monitoring Channels

LCO 3.3.9 Two channels of Neutron Flux monitoring instrumentation shall be OPERABLE.

APPLICABILITY: MODES 3, 4, and 5 with no more than one Control Rod capable of withdrawal.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more channels inoperable.	A.1	Suspend all operations involving positive reactivity additions.	Immediately
	AND		
	A.2	Perform SDM verification in accordance with SR 3.1.1.1, if T_{avg} > 525°F, or SR 3.1.2.1, if $T_{avg} \leq$ 525°F.	4 hours AND Once per 12 hours thereafter

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.9.1	Perform CHANNEL CHECK on each required neutron flux channel.	12 hours
SR 3.3.9.2	Perform CHANNEL CALIBRATION on each required neutron flux channel.	18 months

3.3 INSTRUMENTATION

3.3.10 Spent Fuel Pool Radiation Monitor

LCO 3.3.10 Two Spent Fuel Pool (SFP) Radiation Monitors shall be OPERABLE.

APPLICABILITY: Whenever fuel is in the SFP area.

----------NOTE----------
LCOs 3.0.3 and 3.0.4 are not Applicable

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or Two Spent Fuel Pool Radiation Monitors inoperable.	A.1 Suspend fuel movement in the SFP area.	Immediately
	AND	
	A.2.1 Restore Monitors to Operable Status.	72 hours
	OR	
	A.2.2 Provide equivalent monitoring capability.	72 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.10.1	Perform a CHANNEL CHECK on each SFP Radiation Monitoring channel.	24 hours
SR 3.3.10.2	Perform a CHANNEL FUNCTIONAL TEST on each SFP Radiation Monitoring channel.	31 days
SR 3.3.10.3	Perform a CHANNEL CALIBRATION on each SFP Radiation Monitoring channel.	18 months

ATTACHMENT 2

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

STS CONVERSION TECHNICAL SPECIFICATION CHANGE REQUEST

3.3 INSTRUMENTATION PART

Bases for the Revised Technical Specifications

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Protective System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a reactor trip to protect against violating the core specified acceptable fuel design limits and breaching the reactor coolant pressure boundary during Anticipated Operational Occurrences (AOOs). By tripping the reactor, the RPS also assists the Engineered Safety Features systems in mitigating accidents.

The protection and monitoring systems have been designed to ensure safe operation of the reactor. This is achieved by specifying Limiting Safety System Settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this Specification as the Allowable Value, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).

During AOOs, which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling;
- Fuel centerline melting shall not occur; and
- The Primary Coolant System (PCS) pressure SL of 2750 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 50 (Ref. 1) and 10 CFR 100 (Ref. 2) criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 (Ref. 2) limits. Different accident categories allow a different fraction of these limits based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

BACKGROUND (continued)

The RPS is segmented into three interconnected modules. These modules are:

- Measurement channels;
- Bistable trip units; and
- RPS Logic

This LCO addresses measurement channels and bistable trip units. It also addresses the automatic bypass removal feature for those trips with operating bypasses. The RPS Logic is addressed in LCO 3.3.2, "Reactor Protective System (RPS) Logic and Trip Initiation."

The role of each of these modules in the RPS, including those associated with the logic, is discussed below.

Measurement Channels

Measurement channels, consisting of field transmitters or process sensors and associated instrumentation, provide a measurable electronic signal based upon the physical characteristics of the parameter being measured.

The excore nuclear instrumentation (wide range and power range) and the thermal margin monitor are considered components in the measurement channels.

The wide range Nuclear Instruments (NIs) provide a Hi Startup Rate Trip. There are only two wide range NI channels. The wide range channel signal processing electronics is physically mounted in RPS cabinet channels C (NI-003) and D (NI-004). Separate bistables mounted within the Channel C wide range channel drawer supply High Startup Rate trip signals to RPS channels A and C. Separate bistables mounted within the Channel D wide range channel drawer provide High Startup Rate trip signals to RPS channels B and D.

Two RPS trips use a power level designated as Q power as an input. Q power is the higher of NI power from the power range NI drawer and primary calorimetric power (ΔT power) based on PCS hot leg and cold leg temperatures. Trips using Q power as an input include the Variable High Power Trip (VHPT) and the Thermal Margin/Low Pressure (TM/LP) trips, both of which employ the thermal margin monitor for trip generation.

The thermal margin monitor provides the complex signal processing necessary to calculate the TM/LP trip setpoint, VHPT trip setpoint and trip comparison, and Q power calculation.

BACKGROUND (continued)

The excore power range NIs (NI 005 through NI 008) and the thermal margin monitors are mounted in the RPS cabinet, with one channel of each in each of the four RPS bays.

With the exception of the wide range NIs, which employ two channels, and Loss of Load, which employs a single pressure sensor, four identical measurement channels with electrical and physical separation are provided for each parameter used in the direct generation of trip signals. These are designated channels A through D. Measurement channels provide input to one or more RPS bistables within the same RPS channel. In addition, some measurement channels are used as inputs to Engineered Safety Features System (ESF) bistables, and most provide indication in the control room.

When a channel monitoring a parameter exceeds a predetermined setpoint, indicating an unsafe condition, the bistable monitoring the parameter in that channel will trip. Tripping two or more channels of bistables monitoring the same parameter de-energizes Matrix Logic, which in turn de-energizes the Initiation Logic. This causes all four DC clutch power supplies to de-energize, interrupting power to the control rod drive mechanism DC clutches, allowing the full length control rods to insert into the core.

Three of the four measurement and bistable channels are necessary to meet the redundancy and testability of GDC 21 in 10 CFR 50, Appendix A (Ref. 1). The fourth channel provides additional flexibility by allowing one channel to be removed from service (trip channel bypass) for maintenance or testing while still maintaining a minimum 2-out-of-3 logic. Thus, even with a channel inoperable, no single additional failure in the RPS can either cause an inadvertent trip or prevent a required trip from occurring.

Since no single failure will either cause or prevent a protective system actuation, this arrangement meets the requirements of IEEE 279-1971 (Ref. 3).

In the case of wide range power and loss of load, where fewer than four sensor channels are employed, the reactor trips provided are not required by the plant Safety Analysis. As such, they need not meet the above criteria. In these cases, however, the sensor channels provide trip input signals to all four RPS channels.

BACKGROUND (continued)

Most of the RPS trips are generated by comparing a single measurement to a fixed bistable setpoint. Certain Functions, however, make use of more than one measurement to provide a trip. The following trips use multiple measurement channel inputs:

- Variable High Power Trip (VHPT)

  The VHPT uses Q power as its only input. Q power is the higher of NI power and ΔT power. It has a trip setpoint that tracks power levels downward so that it is always within a fixed increment above current power, subject to a minimum value.

  On power increases, the trip setpoint remains fixed unless manually reset, at which point it increases to the new setpoint, a fixed increment above Q power at the time of reset, subject to a maximum value. Thus, during power escalation, the trip setpoint must be repeatedly reset to avoid a reactor trip.

- Thermal Margin/Low Pressure (TM/LP)

  Q power is only one of several inputs to the TM/LP trip. Other inputs include internal ASI and cold leg temperature based on the higher of two cold leg resistance temperature detectors. The TM/LP trip setpoint is a complex function of these inputs and represents a minimum acceptable PCS pressure to be compared to actual PCS pressure in the TM/LP trip unit.

Bistable Trip Units

Bistable trip units, mounted in the RPS cabinet, receive an analog input from the measurement channels, compare the analog input to trip setpoints, and provide contact output to the Matrix Logic. They also provide local trip indication and remote annunciation.

There are four channels of bistable trip units, designated A through D, for each RPS Function, one for each measurement channel. Bistable output relays de-energize when a trip occurs.

The contacts from these bistable relays are arranged into six coincidence matrices, comprising the Matrix Logic. If bistables monitoring the same parameter in at least two channels trip, the Matrix Logic will generate a reactor trip (2-out-of-4 logic).

BACKGROUND (continued)

Four of the RPS trip Function measurement channels provide contact outputs to the RPS, so the comparison of an analog input to a trip setpoint is not necessary. In these cases, the bistable trip unit is replaced with an auxiliary trip unit. The auxiliary trip units provide contact multiplication so the single input contact opening can provide multiple contact outputs to the coincidence logic as well as trip indication and annunciation.

Trips employing auxiliary trip units include the Variable High Power Trip, which receives contact inputs from the thermal margin monitors; the High Startup Rate trip which employs contact inputs from bistables mounted in the two wide range drawers; the Loss of Load trip which receives contact inputs from one of two auxiliary relays which are operated by a single relay sensing turbine EHC auto stop oil pressure; and the Containment High Pressure (CHP) trip, which employs CHP pressure switch contacts.

All RPS trips, with the exception of the Loss of Load and CHP trip, generate a pretrip alarm as the trip setpoint is approached.

The trip setpoints used in the bistable trip units are based on the analytical limits stated in Reference 4. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, and instrument drift, Allowable Values specified in Table 3.3.1-1, in the accompanying LCO, are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in the CPCo EGAD "Setpoint Methodology" (Ref. 6). The nominal trip setpoint entered into the bistable is normally still more conservative than that specified by the Allowable Value, to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the interval between surveillances. A channel is inoperable if its actual setpoint is not within its required Allowable Value.

Setpoints in accordance with the Allowable Value will ensure that SLs of Chapter 2.0 are not violated during AOOs and the consequences of DBAs will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed.

Note that in the accompanying LCO 3.3.1, the Allowable Values of Table 3.3.1-1 are the LSSS.

BACKGROUND (continued)

RPS Logic

The RPS Logic, addressed in LCO 3.3.2, consists of both Matrix and Initiation Logic and employs a scheme that provides a reactor trip when bistables in any 2-out-of-4 channels sense the same input parameter trip. This is called a 2-out-of-4 trip logic. This logic and the clutch power supply configuration are shown in FSAR Figure 7-1 (Ref. 9).

Bistable relay contact outputs from the four channels are configured into six logic matrices. Each logic matrix checks for a coincident trip in the same parameter in two bistable channels. The matrices are designated the AB, AC, AD, BC, BD, and CD matrices to reflect the bistable channels being monitored. Each logic matrix contains four normally energized matrix relays. When a coincidence is detected, consisting of a trip in the same Function in the two channels being monitored by the logic matrix, all four matrix relays de-energize.

The matrix relay contacts are arranged into trip paths, with one of the four matrix relays in each matrix opening a contact in one of the four trip paths.

The trip paths thus have six contacts in series, one from each matrix, and perform a logical OR function, de-energizing the M contactors if any one or more of the six logic matrices indicate a coincidence condition.

De-energizing the M contactors removes AC power to the four clutch power supply inputs. Contacts from M Contactors M1 and M2 are in series with each other and in the AC power supply path to clutch power supplies PS1 and PS3. M3 and M4 are similarly arranged with respect to clutch power supplies PS2 and PS4. Approximately half of the control rods receive clutch power from auctioneered clutch power supplies 1 and 2. The remaining control rods receive clutch power from auctioneered clutch power supplies 3 and 4.

Manual reactor trip capability is afforded by two main control board-mounted pushbuttons. One of these (CO-1) opens contacts in series with each of the four trip paths, de-energizing all M contactors. The other pushbutton (CO-2) opens circuit breakers which provide AC input power to the M contractor contacts and downstream clutch power supplies. Thus depressing either pushbutton will cause a reactor trip by diverse means.

When a coincidence occurs in two RPS channels, all four matrix relays in the affected matrix de-energize. This in turn de-energizes all four M contactors, which interrupt AC input power to the four clutch power supplies, allowing the control rods to insert by gravity.

