50-269 DPC OCONEE 1

SUPPLEMENT 1 TO IMPROVED TECHNICAL SPECIFICATIONS SUBMITTAL

REC'D W/LTR DTD 03/26/98....9804060231

ENCLOSURE 1

ITS Section	Subject	ID
1.0	Add generic NSHC for M DOCs that was inadvertently excluded.	3
1.0	Revise DOC A7 for Section 1.0 - refers to MODE 3 instead of MODE 2, incorrectly characterizes change.	14
1.0	Revise heading on page 12 of DOCs for Section 1.0 to say "Less" instead of "More."	15
1.0	Revise DOC L3 on page 12 of DOCs for Section 1.0 - last sentence, change "upon" to "of"	16
1.0	Remove ONS-006 (BWOG-65) - rejected by the TSTF. Restores NUREG examples.	48
1.0	Remove rejected TSTF-39, R1 and incorporate TSTF-205 (R1). Revises definition of CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST.	52

ITS Section	Subject	ID
1.0	Incorporate TSTF-124 which deletes paragraph from CHANNEL CALIBRATION & CHANNEL FUNCTIONAL TEST definitions related to testing of bypass functions. This had already been incorporated as a plant specific change so only change is deletion of JFD & markup.	53
2.0	Revise DOC M3 on page 2 of DOCs for Section 2.0 - should say RCS pressure to be restored within 5 minutes, not 15 minutes.	17
2.0	Revise NUREG markup for 2.2.4 to correctly crossreference DOC M3 instead of M4.	18
2.0	Revise ITS 2.1.1.1 & 2.1.1.2 to include reference to AXIAL POWER IMBALANCE and RCS Variable Low Pressure Protective limits ensuring compliance to SLs.	30
3.0	Revise LCO 3.0.5 to reinstate ", or" consistent with the NUREG. Also, remove incorrect crossref. on NUREG M/U (incorrectly xref from LCO 3.0.4 to CTS 4.0.3)	27
3.0	Remove ONS-012 (BWOG-70), withdrew by BWOG. This change removed incorrect bracketed references. JFD revised to correct on a plant specific basis.	43
3.0	Incorporate TSTF-071, R1 - Adds examples for SFDP.	49

ITS Section	Subject	ID
3.1	Revise ITS to incorporate approved TSTF-110, R2 which eliminates the increased SR Freq associated wtih inoperable alarms. No impact to ITS since Oconee had retained CTS which is the same Freq. as modified NUREG. Only affects markup and JFDs.	36
3.1	Incorporate revision to ONS-017 (BWOG-84) - Revises RA A to require imbalance to be determined within 2 hours when APSR inoperable or not aligned.	41
3.1	Incorporate revision to ONS-013 (BWOG-70) - Revises LCO Note wording.	44
3.1	Revise ITS 3.1.8 to incorporate revision to ONS-027 - Deletes ACT E and modifes ACT D to replace RA D.1 with "Suspend PHYSICS TESTING" with a CT of 30 minutes.	57
3.1	Remove JFD associated with approved ONS generic changes ONS-018 and annotate NUREG Markup with TSTF number 256.	58
3.10	Revise ITS as appropriate to consider the fact that the SSF Submersible Pump is not a QA grade pump. Adds separate SR for Submersible Pump.	29
3.2	Revise to refer to SDM as specified in the COLR rather than value.	2

ITS Section	Subject	ID
3.2	Revise ITS to incorporate approved TSTF-110, R2 which eliminates the increased SR Freq associated wtih inoperable alarms - Revises SR Frequencies associated with 3.2.1, 3.2.2, & 3.2.3	37
3.2	Incorporate revision to ONS-013 (BWOG-70) and replace JFD annotations with TSTF 216 annotations on NUREG Markup. Revises LCO Note wording.	45
3.3	Revise JFD 29 of Att. 6 to remove incorrect reference to ONS-003.	22
3.3	Revise NUREG Bases Markup for 3.3 to refer to ESPS rather than ESFAS, 2nd paragraph, page B 3.3-48	24
3.3	Incorporate revision to ONS-021 (BWOG-85) - Deletes SR 3.3.10.3, clarifies Bases of 3.3.9 & 10 to clarify Channel Check requires one decade of overlap.	42
3.3	Incorporate revision to ONS-001 (BWOG-060) - Removes separate SR for calibrating power range channel and consolidates with SR 3.3.1.3	55
3.3	Remove JFDs associated with approved ONS generic changes ONS-002, 003, 011, 019, 020 and annotate NUREG Markup with TSTF numbers 211, 212, 215, and 217, respectively	56

ITS Section	Subject	ID
3.4	Change relocation of 3.4 LA13 from SLC to QA Topical (testing after maintenance or modification).	4
3.4	Capitalize "RCS" in 3.4 DOC M19.	5
3.4	3.4 DOC M32 (6th line) "exist to be" to "exists to be"	6
3.4	Revise ITS 3.4.13.e to incorporate Amendment 227 (changes primary to secondary leakage limit to 150 gallon per day per SG)	7
3.4	DOC LA12 for Section 3.4 should refer to CTS 4.3.2 not CTS 4.5.2	28
3.4	Incorporate revision to ONS-013 (BWOG-70) - Moves LCO Note back to Applicability Note.	46
3.4	Incorporate revision to ONS-023 (BWOG-78) - Changes the order of the info in Condition C.	47

ITS Section	Subject	ID
3.4	Remove JFD associated with approved ANO generic changes ANO-1-46, 47, & 48 and annotate NUREG Markup with TSTF numbers	60
3.4	Remove JFD associated with approved ONS generic change ONS-024 and annotate NUREG Markup with TSTF number 219.	61
3.5	Add requirement for manual OPERABILITY of LP-9, 10, 12, 14, 15, 16, 17, 18 - Adds SRs.	20
3.5	Revise CTS Markup for ITS 3.5.3, page 4 of 6 to correctly refer to ITS SR 3.5.3.2 instead of "3.5.2.3."	34
3.5	Remove JFD associated with approved ONS generic change ONS-004 and annotate NUREG Markup with TSTF number ...	62
3.6	Remove JFD associated with approved ONS generic change ONS-004 and annotate NUREG Markup with TSTF number ...	64
3.7	DOC M13 is missing and DOC A13 is described as M change. Changed A13 to M13. DOC A13 shown as not used. Revised CTS Markup to show M13 instead of A13.	8

ITS Section	Subject	ID
3.7	Revise DOC LA8 to indicate relocation of marked CTS 4.12.1.a information to Bases for SR 3.7.9.1 & revise Bases for SR 3.7.9.1 to include relocated information.	9
3.7	Incorporate changes to ONS-007 (BWOG-66) - Revises Bases for 3.7.13 to reflect appropriate nomenclature consistent with ITS Spec	26
3.7	Revise ITS 3.4.13.e to incorporate Amendment 227. Revises Bases to say that 150 gpd per SG leakage limit is bound by analysis.	33
3.7	Revise ITS to remove draft generic change ONS-025 due to being withdrawn from BWOG consideration on 11/6/97. This affects the wording of Condition D.	38
3.7	Revise ITS to remove draft generic change ONS-022 (BWOG-77) due to being rejected by the TSTF. Adds modifier "except when all MFCVs and SFCVs are closed and deactivated or isolated by a closed manual valve" back to Applicability.	39
3.7	Revise ITS to remove TSTF-157 based on the high probability that the TSTF will be permanently withdrawn. Adds back NUREG SR 3.7.5.5 which requires alignment check of EFW flow paths.	40
3.7	Remove DOC LA9 from Section 3.7 and show in CTS markup that CTS 4.5.4.1.b.1 is addressed by Section 5.0.	54

ITS Section	Subject	ID
3.7	Remove JFD associated with approved ONS generic change ONS-004 and annotate NUREG Markup with TSTF number.	65
3.7	Remove JFD associated with approved ONS generic change ONS-007 and annotate NUREG Markup with TSTF number 255.	66
3.9	Revise ITS submittal to incorporate approved TSTF-96, R1 - No change to ITS since it replaces draft generic change ONS-009.	35
3.9	Remove JFD associated with approved ONS generic change ONS-008 and annotate NUREG Markup with TSTF number 214.	67
4.0	Revise 4.0 DOC A3 to change "in" to "and" in the second sentence.	32
4.0	Incorporate TSTF-123, R1 - minor change to wording of 4.2.2.	51
5.0	Remove prescriptive information regarding which organization fulfills the operating experience review requirement.	10

ITS Section	Subject	ID
5.0	Correct ITS retyped pages for 5.5.12.a and 5.5.12.b to agree with NUREG Markup.	12
5.0	Revise DOC A30 to correctly refer to ITS 5.5.7 and the Pre-Stressed Concrete Containment Tendon Testing Program.	19
5.0	Revise Section 5.0 DOCs A15, A17 & A18 to change "duplicates" to "duplicate."	25
5.0	Revise ITS 5.6.5.a to add item 12 requiring AXIAL POWER IMBALANCE and RCS Variable Low Pressure Protective limits to be in the COLR.	31
5.0	Remove TSTF-119 - Adds back specific requirement to have written procedures for Fire Protection program implementation.	50
CVR ATT3	Att. 3 revised to remove JFD reference column and show draft generic change disposition - show xref. between ONS#/BWOG#/TSTF#.	75
CVR ATT4	Att. 4 revised to show disposition of generic changes based on latest TSTF status and to include new TSTFs (including ONS draft generic changes).	76

ITS Section	Subject	ID
CVR ATT7	Change Att. 7 of cover letter to reflect change to LA DOCS in Sections 2.0, 3.2, 3.4, 3.5, 3.7, 3.10, and 5.0	68

ENCLOSURE 2

1.0 Use and Application

ITS Section 1.0

ID 3

Subject Add generic NSHC for M DOCs that was inadvertently excluded.

MORE RESTRICTIVE CHANGES

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." Some of the proposed changes involve adding more restrictive requirements to the existing Technical Specifications by either making current requirements more stringent or by adding new requirements which currently do not exist.

These changes may include additional commitments that decrease allowed outage time, increase frequency of surveillance, impose additional surveillance, increase the scope of a specification to include additional plant equipment, increase the applicability of a specification, or provide additional actions. These changes are generally made to conform with the NUREG-1430.

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

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

   The proposed changes provide more stringent requirements than previously existed in the Technical Specifications. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event. If anything the new requirements may decrease the probability or consequences of an analyzed event by incorporating the more restrictive changes. The changes do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed changes provide more stringent requirements than previously existed in the Technical Specifications. The changes do not alter the plant configuration (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The changes do impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis.

Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

ITS Section 1.0

ID 14

Subject Revise DOC A7 for Section 1.0 - refers to MODE 3 instead of MODE 2, incorrectly characterizes change.

Coolant Temperature and bolting status of the reactor vessel head closure studs in the ITS (MODE definition and Table 1.1-1).

The CTS defines the reactivity condition in terms of a subcritical condition (expressed in %Δk/k). The NUREG defines the reactivity condition in terms of K_{eff}. The ONS ITS adopts the K_{eff} convention. The small difference between 1 percent Δk/k and 0.99 K_{eff} is well within the typical accuracy for reactivity predictions. Therefore, the movement of the CTS definitions for Reactor Operating Conditions into the ITS Table 1.1-1 is considered an administrative change and is consistent with the NUREG method of presentation of MODES. The applicability of the Reactor Operating Condition definition changes are evaluated at each occurrence of the defined Reactor Operating Condition in the ONS CTS. Changes to the ONS CTS are discussed on an individual basis with the Specification. Each change is evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis.

A5 Not used.

A6 CTS 1.2.2 defines Hot Shutdown in terms of a subcritical condition (1% Δk/k shutdown) and an average reactor coolant temperature of greater than or equal to 525°F. This Hot Shutdown operating condition definition is modified to correlate with the ITS MODE 3 criteria established in ITS Table 1.1-1. The ITS MODE 3 criteria imposes a minimum average reactor coolant temperature of 250°F. The lower average reactor coolant temperature band could represent more restrictive requirements on the operation of the facility. The applicability of this Reactor Operating Condition definition change is evaluated at each occurrence of the defined Hot Shutdown Applicability in the CTS. Changes to the CTS are discussed on an individual basis with the Specification. Each change is evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis. This change is consistent with the NUREG.

A7 CTS 1.2.4 defines Hot Standby as when T_{avg} is greater than 525°F, the reactor is critical and indicated neutron power on the power range
1 channels is less than 2 percent of rated power. ITS MODE 2 is defined
1 as when $k_{eff} \geq 0.99$ and ≤ 5% RATED THERMAL POWER. The applicability of this Reactor Operating Condition definition change is evaluated at each occurrence of the defined Hot Standby Applicability in the ONS CTS. Changes to the CTS are discussed on an individual basis with the Specification. Each change is evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis. This change is an administrative change and is consistent with the NUREG.

A8 Not used.

ITS Section 1.0

ID 15

Subject Revise heading on page 12 of DOCs for Section 1.0 to say "Less" instead of "More."

1 **TECHNICAL CHANGES - LESS RESTRICTIVE**

L1 CTS 1.5.2 defines a Channel Test as the injection of an internal or external test signal into the channel to verify its proper output response; including alarm and/or trip initiating action where applicable. The ITS Defines a CHANNEL FUNCTIONAL TEST as the injection of a simulated or actual signal into the channel as close to the sensor
1 as practicable to verify OPERABILITY of all devices in the channel
1 required for channel OPERBILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the channel is functionally tested. The provision to permit use an actual signal is a less restrictive requirement upon unit operation and is consistent with the NUREG. Use of an actual signal does not affect the performance of the channel. OPERABILITY can be adequately demonstrated in either case since the channel itself does not discriminate between an "actual" or "simulated" signal.

L2 CTS 1.5.4 requires an Instrument Channel Calibration to encompass the entire channel and does not exclude RTDs and thermocouples. The ITS definition of CHANNEL CALIBRATION allows performing "...an in place qualitative assessment of sensor behavior..." for these devices. This change is a less restrictive requirement upon unit operations and is consistent with the NUREG. A qualitative assessment of sensor behavior is acceptable for RTDs and thermocouples since the operation of these devices is governed by well understood and predictable physical relationships between the temperature of the sensed medium and the output of the RTD or thermocouple. Additionally, the output of RTDs and thermocouples is not adjustable. These devices are reliable and not subject to drift in the same manner as other sensors. As a result a qualitative assessment of sensor behavior is sufficient to determine its OPERABILITY and acceptability for continued use.

L3 CTS 1.2.8 defines Startup as a reduction in the shutdown margin with the intent of going critical. ITS MODE 2 is comparable to the CTS condition of startup. ITS MODE 2 is specified as when $K_{eff} \geq 0.99$ and THERMAL POWER is < 5%. The elimination of requirements associated with the CTS Startup Condition when shutdown margin is being reduced and $K_{eff} < 0.99$ (i.e., when in ITS MODES 3, 4 and 5) is a less restrictive requirement upon unit operation and is consistent with the NUREG. When the unit is in MODES 3, 4 or 5 shutdown margin is controlled by ITS Specification
1 3.4.1. ITS 3.4.1 ensures appropriate control of shutdown margin when in ITS MODES 3, 4 or 5.

L4 CTS 1.2.4 and 1.2.5 require use of the power range neutron channels as the indication to be used to establish the transition between Hot Standby and Power Operation. ITS Table 1.1-1 uses % RATED THERMAL POWER as the measure of reactor power used to establish the transition point between MODES 1 and 2. Although, the power range neutron channels may still be used since they are normalized to the calorimetric power

ITS Section 1.0

ID 16

Subject Revise DOC L3 on page 12 of DOCs for Section 1.0 - last sentence, change "upon" to "of"

1 **TECHNICAL CHANGES - LESS RESTRICTIVE**

L1 CTS 1.5.2 defines a Channel Test as the injection of an internal or external test signal into the channel to verify its proper output response; including alarm and/or trip initiating action where applicable. The ITS Defines a CHANNEL FUNCTIONAL TEST as the injection of a simulated or actual signal into the channel as close to the sensor
1 as practicable to verify OPERABILITY of all devices in the channel
1 required for channel OPERBILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the channel is functionally tested. The provision to permit use an actual signal is a less restrictive requirement upon unit operation and is consistent with the NUREG. Use of an actual signal does not affect the performance of the channel. OPERABILITY can be adequately demonstrated in either case since the channel itself does not discriminate between an "actual" or "simulated" signal.