BASES

BACKGROUND (continued)

Matrix Logic refers to the matrix power supplies, trip channel bypass contacts, and interconnecting matrix wiring between bistable and auxiliary trip units, up to but not including the matrix relays. Contacts in the bistable and auxiliary trip units are excluded from the Matrix Logic definition, since they are addressed as part of the measurement channel.

The Initiation Logic consists of the matrix relays and their associated contacts, all interconnecting wiring, CO-1 manual trip contacts, and M contactors.

Neither the clutch power supplies nor the AC input power source to these supplies is considered as Safety Related, and are not subject to Technical Specifications, other than as addressed by the RPS Logic and Trip Initiation Specification, LCO 3.3.2. Operation may continue with one or two selective clutch power supplies de-energized.

It is possible to change the 2-out-of-4 RPS Logic to a 2-out-of-3 logic for a given input parameter in one channel at a time by trip channel bypassing select portions of the Matrix Logic. Trip channel bypassing a bistable effectively shorts the bistable relay contacts in the three matrices associated with that channel. Thus, the bistables will function normally, producing normal trip indication and annunciation, but a reactor trip will not occur unless two additional channels indicate a trip condition. Trip channel bypassing can be simultaneously performed on any number of parameters in any number of channels, providing each parameter is bypassed in only one channel at a time. A bypass key interlock prevents simultaneous trip channel bypassing of the same parameter in more than one channel. Trip channel bypassing is normally employed during maintenance or testing.

In addition to the trip channel bypasses, there are also operating bypasses on select RPS trips. Some of these bypasses are enabled manually, others automatically, in all four RPS channels when plant conditions do not warrant the specific trip protection. All operating bypasses are automatically removed when enabling bypass conditions are no longer satisfied. Trips with operating bypasses include the High Startup Rate, Low PCS flow, Low SG Pressure, TM/LP and Loss of Load. The Loss of Load trip and High Startup Rate trip operating bypasses are automatically enabled and disabled.

Several instrument channels provide more than one required function. Table B 3.3.1-1 provides a listing of these channels and the specifications which they affect.

Table B 3.3.1-1
Instruments Affecting Multiple Specifications

Required Instrument channels	Affected Specifications
Source Range NI 01/03 & 02/04 Count Rate Signal	3.9.2
Source Range NI 01/03 & 02/04 Count Rate Signal	3.3.9
Source Range NI-01/03 & 02/04 Count Rate Indication @ C-150	3.3.8 #1
Wide Range NI-01/03 & 02/04 Flux level 10^4 interlock	3.3.1 #2, 3, 6, 7, & 9
Wide Range NI-01/03 & 02/04 Start-up Rate	3.3.1 #2
Wide Range NI-01/03 & 02/04 Flux Level Indication	3.3.9
Wide Range NI-01/03 & 02/04 Flux Level Indication	3.3.7 #3
Power Range NI-05 - 08, Power level signal	3.1.7
Power Range NI-05 - 08, Power level signal	3.2.5
Power Range NI-05 - 08, Q-power	3.3.1 #1 & 9
Power Range NI-05 - 08, ASI	3.3.1 #1 & 9
Power Range NI-05 - 08, ASI	3.2.5
Power Range NI-05 - 08, ASI	3.2.4
Power Range NI-05 & 06; 15% interlock	3.3.1 #2 & 10
Power Range NI-05 & 06; 15% interlock	3.2.5
PCS TC, Temperature signal	3.3.1 #9
PCS TC, Temperature indication	3.3.7 #6 & 7
PCS TC, Q-power	3.3.1 #1 & 9
PCS TH, Temperature indication	3.3.7 #1
PCS TH, Q-power	3.3.1 #1 & 9
Pressurizer Pressure PI-0102 A, B, C, D Pressure signal	3.3.1 #8 & 9
Pressurizer Pressure PI-0102 A, B, C, D Pressure signal	3.3.3 #1.a
Pressurizer Pressure PI-0110, Pressure indication	3.3.6 #2
Steam Generator Level LI-0751 & 0752 A, B, C, D Level Signal	3.3.1 #4 & 5
Steam Generator Level LI-0751 & 0752 A, B, C, D Level Signal	3.3.3 #5 a & b
Steam Generator Level LI-0751A, 0752A Level indication	3.3.7 #10 & 11
Steam Generator Pressure PI 0751 & 0752 A, B, C, D Pressure Signal	3.3.1 #6 & 7
Steam Generator Pressure PI 0751 & 0752 A, B, C, D Pressure Signal	3.3.3 #4 a & b
Steam Generator Pressure PI 0751 & 0752 A, B, C, D Pressure Signal	3.3.6 #13 & 14
Steam Generator Pressure (WR) PI-0757 & 0758 A, B Pressure Signal	3.3.7 #8 & 9
Containment Pressure PS-1801, 2, 3, & 4, switch output	3.3.1 #11
Containment Pressure PS-1801, 2A, 3, & 4A, switch output	3.3.3 #2.a
Containment Pressure PS-1801A, 2, 3A, & 4, switch output	3.3.3 #2.b

APPLICABLE SAFETY ANALYSES

Each of the analyzed accidents and transients can be detected by one or more RPS Functions. The accident analysis contained in Reference 4 takes credit for most RPS trip Functions. The High Startup Rate and Loss of Load Functions, which are not specifically credited in the accident analysis are part of the NRC approved licensing basis for the plant. The High Startup Rate and Loss of Load trip are purely equipment protective, and their use minimizes the potential for equipment damage.

The specific safety analyses applicable to each protective Function are identified below:

1. Variable High Power Trip (VHPT)

   The VHPT provides reactor core protection against positive reactivity excursions. The Safety Analysis assumes that this trip is OPERABLE to terminate excessive positive reactivity insertions during power operation and while shutdown.

2. Hi Startup Rate

   The High Startup Rate trip is used to trip the reactor when excore wide range power indicates an excessive rate of change. The High Startup Rate trip minimizes transients for events such as a continuous control rod withdrawal or a boron dilution event from low power levels. The trip may be bypassed when THERMAL POWER is $< 1 \times 10^{-4}$% RTP, when poor counting statistics may lead to erroneous indication. It is also bypassed at > 13% RTP, where moderator temperature coefficient and fuel temperature coefficient make high rate of change of power unlikely. In MODES 3, 4, and 5 when no more than one control rod is capable of being withdrawn, the High Startup Rate trip is not required to be OPERABLE; however, the indication and alarm functions of both channels is required by LCO 3.3.8, "Neutron Flux Monitoring Channels," to be OPERABLE. LCO 3.3.8 ensures the wide range channels are available to detect and alert the operator to a boron dilution event.

   There are only two wide range drawers, with each supplying contact input to auxiliary trip units in two RPS channels.

3. Low PCS Flow

   The Low PCS Flow trip provides protection during events which suddenly reduce the PCS flow rate during power operation, such as loss of power to, or seizure of, a Primary Coolant Pump.

APPLICABLE SAFETY ANALYSES (continued)

4,5. Low Steam Generator A and B Level

The low steam generator level trips are provided to trip the reactor in the event of excessive steam demand and loss of feedwater events.

6,7. Low SG Pressure

The Low Steam Generator Pressure trips provide protection against excessive rates of heat extraction from the steam generators which result in a rapid uncontrolled cooldown of the PCS. These trips are needed to shutdown the reactor and assist the ESF System in the event of a steam or feedwater line break.

8. High Pressurizer Pressure

The High Pressurizer Pressure trip, in conjunction with pressurizer safety valves and main steam safety valves, provides protection against over pressure conditions in the Primary Coolant System (PCS) when at operating temperature. The safety analyses assume the High Pressurizer Pressure trip is OPERABLE during accidents and transients which suddenly reduce PCS cooling (Loss of Load, MSIV closure, etc) or which suddenly increase reactor power (Rod Ejection).

9. Thermal Margin/Low Pressure (TM/LP)

The TM/LP trip is provided to prevent reactor operation when the Departure from Nucleate Boiling Ratio (DNBR) is insufficient. The TM/LP trip protects against slow reactivity or temperature increases, and against pressure decreases.

The trip is initiated whenever the PCS pressure signal drops below a minimum value (P_{min}) or a computed value (P_{var}) as described below, whichever is higher.

The TM/LP trip uses Q Power, ASI, and Tc as inputs.

Q Power, is the higher of core thermal power (ΔT Power) or nuclear power. ΔT power uses hot leg and cold leg RTDs as inputs. Nuclear power uses the power range nuclear instruments as inputs. Both the ΔT and Excore Power signals have provisions for calibration by calorimetric calculations.

APPLICABLE SAFETY ANALYSES (continued)

ASI, AXIAL SHAPE INDEX, is calculated from the upper and lower power range excore detector signals, as explained in the definition section. The signal is corrected for the difference between the flux at the core periphery and the flux at the detectors.

Tc, cold leg temperature, is the higher of the two cold leg signals.

The TM/LP trip setpoint is a complex function of these inputs and represents a minimum acceptable PCS Pressure for the existing temperature and power conditions. It is compared to actual PCS Pressure in the TM/LP Trip Unit.

10. Loss of Load

The Loss of Load trip is provided to prevent lifting the pressurizer and main steam safety valves in the event of a turbine generator trip while at power. The trip is equipment protective. The safety analyses do not assume that this trip functions during any accident or transient. The Loss of Load trip uses a single pressure switch in the turbine Auto Stop Oil circuit to sense a turbine trip for input to all four RPS auxiliary trip units.

11. Containment High Pressure

The Containment High Pressure trip provides a backup reactor trip in the event of a Loss of Coolant Accident, Main Steam Line Break, or Main Feedwater Line Break. The High Containment Pressure trip shares sensors with the Containment High Pressure sensing logic for Safety Injection, Containment Isolation, and Containment Spray. Each of these sensors has a single bellows which actuates two micro switches. One micro switch on each of four sensors provides an input to the RPS.

Interlocks/Bypasses

The bypasses and their Allowable Values are addressed in footnotes to Table 3.3.1-1.

The RPS operating bypasses are:

a. High Startup Rate bypass. The High Startup Rate trip is automatically bypassed at $< 1 \times 10^{-4}$% RTP, as sensed by wide range NI bistables, and at > 13% RTP by the power range NI Level 1 bistable.

APPLICABLE SAFETY ANALYSES (continued)

b. Loss of Load bypass removal. The Loss of Load trip is automatically bypassed when at < 17% RTP as sensed by the power range NI Level 1 bistable. The bypass is automatically removed by this bistable above the setpoint. This same bistable is used to bypass the High Startup Rate trip. The difference in specified setpoint is to allow for hysteresis.

One other bypass is provided in the RPS design, but it cannot be used when the RPS is required to be operable. The Zero Power Mode Bypass (ZPMB) is manually actuated. Manual actuation is enabled when both wide range NI channels are below 10^{-4}%, and the bypass is automatically removed when either channel is above that set point. The ZPMB disables the TM/LP, Low SG Pressure, and PCS low flow trips. This bypass allows control rod testing when PCS pressure, flow, or temperature is too low to allow resetting the trips. The Low Flow trip provides protection against the occurrence of a control rod withdrawal accident occurring when less than four primary coolant pumps are in service. That accident has only been analyzed under four pump flow. If only one control rod is capable of withdrawal, or is PCS boron concentration assures criticality cannot occur even with all rods withdrawn (as required by LCO 3.9.1) the RPS is not required to be OPERABLE. The SHUTDOWN MARGIN requirements of LCOs 3.1.1 and 3.1.2 assure that criticality will not occur upon withdrawal of a single control rod.

The wide range flux level indication actuates bistables which actuate the permissive signal for the Zero Power Mode Bypass (for the TM/LP, Low PCS Flow, and Low SG Pressure trips), and bypass the startup rate trip. Wide range channel NI-03 provides the bypass permissive for RPS channels "A" and "C"; NI-04, for "B" and "D". A separate bistable trip unit is provided for each RPS channel.