L2 CTS 1.5.4 requires an Instrument Channel Calibration to encompass the entire channel and does not exclude RTDs and thermocouples. The ITS definition of CHANNEL CALIBRATION allows performing "...an in place qualitative assessment of sensor behavior..." for these devices. This change is a less restrictive requirement upon unit operations and is consistent with the NUREG. A qualitative assessment of sensor behavior is acceptable for RTDs and thermocouples since the operation of these devices is governed by well understood and predictable physical relationships between the temperature of the sensed medium and the output of the RTD or thermocouple. Additionally, the output of RTDs and thermocouples is not adjustable. These devices are reliable and not subject to drift in the same manner as other sensors. As a result a qualitative assessment of sensor behavior is sufficient to determine its OPERABILITY and acceptability for continued use.

L3 CTS 1.2.8 defines Startup as a reduction in the shutdown margin with the intent of going critical. ITS MODE 2 is comparable to the CTS condition of startup. ITS MODE 2 is specified as when $K_{eff} \geq 0.99$ and THERMAL POWER is < 5%. The elimination of requirements associated with the CTS Startup Condition when shutdown margin is being reduced and $K_{eff} < 0.99$ (i.e., when in ITS MODES 3, 4 and 5) is a less restrictive requirement upon unit operation and is consistent with the NUREG. When the unit is in MODES 3, 4 or 5 shutdown margin is controlled by ITS Specification
1 3.4.1. ITS 3.4.1 ensures appropriate control of shutdown margin when in ITS MODES 3, 4 or 5.

L4 CTS 1.2.4 and 1.2.5 require use of the power range neutron channels as the indication to be used to establish the transition between Hot Standby and Power Operation. ITS Table 1.1-1 uses % RATED THERMAL POWER as the measure of reactor power used to establish the transition point between MODES 1 and 2. Although, the power range neutron channels may still be used since they are normalized to the calorimetric power

measurement, the ITS permits use of the calorimetric measurement itself. The added flexibility to use calorimetric indications as a measure of THERMAL POWER is a less restrictive requirement upon unit operation and is consistent with the NUREG. The use of the calorimetric indication is acceptable since it provides a more direct and more accurate indication of THERMAL POWER.

LESS RESTRICTIVE CHANGE L3

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

CTS 1.2.8 defines Startup as a reduction in the shutdown margin with the intent of going critical. ITS MODE 2 is comparable to the CTS condition of startup. ITS MODE 2 is specified as when K_{eff} $\geq$ 0.99 and THERMAL POWER is < 5%. The elimination of requirements associated with the CTS Startup Condition when shutdown margin is being reduced and K_{eff} < 0.99 (i.e., when in ITS MODES 3, 4 and 5) is a less restrictive requirement upon unit operation and is consistent with the NUREG. When the unit is in MODES 3, 4 or 5 shutdown margin is controlled by ITS Specification 3.4.1. ITS
1 3.4.1 ensures appropriate control of shutdown margin when in ITS MODES 3, 4 or 5.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change allows operation when K_{eff} is < 0.99 without restrictions associated with being in the Startup condition. This change does not result in any hardware changes. Restrictions associated with being in the Startup condition when K_{eff} is < 0.99 are not considered as initiators of any previously analyzed accident. As such, the probability of an accident is independent of being in the Startup condition when K_{eff} is < 0.99. Also, the change does not change the assumed response of equipment in performing specified mitigation functions from that originally considered. The consequences are not changed since the equipment functions the same. Therefore, the change does not significantly increase the probability or consequences of an accident.

ITS Section 1.0

ID 48

Subject Remove ONS-006 (BWOG-65) - rejected by the TSTF.
Restores NUREG examples.

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
1	C. One Function X train inoperable. AND	C.1 Restore Function X train to OPERABLE status. OR	72 hours
1	One Function Y train inoperable.	C.2 Restore Function Y train to OPERABLE status.	72 hours

(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
1 1 1	A. One channel inoperable.	A.1 Perform SR 3.x.x.x. OR A.2 Reduce THERMAL POWER to ≤ 50% RTP.	Once per 8 hours 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 (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 hours 72 hours

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(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. OR A.2 Reduce THERMAL POWER to ≤ 50% RTP.	Once per 8 hours 8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

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(continued)

in order to maintain consistency in the approach to determining offsite doses for certain accident analyses.

12 Not used.

13 The definitions of EMERGENCY FEEDWATER INITIATION AND CONTROL (EFIC) RESPONSE TIME, ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME, and REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME were not incorporated. These terms and the referenced testing were not incorporated into ITS because they are not consistent with CTS. Response time testing of these systems, as required by specifications in NUREG 1430, is not required by CTS Specifications.

14 Not used.

15 Not used.

16 Not used.

17 Not used.

18 Not used.

19 The specific chapter reference in part "a." of the PHYSICS TESTS definition is deleted. ONS is not a Standard Review Plant. Consequently Physic Testing is described in more than one UFSAR Chapter. Removal of the reference to a specific chapter simply ensures that physics testing described in other Chapters of the UFSAR is encompassed by this definition.

20 ONS will maintain the RCS Pressure and Temperature Curves and Limits in the ITS and will not implement a PTLR at this time. Since a PTLR is not implemented, the definition serves no purpose and has been deleted.

1 21 Not used.

1 22 Not used.

23 The second reference provided in the NUREG definition for Dose Equivalent I-131 is deleted. UFSAR 15.14.7 uses the dose conversion factors specified in TID-14844 for the determination of Dose Equivalent I-131.

24 The subscript AVG is deleted to preclude in any potential confusion with Oconee instrumentation that measures reactor coolant temperature. Oconee has a T_{AVG} indication. However, this instrument cannot be used to measure reactor coolant temperature in the range specified in ITS Table 1.1-1.

25 The NUREG Definitions for NUCLEAR HEAT FLUX HOT CHANNEL FACTOR $F_Q(Z)$ and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F\Delta_H^N$) are not adopted since the Specification, 3.2.5, "Power Peaking Factors" which uses these defined terms is not adopted.

26 An appropriate value of 250°F is selected for the MODE 4/MODE 3 transition temperature. There is no CTS equivalent value since the CTS does not include a similar MODE transition point. This temperature is appropriate since it is only slightly above the upper limit for using the Low Pressure Injection System in decay heat removal (DHR) alignment. This value permits use of MODE 4 as a transition MODE wherein required RCS flow may be maintained using either the RCS loops or the DHR loops.

27 The definition for L_a is not adopted in the ITS. The implementation of 10 CFR 50 Appendix J, Option B has resulted in modifications to the ISTS which capture the Leakage Rate Testing Program requirements in ITS paragraph 5.5.2. ITS section 5.5.16 provides a description of L_a which is consistent with the definition of L_a in ISTS Section 1.1, Definitions.

ITS Section 1.0

ID 52

Subject Remove rejected TSTF-39, R1 and incorporate TSTF-205 (R1). Revises definition of CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST.

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.
ALLOWABLE THERMAL POWER	ALLOWABLE THERMAL POWER shall be the maximum reactor core heat transfer rate to the reactor coolant permitted by consideration of the number and configuration of reactor coolant pumps (RCPs) in operation.
AXIAL POWER IMBALANCE	AXIAL POWER IMBALANCE shall be the power in the top half of the core, expressed as a percentage of RATED THERMAL POWER (RTP), minus the power in the bottom half of the core, expressed as a percentage of RTP.
AXIAL POWER SHAPING RODS (APSRs)	APSRs shall be the control components with part length absorbers used to control the axial power distribution of the reactor core. The APSRs are positioned manually by the operator and are not trippable.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel 1 monitors. The CHANNEL CALIBRATION shall encompass 1 all devices in the channel required for channel 1 OPERABILITY 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.

(continued)

1.1 Definitions

CHANNEL CALIBRATION (continued)

The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated.

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 the injection of a simulated or actual signal into the channel
1 as close to the sensor as practicable to verify
1 OPERABILITY of all devices in the channel required
1 for channel OPERABILITY.

The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or
1 total channel steps so that the entire channel is functionally tested.

CONTROL RODS

CONTROL RODS shall be all full length safety and regulating rods.

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, 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 specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific 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

(continued)

A1 <except as marked>

LA2

1.4.4 Reactor Protective System Logic

This system utilizes reactor trip module relays (coils and contacts) in all four of the protective channels as shown in Figure 7.2-1 of the FSAR, to provide reactor trip signals for de-energizing the six control rod drive trip breakers. The control rod drive trip breakers are arranged to provide a one out of two times two logic. Each element of the one out of two times two logic is controlled by a separate set of two out of four logic contacts from the four reactor protective channels.

1.4.5 Engineered Safety Features System

This system utilizes relay contact output from individual channels arranged in three analog sub-systems and two two-out-of-three logic sub-systems as shown in Figure 7.3-1 of the FSAR. The logic sub-system is wired to provide appropriate signals for the actuation of redundant Engineered Safety Features equipment on a two-of-three basis for any given parameter.

A23

1.4.6 Degree of Redundancy

The difference between the number of operable channels and the number of channels which when tripped will cause an automatic system trip.

A2

Supp. 1

~~1.5 INSTRUMENTATION SURVEILLANCE~~

OPERABILITY of all devices in the channel required for channel OPERABILITY

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the channel is tested

~~1.5.1~~ CHANNEL FUNCTIONAL TEST ~~Trip Test~~

A trip test is a test of logic elements in a protective channel to verify their associated trip action.

A9 A27

~~1.5.2~~ Channel FUNCTIONAL Test

<CHANNEL FUNCTIONAL TEST definition as depicted in the ITS>

A channel test is the injection of an internal or external test signal into the channel to verify its proper output response; including alarm and/or trip initiating action where applicable.

as close to the sensor as practicable

or actual

L1

~~1.5.3~~ CHANNEL CHECK ~~Instrument~~ Channel Check

<CHANNEL CHECK definition per ITS>

M3

An instrument channel check is a verification of acceptable instrument performance by observation of its behavior and/or state; this verification includes comparison of output and/or state of independent channels measuring the same variable.

~~1.5.4~~ CHANNEL CALIBRATION ~~Instrument~~ Channel Calibration

<CHANNEL CALIBRATION as depicted in the ITS>

An instrument channel calibration is a test, and adjustment (if necessary), to establish that the channel output responds with ~~acceptable~~ necessary range and accuracy to known values of the parameter which the channel measures or an accurate simulation of these values. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include the channel Functional test.

A23

all devices in the channel required for channel OPERABILITY and

Supp. 1

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.

L2

A24

The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping, or total channel steps so the entire channel is calibrated.

A9 CTS 1.5.1 defines a Trip Test as a test of logic elements in a protective channel to verify their associated trip action. CTS 1.5.2 defines a Channel Test as the injection of an internal or external test signal into the channel to verify its proper output response; including alarm and/or trip initiating action where applicable. The ITS definition of CHANNEL FUNCTIONAL TEST requires the injection of a simulated or actual signal into the channel as close to the sensor as
1 practicable to verify OPERABILITY of all devices in the channel required
1 for channel OPERABILITY. CTS 1.5.1 and 1.5.2 definitions for Channel Test and Trip Test, when combined, are comparable to the NUREG definition of CHANNEL FUNCTIONAL TEST. Therefore, the CHANNEL FUNCTIONAL TEST definition from the NUREG has been adopted in its entirety. This change is administrative and is consistent with the NUREG.

A10 Not used.

A11 CTS definitions comparable to the ITS definitions for ACTIONS, ALLOWABLE THERMAL POWER, AXIAL POWER SHAPING RODS (APSRS), CONTROL RODS, DOSE EQUIVALENT I-131, LEAKAGE, MODE, PHYSICS TEST, SHUTDOWN MARGIN and THERMAL POWER do not exist. ITS states ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times. ITS states ALLOWABLE THERMAL POWER shall be the maximum reactor core heat transfer rate to the reactor coolant permitted by consideration of the number and configuration of reactor coolant pumps (RCPs) in operation. ITS states APSRs shall be the control components with part length absorbers used to control the axial power distribution of the reactor core. The APSRs are positioned manually by the operator and do not trip. ITS states CONTROL RODS shall be all full length safety and regulating rods. ITS states 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 I-131 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."

ITS states LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except 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 not to be pressure boundary LEAKAGE; or

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

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE.

c. Pressure Boundary LEAKAGE

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

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

ITS states 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 the UFSAR;

b. Authorized under the provisions of 10 CFR 50.59; or

c. Otherwise approved by the Nuclear Regulatory Commission.

ITS states 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 (safety and regulating) are fully inserted except for the single CONTROL ROD of highest reactivity worth, which is assumed to be fully withdrawn. With any CONTROL ROD not capable of being fully inserted, the reactivity worth of these CONTROL 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 APSR position.

ITS states THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

This change includes definitions in the ITS that are established in the NUREG but which do not exist as definitions in the CTS. The addition of the definitions is made to make the Improved Technical Specifications (ITS) consistent with the B&W Standard Technical Specification. The addition of the definitions by itself does not add limitations or requirements on the facility and is therefore considered to be an administrative change. This change is consistent with the NUREG.

A12 Not used.

A13 CTS 1.2.4 and 1.2.5 establishes the transition power level between the Hot Standby and Power Operation Reactor Operating Conditions as 2% rated power as indicated on the power range channels (nuclear instrumentation). CTS 1.2.8 does not include a requirement regarding power level. CTS 1.2.5 is comparable to ITS MODE 1 and CTS 1.2.4 combined with CTS 1.2.8 are comparable to ITS MODE 2. ITS MODE 1 and MODE 2 establishes the transition power level as 5% RATED THERMAL POWER in accordance with Table 1.1-1 of the NUREG. The 5% RTP MODE transition criteria is adopted for the purpose of maintaining consistency with the NUREG.

While the change in definition could be a less restrictive change, its affect cannot be adequately evaluated without considering how it is applied in each CTS occurrence. Therefore, the applicability of the Reactor Operating Condition definition changes are evaluated at each occurrence of the defined Reactor Operating Condition in the CTS. Changes to the CTS are discussed on an individual basis with the Specification. Each change is evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis. Where the overall affect was less or more restrictive an appropriate discussion is provided.

A14 The CTS 1.2.6 definition for Refueling Shutdown includes when, even with all rods removed, the reactor would be subcritical by at least 1 percent $\Delta k/k$ and the coolant temperature at the low pressure injection pump suction is no more than 140°F. ITS MODE 6 is when one or more reactor head bolts are not fully tensioned. This change results in the deletion from the definition of the requirement that the reactor be maintained subcritical by 1% $\Delta k/k$ even with all control rods removed and the coolant temperature at the decay heat removal pump suction is at the refueling temperature (normally 140°F). However, ITS LCO 3.9.1 provides controls upon SDM when in MODE 6. The adoption of ITS Specification 3.9.1 evaluates the implications of this change in definition and categorizes the adoption of ITS Specification 3.9.1 and its Bases as more restrictive or less restrictive as appropriate. This change is administrative and is consistent with the NUREG.

A15 CTS provisions comparable to ITS 1.2 do not exist. ITS 1.2 establishes the use and convention for the Logical Connectors used throughout the Improved Technical Specifications (ITS). In addition, ITS 1.2 demonstrates through example the usage of the Logical Connectors. This is an administrative change made to conform to the NUREG convention and is consistent with the NUREG. Technical changes associated with the adoption of these conventions are included in separate discussion of change associated with the individual specifications.

A16 CTS provisions comparable to ITS 1.3 do not exist. ITS 1.3 establishes the use and convention for Completion Times associated with the LCOs throughout the ITS. In addition, ITS 1.3 demonstrates through example the correct interpretation and usage of the Completion Times. This is an administrative change made to conform to the NUREG convention and is consistent with the NUREG. Technical changes associated with the adoption of these conventions are included in separate discussion of change associated with the individual specifications.

A17 CTS provisions comparable to ITS 1.4 do not exist. ITS 1.4 establishes the use and convention of Frequency requirements associated with the Surveillance Requirements throughout the ITS. In addition, ITS 1.4 demonstrates through example the correct interpretation and usage of the Frequency requirements. This is an administrative change made to conform to the NUREG convention and is consistent with the NUREG. Technical changes associated with the adoption of these conventions are included in separate discussion of change associated with the individual specifications.

A18 CTS 1.2.1, 1.2.2 and 1.2.4 use the term T_{avg}. ITS Table 1.1-1 uses the term "Average Reactor Coolant Temperature." No technical or interpretational change exists. This change is administrative and is consistent with the NUREG.

A19 CTS 1.2.4 and 1.2.5 provide definitions comparable to the ITS requirements defining MODE 2 and MODE 1 respectively. These CTS definitions specify power limits in terms of indicated neutron power. ITS Table 1.1-1 prescribes power limits in terms % RATED THERMAL POWER. Indicated neutron power is normalized to calorimetric values. Therefore, indicated neutron power is equivalent to % RATED THERMAL POWER. Therefore this change is administrative and is consistent with the NUREG.

A20 A CTS definition comparable to ITS MODE 4 does not exist. ITS defines MODE 4 as K_{eff} < 0.99 K_{eff} and average reactor coolant temperature < 250°F and > 200°F. When in this condition, CTS has no defined Reactor Operating Condition. This change is consistent with the NUREG presentation of MODES. The applicability of the Reactor Operating Condition definition changes are evaluated at each occurrence of the defined Reactor Operating Condition in the CTS. Changes to the CTS are

discussed on an individual basis with the Specification. Each change is evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis. This change administrative and is consistent with the NUREG.

A21 A provision comparable to last sentence to the ITS definition of CORE ALTERATION does not exist. This sentence states that suspension of CORE ALTERATIONS does not preclude completion of movement of a component to a safe position. The is no CTS requirement directing the suspension of Refueling Operations. Therefore, no relief is provided by the addition of this statement. This change is an administrative change and is consistent with the NUREG.

A22 Not used.

1 A23 CTS 1.5.4 requires the calibration to encompass "the entire channel,
1 including the equipment actuation, alarm, or trip and shall be deemed to
1 include the channel test." CTS 1.5.2 requires the channel test to
1 verify "proper output response, including alarm and/or trip initiating
1 actions where applicable." The ITS definition for CHANNEL CALIBRATION
1 and CHANNEL FUNCTIONAL TEST replaces these phrases with "all devices in
1 the channel required for channel OPERABILITY" and "OPERABILITY of all
1 devices in the channel required for channel OPERABILITY," respectively.
1 This change removes any ambiguity as to whether the list of items to
1 include in the test or calibration is inclusive or representative,
1 clarifying that the components that are required to be tested or
1 calibrated are those that are necessary for the channel to perform its
1 safety function. This change is administrative since it does not change
1 the requirement and is consistent with the NUREG as modified by
1 TSTF-205.

A24 An explicit CTS provision permitting Instrument Channel Calibration to be performed by any sequential, overlapping or total channel steps does not exist. The ITS definition of CHANNEL CALIBRATION explicitly permits the testing to be performed by any sequential, overlapping or total channel steps so the entire channel is calibrated. CTS does not preclude such testing using any sequential, overlapping or total channel steps so the entire channel is calibrated. Therefore, this change is an administrative change and is consistent with the NUREG.

A25 CTS 1.2.1 and 1.2.6 contain a reference to a separate CTS specification which provides requirements regarding RCS pressure. CTS 1.6.2 provides a information regarding a description of neutron power range channel inputs into imbalance instrumentation and information regarding a reference to a separate CTS specification which contains imbalance limits and setpoints. The description and references are not retained in the ITS. No technical or interpretational change exists. This change is administrative and is consistent with the NUREG.

A26 The ONS 1, 2, & 3 CTS Bases are completely replaced by revised bases that reflect the format and applicable content of proposed ITS Section 3.4. The revised Bases are shown in the proposed ONS ITS Bases for Section 3.4.

A27 An explicit CTS provision permitting Instrument Channel Testing to be performed by any sequential, overlapping or total channel steps does not exist. The ITS definition of CHANNEL FUNCTIONAL TEST explicitly permits the testing to be performed by any sequential, overlapping or total channel steps so the entire channel is functionally tested. CTS does not preclude such testing using any sequential, overlapping or total channel steps so the entire channel is calibrated. Therefore, this change is an administrative change and is consistent with the NUREG.

1 **TECHNICAL CHANGES - LESS RESTRICTIVE**

L1 CTS 1.5.2 defines a Channel Test as the injection of an internal or external test signal into the channel to verify its proper output response; including alarm and/or trip initiating action where applicable. The ITS Defines a CHANNEL FUNCTIONAL TEST as the injection of a simulated or actual signal into the channel as close to the sensor
1 as practicable to verify OPERABILITY of all devices in the channel
1 required for channel OPERBILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the channel is functionally tested. The provision to permit use an actual signal is a less restrictive requirement upon unit operation and is consistent with the NUREG. Use of an actual signal does not affect the performance of the channel. OPERABILITY can be adequately demonstrated in either case since the channel itself does not discriminate between an "actual" or "simulated" signal.

L2 CTS 1.5.4 requires an Instrument Channel Calibration to encompass the entire channel and does not exclude RTDs and thermocouples. The ITS definition of CHANNEL CALIBRATION allows performing "...an in place qualitative assessment of sensor behavior..." for these devices. This change is a less restrictive requirement upon unit operations and is consistent with the NUREG. A qualitative assessment of sensor behavior is acceptable for RTDs and thermocouples since the operation of these devices is governed by well understood and predictable physical relationships between the temperature of the sensed medium and the output of the RTD or thermocouple. Additionally, the output of RTDs and thermocouples is not adjustable. These devices are reliable and not subject to drift in the same manner as other sensors. As a result a qualitative assessment of sensor behavior is sufficient to determine its OPERABILITY and acceptability for continued use.

L3 CTS 1.2.8 defines Startup as a reduction in the shutdown margin with the intent of going critical. ITS MODE 2 is comparable to the CTS condition of startup. ITS MODE 2 is specified as when $K_{eff} \geq 0.99$ and THERMAL POWER is < 5%. The elimination of requirements associated with the CTS Startup Condition when shutdown margin is being reduced and $K_{eff} < 0.99$ (i.e., when in ITS MODES 3, 4 and 5) is a less restrictive requirement upon unit operation and is consistent with the NUREG. When the unit is in MODES 3, 4 or 5 shutdown margin is controlled by ITS Specification
1 3.4.1. ITS 3.4.1 ensures appropriate control of shutdown margin when in ITS MODES 3, 4 or 5.

L4 CTS 1.2.4 and 1.2.5 require use of the power range neutron channels as the indication to be used to establish the transition between Hot Standby and Power Operation. ITS Table 1.1-1 uses % RATED THERMAL POWER as the measure of reactor power used to establish the transition point between MODES 1 and 2. Although, the power range neutron channels may still be used since they are normalized to the calorimetric power

LESS RESTRICTIVE CHANGE L1

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

CTS 1.5.2 defines a Channel Test as the injection of an internal
or external test signal into the channel to verify its proper
output response; including alarm and/or trip initiating action
where applicable. The ITS Defines a CHANNEL FUNCTIONAL TEST as
the injection of a simulated or actual signal into the channel as
1 close to the sensor as practicable to verify OPERABILITY of all
1 devices in the channel required for channel OPERBILITY. The
CHANNEL FUNCTIONAL TEST may be performed by means of any series of
sequential, overlapping, or total channel steps so that the
channel is functionally tested. The provision to permit use an
actual signal is a less restrictive requirement upon unit
operation and is consistent with the NUREG. Use of an actual
signal does not affect the performance of the channel.
OPERABILITY can be adequately demonstrated in either case since
the channel itself does not discriminate between an "actual" or
"simulated" signal.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change allows use of an actual signal to be used in the CHANNEL FUNCTIONAL TEST. This change does not result in any hardware changes. The CHANNEL FUNCTIONAL TEST is not considered as the initiator of any previously analyzed accident. As such, the probability of an accident is independent of the signal used to perform CHANNEL FUNCTIONAL TESTING. Also, the change does not change the assumed response of the equipment in performing its specified mitigation functions from that originally considered. The consequences are not changed since the instrument channel functions the same, regardless of the signal used to perform the CHANNEL FUNCTIONAL TEST. Therefore, the change does not significantly increase the probability or consequences of an accident.

CTS

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.	Doc A11
ALLOWABLE THERMAL POWER	ALLOWABLE THERMAL POWER shall be the maximum reactor core heat transfer rate to the reactor coolant permitted by consideration of the number and configuration of reactor coolant pumps (RCPs) in operation.	Doc A11
AXIAL POWER IMBALANCE	AXIAL POWER IMBALANCE shall be the power in the top half of the core, expressed as a percentage of RATED THERMAL POWER (RTP), minus the power in the bottom half of the core, expressed as a percentage of RTP.	1.6.2 5
AXIAL POWER SHAPING RODS (APSRs)	APSRs shall be control components with part length absorbers used to control the axial power distribution of the reactor core. The APSRs are positioned manually by the operator and are not trippable.	Doc A11
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass ~~the entire channel, including the required sensor, alarm, display, and trip functions, and shall include~~ all devices in the channel required for channel OPERABILITY and 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~~	1.5.4 TSTF 205 TSTF 19, R1 Supp 1

(continued)

1.1 Definitions

TSTF 19,R1

CHANNEL CALIBRATION (continued)

~~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 that the entire channel is calibrated. 1.5.4

Supp 1

~~The CHANNEL CALIBRATION shall also include testing of safety related Reactor Protection System (RPS), Engineered Safety Feature Actuation System (ESFAS), and Emergency Feedwater Initiation and Control (EFIC) bypass functions for each channel affected by the bypass operation.~~ TSTF 124

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. 1.5.3

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be 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, display, and trip functions~~ of all devices in the channel required for channel OPERABILITY. ~~The ESFAS CHANNEL FUNCTIONAL TEST shall also include testing of ESFAS safety related bypass functions for each channel affected by bypass operation.~~ 1.5.1 1.5.2

TSTF-205

Supp 1

TSTF 124

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

Doc A11

CONTROL RODS

CONTROL RODS shall be all full length safety and regulating rods ~~that are used to shut down the reactor and control power level during maneuvering operations~~.

9

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE 1.2.7

TSTF 205

Supp 1

Supp 1

(continued)

ITS Section 1.0

ID 53

Subject Incorporate TSTF-124 which deletes paragraph from CHANNEL CALIBRATION & CHANNEL FUNCTIONAL TEST definitions related to testing of bypass functions. This had already been incorporated as a plant specific change so only change is deletion of JFD & markup.

1.1 Definitions

TSTF 19,R1

CHANNEL CALIBRATION (continued)	~~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 that the entire channel is calibrated.	1.5.4
Supp 1	~~The CHANNEL CALIBRATION shall also include testing of safety related Reactor Protection System (RPS), Engineered Safety Feature Actuation System (ESFAS), and Emergency Feedwater Initiation and Control (EFIC) bypass functions for each channel affected by the bypass operation.~~	TSTF 124
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.	1.5.3
CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be 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, display, and trip functions~~ of all devices in the channel required for channel OPERABILITY. ~~The ESFAS CHANNEL FUNCTIONAL TEST shall also include testing of ESFAS safety related bypass functions for each channel affected by bypass operation.~~	1.5.1 1.5.2 TSTF-205 TSTF 124
Supp1		
CONTROL RODS	CONTROL RODS shall be all full length safety and regulating rods ~~that are used to shut down the reactor and control power level during maneuvering operations~~.	Doc A11 9
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE	1.2.7

Supp 1 TSTF 205

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

Supp 1

(continued)

TECHNICAL SPECIFICATIONS

NOTE: The first four justifications for these changes from NUREG-1430 were generically used throughout the individual LCO section markups. Not all generic justifications are used in each section.

1. The brackets are removed and the proper plant specific information or value is provided.

2. Not used.

3. Not used.

4. Not used.

5 The definition of AXIAL POWER SHAPING RODS (APSRs) has been modified to specify that these are control components with part-length absorbers. The ONS design provides control components with part-length absorbers. This specifically excludes the full length control components (regulating rods) when they are being used to control the axial power distribution of the reactor.

6 Not used.

1 7 Not used.

1 8 Not used.

9 The definition of CONTROL RODS is modified to eliminate the overly prescriptive description of the function of the CONTROL RODS. The definition as written, if literally interpreted, prevents these reactivity control components from being used to startup the reactor, control xenon oscillations and control reactor imbalance, etc. Additionally, the NUREG and ITS usage of the term CONTROL RODS is intended to classify a component for the purposes of application of LCOs and SRs and not to prescribe the function of the devices. [ONS-005]

10 Not used.

11 The CLB regarding the half lives of nuclides included in the E-bar determination is retained. The definition is changed from including nuclides with half lives longer than 15 minutes to including nuclides with half lives longer than 30 minutes. The CLB definition is retained

2.0 Safety Limits

ITS Section 2.0

ID 17

Subject Revise DOC M3 on page 2 of DOCs for Section 2.0 - should say RCS pressure to be restored within 5 minutes, not 15 minutes.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 The CTS applicability for the Reactor Core Safety Limits (CTS 2.1) is "during power operation." CTS 1.2.5 defines Power Operation as "...when the indicated neutron power range is above 2 percent of rated power as indicated on the power range channels." The ITS Applicability for Reactor Core Safety Limits is MODES 1 and 2. Thus, the ITS APPLICABILITY is more restrictive since MODE 2 is defined as ≤ 5% RTP with $k_{eff} \geq 0.99$. The inclusion of MODE 2 is appropriate because the reactor is critical in MODE 2 and limiting accidents and transients are postulated to begin in these MODES. The proposed change is consistent with the NUREG.

M2 CTS 6.3.1 requires the reactor to be shut down immediately and maintained in a safe shutdown condition until the Commission authorizes resumption of operation when a Safety Limit is violated. ITS 2.2.2 requires restoring compliance within limits when the Safety Limit for RCS Pressure (ITS 2.1.2) is exceeded. The addition of this requirement is appropriate in that it reduces the potential for exceeding the design pressure. The addition of this more restrictive action is consistent with the NUREG.

M3 CTS 6.3.1 does not establish specific required actions should the RCS Pressure Safety Limit be violated in MODES 3, 4, and 5. ITS 2.2.3
1
requires RCS pressure be restored within 5 minutes when the Safety Limit is violated. This represents a more restrictive requirement than that currently imposed. This more restrictive requirement is considered appropriate since exceeding the Safety Limit in MODE 3, 4, or 5 is potentially more severe than exceeding this Safety Limit in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. The proposed change is consistent with the NUREG.

ITS Section 2.0

ID 18

Subject Revise NUREG markup for 2.2.4 to correctly crossreference DOC M3 instead of M4.

CTS

2.0 SLs

2.2 SL Violations (continued)

2.2.~~3~~2 In MODE 1 or 2, if SL 2.1.2 is violated ~~not met~~, restore compliance within limits and be in MODE 3 within 1 hour. 6.3.1

4

Supp 1

2.2.~~4~~3 In MODES 3, 4, and 5, if SL 2.1.2 is violated ~~not met~~, restore RCS pressure to ≤ ~~[~~2750~~]~~ psig within 5 minutes. DOC M3

2

~~2.2.5 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.~~

~~2.2.6 Within 24 hours, notify the [Vice President – Nuclear Operations].~~

~~2.2.7 Within 30 days, a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC and the [Plant Superintendent and Vice President – Nuclear Operations].~~ TSTF-005, R1

2.2.8 Operation of the plant shall not be resumed until authorized by the NRC.

ITS Section 2.0

ID 30

Subject Revise ITS 2.1.1.1 & 2.1.1.2 to include reference to AXIAL POWER IMBALANCE and RCS Variable Low Pressure Protective limits ensuring compliance to SLs.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the maximum local fuel pin centerline temperature shall be ≤ 5080 - (6.5 x 10^{-3} x (Burnup,
1 MWD/MTU))°F. Operation within this limit is ensured by
1 compliance with the Axial Power Imbalance Protective
1 Limits as specified in the Core Operating Limits Report.