The same bistables that provide the Zero Power Mode Bypass permissive also automatically bypass the High Startup Rate trips below the setpoint and enable them above. When at very low power levels, the nuclear instrument signals are not steady; if the Startup Rate trips were not bypassed, spurious trips could occur during start up operations.

The High Startup Rate trip is automatically bypassed when power range indicated power exceeds a nominal 15% RATED POWER. Allowing for bistable hysteresis this bypass may be as low as 13%. The trip is not useful above that power level since reactivity insertions at power would induce an immediate change in power level and eventually be terminated by the VHPT without attaining any significant startup rate. This bypass is automatically removed when the associated power range indication decreases below the bistable setpoint.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Power range NI-05 provides the bistable for RPS channel "A", NI-06 for "B", NI-07 for "C", and NI-08 for "D". These same power range bistable amplifiers also bypass the Loss of Load trip below the setpoint and enable the ASI alarm function above the setpoint. In addition, these bistables in NI-05 and NI-06 bypass the Turbine Trip on Generator Trip function below setpoint.

LCO

The LCO requires all instrumentation performing an RPS Function to be OPERABLE. Failure of any required portion of the instrument channel renders the affected channel(s) inoperable and reduces the reliability of the affected Functions. The specific criteria for determining channel OPERABILITY differ slightly between Functions. These criteria are discussed on a Function by Function basis below.

Actions allow maintenance (trip channel) bypass of individual channels, but the bypass key used to bypass a single channel cannot be simultaneously used to bypass that same parameter in other channels. This interlock prevents operation with a second channel in the same Function bypassed. The unit is restricted to 7 days in a trip channel bypass or otherwise untripped condition before either restoring the Function to four channel operation (2-out-of-4 logic) or placing the channel in trip (1-out-of-3 logic).

Only the Allowable Values are specified for each RPS trip Function in the LCO. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable, provided that operation and testing are consistent with the assumptions of the setpoint calculations. Each Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument uncertainties appropriate to the trip Function. These uncertainties are defined in Reference 6.

The following Bases for each trip Function identify the above RPS trip Function criteria items that are applicable to establish the trip Function OPERABILITY.

LCO (continued) This LCO requires that all four channels of all trip functions be OPERABLE when in MODES 1 and 2, and in MODES 3, 4, and 5 whenever more than one control rod is capable of being withdrawn. Exceptions are noted in the individual LCO bases below:

1. Variable High Power Trip (VHPT)

The Allowable Value is high enough to provide an operating envelope that prevents unnecessary Variable High Power trips during normal plant operations. The Allowable Value is low enough for the system to function adequately during reactivity addition events.

The VHPT is designed to limit maximum reactor power to its maximum design and to terminate power excursions initiating at lower powers without power reaching this full power limit. During a plant startup, the VHPT trip setpoint is initially at its minimum value, 30%. It remains fixed until manually reset, at which point it increases to ≤ 15% above existing Q Power.

The power increase may then continue until the new setpoint is approached at which time the VHPT setpoint is again manually reset to 15% above the existing Q Power. This pattern continues until the VHPT setpoint reaches its maximum setting of 106.5%. Thus, during power escalation, the VHPT trip setpoint is never more than 15% above existing power. This limits the magnitude of any inadvertent reactivity insertion or power increase. On a power decrease, the VHPT trip setpoint automatically tracks power levels downwards so that it is always a nominal 15% above the existing power.

During normal plant operation a VHPT is initiated when the reactor power level reaches its maximum value of 106.5% RTP. Adding to this the possible variation in trip point due to calibration and instrument errors, the maximum actual steady state power at which a trip would be actuated is 115%, which was used for the purpose of safety analysis.

2. High Startup Rate

This LCO requires four channels of High Startup Rate to be OPERABLE.

The high startup rate trip serves as a backup to the administratively enforced startup rate limit. The Function is not credited in the accident analyses; therefore, an Allowable Value for the trip cannot be derived from analytical limits and is not specified.

LCO (continued)

The four channels of High Startup Rate RPS trips are derived from two wide range NI signal processing drawers. Thus, a failure in one wide range channel may render two RPS channels inoperable. It is acceptable to continue operation under this condition because the High Startup Rate trip is not required by the Safety Analysis.

3. Low PCS Flow

A reactor trip is provided to protect the core against DNB should the coolant flow suddenly decrease significantly.

This trip is set high enough to maintain fuel integrity during a loss of flow condition. The setting is low enough to allow for normal operating fluctuations from offsite power.

Flow in each of the four coolant loops is determined from pressure drop from inlet to outlet of the steam generators. The total flow through the reactor core is determined, for the RPS flow channels, by summing the loop pressure drops across the steam generators and correlating this pressure sum with the sum of steam generator differential pressures which exists at 100% flow (four pump operation at full power T_{ave}). Full PCS flow is that flow which exists at RATED POWER, at full power T_{ave}, with four pumps operating.

The Low Flow Trip setting of 95% insures that the reactor cannot operate when the flow rate is less than 93% of the nominal value considering instrument errors.

The trip may be manually bypassed, for testing, when the reactor is shutdown and only one rod is capable of withdrawal. This bypass is part of the ZPMB circuitry, which also bypasses the TM/LP and SG Pressure-Low trips. Use of this bypass is restricted to those situations in which no more than one control rod is capable of withdrawal in order that the RPS continue to provide lower mode protection in the event of a control rod withdrawal with fewer than four PCPs operating.

4,5. Low Steam Generator Level

The Allowable Value assures that there will be sufficient water inventory in the steam generator at the time of trip to allow a safe and orderly plant shutdown and to prevent steam generator dryout assuming minimum auxiliary feedwater capacity.

LCO (continued)

The 25.9% narrow range minimum setting listed in Table 3.3.1-1 assures that the heat transfer surface (tubes) is covered with water when the reactor is critical. The 25.9% indicated level corresponds to the location of the feed ring, at 46.7" above the lower instrument tap. The narrow range instrument spans 180" for its 100% range.

Each steam generator level is sensed by measuring the differential pressure between the top and bottom of the downcomer annulus in the steam generator. These trips share four level sensing channels on each steam generator with the Auxiliary Feedwater Actuation Signal.

6,7. Low Steam Generator Pressure

The Allowable Value of 500 psia is sufficiently below the rated load operating point of 739 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used in the accident analysis.

Since excessive steam demand causes the PCS to cool down, resulting in positive reactivity addition to the core, a reactor trip is required to offset that effect.

The trip may be manually bypassed, for testing, when the reactor is shutdown and only one rod is capable of withdrawal. This bypass is part of the ZPMB circuitry, which also bypasses the TM/LP and Low Flow trips. Use of this bypass is restricted to those situations in which no more than one control rod is capable of withdrawal in order that the RPS continue to provide lower mode protection in the event of a control rod withdrawal with fewer than four PCPs operating.

The SG pressure channels are shared with the Steam Generator Low Pressure signals which isolate the steam and feedwater lines.

The safety analysis includes a ± 22 psi uncertainty allowance.

LCO (continued)

8. High Pressurizer Pressure

The Allowable Value is set high enough to allow for pressure increases in the PCS during normal operation (i.e., plant transients) not indicative of an abnormal condition. The setting is below the lift setpoint of the pressurizer safety valves and low enough to initiate a reactor trip when an abnormal condition is indicated.

The High Pressurizer pressure trip shares four safety grade instrument channels with the TM/LP trip and the low pressurizer pressure Safety Injection Signal.

The safety analysis includes a ± 22 psi uncertainty allowance.

9. Thermal Margin/Low Pressure (TM/LP)

The TM/LP trip system monitors core power, reactor coolant maximum inlet temperature, (T_c), core coolant system pressure and axial shape index. The Low Pressure Trip limit (P_{var}) is calculated using the equations given in Table 3.3.1-1.

The calculated limit (P_{var}) is then compared to a fixed Low Pressure Trip limit (P_{min}). The auctioneered highest of these signals becomes the trip limit (P_{trip}). P_{trip} is compared to the measured PCS pressure and a trip signal is generated when the measured pressure for that channel is less than or equal to P_{trip}. A pre-trip alarm is also generated when P is less than or equal to the pre-trip setting $P_{trip} + \Delta P$.

The TM/LP trip set points are derived from the 4-pump operation core thermal limits through application of appropriate allowances for measurement uncertainties and processing errors. A pressure allowance of 165 psi is assumed to account for instrument drift in both power and inlet temperatures, calorimetric power measurement, inlet temperature measurement, and primary system pressure measurement. Uncertainties accounted for that are not a part of the 165 psi term include allowances for assembly power tilt, fuel pellet manufacturing tolerances, core flow measurement uncertainty and core bypass flow, inlet temperature measurement time delays, and ASI measurement. Each of these allowances and uncertainties is included in the development of the TM/LP trip set point used in the accident analysis.

LCO (continued)

The trip may be manually bypassed, for testing, when the reactor is shutdown and only one rod is capable of withdrawal. This bypass is part of the ZPMB circuitry, which also bypasses the Low Flow and SG Pressure-Low trips. Use of this bypass is restricted to those situations in which no more than one control rod is capable of withdrawal in order that the RPS continue to provide lower mode protection in the event of a control rod withdrawal with fewer than four PCPs operating.

10. Loss of Load

The LCO requires four Loss of Load trip channels to be OPERABLE in MODE 1, above 17% power.

The Loss of Load trip is automatically disabled when power is below a nominal 15% RATED POWER, or 17% power allowing for bistable hysteresis, to allow startup and shutdown of the turbine generator. At low power the transient from a turbine trip would not cause safety valve operation. The Loss of load trip is automatically enabled and bypassed by the same power range bistable amplifiers that disable and enable the High Startup Rate trip. When power range channel NI-005 exceeds the setpoint, Loss of Load channels "A" and "C" are automatically enabled and High Startup Rate channels "A" and"C" are automatically disabled. Power range NI-006 bistable controls RPS channels "B" and "D" trips similarly.

The discrepancy between the nominal bypass power level of < 15% and the >17% power APPLICABILITY statement in Table 3.3.1-1 accounts for bistable hysteresis.

The Loss of Load trip uses a single pressure switch in the turbine Auto Stop Oil circuit to sense a turbine trip for input to all four RPS auxiliary trip units. The Function is not credited in the accident analyses; therefore, an Allowable Value for the trip cannot be derived from analytical limits and is not specified.

11. Containment High Pressure

The Allowable Value is high enough to allow for small pressure increases in containment expected during normal operation (i.e., plant heatup) that are not indicative of an abnormal condition. The High Containment Pressure trip provides a backup reactor trip in the event of a Loss of Coolant Accident, Main Steam Line Break, or Main Feedwater Line Break. The High Containment Pressure trip shares sensors with the Containment High Pressure sensing logic for Safety Injection, Containment Isolation, and Containment Spray.

BASES

LCO (continued)

Interlocks/Bypasses

The LCO on bypass permissive removal channels requires that the automatic bypass removal feature of all four operating bypass channels be OPERABLE for each RPS Function with an operating bypass in the MODES addressed in the specific LCO for each Function. All four bypass removal channels must be OPERABLE to ensure that none of the four RPS channels are inadvertently bypassed.

The LCO applies to the bypass removal feature only. If the bypass enable Function is failed so as to prevent entering a bypass condition, operation may continue.

The interlock Allowable Values are based on analysis requirements for the bypassed functions. These are discussed above as part of the LCO discussion for the affected Functions.

APPLICABILITY

This LCO is applicable in MODES 1 and 2, and in MODES 3, 4, and 5 when more than one control rod is capable of withdrawal and PCS boron concentration is less than that required by LCO 3.9.1. Loss of Load is only applicable in MODE 1 because it may be automatically bypassed at < 17% RTP, where it is no longer needed.