2.1.1.2 In MODES 1 and 2, the departure from nucleate boiling ratio shall be maintained greater than the limit of 1.18
1 for the BWC correlation. Operation within this limit is
1 ensured by compliance with the Axial Power Imbalance
1 Protective Limits and RCS Variable Low Pressure
1 Protective Limits as specified in the Core Operating
1 Limits Report.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2750 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 In MODE 1 or 2, if SL 2.1.1.1 or SL 2.1.1.2 is violated, be in MODE 3 within 1 hour.

2.2.2 In MODE 1 or 2, if SL 2.1.2 is violated, restore compliance within limits and be in MODE 3 within 1 hour.

2.2.3 In MODES 3, 4, and 5, if SL 2.1.2 is violated, restore RCS pressure to ≤ 2750 psig within 5 minutes.

A1 <Except as marked>

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

2.1 ~~SAFETY LIMITS,~~ REACTOR CORE SLs

Applicability

Applic MODES 1+2

Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

In MODES 1 + 2 — M1

Objective

To maintain the integrity of the fuel cladding.

Specification

≤ — A3

Supp. 1

2.1.1.1 The maximum local fuel pin centerline temperature shall be less than 5080 - $(6.5 \times 10^{-3}) \times$(Burnup, MWD/MTU) °F. Operation within this limit is ~~assured~~ ensured by compliance with the Axial Power Imbalance Protective Limits as specified in the Core Operating Limits Report.

Suppl

2.1.1.2 The DNBR shall be maintained greater than the correlation limits of ~~1.3 for BAW-2 and~~ 1.18 for BWC. Operation within this limit is ~~assured~~ by compliance with the Axial Power Imbalance Protective Limits and Variable Low RCS Pressure Protective Limits as specified in the Core Operating Limits Report. — A4

Bases — A2

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions and anticipated transients. This is accomplished by operating within the nuclear boiling heat transfer regime where the heat transfer coefficient is large and the cladding temperature is only slightly greater than the coolant temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation, but neutron power and reactor coolant pressure and temperature can be related to DNB using a critical heat flux (CHF) correlation. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual local heat flux, is indicative of the margin to DNB.

The BAW-2 and BWC CHF correlations[1,2] have been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC). A DNBR of 1.30 (BAW-2) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur.

TECHNICAL CHANGES - REMOVAL OF DETAILS

1 LA1 Not used.

LA2 CTS 6.3 requires actions prescribed by 6.3.1 - 6.3.4 to be taken when a Safety Limit is violated. CTS 6.3.2 requires a Safety Limit violation to be reported to the Commission, the Site Vice President, and the Director of the Nuclear Safety Review Board. The CTS 6.3.3 requirement to have the written report of the Safety Limit Violation reviewed by the Operations Superintendent and the Station Manager and submitted to the Site Vice President and the Director of the Nuclear Safety Review Board is relocated to the Quality Assurance Topical Report. The CTS 6.3.2 requirements for reporting the Safety Limit violation to the Site Vice President and the Director of the Nuclear Safety Review Board are relocated to the Quality Assurance Topical Report. These details are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of these details to the Quality Assurance Topical Report provides reasonable assurance that the details are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Therefore, relocation of these details is acceptable. Changes to the Quality Assurance Topical Report are controlled by the provisions of 10 CFR 50.54. This change is consistent with the NUREG, as modified by TSTF-005, Revision 1.

CTS

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

X [Burnup] (1)

2.1.1.1 In MODES 1 and 2, the maximum local fuel pin centerline temperature shall be ≤ ~~[~~5080 - (6.5 x 10^{-3} ~~/~~MWD/MTU)~~]~~ °F. Operation within this limit is ensured by compliance with the AXIAL POWER IMBALANCE protective limits ~~preserved by the Reactor Protection System setpoints in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation,"~~ as specified in the COLR. (3) 2.1 TSTF 126

Supp.1 (3) and RCS Variable Low Pressure Protective Limits

2.1.1.2 In MODES 1 and 2, the departure from nucleate boiling ratio shall be maintained greater than the limits of ~~[1.3 for the BAW-2 correlation and~~ 1.18 for the BWC correlation~~]~~. Operation within this limit is ensured by compliance ~~with SL 2.1.1.3 and~~ with the AXIAL POWER IMBALANCE protective limits ~~preserved by the RPS setpoints in LCO 3.3.1,~~ as specified in the COLR. (2) 2.1 TSTF 126

~~2.1.1.3 In MODES 1 and 2, Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the SL shown in Figure 2.1.1-1.~~ (3)

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ ~~[~~2750~~]~~ psig. (2) 2.2.1

2.2 SL Violations

With any SL violation, the following actions shall be completed: 6.3

2.2.1 In MODE 1 or 2, if SL 2.1.1.1 or SL 2.1.1.2 is violated, be in MODE 3 within 1 hour. 6.3.1

~~2.2.2 In MODE 1 or 2, if SL 2.1.1.3 is violated, restore RCS pressure and temperature within limits and be in MODE 3 within 1 hour.~~ (3)

(continued)

TECHNICAL SPECIFICATIONS

1 The plant specific information from current technical specification (CTS) 2.1 for maximum local fuel pin centerline temperature was inserted in SL 2.1.1.1. This information is consistent with the ONS current licensing basis.

2 Brackets removed and appropriate plant specific information provided. CTS 2.1 currently specifies that the DNBR shall be maintained greater than the correlation limits of 1.3 for BAW-2 and 1.18 for BWC. However, since the BAW-2 correlation is no longer applicable (Oconee no longer uses Mark B fuel) it is not included in the ITS. Refer to Discussion of Change (DOC) A4 (Attachment 2 for this section).

1 3 NUREG Safety Limit 2.1.1.3 is not included in the ONS ITS. CTS does not
1 have a curve similar to that provided by NUREG 2.1.1.3 and uses a curve
1 based on the RCS Variable Low Pressure Protective Limit. Also, because
1 of this difference, TSTF-126 will not be adopted and the existing CTS
1 statement that compliance with the AXIAL POWER IMBALANCE Protective
1 Limits and the RCS Variable Low Pressure Protective Limits specified in
1 the COLR ensures compliance with the SL will be retained in SL 2.1.1.2.
1 The existing CTS statement that compliance with the AXIAL POWER
1 IMBALANCE Protective Limits specified in the COLR ensures compliance
1 with the SL will be retained in SL 2.1.1.1.

4 The wording in NUREG 2.2.3 and 2.2.4 was modified to be consistent with the wording used in NUREG 2.2.1. The words "not met" were replaced with the word "violated." This change precludes the potential misinterpretation of an unintended distinction, is administrative in nature and has been made for consistency with similar ITS Specifications. The proposed wording change is consistent with the other Standard Technical Specification NUREGs and Crystal River Unit 3 Technical Specifications, a B&W plant that has already converted to ITS.

3.0 - LCO APPLICABILITY

ITS Section 3.0

ID 27

Subject Revise LCO 3.0.5 to reinstate ", or" consistent with the NUREG. Also, remove incorrect crossref. on NUREG M/U (incorrectly xref from LCO 3.0.4 to CTS 4.0.3)

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued)

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.

Exceptions to this Specification are stated in the individual Specifications.

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
1 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, an evaluation shall be performed in accordance with Specification 5.5.16, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and

(continued)

CTS

3.0 LCO APPLICABILITY

Supp 1

LCO 3.0.4 (continued)

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.

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 operation in the MODE or other specified condition in the Applicability only for a limited period of time.~~

TSTF 104

LCO ~~SR~~ 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.

Edit

~~[Reviewer'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, 3, and 4. 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.]~~

9

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.

DOC A5

Supp 1

(continued)

ITS Section 3.0

ID 43

Subject Remove ONS-012 (BWOG-70), withdrew by BWOG. This change removed incorrect bracketed references. JFD revised to correct on a plant specific basis.

TECHNICAL SPECIFICATIONS

NOTE: The first four justifications for these changes from NUREG-1430 were generically used throughout the individual LCO section markups. Not all generic justifications are used in each section.

1. The brackets are removed and the proper plant specific information or value is provided.

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

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

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

5 NUREG LCO 3.0.7 refers to LCOs 3.1.10, 3.1.11 and 3.4.19. These LCOs
1 are not in the ITS. Therefore, reference to them has been deleted. The
1 brackets are removed and the plant specific reference is provided.

6 Not used.

7 Not used.

8 Not used.

9 Bracketed reviewers Note is deleted.

10 LCO 3.0.3 requirements regarding the time to be in MODE 3 is changed to 12 hours to reflect the CLB. Should a Condition require multiple unit shutdown, the additional time provides for minimizing challenges to plant systems and personnel associated with a multi unit shutdown. Additionally, for a single unit shutdown, the 12 hours provides some additional time to potentially correct any Condition necessitating the shutdown. Any increased risk associated with the extension of time to be in MODE 3 is at least partially offset by a reduction in risk associated with averted plant shutdowns and the averted potential for shutdown transients. Additionally, the LCO 3.0.3 time to attain MODE 4 is modified to 18 hours to reflect the additional time allowed to attain MODE 3. The time to attain MODE 5 is not modified.

ITS Section 3.0

ID 49

Subject Incorporate TSTF-071, R1 - Adds examples for SFDP.

BASES

LCO 3.0.6 (continued)

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 remaining OPERABLE support systems are OPERABLE, thereby ensuring safety function is retained.

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or (EXAMPLE B3.06-1)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or (EXAMPLE B3.06-2)

c. A required system redundant to support system(s)for the supported systems (a) and (b) above is also inoperable. (EXAMPLE B3.06-3)

<u>EXAMPLE B3.06-1</u>

If System 2 of Train A is inoperable, and System 5 of Train B is inoperable, a loss of safety function exists in supported System 5.

<u>EXAMPLE B3.06-2</u>

If System 2 of Train A is inoperable, and System 11 of Train B is inoperable, a loss of safety function exists in System 11 which is in turn supported by System 5.

<u>EXAMPLE B3.06-3</u>

If System 2 of Train A is inoperable, and System 1 of Train B is inoperable, a loss of safety function exists in Systems 2, 4, 5, 8, 9, 10 and 11.

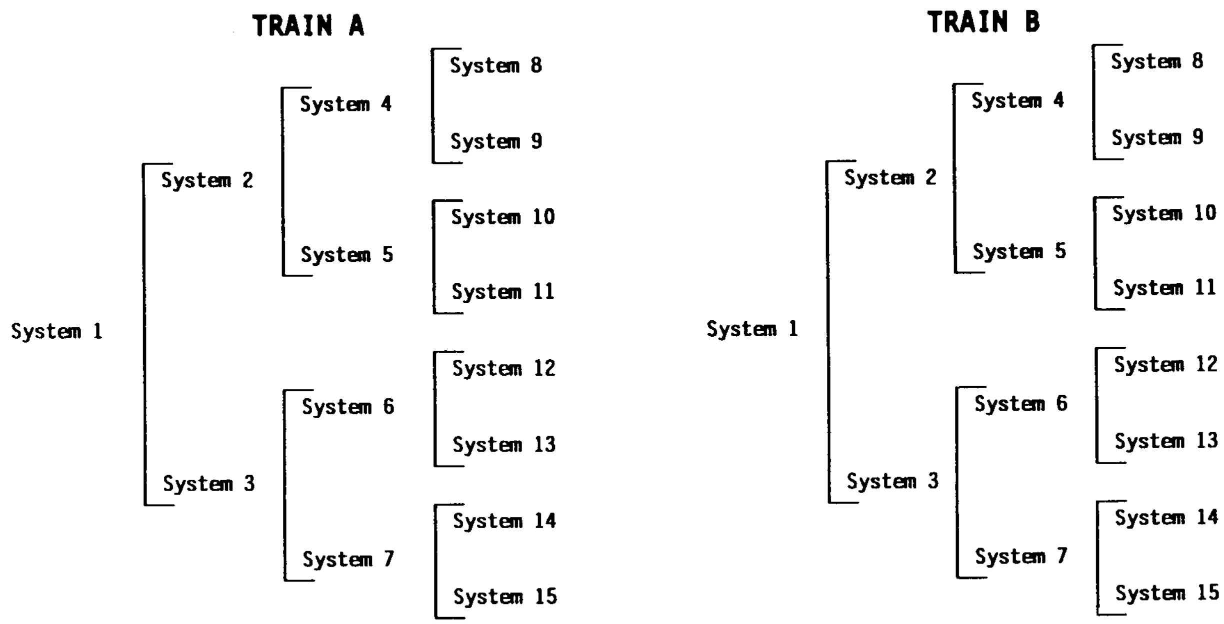
If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

(continued)

BASES

LCO 3.0.6
(continued)

EXAMPLES



LCO 3.0.7 There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCO 3.1.8 allows specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

(continued)

BASES

LCO 3.0.7 (continued)	The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

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 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 an Exception LCO are only applicable when the Exception LCO is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

(continued)

BASES

SR 3.0.1 (continued)

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.

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 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 example of this process are:

a. Emergency feedwater (EFW) pump turbine maintenance during refueling that requires testing at steam pressures > 300 psi. However, if other appropriate testing is satisfactorily completed, the EFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed while the plant reaches the steam pressure required to perform the EFW pump testing.

b. High Pressure Injection (HPI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPI 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

(continued)

BASES

SR 3.0.2 (continued)

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).

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 "inaccordance with 10 CFR 50, Appendix J, as modified byapproved exemptions." The requirements of regulations takeprecedence 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.

(continued)

BASES

SR 3.0.2 (continued)

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.

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 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 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

(continued)

BASES

SR 3.0.3 (continued)

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.

Satisfactory 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. 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

(continued)

BASES

SR 3.0.4 (continued)

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 entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit 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 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability associated with operation in MODES 1, 2, 3, or 4. The requirements of SR 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

BASES

LCO 3.0.6 (continued)

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 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 in Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

16

Specification 5.5.15 "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 remaining OPERABLE support systems are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and

TSTF-71, R1

Supp.1

INSERT B 3.08A

(continued)

INSERT B 3.0-8A

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or (EXAMPLE B3.06-1)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or (EXAMPLE B3.06-2)

c. A required system redundant to support system(s)for the supported systems (a) and (b) above is also inoperable. (EXAMPLE B3.06-3)

EXAMPLE B3.06-1

If System 2 of Train A is inoperable, and System 5 of Train B is inoperable, a loss of safety function exists in supported System 5.

EXAMPLE B3.06-2

If System 2 of Train A is inoperable, and System 11 of Train B is inoperable, a loss of safety function exists in System 11 which is in turn supported by System 5.

EXAMPLE B3.06-3

If System 2 of Train A is inoperable, and System 1 of Train B is inoperable, a loss of safety function exists in Systems 2, 4, 5, 8, 9, 10 and 11.

Supp. 1

TSTF-71A

INSERT B 3.0-9A

BASES

LCO 3.0.6 (continued)

Required Actions of the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.7

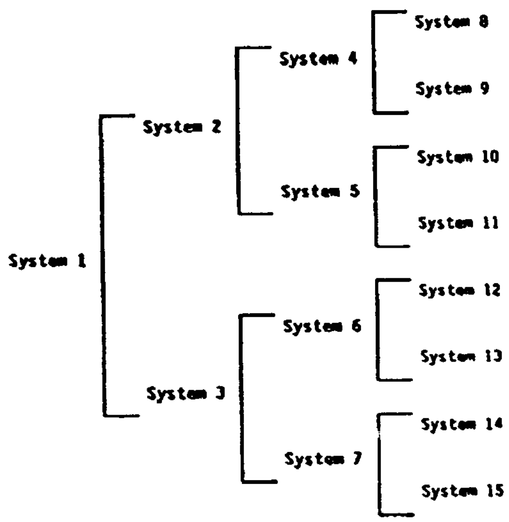
There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, ~~and to perform special evolutions~~. Test Exception LCO~~s~~ ~~[~~3.1.9 ~~3.1.10, 3.1.11, and 3.4.9]~~ allow~~s~~ specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect. 8 1

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

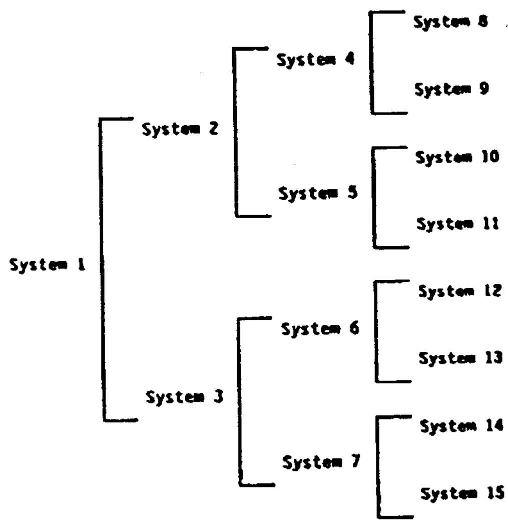
INSERT B 3.0-9A

EXAMPLES

TRAIN A



TRAIN B



3.1 Reactivity Control

ITS Section 3.1

ID 36

Subject Revise ITS to incorporate approved TSTF-110, R2 which eliminates the increased SR Freq associated wtih inoperable alarms. No impact to ITS since Oconee had retained CTS which is the same Freq. as modified NUREG. Only affects markup and JFDs.

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.1.4.1	Verify individual CONTROL ROD positions are within ~~[6.5]~~ 9% of their group average height. (1)	~~4 hours when the asymmetric CONTROL ROD alarm is inoperable AND~~ 12 hours ~~when the asymmetric CONTROL ROD alarm is OPERABLE~~ TSTF 110, R2	DOC M9
SR 3.1.4.2	Verify CONTROL ROD freedom of movement (trippability) by moving each individual CONTROL ROD that is not fully inserted ~~≥ 3%~~ ~~in any direction~~. (8)	92 days	T 4.1-2 Item 1
SR 3.1.4.3	---NOTE--- With rod drop times determined with at least one but less than four reactor coolant pumps operating, operation may proceed provided operation is restricted to the pump combination operating during the rod drop time determination or a pump combination providing less total reactor coolant flow. (22) --- Verify the rod drop time for each CONTROL ROD, from the fully withdrawn position, is ≤ ~~[1.66]~~ 1.4 seconds from power interruption at the CONTROL ROD drive breakers to 3/4 insertion (25% withdrawn position) ~~with $T_{avg} \geq 525°F$~~. (1) (25)	Prior to reactor criticality after each removal of the reactor vessel head	DOC L6 4.7.1

Supp. 1

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify position of each APSR is within [6.5]% of the group average height. R ①	~~4 hours when the asymmetric CONTROL ROD alarm is inoperable~~ ~~AND~~ 12 hours ~~when the asymmetric CONTROL ROD alarm is OPERABLE~~

DOC M13

TSTF 110, R2

Supp. 1

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.7.1	Verify ~~the~~ absolute position indicator channel~~s~~ and ~~the~~ relative ~~position~~ ~~indicator~~ channels agree ~~within the limit specified in the COLR.~~ Perform CHANNEL CHECK of required position indicator channel. (12)	~~4 hours when the asymmetric CONTROL ROD alarm is inoperable AND 12 hours when the asymmetric CONTROL ROD alarm is OPERABLE~~

Table 4.1-1 Items 23 & 24

TSTF 110, R2

Supp. 1

TECHNICAL SPECIFICATIONS

1 The brackets are removed and the proper plant specific information or value is provided.

2 Changes are made (additions, deletions, and/or changes to the NUREG) to reflect the facility specific nomenclature, number (including renumbering due to deletion or addition), reference, system description, or analysis description.

3 In SR 3.1.3.1, the phrase "within the upper limit specified in the COLR" is changed to "within the limits specified in the COLR." The term "upper" is not needed since the ITS does not include a lower MTC limit (refer to JFD 4 below). The COLR provides the appropriate limits at all power levels.

4 NUREG SR 3.1.3.2 is not included as a requirement in the ITS. CTS does not specify a lower limit on MTC. Although ONS has performed a verification of EOC MTC as a one time commitment, this verification is not performed each cycle. The change is consistent with the current licensing basis.

5 The NUREG value for nuclear overpower trip setpoint is replaced with the appropriate plant specific value.

6 NUREG-1430 Specification 3.1.4 Required Action A.2.5 and associated Completion Time is not included in the ONS ITS. This item is not a CTS requirement for this Condition; and therefore, is not being adopted. By not adopting these changes, requirements consistent with current license bases are being maintained. Removing the requirement to perform NUREG SR 3.2.5.1 while operating with a misaligned CONTROL ROD is appropriate since the CTS and ITS require the performance of NUREG SR 3.2.4.1, determination of QUADRANT POWER TILT, while operating with a misaligned CONTROL ROD above 20% RTP. Performance of NUREG SR 3.2.4.1 ensures that any adverse trends in QPT, whether due to a misaligned CONTROL ROD or any other cause, are recognized within 2 hours. It further provides reasonable assurance that the potentially non-conservative CONTROL ROD position is not adversely affecting the power distribution within the core.

1 7 Not used.

8 The portion of NUREG-1430 SR 3.1.4.2 that requires moving each CONTROL ROD "≥ 3%" is not included in the ONS ITS. CTS 4.1-2, Item 1 does not specify how much to move the CONTROL RODS. The minimum distance a CONTROL ROD must be moved during testing is different between units. This is due to design differences between Units 1 & 2 and Unit 3. The CONTROL RODS on Units 1 & 2 are moved further to exercise the CRD fully through an area of thermal binding inherent in the design. This minimum distance is currently specified by procedure. This change preserves the

BASES

ACTIONS (continued)

D.1.1 and D.1.2

When one or more rods are untrippable, the SDM may be adversely affected. Under these conditions, it is important to determine the SDM and, if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration to restore SDM.

In this situation, SDM verification must include the worth of the untrippable rod as well as a rod of maximum worth.

D.2

unit — 4

If the untrippable rod(s) cannot be restored to OPERABLE status, the ~~plant~~ must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the ~~plant~~ must be brought to at least MODE 3 within ~~6~~ 12 hours. 1 unit — 4

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. unit — 4

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

CONTROL — 4

Verification that individual CONTROL rods are aligned within [6.5]% 2 of their group average height limits at a 12 hour Frequency allows the operator to detect a rod that is beginning to ~~deviate from its expected position.~~ If the asymmetric CONTROL ROD alarm is inoperable, a frequency of 4 hours is reasonable to prevent large deviations in CONTROL ROD alignment from occurring without detection. TSTF 110,R1 The specified ~~Frequency takes into account other~~ rod position information that is continuously available to the operator in the control room, so that during actual CONTROL rod motion, deviations can immediately be detected. 4

Supp.1

(continued)

BASES

ACTIONS

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

Required Action A.1 ~~is~~ only practical ~~for~~ instances where small movements of the APSR group are sufficient to re-establish APSR alignment. — 1

A.2

The reactor may continue in operation with the APSR inoperable or misaligned if further movement of the APSR group is prohibited, so that ~~the~~ misalignment does not increase and cause the limits on AXIAL POWER IMBALANCE to be exceeded. The required Completion Time of up to 2 hours will not cause significant xenon redistribution to occur. — 25

B.1

The ~~plant~~ unit must be brought to a MODE in which the LCO does not apply if the Required Actions and associated Completion Times cannot be met. To achieve this status, the ~~plant~~ unit must be brought to at least MODE 3 within ~~6~~ 12 hours. The Completion Time of ~~6~~ 12 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging ~~plant~~ unit systems. In MODE 3, APSR ~~group~~ alignment limits are not required because the reactor is not generating significant THERMAL POWER and excessive local LHRs cannot occur from APSR misalignment. — 12 — 1 — 4

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Verification at a 12 hour Frequency that individual APSR positions are within 6.5% of the group average height limits allows the operator to detect an APSR beginning to deviate from its expected position. If the asymmetric CONTROL ROD alarm is inoperable, a 4 hour Frequency is reasonable to prevent large deviations in APSR alignment from occurring without detection. In addition, APSR position is continuously available to the operator in the control room so that during actual ~~rod~~ APSR motion, deviations can immediately be detected. — 2 — TSTF 110, R2 — 4

Supp. 1

(continued)

BASES

ACTIONS (continued)

A.1
~~C.1~~

required / is

If ~~both~~ the ~~absolute~~ position indicator channel ~~and relative position indicator channel are~~ inoperable for one or more rods, ~~or if the Required Actions and associated Completion Times are not met~~, the position of the ~~rod(s)~~ is not known with certainty. Therefore, each affected ~~rod~~ must be declared inoperable, and the limits of LCO 3.1.4 or LCO 3.1.6 apply. The required Completion Time for declaring the rod(s) inoperable is immediately. Therefore, LCO 3.1.4 or LCO 3.1.6 is entered immediately, and the required Completion Times for the appropriate Required Actions in those LCOs apply without delay.

CONTROL ROD or APSR

1

4

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

INSERT B 3.1-44A

1

Supp.1

~~Verification is required that the Absolute Position Indicator channels and Relative Position Indicator channels agree within the limit given in the COLR. This verification ensures that the Relative Position Indicator channels, which are regarded as the potentially less reliable means of position indication, remain OPERABLE and accurate.~~ The required Frequency of 12 hours is adequate for verifying that no degradation in system OPERABILITY has occurred. ~~If the asymmetric CONTROL ROD alarm is inoperable, then the Surveillance is performed every 4 hours. This required Frequency is adequate for ensuring that the CONTROL RODS and APSRs do not exceed their alignment limits.~~

1

TSTF 110, R2

REFERENCES

1. UFSAR, Section 3.1, ~~10 CFR 50, Appendix A, GDC 13~~. Chapter 15.
2. UFSAR, ~~Section [14.1.2.2], Section [14.1.2.3], Section [14.1.2.6], Section [14.1.2.7], Section [14.2.2.4], and Section [14.2.2.5]~~.
3. 10 CFR 50.36.

4

ITS Section 3.1

ID 41

Subject Incorporate revision to ONS-017 (BWOG-84) - Revises RA A to require imbalance to be determined within 2 hours when APSR inoperable or not aligned.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 AXIAL POWER SHAPING ROD (APSR) Alignment Limits

LCO 3.1.6 Each APSR shall be OPERABLE and aligned within 6.5% of its group average height.

APPLICABILITY: MODES 1 and 2.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
1 1 1 1 1 1 1	A. One APSR inoperable, not aligned within its limits, or both.	A.1 Perform SR 3.2.2.1.	2 hours AND 2 hours after each APSR movement
	B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

BASES (continued)

APPLICABILITY

The requirements on APSR OPERABILITY and alignment are applicable in MODES 1 and 2, because these are the only MODES in which significant neutron (or fission) power is generated, and the OPERABILITY and alignment of APSRs have the potential to affect the safety of the unit. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down and excessive local LHRs cannot occur from APSR misalignment.

1 ACTIONS

A.1

1

1 The ACTIONS described below are required if one APSR is declared inoperable due to inoperable position indication or is misaligned. The unit is not allowed to operate with more than one inoperable or misaligned APSR. This would require the reactor to be placed in MODE 3, in accordance with LCO 3.0.3.

An alternative to realigning a single inoperable or misaligned APSR to the group average position is to align the remainder of the APSR group to the position of the inoperable or misaligned APSR. This restores the alignment requirements. Deviations up to 2 hours will not cause significant xenon redistribution to occur.

The reactor may continue in operation with the APSR
1 inoperable or misaligned if the limits on AXIAL POWER
1 IMBALANCE are surveilled within 2 hours to determine if the
1 AXIAL POWER IMBALANCE is still within limits. Also, since
1 any additional movement of the APSRs may result in
1 additional imbalance, Required Action A.1 also requires the
1 AXIAL POWER IMBALANCE surveillance to be performed again
1 within 2 hours after each APSR movement. The required
Completion Time of up to 2 hours will not cause significant xenon redistribution to occur.

B.1

The unit must be brought to a MODE in which the LCO does not apply if the Required Actions and associated Completion Times cannot be met. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours. The Completion Time of 12 hours is reasonable, based on

(continued)

proposed ONS ITS. This loss in flexibility is considered more restrictive and is consistent with the NUREG. Measuring control rod insertion times during flow conditions is more conservative and more appropriate since testing with reactor pumps operating simulates a reactor trip under actual conditions.

M11 CTS 4.7.1.c requires CONTROL ROD drop time testing following each refueling outage. ITS SR 3.1.4.3 Frequency is "Prior to reactor criticality after each removal of the reactor vessel head." Measuring CONTROL ROD drop times prior to reactor criticality after each removal of the reactor vessel head ensures the reactor internals and control rod drive mechanism will not interfere with CONTROL ROD motion or CONTROL ROD drop time and allows the operator to determine that the maximum CONTROL ROD drop time permitted is consistent with the assumed CONTROL ROD drop time used in the safety analysis. This frequency is considered more restrictive since it explicitly requires completion of this testing prior to criticality and is consistent with the NUREG.

M12 CTS 3.1.9 provides low power physics testing restrictions. Since no specific actions are provided to address failure to meet these restrictions, CTS would require entry into LCO 3.0, which requires the unit be placed in Hot Shutdown (comparable to ITS MODE 3) within 12 hours. ITS 3.1.8 ACTION A, B, and C are added. ITS ACTION A requires the control rod drive trip breakers be opened immediately if the thermal power limit is exceeded. ITS 3.1.8 Action B requires boration be initiated within 15 minutes and physics tests exceptions be suspended within 1 hour if the SDM limit is exceeded. ITS 3.1.8 ACTION C requires physics tests exceptions be suspended within 1 hour if the nuclear overpower trip setpoint is not within limits or the nuclear instrumentation high startup rate control rod withdrawal inhibit is inoperable. The proposed actions are more restrictive since their Completion Times are shorter than the 12 hours of CTS LCO 3.0. The proposed ACTIONS are consistent with the NUREG

CTS 3.1.7 imposes an MTC limit above 95% RTP. ITS LCO 3.1.3 has an applicability in MODES 1 and 2 for MTC. The physics test exception for MTC provided by ITS 3.1.8 is necessary only due to the more restrictive requirement on MTC imposed by ITS LCO 3.1.3. The proposed change is consistent with the NUREG.

M13 CTS specify alignment limits for movable control assemblies (safety and regulating) but do no specify alignment limits for axial power shaping rods (APSRs). ITS 3.1.6 specifies that each APSR be within 6.5% of its
1 group average height during MODES 1 and 2. Action A requires AXIAL
1 POWER IMBALANCE to be determined within 2 hours when one APSR is
1 inoperable or not aligned to within 6.5% of its group height and within
1 2 hours after each APSR movement. Action B requires the reactor to be in MODE 3 if the Required Action and associated Completion Time is not met. SR 3.1.6.1 requires verification that the LCO is met periodically (every 12 hours). The LCO, Required Actions and Surveillance

Requirements provide a conservative approach to ensure that the continued operation remains within the bounds of the safety analysis. The addition of these requirements are more restrictive on plant operation and are consistent with the NUREG.

M14 CTS 3.1.3.5 does not allow operation without safety rods fully withdrawn. No specific actions are provided to address the Condition where two or more safety rods are not fully withdrawn and are less than 9 inches misaligned, CTS would require entry into CTS LCO 3.0, which requires the unit be placed in Hot Shutdown (comparable to ITS MODE 3) within 12 hours. ITS ACTION B is added to provide action for the condition where more than one safety rod is not fully withdrawn. ITS 3.1.5 Required Action B.1.1 requires SDM to be verified within limits or Required Action B.1.2 requires boration be initiated to restore SDM in one hour when more than one safety rod is not fully withdrawn. ITS 3.1.5 Required Action B.2 requires the unit be in MODE 3 in 12 hours. The addition of the intermediate requirements to verify SDM limits within limits or initiate boration to restore SDM is more restrictive. This action is considered appropriate since when more than one safety rod is not fully withdrawn there is a possibility that the required SDM may be adversely affected and it is important to determine SDM and initiate boration if necessary. Ensuring the SDM meets the minimum requirement within 1 hour is adequate to determine that further degradation of the SDM is not occurring. The proposed change is consistent with the NUREG.

M15 CTS 3.1.9.3 provides certain provisions for physics testing provided SDM is $\geq 1.0\% \Delta k/k$ with the exception that the stuck rod worth criterion does not apply during rod worth measurements. A similar exception is not included in LCO 3.1.8. As such, the proposed LCO is considered more restrictive and is consistent with the NUREG. This more restrictive requirement provides additional assurance that fuel damage criteria would be preserved if an accident were to occur during PHYSICS TESTS.