The trips are designed to take the reactor subcritical, maintaining the SLs during AOOs and assisting the ESF in providing acceptable consequences during accidents.

If no more than one control rod can be withdrawn the RPS function is already fulfilled (the safety analyses and the SHUTDOWN MARGIN definition both use the assumption that the highest worth withdrawn control rod will fail to insert on a trip) and the safety analyses assumptions and SHUTDOWN MARGIN requirements will be met without the RPS trip function.

ACTIONS

The most common causes of channel inoperability are outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant specific setpoint analysis. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the process instrument is set up for adjustment to bring it to within specification. If the trip setpoint is less conservative than the Allowable Value in Table 3.3.1-1, the channel is declared inoperable immediately, and the appropriate Condition(s) must be entered immediately.

ACTIONS (continued)

In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or RPS bistable trip unit is found inoperable, then all affected Functions provided by that channel must be declared inoperable, and the plant must enter the Condition for the particular protection Function affected.

When the number of inoperable channels in a trip Function exceeds that specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 is immediately entered if applicable in the current MODE of operation.

A Note has been added to the ACTIONS to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function. The Completion Times of each inoperable Function will be tracked separately for each Function, starting from the time the Condition was entered.

A.1

Condition A applies to the failure of a single channel in any RPS automatic trip Function, except High Startup Rate or Loss of Load. RPS coincidence logic is normally 2-out-of-4.

This action does not apply to the High Startup Rate or Loss of Load trips because the safety analyses take no credit for the functioning of these trips, they are installed for equipment protection only. In addition, there are fewer than four instrument channels for these functions, so that a single failure may defeat the Function in multiple RPS channels.

If one RPS bistable trip unit or associated instrument channel is inoperable, startup or power operation is allowed to continue. Though not explicitly required, the inoperable channel should be bypassed or tripped. If it is neither bypassed nor tripped, leaving the inoperable trip function in an untripped condition, then the operator must be careful not to inadvertently bypass the same Function in another channel. The provision of four trip channels allows one channel to be bypassed (removed from service) during operations, placing the RPS in 2-out-of-3 coincidence logic. It is preferable to place an inoperable channel in bypass rather than trip, since no additional random failure of a single channel can either spuriously trip the reactor or prevent it from tripping.

The failed channel is restored to OPERABLE status or is placed in trip within 7 days. Restoring the channel to OPERABLE status restores the full capability of the Function.

ACTIONS (continued)

Required Action A.1 places the Function in a 1-out-of-3 configuration. In this configuration, common cause failure of dependent channels cannot prevent trip.

The Completion Time of 7 days is based on operating experience, which has demonstrated that a random failure of a second channel occurring during the 7 day period is a low probability event.

B.1 and B.2

Condition B applies to the failure of two channels in any RPS automatic trip Function except High Startup Rate or Loss of Load.

Condition B does not apply to the High Startup Rate or Loss of Load trips. The safety analyses take no credit for the functioning of these trips; they are installed for equipment protection only. If Condition B were applicable to these non-safety grade trips, failure of one Startup Rate instrument during power operation, for instance, would limit plant operation to 7 days even though the trips are automatically bypassed. In addition, there are fewer than four instrument channels for these functions, so that a single failure may defeat the Function in multiple RPS channels. Post maintenance CHANNEL FUNCTIONAL TESTS of the inoperable channel would likely result in a reactor trip if performed at power.

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODES even though two channels are inoperable, with one channel tripped. MODE changes in this configuration are allowed to permit maintenance and testing on one of the inoperable channels. In this configuration, the protection system is in a 1-out-of-2 logic, and the probability of a common cause failure affecting both of the OPERABLE channels during the 7 days permitted is remote.

Required Action B.1 provides for placing one inoperable channel in trip within the Completion Time of 1 hour. Though not explicitly required, the other inoperable channel should be bypassed. If it is not bypassed, leaving one inoperable trip function in an untripped condition, then the operator must be careful not to inadvertently bypass the same Function in another channel. This could defeat three of the four RPS channels, rendering the RPS inoperable.

ACTIONS (continued)

This Completion Time is sufficient to allow the operator to take all appropriate actions for the failed channels while ensuring that the risk involved in operating with the failed channels is acceptable. With one channel of protective instrumentation bypassed or inoperable in an untripped condition, the RPS is in a 2-out-of-3 logic; but with another channel failed, the RPS may be operating in a 2-out-of-2 logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the RPS in a 1-out-of-2 logic. If any of the other OPERABLE channels receives a trip signal, the reactor will trip.

One channel should be restored to OPERABLE status within 7 days for reasons similar to those stated under Condition A. After one channel is restored to OPERABLE status, the provisions of Condition A still apply to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action B.2 must be placed in trip if more than 7 days have elapsed since the initial channel failure.

C.1

Condition C applies to the failure of a single Loss of Load or High Startup Rate trip unit or associated instrument channel. RPS coincidence logic is normally 2-out-of-4.

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODES even though one channel is inoperable. MODE changes in this configuration are allowed to permit maintenance and testing on one of the inoperable channels. In this configuration, the protection system may be in a 1-out-of-3 or 2-out-of-3 logic.

If one channel fails it must be restored to OPERABLE status prior to entering MODE 2 following the next MODE 5 entry. If the plant is in MODE 5 at the time the channel becomes inoperable, then the failed channel must be restored to OPERABLE status prior to startup. The Completion Time is based on the fact that the safety analyses take no credit for the functioning of these trips. In addition, there are fewer than four instrument channels for these functions, so that Post maintenance CHANNEL FUNCTIONAL TESTS of the inoperable channel would likely result in a reactor trip if performed at power.

ACTIONS (continued)

D.1, and D.2

Condition D applies to two Loss of Load or two High Startup Rate trip units or associated instrument channels inoperable.

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODES even though two channels are inoperable, with one channel tripped. MODE changes in this configuration are allowed to permit maintenance and testing on one of the inoperable channels. In this configuration, the protection system is in a 1-out-of-2 logic, and the probability of a common cause failure affecting both of the OPERABLE channels during the 7 days permitted is remote.

Required Action D.1 provides for placing one inoperable channel in trip within the Completion Time of 1 hour. Though not explicitly required, the other inoperable channel should be bypassed. If it is not bypassed, leaving one inoperable trip function in an untripped condition, then the operator must be careful not to inadvertently bypass the same Function in another channel. This could defeat three of the four RPS channels, rendering the RPS inoperable.

This Completion Time is sufficient to allow the operator to take all appropriate actions for the failed channels while ensuring that the risk involved in operating with the failed channels is acceptable. With one channel bypassed or inoperable in an untripped condition, the RPS is in a 2-out-of-3 logic; but with another channel failed, the RPS may be operating in a 2-out-of-2 logic. This should be corrected. To correct the problem, the second channel is placed in trip. This places the RPS in a 1-out-of-2 logic. If any of the other OPERABLE channels receives a trip signal, the reactor will trip.

One channel should be restored to OPERABLE status prior to entering MODE 2 following MODE 5 entry for reasons similar to those stated under Condition C. After one channel is restored to OPERABLE status, the provisions of Condition C still apply to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action D.2 must also be restored to OPERABLE before startup.

ACTIONS (continued)

E.1, and E.2

Condition E applies to one or two automatic bypass removal channels inoperable. If the bypass removal channel for any operating bypass cannot be restored to OPERABLE status, the associated RPS channel may be considered OPERABLE only if the bypass is not in effect. Otherwise, the affected RPS channels must be declared inoperable, and the bypass either removed or the bypass removal channel repaired. This is addressed by requiring entry into the appropriate CONDITION for the channels rendered inoperable by the bypass channel failure.

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODES even though two channels are inoperable, with one channel bypassed and one tripped. MODE changes in this configuration are allowed to permit maintenance and testing on one of the inoperable channels.

F.1, F.2.1, and F.2.2

Condition F is entered when the Required Action and associated Completion Time of Condition A, B, C, D, or E are not met.

If the Required Actions associated with these Conditions cannot be completed within the required Completion Times, the reactor must be brought to a MODE in which the Required Actions do not apply. The Required Action F.1 allowed Completion Time of 6 hours to be in MODE 3 is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

Required Actions F.2.1 and F.2.2 allow 6 hours to ensure that no more than one control rod is capable of being withdrawn or to ensure PCS boron is at least that required in the COLR for LCO 3.9.1. This completion time is reasonable to place the plant in a MODE where the Required Actions do not apply.

G.1

If the control room temperature exceeds 90°F, all Thermal Margin Monitor channels become inoperable due to exceeding their temperature qualification limit. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

The SRs for any particular RPS Function are found in the SR column of Table 3.3.1-1 for that Function. Most Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION.

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limits.

The Containment Pressure and Loss of Load channels are pressure switch actuated; they have no associated control room indicator and do not require a CHANNEL CHECK.

The RPS input channels consist of the following instruments:

Power Range Nuclear Power and Axial Shape Index
ΔT Power and associated PCS temperature channels.
Start Up Rate and Wide Range Power
Pressurizer Pressure
Primary Coolant System Flow
Turbine Generator Auto Stop Oil Pressure
Steam Generator Level
Steam Generator Pressure
Containment Pressure

A CHANNEL CHECK is also required each 12 hours for the TM/LP calculated setpoint indicator channels.

SURVEILLANCE REQUIREMENTS (continued)

The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.

SR 3.3.1.2

A daily calibration (heat balance) is performed when THERMAL POWER is ≥ 15%. The daily calibration shall consist of adjusting the "nuclear power calibrate" potentiometers to agree with the calorimetric calculation if the absolute difference is > 2.0%. The "ΔT power calibrate" potentiometers are then used to null the "nuclear power—ΔT power" indicators on the RPS Reactor Power Calibration and Indication panel. Performance of the daily calibration ensures that the two inputs to the Q power measurement are indicating accurately with respect to the much more accurate secondary calorimetric calculation.

The Frequency of 24 hours is based on plant operating experience and takes into account indications and alarms located in the control room to detect deviations in channel outputs.

The Frequency is modified by a Note indicating this Surveillance must be performed within 12 hours after THERMAL POWER is ≥ 15% RTP. The secondary calorimetric is inaccurate at lower power levels. The 12 hours allows time for plant stabilization, data taking, and instrument calibration.

SR 3.3.1.3

A CHANNEL FUNCTIONAL TEST is performed on each RPS instrument channel, except Loss of Load and Power Rate of Change, every 92 days to ensure the entire channel will perform its intended function when needed.

In addition to power supply tests, The RPS CHANNEL FUNCTIONAL TEST consists of three overlapping tests as described in Reference 7. These tests verify that the RPS is capable of performing its intended function, from bistable input through the de-energization of the clutch power supplies. They include:

SURVEILLANCE REQUIREMENTS (continued)

Bistable Tests

The bistable setpoint must be found to trip within the Allowable Values specified in the LCO and left set consistent with the assumptions of the plant specific setpoint analysis (Ref. 5).

A test signal is superimposed on the input in one channel at a time to verify that the bistable trips within the specified tolerance around the setpoint. This is done with the affected RPS channel trip channel bypassed. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint analysis.

The as found and as left values must also be recorded and reviewed for consistency with the assumptions of the Frequency extension analysis. The requirements for this review are outlined in Reference 8.

Matrix Logic Tests

Matrix Logic tests are addressed in LCO 3.3.2. This test is performed one matrix at a time. It verifies that a coincidence in the two input channels for each Function removes power from the matrix relays. During testing, power is applied to the matrix relay test coils and prevents the matrix relay contacts from assuming their de-energized state. This test will detect any short circuits around the bistable contacts in the coincidence logic, such as may be caused by faulty bistable relay or trip channel bypass contacts.

Trip Path Tests

Trip Path (Initiation Logic) tests are addressed in LCO 3.3.2. These tests are similar to the Matrix Logic tests, except that test power is withheld from one matrix relay at a time, allowing the initiation circuit to de-energize, de-energizing selected clutch power supplies.

The Frequency of 92 days is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 8).

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.4

A calibration of the excore nuclear instrumentation power range channels using the internal test circuitry must be performed every 31 days. The internal test circuitry excludes the detectors from calibration, but ensures that the channels are reading accurately and within tolerance. The Surveillance verifies that the channel responds to a calibrated internal test signal within the necessary range and accuracy. This leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests.

The as found and as left values must also be recorded and reviewed for consistency with the assumptions of the frequency extension analysis. The requirements for this review are outlined in Reference 8.

The neutron detectors are excluded from calibration because they are passive devices with minimal drift and because of the difficulty of simulating a meaningful signal. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration (SR 3.3.1.2). In addition, associated control room indications are continuously monitored by the operators.

The Frequency of 31 days is acceptable, based on plant operating experience, and takes into account indications and alarms available to the operator in the control room.

SR 3.3.1.5

The constants in each thermal margin monitor must be verified every 92 days. This test verifies that the programmable constants used to calculate the setpoints generated by the digital circuitry of the TMM are correct. It is nearly equivalent to a CHANNEL FUNCTIONAL TEST on an analog circuit. Because the constants are entered digitally, there is no setpoint drift. For this reason, a 92 day frequency is adequate.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.6

A CHANNEL FUNCTIONAL TEST on the Loss of Load, and High Startup Rate channels is performed prior to a reactor startup to ensure the entire channel will perform its intended function if required.

The High Startup Rate Trip is actuated by either of the Wide Range Nuclear Instrument Startup Rate channels. NI-01/03 sends a trip signal to RPS channels "A" and "C"; NI-02/04 to "B" and "D". Since each Startup Rate channel would cause a trip on two RPS channels, the Startup Rate Trip is not tested when the reactor is critical. The High Startup Rate trip Function is required during startup operation and is bypassed when shut down or above a nominal 15% RTP, (13% allowing for bistable hysteresis).

The four Loss of Load Trip channels are all actuated by a single pressure switch monitoring Turbine Auto Stop Oil pressure. It is not testable with the reactor critical.

Operating experience has shown that these components usually pass the Surveillance when performed at a Frequency of once per 7 days prior to each reactor startup.

SR 3.3.1.7

SR 3.3.1.7 is the performance of a CHANNEL CALIBRATION on the RPS measurement channels every 18 months.

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with the setpoint analysis.

The as found and as left values must also be recorded and reviewed for consistency with the assumptions of the frequency extension analysis. The requirements for this review are outlined in Reference 8.

The Frequency is based upon the assumption of an 18 month calibration interval for the determination of the magnitude of equipment drift.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The RPS input channels consist of the following instruments:

Power Range Nuclear Power and Axial Shape Index
ΔT Power and associated PCS temperature channels.
Start Up Rate and Wide Range Power
Pressurizer Pressure
Primary Coolant System Flow
Turbine Generator Auto Stop Oil Pressure
Steam Generator Level
Steam Generator Pressure
Containment Pressure

As part of the CHANNEL CALIBRATION of the Wide Range Nuclear Instrumentation, the automatic removal of the Zero Power Mode Bypass of Low PCS Flow, TM/LP, and Low SG Pressure trips, and of the automatic bypassing of the Loss of Load and High Startup Rate trips should be verified to assure that these trips are available when required.

The neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices with minimal drift and because of the difficulty of simulating a meaningful signal. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration (SR 3.3.1.2).

SR 3.3.1.8

This SR verifies that the control room temperature is within the temperature limits for the most restrictive component, the Thermal Margin Monitor. The 12 hour frequency is appropriate based on engineering judgement and past operation experience.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 21
2. 10 CFR 100
3. IEEE Standard 279-1971, April 5, 1972
4. FSAR, Chapter 14
5. 10 CFR 50.49
6. CPCo EGAD "Setpoint Methodology"
7. FSAR, Section 7.2
8. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989
9. FSAR Figure 7-1
10. SER, dated 03/28/90, "Reactor Protection System Upgrade - Thermal Margin Monitor Audit Confirmatory Item - Temperature Qualification"

B 3.3 **INSTRUMENTATION**

B 3.3.2 Reactor Protective System (RPS) Logic and Trip Initiation

BASES

BACKGROUND

The RPS initiates a reactor trip to protect against violating the core specified acceptable fuel design limits and reactor coolant pressure boundary integrity during Anticipated Operational Occurrences (AOOs). By tripping the reactor, the RPS also assists the Engineered Safety Features (ESF) systems in mitigating accidents.

The protection and monitoring systems have been designed to ensure safe operation of the reactor. This is achieved by specifying Limiting Safety System Settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this Specification as the Allowable Value, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents.

During AOOs, which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The departure from nucleate boiling ratio shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling;
- Fuel centerline melting shall not occur; and
- The Reactor Coolant System pressure SL of 2750 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 50 (Ref. 1) and 10 CFR 100 (Ref. 2) criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 (Ref. 2) limits. Different accident categories allow a different fraction of these limits based on probability of occurrence.

Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

BACKGROUND (continued) The RPS is segmented into three interconnected modules. These modules are:

- Measurement channels;
- Bistable trip units; and
- RPS Logic;

This LCO addresses the RPS Logic, including Manual Trip capability. LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation" provides a description of the role of this equipment in the RPS. This is summarized below:

RPS Logic

The RPS Logic consists of both Matrix and Initiation Logic and employs a scheme that provides a reactor trip when bistables in any 2-out-of-4 channels sense the same input parameter trip. This is called a 2-out-of-4 trip logic. This logic and the clutch power supply configuration are shown in FSAR Figure 7-1 (Ref. 5).

Bistable relay contact outputs from the four channels are configured into six logic matrices. Each logic matrix checks for a coincident trip in the same parameter in two bistable channels. The matrices are designated the AB, AC, AD, BC, BD, and CD matrices to reflect the bistable channels being monitored. Each logic matrix contains four normally energized matrix relays. When a coincidence is detected, consisting of a trip in the same Function in the two channels being monitored by the logic matrix, all four matrix relays de-energize.

The matrix relay contacts are arranged into trip paths, with one of the four matrix relays in each matrix opening contacts in one of the four trip paths. Each trip path provides power to one of the four normally energized clutch power supply "M contactors" (M1, M2, M3, and M4).

The trip paths thus have six contacts in series from each matrix, and perform a logical OR function, de-energizing the M contactors if any one or more of the six logic matrices indicate a coincidence condition.

BACKGROUND (continued)

De-energizing the M contactors removes AC power to the four clutch power supply inputs. Contacts from M Contactors M1 and M2 are in series with each other and in the AC power supply path to clutch power supplies PS1 and PS3. M3 and M4 are similarly arranged with respect to clutch power supplies PS2 and PS4. Approximately half of the control rods receive power from auctioneered clutch power supplies 1 and 2. The remaining control rods receive clutch power from auctioneered clutch power supplies 3 and 4.

Manual reactor trip capability is afforded by two main control board-mounted pushbuttons. One of these (CO-1) opens contacts in series with each of the four trip paths, de-energizing all M contactors. The other pushbutton (CO-2) opens circuit breakers which provide AC input power to the M contractor contacts and downstream clutch power supplies. Thus depressing either pushbutton will cause a reactor trip. When a coincidence occurs in two RPS channels, all four matrix relays in the affected matrix de-energize. This in turn de-energizes all four M contactors, which interrupt AC input power to the four clutch power supplies, allowing the control rods to insert by gravity.

Matrix Logic refers to the matrix power supplies, trip channel bypass contacts, and interconnecting matrix wiring between bistable and auxiliary trip units, up to but not including the matrix relays. Contacts in the bistable and auxiliary trip units are excluded from the Matrix Logic definition, since they are addressed as part of the measurement channel.

The Initiation Logic consists of the matrix relays and their associated contacts, all interconnecting wiring, CO-1 manual trip contacts, and M contactors.

Manual Trip circuitry includes both manual reactor trip pushbuttons CO-1 and CO-2, and the interconnecting wiring necessary to effect de-energization of the clutch power supplies.

Neither the clutch power supplies nor the AC input power source to these supplies is considered as Safety Related other than as addressed by this LCO. Operation may continue with one or two selective clutch power supplies de-energized.

BACKGROUND (continued)

It is possible to change the 2-out-of-4 RPS Logic to a 2-out-of-3 logic for a given input parameter in one channel at a time by trip channel bypassing select portions of the matrix logic. Trip channel bypassing a bistable effectively shorts the bistable relay contacts in the three matrices associated with that channel. Thus, the bistables will function normally, producing normal trip indication and annunciation, but a reactor trip will not occur unless two additional channels indicate a trip condition. Trip channel bypassing can be simultaneously performed on any number of parameters in any number of channels, providing each parameter is bypassed in only one channel at a time. A bypass key interlock prevents simultaneous trip channel bypassing of the same parameter in more than one channel. Trip channel bypassing is normally employed during maintenance or testing.

Functional testing of the entire RPS, from bistable input through the de-energization of individual sets of clutch power supplies can be performed either at power or shutdown and is normally performed on a quarterly basis. FSAR, Section 7.2 (Ref. 3), explains RPS testing in more detail.

APPLICABLE SAFETY ANALYSES

Reactor Protective System (RPS) Logic

The RPS Logic provides for automatic trip initiation to maintain the SLs during AOOs and assist the ESF systems in ensuring acceptable consequences during accidents. All transients and accidents that call for a reactor trip assume the RPS Logic is functioning as designed.

Manual Trip

There are no accident analyses that take credit for the Manual Trip; however, the Manual Trip is part of the RPS circuitry. It is used by the operator to shut down the reactor whenever any parameter is rapidly trending toward its trip setpoint. A Manual Trip accomplishes the same results as any one of the automatic trip Functions.

LCO

Reactor Protective System (RPS) Logic

Failures of individual bistable relays and their contacts are addressed in LCO 3.3.1. This Specification addresses failures of the Matrix Logic not addressed in the above, such as the failure of matrix relay power supplies or the failure of the trip channel bypass contact in the bypass condition.

Loss of a single vital bus will de-energize one of the two power supplies in each of three matrices. Because of power supply auctioneering, all four matrix relays will remain energized in each affected matrix. This de-energization of up to three matrix power supplies due to a single failure is to be treated as a single channel failure.

Each of the four Initiation Logic channels de-energizes one set of clutch power supplies if any of the six coincidence matrices de-energize their associated matrix relays. They thus perform a logical OR function. Initiation Logic channels 1 and 2 receive AC power from Vital bus 3. Initiation Logic channels 3 and 4 receive AC input power from Vital bus 4. Because of clutch power supply output auctioneering, it is possible to de-energize either input bus without de-energizing control rod clutches.

1. Matrix Logic

   This LCO requires six channels of Matrix Logic to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when more than one control rod is capable of being withdrawn.

2. Initiation Logic

   This LCO requires four channels of Initiation Logic to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when more than one control rod is capable of being withdrawn.

3. Manual Trip

   The LCO requires both Manual Trip channels to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when more than one control rod is capable of being withdrawn.

   Two independent pushbuttons are provided. Each pushbutton is considered a channel. Depressing either pushbutton interrupts power to all four clutch power supplies, tripping the reactor.

BASES

APPLICABILITY The RPS Matrix Logic and Manual Trip are required to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when more than one control rod is capable of being withdrawn from the core and PCS boron concentration is less than that required by LCO 3.9.1. This ensures the reactor can be tripped when necessary, but allows for maintenance and testing when the reactor trip is not needed.