M16 CTS 3.1.3.5, 3.5.2.4.a, 3.5.2.5.b, 3.5.2.5.c, and 3.5.2.6 excepts these individual specifications during physics testing with the statement: "except for physics testing." No differentiation is made in the CTS of the applicability of these exceptions with respect to the unit's THERMAL POWER level. Since these test exceptions are currently not used and are not needed at power levels > 5% RTP, ONS has opted not include the MODE 1 exceptions in the proposed ONS ITS. Since the CTS allows these exceptions at power levels > 5% RTP (MODE 1), the deletion of these exceptions is considered more restrictive. The proposed change is not consistent with NUREG 3.1.8, which provides MODE 1 physics tests exceptions. The proposed change is more conservative since no exceptions are allowed during PHYSICS TEST conducted in MODE 1, and therefore continue to ensure that operation remains within the bounds of the safety analysis.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 AXIAL POWER SHAPING ROD (APSR) Alignment Limits

CTS

LCO 3.1.6 Each APSR shall be OPERABLE and aligned within ~~[6.5]%~~ 9% of its group average height. (1) DOC M13

TSTF-159, R1

Supp. 1

APPLICABILITY: MODES 1 and 2 ~~when the APSRs are not fully withdrawn~~. DOC M13

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One APSR inoperable, not aligned within its limits, or both. Perform SR 3.2.2.1 TSTF-	A.1 ~~Align the APSR group to within [6.5]% of the inoperable or misaligned rod, while maintaining the APSR insertion limits in the COLR.~~ Restore APSR alignment. TSTF 143 *AND* A.2 ~~Prevent movement of the APSR group, while the rod remains inoperable or misaligned.~~	2 hours DOC M13 *AND* 2 hours after each APSR movement
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	~~6~~ 12 hours (27) DOC M13

Supp. 1

unit. The short time frame in which the unit is expected to be conducting PHYSICS TESTS requiring the exception to one or more LCOs does not warrant the frequent verification requirements. Further, this SR provides a verification of RPS system performance at a Frequency significantly shorter than that required of the RPS when operating in MODES 1 at RATED THERMAL POWER (ref. NUREG-1430 LCO 3.3.1). No basis exists that indicates the RPS trip function, or its calibration, would behave differently than that observed during power operation.

16 CTS Specification 3.1.9.1 requires that the nuclear overpower trip be set at less than or equal to 5% RTP during the conduct of low power PHYSICS TESTS. Therefore, NUREG LCO 3.1.9.b (ITS LCO 3.1.8) is modified to specify that the Nuclear Overpower Trip Setpoint be set at 5% RTP rather than the 25% RTP value established by the NUREG. NUREG SR 3.1.9.2 is modified accordingly. NUREG LCO 3.1.9.b (ITS LCO 3.1.8.b) is modified to read that the "Nuclear overpower trip setpoint is set to less than or equal to 5% RTP," similar to the terminology of NUREG Specification 3.3.1.

1 17 Not used.

18 Specific reference to the source range and intermediate range nuclear instrumentation in NUREG LCO 3.1.9.c and Condition C is incorrect (and unnecessary) for the ONS design. At ONS, only wide range nuclear instrumentation provides a high startup rate CONTROL ROD withdrawal inhibit function.

1 19 Not used.

1 20 Not used.

21 Not used.

22 The change to SR 3.1.4.3 maintains CONTROL ROD drop time testing requirements consistent with CTS 4.7.1 requirements at reactor coolant flow conditions. This change does not add new requirements nor does it change or remove any existing requirements. The NOTE in SR 3.1.4.3 was modified to allow continued operation with reactor coolant pump combinations which provide less total reactor coolant system flow than the combination used during drop time testing. Allowing for continued operation provided the total reactor coolant flow is less than the total flow during testing is appropriate due to the bounding nature of the test flow conditions. Without this change to the Note, reducing the number of running RCPs from 3 to 2, with drop time testing having been performed with 3 RCPs running, would require that all CONTROL RODS be declared inoperable. This declaration would be unnecessarily restrictive since a reduction in RCS flow can only improve rod drop times.

(4) <Except as marked>

BASES

equipment used to determine this value.

LCO (continued)

value is established based on the distance between reed switches, with additional allowances for uncertainty in the ~~absolute position indicator amplifiers, group maximum or minimum synthesizer, and asymmetric alarm or fault detector outputs.~~ ~~The position of an inoperable rod is not included in the calculation of the rod group's average position.~~ (14)

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

APPLICABILITY

The requirements on APSR OPERABILITY ~~and alignment~~ are applicable in MODES 1 and 2, ~~when the APSRs are not fully withdrawn~~ (1) because these are the only MODES in which significant neutron ~~(or fission)~~ power is generated, and the OPERABILITY and alignment of ~~rods~~ APSRs have the potential to affect the safety of the ~~plant~~ unit. ~~OPERABILITY and alignment of the APSRs are not required when they are fully withdrawn because they do not influence core power peaking.~~ (1) In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down ~~and not producing fission power~~ and excessive local LHRs cannot occur from APSR misalignment.

ACTIONS

The ACTIONS described unit below are required if one APSR is declared (25) inoperable due to inoperable position indication or is misaligned (25). The ~~plant~~ unit is not allowed to operate with more than one inoperable or misaligned (25) APSR. This would require the reactor to be ~~shut down~~ placed in MODE 3, in accordance with LCO 3.0.3.

A.1 ~~and A.2~~

TSTF

Supp. 1

An alternative to realigning a single inoperable or misaligned APSR to the group average position is to align the remainder of the APSR group to the position of the ~~misaligned or inoperable~~ APSR (1) ~~while maintaining APSR insertion, in accordance with the limits in the COLR.~~ This restores the alignment requirements. Deviations up to 2 hours will not cause significant xenon redistribution to occur. ~~Required Action A.1 assumes the APSR group movement does not cause the limits of LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," to be exceeded. For this reason,~~ (1)

(continued)

BASES

ACTIONS

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

Required Action A.1 ~~is~~ only practical for instances where small movements of the APSR group are sufficient to re-establish APSR alignment.

inoperable or 25

TSTF

Supp. 1

The reactor may continue in operation with the APSR misaligned if further movement of the APSR group is prohibited, so that ~~the~~ misalignment does not increase and cause the limits on AXIAL POWER IMBALANCE to be exceeded. The required Completion Time of up to 2 hours will not cause significant xenon redistribution to occur.

INSERT B 3.1-36A

B.1

12 1

The ~~plant~~ unit must be brought to a MODE in which the LCO does not apply if the Required Actions and associated Completion Times cannot be met. To achieve this status, the ~~plant~~ unit must be brought to at least MODE 3 within ~~6~~ 12 hours. The Completion Time of ~~6~~ 12 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging ~~plant~~ systems. In MODE 3, APSR ~~group~~ alignment limits are not required because the reactor is not generating significant THERMAL POWER and excessive local LHRs cannot occur from APSR misalignment.

1 4

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Verification at a 12 hour Frequency that individual APSR positions are within [6.5]% of the group average height limits allows the operator to detect an APSR beginning to deviate from its expected position. If the asymmetric CONTROL ROD alarm is inoperable, a 4 hour Frequency is reasonable to prevent large deviations in APSR alignment from occurring without detection. In addition, APSR position is continuously available to the operator in the control room so that during actual ~~rod~~ APSR motion, deviations can immediately be detected.

2

TSTF 110, R2

Supp. 1

4

(continued)

INSERT B 3.1-36A

the limits on AXIAL POWER IMBALANCE are surveilled within 2 hours to determine if the AXIAL POWER IMBALANCE is still within limits. Also, since any additional movement of the APSRs may result in additional imbalance, Required Action A.1 also requires the AXIAL POWER IMBALANCE surveillance to be performed again within 2 hours after each APSR movement

ITS Section 3.1

ID 44

Subject Incorporate revision to ONS-013 (BWOG-70) - Revises LCO Note wording.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Safety Rod Position Limits

LCO 3.1.5 Each safety rod shall be fully withdrawn.

-----------------------------NOTE-----------------------------
1 Not required for any safety rod positioned to perform SR 3.1.4.2.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One safety rod not fully withdrawn.	A.1	Withdraw the rod fully.	1 hour
	OR		
	A.2.1.1	Verify SDM is within the limit specified in the COLR.	1 hour
		OR	
	A.2.1.2	Initiate boration to restore SDM to within limit.	1 hour
	AND		
	A.2.2	Declare the rod inoperable.	1 hour

(continued)

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Safety Rod ~~Insertion~~ Position Limits (2)

LCO 3.1.5 Each safety rod shall be fully withdrawn. 3.1.3.5 3.1.11 3.5.2.2.a

APPLICABILITY: MODES 1 and 2. 3.1.3.5 3.1.11 3.5.2 APP

Supp. 1

-----NOTE-----
~~This LCO is not applicable while performing~~ SR 3.1.4.2.
Not required for any safety rod ~~inserted~~ positioned to perform (2) TSTF 216 3.5.2.5.a

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One safety rod not fully withdrawn.	A.1 Withdraw the rod fully.	1 hour	3.5.2.2.d.1 DOC L2
	OR		
	A.2.1.1 Verify SDM ~~is ≥ 1% Δk/k.~~ ~~to be~~ is within the limit ~~provided~~ specified in the COLR. (29) TSTF-009 Rev. 1	1 hour	3.5.2.2.d.2.a 3.5.2.1.a
	OR		
	A.2.1.2 Initiate boration to restore SDM to within limit.	1 hour	3.5.2.1.b
	AND		
	A.2.2 Declare the rod inoperable.	1 hour	DOC L2 3.5.2.2.b

(continued)

1 23 Not used.

24 NUREG 3.1.4, Required Action A.2.3 is modified to reflect the CTS requirement (CTS 3.5.2.2.d.2.b) to reduce both nuclear overpower trip setpoints. This is further modified as "...setpoints, based on flux and flux/flow imbalance" for clarity.

25 NUREG SR 3.1.4.3 requires rod drop times be verified with RCS temperature (T_{avg}) ≥ 525°F prior to criticality after each removal of the reactor vessel. At ONS, the SR is performed concurrent with heatup and pressurization. Currently there are no temperature limitations in CTS or the implementing procedure other than the temperature necessary for reactor coolant pump operation. Therefore, the NUREG temperature requirement is not included in the ONS ITS.

26 NUREG SR 3.1.9.1, which requires THERMAL POWER be verified ≤ 5% RTP, is not included in the ITS since the nuclear power overpower trip setpoint is also ≤ 5% RTP. The overpower trip setpoint will be verified prior to performance of PHYSICS TESTS and can be relied upon to verify THERMAL POWER is ≤ 5% RTP.

27 The current licensing basis (CLB) permits 12 hours to place a unit in Hot Shutdown when an LCO is not met. To maintain consistency with current procedures, training and staffing requirements, the 12 hours permitted to place a unit in MODE 3 is retained in the ITS.

Although ITS 3.1.2 Required Action B.1 requires placing the unit in MODE 2 instead of MODE 3, the time requirement is consistent with other ITS requirements involving placing the unit in MODE 3. The small time necessary to achieve MODE 3 from MODE 2 does not warrant a unique time allowance.

28 NUREG Specification 3.1.8, which provides PHYSICS TESTS exceptions in MODE 1, is not included in the ONS ITS. ONS currently does not take these exceptions in MODE 1 and does not foresee needing these exceptions in the future.

29 Editorial changes were made to TSTF-009, Revision 1, of the NUREG for consistency to the wording of similar references to limits located in the COLR made from other NUREG Specifications. For example, reference to limits "provided" in the COLR were changed to limits "specified" in the COLR.

30 NUREG SR 3.1.1.1 is re-worded for consistency with TSTF-009, Revision 1. Therefore, the SR requires the SDM is verified to be within limit consistent with the LCO requirement as modified by the TSTF.

ITS Section 3.1

ID 57

Subject Revise ITS 3.1.8 to incorporate revision to ONS-027 - Deletes ACT E and modifes ACT D to replace RA D.1 with "Suspend PHYSICS TESTING" with a CT of 30 minutes.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Nuclear overpower trip setpoint is not within limit. <u>OR</u> Nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit inoperable.	C.1 Suspend PHYSICS TESTS exceptions.	1 hour
D. RCS lowest loop average temperature not within limit.	D.1 Suspend PHYSICS TESTS exceptions.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.8.1 Verify the RCS lowest loop average temperature is ≥ 520°F.	30 minutes

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Completion Time of one hour is provided for the operator to restore compliance with the excepted LCOs.

C.1

If the nuclear overpower trip setpoint is > 5% RTP, then 1 hour is allowed for the operator to restore the nuclear overpower trip setpoint within limits or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification, in order to ensure that continuity of reactor operation is within initial condition limits. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions.

If the nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit functions are inoperable, then 1 hour is allowed for the operator to restore the functions to OPERABLE status or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions.

D.1

1 If the RCS lowest loop average temperature is not within
1 limit, then 30 minutes is allowed for the operator to
restore the RCS lowest loop average temperature to within limit or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification, in order to ensure that continuity of reactor operation is within initial condition limits. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification that the RCS lowest loop average temperature is ≥ 520°F will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.2

Verification that the nuclear overpower trip setpoint is within the limit specified for PHYSICS TESTS ensures that core protection at the reduced power level is established during PHYSICS TESTS. Performing the verification once within 8 hours prior to the performance of PHYSICS TESTS allows the operator adequate time for verifying the established trip setpoint margin before PHYSICS TESTS.

SR 3.1.8.3

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. CONTROL ROD position;

c. RCS average temperature;

d. Fuel burnup;

e. Xenon concentration; and

f. Moderator temperature coefficient (MTC).

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.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50.59.
3. UFSAR, Section 4.3.4.
4. UFSAR, Sections 14.3, 14.4 and 14.6.
5. UFSAR, Section 14.4, Table 14-2.
6. 10 CFR 50.36.

1 ACTION D is added to provide appropriate Required Actions when RCS
lowest loop average temperature is not within limit. SR 3.1.8.1 is
added to require verifying RCS lowest loop average temperature is within
1 the limit periodically. ITS 3.1.8 ACTION D requires the RCS lowest loop
1 average temperature to be restored within limit or PHYSICS TESTS
1 exceptions be suspended within 30 minutes if the RCS lowest loop average
1 temperature is not within limit. Suspension of PHYSICS TESTS exceptions
1 requires restoration of each of the applicable individual LCOs to within
1 specification, in order to ensure that continuity of reactor operation
1 is within the initial condition limits. This Completion Time is
1 consistent with the individual LCO addressed by the PHYSICS TESTS
1 exception and ensures the unit will not remain in an unacceptable
condition for an extended period of time. ITS SR 3.1.8.1 requires the
lowest RCS loop average temperature be verified every 30 minutes during
PHYSICS TESTS. This ensures the unit is not operating in a condition
that could invalidate the safety analyses.