In MODES 3, 4, and 5 with no more than one control rod capable of withdrawal, or when PCS boron concentration is at least that required by LCO 3.9.1, these Functions do not have to be OPERABLE. However, two wide range neutron flux monitoring channels must be OPERABLE to ensure proper indication of neutron population and to indicate a boron dilution event. This is addressed in LCO 3.3.9, "Neutron Flux Monitoring Channels."

ACTIONS When the number of inoperable channels in a trip Function exceeds that specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 is immediately entered if applicable in the current MODE of operation.

A.1

Condition A applies if one Matrix Logic channel is inoperable in any applicable MODE. Loss of a single vital instrument bus will de-energize one of the two matrix power supplies in up to three matrices. This is considered a single matrix failure. The auctioneered matrix relays will remain energized. The above statement is supported by a Note under Condition A.

The channel must be restored to OPERABLE status within 48 hours. The Completion Time of 48 hours provides the operator time to take appropriate actions and still ensures that any risk involved in operating with a failed channel is acceptable. Operating experience has demonstrated that the probability of a random failure of a second Matrix Logic channel is low during any given 48 hour interval. If the channel cannot be restored to OPERABLE status within 48 hours, Condition D is entered.

B.1

Condition B applies to one Initiation Logic channel inoperable in all applicable MODES. These Required Actions require de-energizing the affected clutch power supplies. This removes the need for the affected channel by performing its associated safety function. With the clutch power supplies associated with one initiation logic channel de-energized, the remaining two clutch power supplies prevent control rod clutches from de-energizing.

ACTIONS (continued)

The remaining clutch power supplies are in a 1-out-of-2 logic with respect to the remaining initiation logic channels in the clutch power supply path. This meets redundancy requirements, but testing on the OPERABLE channels cannot be performed without causing a reactor trip. Loss of AC power to the two trip paths associated with a single pair of clutch power supplies is not considered a trip path failure, as the associated clutch power supplies will de-energize.

Required Action B.1 provides for de-energizing the affected clutch power supplies associated with the inoperable channel within a Completion Time of 1 hour. This Required Action is conservative, since the redundant initiation logic channel associated with the same set of clutch power supplies will de-energize the affected clutch power supplies if required during the one hour allowed.

C.1

Condition C applies to the failure of one manual reactor trip channel. With one manual reactor trip channel inoperable, operation may continue until the reactor is shut down for other reasons. Repair during operation is not required because one OPERABLE channel is all that is required for safe operation. No safety analyses assume operation of the Manual trip. In addition, the Manual Trip channels are not testable without actually causing a reactor trip, so even if the difficulty were corrected, the post maintenance testing necessary to declare the channel OPERABLE could not be completed during operation.

D.1, D.2.1, and D.2.2

Condition D is entered if Required Actions associated with Condition A, B, or C are not met within the required Completion Time, or if more than one Manual Trip, Matrix Logic, or Initiation Logic channel is inoperable.

If the Required Actions associated with these Conditions cannot be completed within the required Completion Times, the reactor must be brought to a MODE in which the Required Actions do not apply. The Required Action D.1 allowed Completion Time of 6 hours to be in MODE 3 is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

Required Actions D.2.1 and D.2.2 allow 6 hours to ensure that no more than one control rod is capable of being withdrawn or to ensure PCS boron is at least that required in the COLR for LCO 3.9.1. This completion time is reasonable to place the plant in a MODE where the Required Actions do not apply.

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1

A CHANNEL FUNCTIONAL TEST on each RPS Logic channel is performed every 92 days to ensure the entire channel will perform its intended function when needed.

In addition to power supply tests, the RPS CHANNEL FUNCTIONAL TEST consists of three overlapping tests as described in Reference 3. These tests verify that the RPS is capable of performing its intended function, from bistable input through the de-energizing of the clutch power supplies. The first test, the bistable test, is addressed by SR 3.3.1.3 in LCO 3.3.1.

This SR addresses the two tests associated with the RPS Logic: Matrix Logic and Initiation Logic (Trip Path).

Matrix Logic Tests

These tests are performed one matrix at a time. They verify that a coincidence in the two input channels for each Function removes power from the matrix relays. During testing, power is applied to the matrix relay test coils and prevents the matrix relay contacts from assuming their de-energized state. The Matrix Logic tests will detect any short circuits around the bistable contacts in the coincidence logic such as may be caused by faulty bistable relay or trip channel bypass contacts.

Trip Path Tests

These tests are similar to the Matrix Logic tests, except that test power is withheld from one matrix relay at a time, allowing the initiation circuit to de-energize, de-energizing the affected set of clutch power supplies.

The Frequency of 92 days is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 4).

SR 3.3.2.2

A CHANNEL FUNCTIONAL TEST on the Manual Trip channels is performed prior to a reactor startup to ensure the entire channel will perform its intended function if required. The Manual Trip Function can only be tested when the plant is shutdown. The simplicity of this circuitry and the absence of drift concern makes this Frequency adequate. Additionally, operating experience has shown that these components usually pass the Surveillance when performed once within 7 days prior to each reactor startup.

BASES

REFERENCES

1. 10 CFR 50, Appendix A
2. 10 CFR 100
3. FSAR, Section 7.2
4. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989
5. FSAR Figure 7-1

B 3.3 **INSTRUMENTATION**

B 3.3.3 Engineered Safety Features Instrumentation (ESFI)

BASES

BACKGROUND

The ESF Instrumentation initiates necessary safety systems, based upon the values of selected unit parameters, to protect against violating core design limits and the Primary Coolant System (PCS) pressure boundary and to mitigate accidents.

The ESFI contains devices and circuitry that generate the following signals when the monitored variables reach levels that are indicative of conditions requiring protective action. Also listed are the inputs to each ESFI Actuation Signal:

1. Safety Injection Actuation Signal (SIS);
   a. Containment High Pressure (CHP)
   b. Pressurizer Pressure - Low
2. Containment High pressure;
   a. Containment Pressure - High - Left Train
   b. Containment Pressure - High - Right Train
3. Containment High Radiation (CHR)
   a. Containment Area Radiation - Hi
4. Steam Generator Low Pressure (SGLP)
   a. "A" SG Pressure - Low
   b. "B" SG Pressure - Low
5. Auxiliary Feedwater Actuation
   a. "A" SG Level - Low
   b. "B" SG Level - Low
6. Recirculation Actuation Signal (RAS);
   a. Safety Injection Refueling Water Tank (SIRWT) Level - Low

BACKGROUND (continued)

In the above list of actuation signals, the CHP and SIRWT Level signals are derived from pressure and level switches.

Equipment actuated by each of the above signals is identified in the FSAR Section 7.3 (Ref. 1).

The ESF circuitry, with the exception of Recirculation Actuation Signal (RAS), employs 2-out-of-4 logic. Four independent measurement channels are provided for each function used to generate ESF actuation signals. When any two channels of the same function reach their setpoint, actuating relays are energized which, in turn, initiate the protective actions. Two separate and redundant trains of actuating relays, each powered from separate power supplies, are utilized. These separate relay trains operate redundant trains of ESF equipment.

RAS logic consists of output contacts of the relays actuated by the SIRWT level switches arranged in a "1-out-of-2 taken twice" logic. The contacts are arranged so that at least one low level signal powered from each station battery is required to initiate RAS. Loss of a single battery, therefore, cannot either cause or prevent RAS initiation.

Measurement Channels

Measurement channels, consisting of field transmitters or process sensors and associated instrumentation, provide a measurable electronic signal based upon the physical characteristics of the parameter being measured.

Four identical measurement channels are provided for each parameter used in the generation of trip signals. These are designated Channels A through D. Measurement channels provide input to ESF bistable trip units within the same ESFI channel. In addition, some measurement channels are also be used as inputs to Reactor Protective System (RPS) bistables, and most provide indication in the control room. Measurement channels used as an input to the RPS or ESFI are not used for control Functions. Those ESFI sensors shared with the RPS are identified in Table B 3.3.1-1.

When a channel monitoring a parameter indicates an unsafe condition, the bistable monitoring the parameter in that channel will trip. In the case of SIRWT and CHP, the sensors are latching auxiliary relays from level and pressure switches respectively, which do not develop an analog input to separate bistable. Tripping two or more channels of bistables monitoring the same parameter will actuate both trains of Actuation Logic of the associated Engineered Safety Features (ESF) equipment.

BACKGROUND (continued)

Three of the four measurement and bistable channels are necessary to meet the redundancy and testability of GDC 21 in Appendix A to 10 CFR 50 (Ref. 2). The fourth channel provides additional flexibility by allowing one channel to be removed from service in a bypass condition for maintenance or testing while still maintaining a minimum 2-out-of-3 logic. There are, however, no built-in provisions for channel bypasses in the ESFI design.

Since no single failure will either cause or prevent a protective system actuation and no protective channel feeds a control channel, this arrangement meets the requirements of IEEE Standard 279-1971 (Ref. 4).

Bistables

Bistables receive an analog input from the measurement channels, compare the analog input to trip setpoints, and provide contact output to the Actuation Logic. They also provide local trip indication and remote annunciation.

There are four channels of bistables, designated A through D, for each ESF Function, one for each measurement channel.

The trip setpoints and Allowable Values used in the bistables are based on the analytical limits stated in Reference 5. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, and instrumentation uncertainties, Allowable Values specified in Table 3.3.3-1, in the accompanying LCO, are conservatively adjusted with respect to the analytical limits. A detailed description of the method used to calculate the trip setpoints, including their explicit uncertainties, is provided in the CPCo EGAD "Setpoint Methodology" (Ref. 7). The actual nominal trip setpoint entered into the bistable is normally still more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION . If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

Setpoints in accordance with the Allowable Value will ensure that Safety Limits of Chapter 2.0, "SAFETY LIMITS (SLs)," are not violated during Anticipated Operational Occurrences (AOOs) and that the consequences of Design Basis Accidents (DBAs) will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed.

BACKGROUND (continued)

ESF Logic

Each of the six ESF actuating signals in Table 3.3.3-1 operates two trains of actuating relays. Each train is capable of initiating a safeguards equipment load group to meet the minimum requirements to provide all functions necessary to operate the system associated with the plant's capability to cope with abnormal events.

The logic circuitry includes bypass provisions such that the SGLP function may be bypassed if 3-out-of-4 SG pressure channels are below a bypass permissive setpoint. Similarly, the SIS on Pressurizer Pressure - Low may be bypassed when 3-out-of-4 channels are below a permissive setpoint.

APPLICABLE SAFETY ANALYSES

Each of the analyzed accidents can be detected by one or more ESF Functions. One of the ESF Functions is the primary actuation signal for that accident. An ESFI Function may be the primary actuation signal for more than one type of accident. An ESFI Function may also be a secondary, or backup, actuation signal for one or more other accidents. Functions such as Manual Initiation, not specifically credited in the accident analysis, serve as backups to Functions and are part of the NRC approved licensing basis for the plant.

ESF Instrumentation protective Functions are as follows:

1. Safety Injection Signal

The SIS ensures acceptable consequences during Loss Of Coolant Accident (LOCA) events, including steam generator tube rupture, and Main Steam Line Breaks (MSLBs) or Feedwater Line Breaks (FWLBs) (inside containment). To provide the required protection, SIS is actuated by manual initiation, by a CHP signal, or by 2-out-of-4 Pressurizer Pressure channels decreasing below the setpoint. SIS initiates the following actions:

a) Start HPSI & LPSI pumps

b) Enable Containment Spray Pump Start on CHP

c) Initiate Safety Injection Valve operations

Each Manual Actuation channel consists of one pushbutton which directly starts the SIS actuation logic for the associated train.