The additional restrictions are necessary to ensure assumptions associated with suspending requirements are met during PHYSICS TESTS initiated in MODE 2.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.~~9~~ 8 PHYSICS TESTS Exceptions – MODE 2 2

LCO 3.1.~~9~~ 8 During performance of PHYSICS TESTS, the requirements of

LCO 3.1.3, "Moderator Temperature Coefficient (MTC)"; DOC M12
~~LCO 3.1.4, "CONTROL ROD Group Alignment Limits";~~ 13
LCO 3.1.5, "Safety Rod ~~Insertion~~ Position Limits"; 3.1.3.5
~~LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits";~~ 13
LCO 3.2.1, "Regulating Rod ~~Insertion~~ Position Limits," ~~for the restricted operation region only~~; and 3.5.2.5.b+c
~~[~~LCO 3.4.2, "RCS Minimum Temperature for Criticality"~~]~~ 14 3.1.3.1

may be suspended, provided:

a. THERMAL POWER is ≤ 5% RTP; 3.1.9.1.a 3.1.9.1.b

b. ~~Reactor trip setpoints on the OPERABLE~~ nuclear overpower trip setpoint is ~~channels are~~ set to ≤ ~~2~~5% RTP; 16 3.1.9.1.a 3.1.9.1.b

c. Nuclear instrumentation ~~source range and intermediate range high startup~~ rate CONTROL ROD withdrawal inhibit ~~are~~ is OPERABLE; ~~and~~ 18 3.1.9.2

d. SDM is ~~≥ [1.0]% Δk/k~~ within the limit ~~provided~~ specified in the COLR. TSTF-009 Rev. 1 29 3.1.9.3

e. RCS lowest loop average temperature is ≥ 520°F. DOC M20 TSTF-(TBD)

Supp. 1

APPLICABILITY: ~~MODE 2~~ during PHYSICS TESTS initiated in MODE 2. TSTF 256 DOC A17

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER not within limit.	A.1 Open control rod drive trip breakers.	Immediately

DOC M12

(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	Doc M12
	AND			
	B.2	Suspend PHYSICS TESTS exceptions.	1 hour	
C. Nuclear overpower trip setpoint is not within limit. OR Nuclear instrumentation ~~source and intermediate range~~ high startup rate CONTROL ROD withdrawal inhibit inoperable. 18	C.1	Suspend PHYSICS TESTS exceptions.	1 hour	Doc M12

Supp.1 INSERT 3.1-22A TSTF-(TBD)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY	
~~SR 3.1.9.1 Verify THERMAL POWER is ≤ 5% RTP.~~	~~1 hour~~	26
SR 3.1.~~9~~ 8.2 2 Verify nuclear overpower trip setpoint is ≤ ~~25%~~ 5 RTP. 16	Once within 8 hours prior to performance of PHYSICS TESTS. 15	Doc M19.

(continued)

INSERT 3.1-22B TSTF-(TBD)

Supp. 1

CTS

INSERT 3.1-22A

Supp. 1

D. RCS lowest loop average temperature not within limit.	D.1 Suspend PHYSICS TESTS exceptions.	30 minutes	DOC M20

INSERT 3.1-22B

SR 3.1.8.1 Verify the RCS lowest loop average temperature is ≥ 520°F.	30 minutes	DOC M20

intent of this SR which is to ensure that the CONTROL RODS are capable of inserting into the core in the event of a reactor trip.

1 9 Not used.

10 Not used.

11 Not used.

12 NUREG 3.1.7 LCO and Actions are modified to maintain CTS requirements for CONTROL ROD position indication channels. CTS 3.5.2.2.b.2 requires only one OPERABLE channel of position indication per control rod. If this required channel is inoperable, the associated control rod must be declared inoperable and the Actions of the control rod's governing Specification must be completed. The CTS requirements are maintained by the indicated changes to NUREG Specification 3.1.7.

SR 3.1.7.1 is modified to match the requirements of ITS LCO 3.1.7. This change is made to provide for Surveillance Requirements which adequately address the equipment required by the LCO. This change provides clarification of the inconsistency within the CTS with regard to the required channels of position indication and surveillance requirements. CTS Table 4.1-1 Items 23 and 24 requires shiftly checks of both the absolute and relative rod position indication channels, while CTS 3.5.2.2.b.2 allows unrestricted operation with either or potentially both of these channels inoperable. This change requires only the channel providing the required indication to be checked.

13 NUREG 3.1.9 exceptions to NUREG LCO 3.1.4 and 3.1.6 are not adopted in the ITS. These exceptions are not present in the CTS and ONS has no plans to perform PHYSICS TESTS that would require the use of these exceptions in the future.

14 NUREG LCO 3.1.9 (PHYSICS TESTS exceptions) allows LCO 3.2.1 "restricted operation region only" requirements to be suspended during PHYSICS TESTS. This exception is modified in the ITS 3.1.8 to allow suspension of LCO 3.2.1 requirements, consistent with CTS provisions which allow exception to position limit (does not limit to regulating rods inserted in the restricted region only) and overlap and sequence limits. This is acceptable since limits on THERMAL POWER and shutdown capability maintained during the PHYSICS TESTS ensure fuel damage criteria are preserved even if an accident were to occur with the LCO suspended.

15 The Frequency of NUREG SR 3.1.9.2. (ITS SR 3.1.8.2) is changed to specify "Once within 8 hours prior to performance of PHYSICS TESTS." This Frequency requires the nuclear overpower trip setpoint be verified prior to the onset of PHYSICS TESTS which ensures that the established LCO conditions are satisfied, with respect to the trip function. The requirement to perform this NUREG SR with a Frequency of 8 hours is excessively restrictive and unduly burdensome on the operation of the

INSERT B 3.1-56A

D.1

Supp. 1

If the RCS lowest loop average temperature is not within limit then 30 minutes is allowed for the operator to restore the RCS lowest loop average temperature to within limit or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification, in order to ensure that continuity of reactor operation is within initial condition limits. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions.

INSERT B 3.1-56B

SR 3.1.8.1

Verification that the RCS lowest loop average temperature is ≥ 520°F will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

ITS Section 3.1

ID 58

Subject Remove JFD associated with approved ONS generic changes ONS-018 and annotate NUREG Markup with TSTF number 256.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.~~9~~ 8 PHYSICS TESTS Exceptions – MODE 2 2

LCO 3.1.~~9~~ 8 During performance of PHYSICS TESTS, the requirements of

LCO 3.1.3, "Moderator Temperature Coefficient (MTC)"; Doc M12
~~LCO 3.1.4, "CONTROL ROD Group Alignment Limits";~~ 13
LCO 3.1.5, "Safety Rod ~~Insertion~~ Position Limits"; 3.1.3.5
~~LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits";~~ 13
LCO 3.2.1, "Regulating Rod ~~Insertion~~ Position Limits" ~~for the restricted operation region only~~; and 3.5.2.5.6+c
~~LCO 3.4.2, "RCS Minimum Temperature for Criticality"~~ 14 3.1.3.1

may be suspended, provided:

a. THERMAL POWER is ≤ 5% RTP; 3.1.9.1.a 3.1.9.1.b

b. ~~Reactor trip setpoints on the OPERABLE~~ nuclear overpower ~~channels are~~ trip setpoint is set to ≤ ~~25~~ 5% RTP; 16 3.1.9.1.a 3.1.9.1.b

c. Nuclear instrumentation ~~source range and intermediate range high startup rate~~ CONTROL ROD withdrawal inhibit ~~are~~ is OPERABLE; ~~and~~ 18 15 3.1.9.2

d. SDM is ~~≥ [1.0]% Δk/k~~ within the limit ~~provided~~ specified in the COLR TSTF-009 Rev. 1 29 3.1.9.3

e. RCS lowest loop average temperature is ≥ 520°F. Doc M20

Supp. 1 APPLICABILITY: ~~MODE 2~~ during PHYSICS TESTS initiated in MODE 2. TSTF 256 TSTF-(TBD) Doc A17

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER not within limit.	A.1 Open control rod drive trip breakers.	Immediately

Doc M12

(continued)

unit. The short time frame in which the unit is expected to be conducting PHYSICS TESTS requiring the exception to one or more LCOs does not warrant the frequent verification requirements. Further, this SR provides a verification of RPS system performance at a Frequency significantly shorter than that required of the RPS when operating in MODES 1 at RATED THERMAL POWER (ref. NUREG-1430 LCO 3.3.1). No basis exists that indicates the RPS trip function, or its calibration, would behave differently than that observed during power operation.

16 CTS Specification 3.1.9.1 requires that the nuclear overpower trip be set at less than or equal to 5% RTP during the conduct of low power PHYSICS TESTS. Therefore, NUREG LCO 3.1.9.b (ITS LCO 3.1.8) is modified to specify that the Nuclear Overpower Trip Setpoint be set at 5% RTP rather than the 25% RTP value established by the NUREG. NUREG SR 3.1.9.2 is modified accordingly. NUREG LCO 3.1.9.b (ITS LCO 3.1.8.b) is modified to read that the "Nuclear overpower trip setpoint is set to less than or equal to 5% RTP," similar to the terminology of NUREG Specification 3.3.1.

1 17 Not used.

18 Specific reference to the source range and intermediate range nuclear instrumentation in NUREG LCO 3.1.9.c and Condition C is incorrect (and unnecessary) for the ONS design. At ONS, only wide range nuclear instrumentation provides a high startup rate CONTROL ROD withdrawal inhibit function.

1 19 Not used.

1 20 Not used.

21 Not used.

22 The change to SR 3.1.4.3 maintains CONTROL ROD drop time testing requirements consistent with CTS 4.7.1 requirements at reactor coolant flow conditions. This change does not add new requirements nor does it change or remove any existing requirements. The NOTE in SR 3.1.4.3 was modified to allow continued operation with reactor coolant pump combinations which provide less total reactor coolant system flow than the combination used during drop time testing. Allowing for continued operation provided the total reactor coolant flow is less than the total flow during testing is appropriate due to the bounding nature of the test flow conditions. Without this change to the Note, reducing the number of running RCPs from 3 to 2, with drop time testing having been performed with 3 RCPs running, would require that all CONTROL RODS be declared inoperable. This declaration would be unnecessarily restrictive since a reduction in RCS flow can only improve rod drop times.

1 23 Not used.

ITS Section 3.2

ID 2

Subject Revise to refer to SDM as specified in the COLR rather than value.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B or C not met.	D.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
1	SR 3.2.1.1	Verify regulating rod groups are within the sequence and overlap limits as specified in the COLR.	12 hours
1	SR 3.2.1.2	Verify regulating rod groups meet the position limits as specified in the COLR.	12 hours
1 1	SR 3.2.1.3	Verify SDM to be within the limit specified in the COLR.	Within 4 hours prior to achieving criticality

3.1.11 Shutdown Margin

MODES 1 and 2 A12

Specification

Supp 1 Applic

The available shutdown margin during ~~all system conditions except refueling~~ shall be ~~greater than 1% Δk/k with the highest worth control rod fully withdrawn.~~ within the limit specified in the COLR. LA3

Acc. crit. for SR 3.2.1.3

Bases

~~A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.~~

~~During power operation and startup the SHUTDOWN MARGIN is known to be within limits if all control rods are OPERABLE and withdrawn to or beyond the insertion limits determined in accordance with the approved methodology and provided in the CORE OPERATING LIMITS REPORT per Specification 6.9.~~

~~During refueling conditions equivalent protection is provided in the requirements of Specification 3.8.4.~~

TECHNICAL CHANGES - REMOVAL OF DETAILS

LA1 The regulating rod group overlap limit currently provided in CTS 3.5.2.5.b is moved to the COLR. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describes very specific setpoints which can only be changed on a cycle specific bases. Since these details are likely to change on a refueling cycle frequency, they are being moved to a licensee controlled document. Placing these details in controlled documents provides adequate assurance that they will be maintained. Changes to the COLR are controlled by 10 CFR 50.59. This change is consistent with the NUREG.

LA2 CTS 4.1.5, which requires a power map be made to verify expect power distributions at periodic intervals using the incore instrumentation system, is moved to the UFSAR Chapter 16. This detail is not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains power distribution requirements. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable. Changes to the UFSAR Chapter 16 are controlled by the provisions of 10 CFR 50.59.

1 LA3 The specific value for SDM in CTS 3.1.11 is relocated to the COLR. SDM
1 is a cycle specific variable, similar to other variables, such as,
1 Moderator Temperature Coefficient, Rod Position Limits, AXIAL POWER
1 IMBALANCE operating limits, and QUADRANT POWER TILT limits all of which
1 may be contained in the COLR. Relocation of the specific SDM value
1 provides core design and operational flexibility that can be used for
1 improved fuel management and to solve plant specific issues. The COLR
1 is subject to the control described in ITS Chapter 5, "Administrative
1 Controls," which ensure any changes to the COLR are appropriately
1 reviewed. This change is consistent with the NUREG as modified by TSTF-
1 009, Revision 1.

4

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.2.1.1	Verify regulating rod groups are within the sequence and overlap limits as specified in the COLR.	4 hours ~~when the CONTROL ROD drive sequence alarm is inoperable~~ ~~AND~~ ~~12 hours when the CONTROL ROD drive sequence alarm is OPERABLE~~ 14	DOC M4
SR 3.2.1.2	Verify regulating rod groups meet the ~~insertion~~ position limits as specified in the COLR 4	4 hours ~~when the regulating rod insertion limit alarm is inoperable~~ ~~AND~~ ~~12 hours when the regulating rod insertion limit alarm is OPERABLE~~ 14	DOC M4
Supp l SR 3.2.1.3	Verify SDM ~~≥ 1% Δk/k~~ to be within the limit specified in the COLR.	Within 4 hours prior to achieving criticality	DOC M6

TSTF-009, Rev. 1

ITS Section 3.2

ID 37

Subject Revise ITS to incorporate approved TSTF-110, R2 which eliminates the increased SR Freq associated wtih inoperable alarms - Revises SR Frequencies associated with 3.2.1, 3.2.2, & 3.2.3

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B or C not met.	D.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
1	SR 3.2.1.1 Verify regulating rod groups are within the sequence and overlap limits as specified in the COLR.	12 hours
1	SR 3.2.1.2 Verify regulating rod groups meet the position limits as specified in the COLR.	12 hours
1 1	SR 3.2.1.3 Verify SDM to be within the limit specified in the COLR.	Within 4 hours prior to achieving criticality

3.2 POWER DISTRIBUTION LIMITS

3.2.2 AXIAL POWER IMBALANCE Operating Limits

LCO 3.2.2 AXIAL POWER IMBALANCE shall be maintained within the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 40% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AXIAL POWER IMBALANCE not within limits.	A.1 Reduce AXIAL POWER IMBALANCE to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to $\leq$ 40% RTP.	2 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
1 SR 3.2.2.1 Verify AXIAL POWER IMBALANCE is within limits as specified in the COLR.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
1 **SR 3.2.3.1** Verify QPT is within limits as specified in the COLR.	7 days AND When QPT has been restored to less than or equal to the steady state limit, 1 hour for 12 consecutive hours, or until verified acceptable at ≥ 95% RTP

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

This Surveillance ensures that the sequence and overlap
limits are not violated. A Surveillance Frequency of
1 12 hours is acceptable because little rod motion due to fuel
1 burnup occurs in 12 hours. Also, the Frequency takes into
account other information available in the control room for
monitoring the status of the regulating rods.

SR 3.2.1.2

Verification of the regulating rod position limits as
1 specified in the COLR at a Frequency of 12 hours is
sufficient to detect whether the regulating rod groups may
be approaching or exceeding their group position limits,
because little rod motion occurs due to fuel burnup occurs
1 in 12 hours. Also, the Frequency takes into account other
information available in the control room for monitoring the
status of the regulating rods.

SR 3.2.1.3

Prior to achieving criticality, an estimated critical position for the CONTROL RODS is determined. Verification that SDM meets the minimum requirements ensures that sufficient SDM capability exists with the CONTROL RODS at the estimated critical position if it is necessary to shut down or trip the reactor after criticality. The Frequency of 4 hours prior to criticality provides sufficient time to verify SDM capability and establish the estimated critical position.

REFERENCES

1. UFSAR, Section 3.1.
2. 10 CFR 50.46.
3. UFSAR, Section 15.2.
4. UFSAR, Chapter 15.
5. 10 CFR 50.36.

BASES

SURVEILLANCE REQUIREMENTS (continued)

b. The axial planes in each core half shall be symmetrical about the core midplane; and

c. The detector strings shall not have radial symmetry.

Figure B 3.2.2-1 (Backup Incore Detector System for AXIAL POWER IMBALANCE Measurement) depicts an example of this configuration. This arrangement is chosen to reduce the uncertainty in the measurement of the AXIAL POWER IMBALANCE by the Backup Incore Detector System. For example, the requirement for placing one detector of each of the three strings at the core midplane puts three detectors in the central region of the core where the neutron flux tends to be higher. It also helps prevent measuring an AXIAL POWER IMBALANCE that is excessively large when the reactor is operating at low THERMAL POWER levels. The third requirement for placement of detectors (i.e., radial asymmetry) reduces uncertainty by measuring the neutron flux at core locations that are not radially symmetric.

The Excore Detector System consists of four detectors (one located outside each quadrant of the core). Each detector consists functionally of two six-foot uncompensated ion chambers adjacent to the top and bottom halves of the core. Comparison of the signals from the two detectors gives an indication of the core axial offset or imbalance.