APPLICABLE SAFETY ANALYSES (continued)

The pressurizer pressure instrument channels which provide input to SIS are the same channels which provide an input to the RPS. The RPS receives an analog signal from each Pressurizer Pressure channel; each SIS initiation logic train receives a binary signal from a group of four relays, each actuated by a bistable in one of the four instrument channels. The contacts of these relays are wired into a 2-out-of-4 logic. It is the output of this pressurizer pressure 2-out-of-4 logic circuit that is blocked during plant cooldowns. A similar arrangement of bistables and relays provide the pressurizer pressure-low block permissive signal when three of the four pressurizer pressure - low bypass bistables are below the bypass setpoint. The initiation and block circuits are illustrated in Reference10.

Each SIS logic train is also actuated by a contact pair on one of the CHP initiation relays for the associated CHP train.

Each train of SIS actuation logic consists of a group of "SIS" relays which energize and seal in when the initiation logic is satisfied. These SIS relays actuate alarms and control functions. One of the control functions selects between an immediate actuation circuit, if offsite power is available, and a time sequenced actuation circuit, if only diesel power is available. These actuation circuits initiate motor operated valve opening and pump starting. The SIS actuation logic is illustrated in Reference 10.

3. Containment High Pressure Signal

The CHP signal closes all containment isolation valves not required for ESF operation, ensuring acceptable consequences during LOCAs and MSLBs or FWLBs (inside containment).

CHP is actuated by 2-out-of-4 pressure switches for the associated train reaching their setpoints. CHP initiates the following actions:

a) Containment Spray

b) Safety Injection Signal

c) Main Feedwater Isolation

d) Main Steam Line Isolation

e) Control Room HVAC Emergency Mode

f) Close Containment Isolation Valves

APPLICABLE SAFETY ANALYSES (continued)

Eight containment pressure channels are provided. Each channel consists of one pressure sensing bellows which actuates two micro - switches. Four of these sixteen micro - switches provide input to the RPS; the remainder are divided into two circuits of 2-out-of-4 logic for the CHP logic trains.

Each CHP logic train consists of an arrangement of six micro - switch contacts and a test relay which energize a group of "5P" relays when the 2-out-of-4 logic is satisfied. The CHP logic is illustrated in Reference 11.

In performing the Containment Isolation Function, either a CHP or a CHR will isolate the same containment isolation valves, except that CHR does not close Component Cooling line valves.

4. Containment High Radiation Signal

CHR is actuated by manual action or, during normal operation, by 2-out-of-4 radiation monitors setpoints. During refueling operations the CHR actuation is manually switched to actuate on 1-of-2 low range radiation monitors at a much lower setpoint. CHR initiates the following actions:

a) Control Room HVAC Emergency Mode

b) Close Containment Isolation Valves

c) Block automatic starting of ECCS pump room sump pumps

The containment area radiation monitors which actuate CHR each de-energize an output relay upon reaching their setpoint. The output contacts of these relays are arranged into two trains of 2-out-of-4 logic. Two manual controls each de-energize two of these relays, initiating both trains of CHR. When either train of 2-out-of-4 logic is satisfied, a group of "5R" relays energize to initiate CHR actions.

One containment radiation monitor is located adjacent to each containment air cooler where radioiodines would condense along with water vapor in the event of minor breaches of primary system integrity. Radiation monitor locations in lower level containment will respond to an abnormal accumulation of radioactive coolant, such as from a ruptured letdown line, or Primary Drain Tank leakage. CHR logic is depicted in Reference 12.

During refueling operations, separate switch-selectable radiation monitors initiate CHR, as addressed by LCO 3.3.6.

APPLICABLE SAFETY ANALYSES. (continued)

4. Steam Generator Low Pressure Signal

The SGLP ensures acceptable consequences during an MSLB or FWLB by isolating the steam generator if it indicates a low steam generator pressure. The SGLP, concurrent with or following a reactor trip, minimizes the rate of heat extraction and subsequent cooldown of the PCS during these events.

One SGLP circuit is provided for each SG. Each SGLP circuit is actuated by 2-out-of-4 pressure channels on the associated SG reaching their setpoint. SGLP initiates the following actions:

a) Close the associated Feedwater Regulating valve and its bypass.

b) Close both Main Steam Isolation Valves.

The SG pressure instrument channels which provide input to SGLP are the same channels which provide an input to the RPS. Both the SGLP logic and the RPS receive analog signals from the instrument channel, and both have their own bistables to initiate actuation on low pressure.

Each SGLP logic is made up of output contacts from four pressure bistables from the associated steam generator. When the logic circuit is satisfied, two relays are energized to actuate steam and feedwater line isolation. The SGLP logic is depicted in Reference 13

5. Recirculation Actuation Signal

At the end of the injection phase of a LOCA, the SIRWT will be nearly empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the ECCS pumps is automatically switched to the containment recirculation sump. Switchover from SIRWT to the containment sump must occur before the SIRWT empties to prevent damage to the ECCS pumps and a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sump to support pump suction. Furthermore, early switchover must not occur to ensure sufficient borated water is injected from the SIRWT to ensure the reactor remains shut down in the recirculation mode. A SIRWT Level - Low signal initiates the RAS.

APPLICABLE SAFETY ANALYSES (continued)

RAS is actuated by manually actuating the circuit "Test" switch or by two of the four level sensors in the SIRWT reaching their setpoints. RAS initiates the following actions:

a) Trip LPSI pumps (this trip can be manually bypassed)

b) Switch HPSI & Spray suction from SIRWT to Containment Sump

c) Adjust cooling water to Shutdown Cooling Heat Exchangers

d) Closes the SIRWT Recirc CVs.

The four SIRWT level sensors each de-energize two relays, one per logic train, when tank level reaches the setpoint. Each channel of level sensor and associated output relays is powered from a different Preferred AC bus. Two Preferred AC buses are powered, through inverters, from each station battery. The manual RAS control for each train de-energizes two of these relays, initiating RAS through the logic train.

Each train of RAS logic consists of the output contacts of the relays actuated by the level switches arranged in a "1-out-of-2 taken twice" logic. The contacts are arranged so that at least one low level signal powered from each station battery is required to initiate RAS. Loss of a single battery, therefore, cannot either cause or prevent RAS initiation. When the logic is satisfied, two DC relays are energized to initiate RAS actions and alarms. The RAS logic is illustrated in Reference 3.

The RAS signal is actuated by separate sensors from those which provide tank level indication. The allowable range of 21" to 27" above the tank floor corresponds to 1.1% to 3.3% indicated level. Typically the actual setting is near the midpoint of the allowable range.

Each RAS Train actuates the valves in the injection and spray pump suction lines for the associated train switching the water supply from the SIRW tank to the containment sump for a recirculation mode of operation. The time required to reach the RAS setpoint depends on the initiating event. Following a DBA, RAS would occur after a period of approximately 20 minutes. The setpoint was chosen to provide adequate water in the containment sump for HPSI pump net positive suction head following an accident, but prevent the pumps from running dry during the 60 second switchover.

APPLICABLE SAFETY ANALYSES (continued)

6. Auxiliary Feedwater Actuation Signal

An AFAS initiates feedwater flow to both steam generators if a low level is indicated in either steam generator.

The AFAS maintains a steam generator heat sink during the following events:

- MSLB;
- FWLB;
- Inadvertent opening of a steam generator atmospheric dump valve; and
- Loss of feedwater.

AFAS is actuated by manual action or by 2-out-of-4 level sensors on either steam generator reaching their setpoints. Manual actuation of Auxiliary Feedwater may be accomplished through pushbutton actuation of each AFAS channel or by use of individual pump and valve controls. Each AFAS channel starts the associated AFW pump(s) and opens the associated flow control valves.

The steam generator level instrument channels which provide input to AFAS are the same channels which provide an input to the RPS. Both the AFAS cabinets and the RPS receive analog signals from the instrument channel, and both have their own bistables to initiate actuation on low level.

Each AFAS train contains a 2-out-of-4 logic for each steam generator. One AFAS logic train actuates motor driven AFW pump P-8A and turbine driven pump P-8B and the associated flow control valves; the other actuates motor driven pump P-8C and the associated valves. Each train provides flow to both steam generators. The AFAS logic uses solid state logic circuits.

LCO

The LCO requires all channel components necessary to provide an ESF actuation to be OPERABLE.

The Bases for the LCO on ESF Functions are:

1. Safety Injection Signal

a. Pressurizer Pressure - Low

This LCO requires four channels of SIAS Pressurizer Pressure - Low to be OPERABLE in MODES 1, 2, and 3.

The setpoint was chosen so as to be low enough to avoid actuation during plant operating transients, but to be high enough to be quickly actuated by a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). The settings include an uncertainty allowance of -22 psia and are the settings assumed in the Loss of Coolant Accident analysis.

The Pressurizer - Low signal for each SIS train can be blocked when 3-out-of-4 channels indicate below 1700 psia. This block prevents undesired actuation of SIS during a normal plant cooldown. The block signal is automatically removed when 2-out-of-4 channels exceed the setpoint, in accordance with the bypass philosophy of removing bypasses when the enabling conditions are no longer satisfied.

This LCO requires four channels of the bypass permissive removal for SIS Pressurizer Pressure - Low to be OPERABLE in MODES 1, 2, and 3.

The bypass permissive channels consist of four sensor subsystems and two actuation subsystems. This LCO applies to failures in the four sensor subsystems, including sensors, bistables, and associated equipment. Failures in the actuation subsystems, including the manual bypass switches, are considered Actuation Logic failures and are addressed in LCO 3.3.4.

This LCO applies to the bypass removal feature only. If the bypass enable Function is failed so as to prevent entering a bypass condition, operation may continue.

The block permissive is set low enough so as not to be enabled during normal plant operation, but high enough to allow blocking prior to reaching the trip setpoint.

LCO (continued)

2. Containment High Pressure

This LCO requires four channels of CHP Containment Pressure - High to be OPERABLE for each of the two trains in MODES 1, 2, and 3.

The setpoint was chosen so as to be high enough to avoid actuation by containment temperature or atmospheric pressure changes, but low enough to be quickly actuated by a LOCA or a MSLB in the containment.

3. Containment High Radiation Signal

This LCO requires four channels of CHR to be OPERABLE in MODES 1, 2, and 3.

The setpoint is based on the maximum primary coolant leakage to the containment atmosphere allowed by Specification 3.1.5 and the maximum activity allowed by Specification 3.1.4. N^{16} concentration reaches equilibrium in containment atmosphere due to its short half-life, but other activity was assumed to build up. At the end of a 24-hour leakage period the dose rate is approximately 20 R/h as seen by the area monitors. A large leak could cause the area dose rate to quickly exceed the 20 R/h setting and initiate CHR.

4. Steam Generator Pressure - Low

This LCO requires four channels of Steam Generator Pressure - Low Instrumentation for each steam generator to be OPERABLE in MODES 1, 2, and 3.

The setpoint was chosen to be low enough to avoid actuation during plant operation, but be close enough to full power operating pressure to be actuated quickly in the event of a MSLB. The setting of includes a -22 psi uncertainty allowance and was the setting used in the FSAR Section 14 analysis.

The SGLP signal from each steam generator may be blocked when 3-of-the-4 steam pressure channels indicate below 550 psia. This block prevents undesired actuation during a normal plant cooldown. The block signal is automatically removed when steam pressure exceeds the setpoint.

Each SGLP logic is made up of output contacts from four pressure bistables from the associated steam generator. When the logic circuit is satisfied, two relays are energized to actuate steam and feedwater line isolation. A similar logic circuit is provided for each block circuit.

LCO (continued)

The block is automatically removed when the steam pressure exceeds 550 psig, in accordance with the bypass philosophy of removing bypasses when the enabling conditions are no longer satisfied.

This LCO applies to failures in the four sensor subsystems, including sensors, bistables, and associated equipment. Failures in the actuation subsystems, including the manual bypass switches, are considered Actuation Logic failures and are addressed in LCO 3.3.4.

This LCO applies to the bypass removal feature only. If the bypass enable Function is failed so as to prevent entering a bypass condition, operation may continue.

The block permissive is set low enough so as not to be enabled during normal plant operation, but high enough to allow blocking prior to reaching the trip setpoint.

5. Auxiliary Feedwater Actuation Signal

The AFAS logic actuates Auxiliary Feedwater (AFW) to both steam generators on low level in either SG.

The Auxiliary Feedwater Actuation Signal (AFAS) is initiated by 2-out-of-4 low level signals occurring for either steam generator, as sensed by narrow range level transmitters. The setpoint is the same as that for Reactor Trip. The allowable value was chosen to assure that Auxiliary Feedwater Flow would be initiated while the steam generator could still act as a heat sink and steam source, and to assure that a reactor trip would not occur on low level without the actuation of Auxiliary Feedwater.

This LCO requires four channels for each steam generator of Steam Generator Level - Low to be OPERABLE in MODES 1, 2, and 3.

LCO (continued)

6. Recirculation Actuation Signal

a. Safety Injection Refueling Water Tank Level - Low

This LCO requires four channels of SIRWT Level - Low to be OPERABLE in MODES 1, 2, and 3.

The RAS signal is actuated by separate sensors from those which provide tank level indication. The allowable range of 21" to 27" above the tank floor corresponds to 1.1% to 3.3% indicated level. Typically the actual setting is near the midpoint of the allowable range.

The upper limit on the Allowable Value for this trip is set low enough to ensure RAS does not initiate before sufficient water is transferred to the containment sump. Premature recirculation could impair the reactivity control Function of safety injection by limiting the amount of boron injection.

Premature recirculation could also damage or disable the recirculation system if recirculation begins before the sump has enough water to prevent air containment in the suction. The lower limit on the SIRWT Level - Low trip Allowable Value is high enough to transfer suction to the containment sump prior to emptying the SIRWT.

Four SIRWT level sensors are arranged to provide two independent Recirculation Actuation Signals. Each low level sensors is powered from a separate Preferred AC bus; thus two are ultimately powered from each station battery. Each Recirculation Actuation Signal (RAS) circuit is wired with the contacts from the pair of level sensors powered from the same battery in parallel. These two parallel circuits are wired in series, producing a "1-out-of-2 taken twice" logic. RAS for each train is actuated by either switch from the left battery sensing low level concurrently with either switch from the right battery.

APPLICABILITY All ESF Instrumentation Functions are required to be OPERABLE in MODES 1, 2, and 3. In MODES 1, 2, and 3 there is sufficient energy in the primary and secondary systems to warrant automatic ESF System responses to:

- Close the main steam isolation valves to preclude a positive reactivity addition;
- Actuate AFW to preclude the loss of the steam generators as a heat sink (in the event the normal feedwater system is not available);
- Actuate ESF systems to prevent or limit the release of fission product radioactivity to the environment by isolating containment and limiting the containment pressure from exceeding the containment design pressure during a design basis LOCA or MSLB; and
- Actuate ESF systems to ensure sufficient borated inventory to permit adequate core cooling and reactivity control during a design basis LOCA or MSLB accident.

In MODES 4, 5, and 6, automatic actuation of ESF Instrumentation Functions is not required because adequate time is available for plant operators to evaluate plant conditions and respond by manually operating the ESF components, if required. In LCO 3.6.3, containment isolation vales must remain OPERABLE IN mode 4. It is thus possible to initiate CHP and CHR manually, on a component basis, in MODE 4. LCO 3.3.6 addresses automatic Refueling CHR isolation during core alterations or during movement of irradiated fuel.

In MODES 5 and 6, ESF Instrumentation initiated systems are either reconfigured or disabled for shutdown cooling operation. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant specific setpoint analysis.

Typically, the drift is small and results in a delay of actuation rather than a total loss of function. Determination of setpoint drift is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the process instrument is set up for adjustment to bring it to within specification. If the actual trip setpoint is not within the Allowable Value in Table 3.3.3-1, the channel is inoperable and the appropriate Condition(s) are entered.

In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value in Table 3.3.3-1, or the sensor, instrument loop, signal processing electronics, or ESF Instrumentation bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the plant must enter the Condition statement for the particular protection Function affected.

When the number of inoperable channels in a trip Function exceeds those specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.

A Note has been added to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function in Table 3.3.3-1. Completion Times for each inoperable channel of a Function will be tracked separately.

A.1 and A.2

Condition A applies to one SIRWT Low Level channel inoperable. The SIRWT low level circuitry is arranged in a "1-out-of-2 taken twice" logic rather than the more frequently used 2-out-of-4 logic. Therefore, Required Action A.1 differs from other ESF functions. With a bypassed SIRWT low level channel, an additional failure might disable automatic RAS, but would not initiate a premature RAS. With a tripped channel, an additional failure could cause a premature RAS, but would not disable the automatic RAS.

Since considerable time is available after initiation of SIS until RAS is required and there is quite a tolerance on the time when RAS must be initiated, and since a premature RAS could damage all the ESF pumps, it is preferable to bypass an inoperable channel and risk loss of automatic RAS than to trip a channel and risk a premature RAS.

ACTIONS (continued)

Eight hours is allowed for this action since it must be by circuit modification.

Action A.2 requires that the inoperable channel must be repaired within 7 days to limit the time the unit is operated with an inoperable channel.

B.1

Condition B applies to the failure of a single channel of one or more input parameters in the following ESFI Functions:

1. Safety Injection Signal
   Pressurizer Pressure - Low

2. Containment High Pressure

3. Containment High Radiation

4. Steam Generator Low Pressure

5. Auxiliary Feedwater Actuation Signal
   Steam Generator Level - Low

ESF coincidence logic is normally 2-out-of-4. If one ESF Instrumentation channel is inoperable, startup or power operation is allowed to continue as long as action is taken to restore the design level of redundancy.

If one ESF Instrumentation channel is inoperable, startup or power operation is allowed to continue, providing the inoperable channel is placed in trip within 7 days (Required Action B.1). The provision of four trip channels allows one channel to be failed in a non-trip condition or optionally bypassed up to the 7 day Completion Time allotted to place the channel in Trip, although except for AFAS there are no installed design provisions for this bypass function. Operating with one failed channel in a non-trip condition or bypassed (removed from service) during operations, placing the ESFI in 2-out-of-3 coincidence logic.

If the failed channel cannot be restored to OPERABLE status in 7 days, the associated bistable is placed in a tripped condition. This places the function in a 1-out-of-3 configuration. In this configuration, common cause failure of the dependent channel cannot prevent ESF actuation. The 7 day Completion Time is based upon operating experience, which has demonstrated that a random failure of a second channel occurring during the 7 day period is a low probability event.

ACTIONS (continued)

C.1 and C.2

Condition C applies to the failure of two channels in any of the ESFI functions addressed by Condition B:

With two inoperable channels, one channel should be placed in trip within the 8 hour Completion Time. Eight hours is allowed for this action since it must be accomplished by a circuit modification, or by removing power from a circuit component.

With one channel of protective instrumentation bypassed or failed in a non-trip condition, the ESF Function is in 2-out-of-3 logic, but with another channel failed the ESFI may be operating with a 2-out-of-2 logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the ESFI in a 1-out-of-2 logic. If any of the other OPERABLE channels receives a trip signal, ESFI actuation will occur.

One of the failed channels should be restored to OPERABLE status within 7 days, for reasons similar to those for Condition B. After one channel is restored to OPERABLE status, the provisions of Condition B still apply to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action C.2 must be placed in trip if more than 7 days has elapsed since the initial channel failure.

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODES even though two channels are inoperable, with one channel tripped. MODE changes in this configuration are allowed, to permit maintenance and testing on one of the inoperable channels. In this configuration, the protection system is in a 1-out-of-2 logic, and the probability of a common cause failure affecting both of the OPERABLE channels during the 7 days permitted is remote.

D.1 and D2

Condition D applies to the failure of one or two bypass removal channels.

The bypass removal channels consist of four sensor subsystems and two actuation subsystems. Condition D applies to failures in one or two of the four sensor subsystems, including sensors, bistables, and associated equipment. Failures in the actuation subsystems, including the manual bypass pushbuttons, are considered Actuation Logic failures and are addressed in LCO 3.3.4.

ACTIONS (continued)

In Condition D, it is permissible to continue operation with one bypass permissive removal channel failed, providing the bypass is disabled (Required Action D.1). Since the bypass Function must be manually enabled, the bypass permissive Function will not by itself cause an undesired bypass insertion, even if multiple sensor channels fail. Because the bypass pushbutton seals in the bypass function when depressed, once inserted, bypass removal can only be accomplished by removing a fuse or some similar time consuming action.

If the bypass removal feature in the inoperable channel cannot be defeated, actions to address the inoperability of the affected automatic trip channel must be taken. Required Action D.2 requires declaring the affected Actuation Logic Trains inoperable, and entering the appropriate Condition. If the bypass removal feature cannot be removed, then the affected ESF function (Low Pressure SIS input or SGLP) is inoperable. LCO 3.3.4 addresses Logic Channel inoperability. The 8 hour Completion Time is permitted due to the absence of installed bypass defeat capability, requiring the need for equipment modifications.

E.1 and E.2

If the Required Actions and associated Completion Times of Condition A, B, C, or D, are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The SRs for any particular ESF Function are found in the SRs column of Table 3.3.3-1 for that Function. Most functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION.

SR 3.3.3.1

A CHANNEL CHECK is performed each 12 hours on each ESF input channel which is provided with an indicator to provide a qualitative assurance that the channel is working properly and that its readings are within limits. The CHP Signal and SIRWT level channels have no associated control room indicator, and are not channel checked.

SURVEILLANCE REQUIREMENTS (continued)

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when Surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Offscale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

The Frequency of about once every shift is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of CHANNEL OPERABILITY during normal operational use of displays associated with the LCO required channels.

SR 3.3.3.2

A CHANNEL FUNCTIONAL TEST is performed every 92 days to ensure the entire channel will perform its intended function when needed.

This test is required to be performed each 92 days on ESF input channels provided with on-line testing capability. It is not required for the SIRWT Level channels since they have no built in test capability. The CHANNEL FUNCTIONAL TEST for SIRWT Level channels is performed each 18 months as part of the required CHANNEL CALIBRATION.

The CHANNEL FUNCTIONAL TEST tests the individual sensor subsystems using an analog test input to each bistable.

SURVEILLANCE REQUIREMENTS (continued)

A test signal is superimposed on the input in one channel at a time to verify that the bistable trips within the specified tolerance around the setpoint. Any setpoint adjustment shall be consistent with the assumptions of the current setpoint analysis.

The as found and as left values must also be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis. The requirements for this review are outlined in Reference 9.

SR 3.3.3.3

CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive surveillances. CHANNEL CALIBRATIONS must be performed consistent with the setpoint analysis.

The as found and as left values must also be recorded and reviewed for consistency with the assumptions of the extension analysis. The requirements for this review are outlined in Reference 9.

The Frequency is based upon the assumption of an 18 month calibration interval for the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. FSAR, Section 7.3
2. 10 CFR 50, Appendix A
3. Logic Diagram, SIS Test and RAS, E 17, Sh 5
4. IEEE Standard 279-1971
5. FSAR, Chapter 14
6. 10 CFR 50.49
7. CPCo EGAD "Setpoint Methodology"
8. FSAR, Section 7.2
9. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989
10. Logic Diagram, Safety Injection initiation, E 17, Sh 3
11. Logic Diagram, Containment High Pressure, E 17, Sh 6
12. Logic Diagram, Containment High Radiation, E 17, Sh 7
13. Logic Diagram, SG Low Pressure and MSIS, E 17, Sh 20