SR 3.2.2.1

1 Verification of the AXIAL POWER IMBALANCE indication every
1 12 hours ensures that the AXIAL POWER IMBALANCE limits are
1 not violated and takes into account other information and
1 alarms available to the operator in the control room.
This Surveillance Frequency is acceptable because the mechanisms that can cause AXIAL POWER IMBALANCE, such as xenon redistribution or CONTROL ROD drive mechanism malfunctions that cause slow AXIAL POWER IMBALANCE increases, can be discovered by the operator before the specified limits are violated.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Excore Detector System consists of four detectors (one located outside each quadrant of the core). Each detector consists functionally of two six-foot uncompensated ion chambers adjacent to the top and bottom halves of the core.

<u>SR 3.2.3.1</u>

1 Checking the QPT indication every 7 days ensures that the
1 operator can determine whether the plant computer software
1 and Incore Detector System inputs for monitoring QPT are
1 functioning properly, and takes into account other
1 information and alarms available to the operator in the
1 Control Room. This procedure allows the QPT mechanisms,
1 such as xenon redistribution, burnup gradients, and CONTROL
1 ROD drive mechanism malfunctions, which can cause slow
1 development of a QPT, to be detected. Operating experience
1 has confirmed the acceptability of a Surveillance Frequency
1 of 7 days.

Following restoration of the QPT to within the steady state limit, operation at ≥ 95% RTP may proceed provided the QPT is determined to remain within the steady state limit at the increased THERMAL POWER level. In case QPT exceeds the steady state limit for more than 24 hours or exceeds the transient limit (Condition A, B, or D), the potential for xenon redistribution is greater. Therefore, the QPT is monitored for 12 consecutive hourly intervals to determine whether the period of any oscillation due to xenon redistribution causes the QPT to exceed the steady state limit again.

REFERENCES

1. 10 CFR 50.46.

2. BAW 10122A, "Normal Operating Controls," Rev. 1, May 1984.

3. 10 CFR 50.36.

Specification 3.2.2

A1 <except as marked> <see 3.1> <SEE 3.2.1>

c. Position limits are specified for regulating and axial power shaping control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits shall be maintained within acceptable operating limits for regulating rod position provided in the CORE OPERATING LIMITS REPORT for the particular number of operating reactor coolant pumps (4,3).

If the control rod position limits are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. An acceptable control rod position shall then be attained within two hours. The minimum shutdown margin required by Specification 3.5.2.1 shall be maintained at all times.

Supp. 1 | L4 12

3.2.2 ~~3.5.2.6~~ MODE 1

SR 3.2.2.1 AXIAL ~~Reactor~~ power imbalance shall be monitored on a frequency not to exceed ~~two~~ hours ~~during~~ ~~power operation~~ above 40 percent rated

APP power. Except for physics tests, imbalance shall be maintained <SEE 3.1>

LCO 3.2.2 within the acceptable operating limits for reactor power imbalance provided in the CORE OPERATING LIMITS REPORT.

ACT A If the imbalance is not within the acceptable envelope, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within two hours reactor power

ACT B shall be reduced ~~until imbalance limits are met.~~

3.5.2.7 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the manager or his designated alternate.

<SEE 3.2.1>

to ≤ 40% RTP within the following 2 hours A3 M3

Page 1 of 1

A1 <Except as marked>

RA C.1 3. The reactor thermal power shall be reduced within the next 2 hours to less than 60% of the allowable power for the reactor coolant pump combination and the Nuclear

RA C.2 Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be reduced within ~~the next 4~~ 10 hours to 65.5% of the thermal power value allowable for the reactor coolant pump combination. L1

COND D 5. If the quadrant power tilt exceeds the Transient Limit but is less than the Maximum Limit provided in the Core Operating Limits Report, due to causes other than simultaneous indication of a misaligned control rod then:

1. RA D.1 Reactor thermal power shall be reduced within 2 hours to less than 60% of the allowable power for the reactor coolant pump combination and the Nuclear Overpower Trip

RA D.2 Setpoints, based on flux and flux/flow imbalance, shall be reduced within ~~the next~~ 2 hours to 65.5% of the thermal power value allowable for the reactor coolant pump combination. 10 L1

COND F 4. If the maximum positive quadrant power tilt exceeds the Maximum Limit provided in the Core Operating Limits Report, the reactor shall be ~~shut down within 4 hours. Subsequent reactor operation is permitted for the purpose of measurement, testing, and corrective action provided the thermal power and the Nuclear Overpower Trip Setpoints allowable for the reactor coolant pump combination are restricted by a reduction of 2% of thermal power for each 1% tilt for the maximum tilt observed prior to shutdown.~~ reduced to ≤ 20% RTP A10 M5 A10

Supp.1 L2 7 days AND when QPT has been restored to ≤ the steady state limit, 1 hour for 12 consecutive hours...

SR 3.2.3.1 g. Quadrant power tilt shall be monitored on a minimum frequency of once every ~~2 hours~~ during power operation above ~~15%~~ full power.

APP MODE 1 20% L3

3.5.2.5 Control Rod Positions

<SEE 3.1> a. Technical Specification 3.1.3.5 does not prohibit the exercising of individual safety rods as required by Table 4.1-2 or apply to inoperable safety rod limits in Technical Specification 3.5.2.2.

<see 3.1> b. Except for physics tests, operating rod group overlap shall be 25% ± 5% between two sequential groups. If this limit is exceeded, corrective measures shall be taken immediately to achieve an acceptable overlap. Acceptable overlap shall be attained within two hours or the reactor shall be placed in a hot shutdown condition within an additional 12 hours. <SEE 3.2.1>

Add Action E M2

Required Action D.2 allows 10 hours for the same action. Therefore, ITS allows 4 more hours to reduce the nuclear overpower trip setpoints.

The 10 hour Completion Times for reducing nuclear overpower trip setpoints when QPT limits are exceed are considered appropriate in light of the 2 hour Completion Times associated with ITS Required Actions A.1, C.1 and D.1 and their required reduction in THERMAL POWER. The adoption of the 4 to 6 additional hours provides sufficient time for an orderly execution of the tasks associated with Required Actions A.1 and A.2, B.1 and B.2, and C.1 and C.2. The proposed change is consistent with the NUREG.

L2 CTS 3.5.2.4.g requires quadrant power tilt (QPT) to be monitored on a minimum frequency of once every 2 hours. ITS SR 3.2.3.1 requires QPT verification every 7 days, and when QPT has been restored to less than or equal to the steady state limit, 1 hour for 12 consecutive hours, or until verified acceptable at $\geq$ 95% RTP. Checking the QPT indication every 7 days ensures that the operator can determine whether the plant computer software and the Incore Detector System inputs for monitoring QPT are functioning properly and takes into account other information and alarms available to the operator in the Control Room. Performing the SR at the 7 day Frequency allows QPT mechanisms, such as xenon redistribution, burnup gradients, and CONTROL ROD drive mechanism malfunctions, which can cause slow development of a QPT, to be detected. Following restoration of the QPT to within the steady state limit, operation at $\geq$ 95% RTP may proceed provided the QPT is determined to remain within the steady state limit at the increased THERMAL POWER level. In case QPT exceeds the steady state limit for more than 24 hours or exceeds the transient limit (Condition A, B, or D), the potential for xenon redistribution is greater. Therefore, the QPT is monitored for 12 consecutive hourly intervals to determine whether the period of any oscillation due to xenon redistribution would cause the QPT to exceed the steady state limit again. This change is consistent with the NUREG.

L3 The CTS 3.5.2.4.a & g applicability for the CTS Quadrant Power Tilt is "during power operation above 15% of rated power." The ITS LCO 3.2.4 applicability is 20% RTP. This is less restrictive since the unit is allowed to operate at a slightly higher power level prior to imposing QPT limits. Both of these Applicabilities are based on the lower mode of operability of the incore detector system. Operation at or below 20% RTP with QPT not imposed is acceptable because the worst case QPT does not result in an LHR high enough to cause violation of the LOCA LHR limit or the initial condition DNB allowable peaking limit during accidents initiated from this power level. The proposed change is consistent with the NUREG.

1 L4 CTS 3.5.2.6 requires AXIAL POWER IMBALANCE to be monitored on a
1 frequency not to exceed two hours. ITS SR 3.2.2.1 requires AXIAL POWER
1 IMBALANCE to be monitored on a frequency not to exceed 12 hours. This
1 frequency ensures that the AXIAL POWER IMBALANCE limits are not violated
1 and takes into account other information and alarms available to the
1 operator in the Control Room. This Surveillance Frequency is acceptable
1 because the mechanisms that can cause AXIAL POWER IMBALANCE, such as
1 xenon redistribution or CONTROL ROD drive mechanism malfunctions that
1 cause slow AXIAL POWER IMBALANCE increases, can be discovered by the
1 operator before the specified limits are violated. The proposed change
1 is consistent with the NUREG.

LESS RESTRICTIVE CHANGE L2

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

1 CTS 3.5.2.4.g requires quadrant power tilt (QPT) to be monitored
1 on a minimum frequency of once every 2 hours. ITS SR 3.2.3.1
1 requires QPT verification every 7 days, and when QPT has been
1 restored to less than or equal to the steady state limit, 1 hour
1 for 12 consecutive hours, or until verified acceptable at ≥ 95%
1 RTP. Checking the QPT indication every 7 days ensures that the
1 operator can determine whether the plant computer software and the
1 Incore Detector System inputs for monitoring QPT are functioning
1 properly and takes into account other information and alarms
1 available to the operator in the Control Room. Performing the SR
1 at the 7 day Frequency allows QPT mechanisms, such as xenon
1 redistribution, burnup gradients, and CONTROL ROD drive mechanism
1 malfunctions, which can cause slow development of a QPT, to be
1 detected. Following restoration of the QPT to within the steady state limit, operation at ≥ 95% RTP may proceed provided the QPT is determined to remain within the steady state limit at the increased THERMAL POWER level. In case QPT exceeds the steady state limit for more than 24 hours or exceeds the transient limit (Condition A, B, or D), the potential for xenon redistribution is greater. Therefore, the QPT is monitored for 12 consecutive hourly intervals to determine whether the period of any oscillation due to xenon redistribution would cause the QPT to exceed the steady state limit again. This change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

The proposed change extends the interval for monitoring quadrant
1 power tilt (QPT) from once every 2 hours to once every 7 days.
1 Checking the QPT indication every 7 days ensures that the operator
1 can determine whether the plant computer software and the Incore
1 Detector System inputs for monitoring QPT are functioning properly
1 and takes into account other information and alarms available to

1 the operator in the Control Room. Performing the SR at the 7 day
1 Frequency allows QPT mechanisms, such as xenon redistribution,
1 burnup gradients, and CONTROL ROD drive mechanism malfunctions,
1 which can cause slow development of a QPT, to be detected. The change affects only the frequency of monitoring and does not result in any change in the response of equipment to an accident. Therefore, the proposed change does not involve a significant increase the probability or consequences of any accident previously analyzed.

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

This change does not result in any changes to the equipment design or capabilities or to the operation of the plant. Further, since the change impacts only the frequency of monitoring and does not result in any change in the response of equipment to an accident, the change does not create the possibility of a new or different kind of accident from any previously analyzed accident.

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

The proposed change extends the interval for monitoring quadrant
1 power tilt (QPT) from once every 2 hours to once every 7 days.
The frequency of monitoring QPT continues to allow the QPT mechanisms, such as xenon redistribution, burnup gradients, and CONTROL ROD drive mechanism malfunctions, which can cause slow development of a QPT, to be detected. Therefore, these extensions in QPT monitoring frequency do not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGE L4

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS 3.5.2.6 requires AXIAL POWER IMBALANCE to be monitored on a frequency not to exceed two hours. ITS SR 3.2.2.1 requires AXIAL POWER IMBALANCE to be monitored on a frequency not to exceed 12 hours. This frequency ensures that the AXIAL POWER IMBALANCE limits are not violated and takes into account other information and alarms available to the operator in the Control Room. This Surveillance Frequency is acceptable because the mechanisms that can cause AXIAL POWER IMBALANCE, such as xenon redistribution or CONTROL ROD drive mechanism malfunctions that cause slow AXIAL POWER IMBALANCE increases, can be discovered by the operator before the specified limits are violated. The proposed change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Power Company has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change extends the interval for monitoring AXIAL POWER IMBALANCE from 2 hours to 12 hours. This Surveillance Frequency is acceptable because the mechanisms that can cause AXIAL POWER IMBALANCE, such as xenon redistribution or CONTROL ROD drive mechanism malfunctions that cause slow AXIAL POWER IMBALANCE increases, can be discovered by the operator before the specified limits are violated. The frequency of monitoring AXIAL POWER IMBALANCE is not considered an initiator for any previously analyzed accidents. As such, the change does not significantly increase the probability of occurrence of any analyzed event. The change affects only the frequency of monitoring and does not result in any change in the response of equipment to an accident. Therefore, the proposed change does not involve a significant increase the consequences of any accident previously analyzed.

1 **2. Does the change create the possibility of a new or different kind**
1 **of accident from any accident previously evaluated?**
1
1 The proposed change does not necessitate a physical alteration of
1 the plant (no new or different type of equipment will be
1 installed) or changes in parameters governing normal plant
1 operation. Thus, this change does not create the possibility of a
1 new or different kind of accident from any accident previously
1 evaluated.
1
1 **3. Does this change involve a significant reduction in a margin of**
1 **safety?**
1
1 The proposed change extends the interval for monitoring AXIAL
1 POWER IMBALANCE from once every 2 hours to once every 12 hours.
1 Verification of the AXIAL POWER IMBALANCE indication every
1 12 hours continues to ensure that the AXIAL POWER IMBALANCE limits
1 are not violated. Therefore, the proposed change does not involve
1 a significant reduction in a margin of safety.

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
Supp. 1	SR 3.2.1.1	Verify regulating rod groups are within the sequence and overlap limits as specified in the COLR.	~~4 hours when the CONTROL ROD drive sequence alarm is inoperable~~ ~~AND~~ ~~12 hours when the CONTROL ROD drive sequence alarm is OPERABLE~~	DOC M4 TSTF 110, R2
Supp. 1	SR 3.2.1.2	Verify regulating rod groups meet the ~~insertion~~ position (4) limits as specified in the COLR	~~4 hours when the regulating rod insertion limit alarm is inoperable~~ ~~AND~~ ~~12 hours when the regulating rod insertion limit alarm is OPERABLE~~	DOC M4 TSTF 110, R2
Supp 1	SR 3.2.1.3	Verify SDM ~~≥ 1% Δk/k~~ to be within the limit specified in the COLR. TSTF-009, Rev. 1	Within 4 hours prior to achieving criticality	DOC M6

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.~~3~~ 2.1 -5 Verify AXIAL POWER IMBALANCE is within limits as specified in the COLR.	~~1 hour when AXIAL POWER IMBALANCE alarm is inoperable~~ ~~AND~~ 12 hours ~~when AXIAL POWER IMBALANCE alarm is OPERABLE~~

3.5.2.6

TSTF 110, R2

Supp. 1

QPT
3.2.~~4~~3 -5

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.~~4~~3 -5 .1	Verify QPT is within limits as specified in the COLR.	~~12 hours when the QPT alarm is inoperable~~ ~~AND~~ 7 days ~~when the QPT alarm is OPERABLE~~ AND When QPT has been restored to less than or equal to the steady state limit, 1 hour for 12 consecutive hours, or until verified acceptable at ≥ 95% RTP

3.5.2.4.g

TSTF 110,R2

Supp. 1

restored to the acceptable region. ITS 3.2.1 ACTION A provides a specific Required Action to restore the regulating rod groups to within limits with a Completion Time of 2 hours. Two hours is based on CTS 3.5.2.5.b which allows two hours to initiate a reactor shutdown for regulating groups inserted out-of-sequence. This is considered a reasonable period of time for restoration based on operating experience.

10 The addition of a new Condition A for Specification 3.2.1 (described in JFD 7 and 9 above) and the reordering of NUREG 3.2.1 Condition A, require that NUREG Specification 3.2.1 Condition D be revised to read "Required Actions and associated Completion Times of Conditions A, B or C not met" to ensure that appropriate actions are provided should Conditions A or B, in addition to C, not be satisfied. The appropriate action is to remove the unit from the LCO APPLICABILITY which is accomplished by having the unit proceed to MODE 3 with a Completion Time of 12 hours.

11 Not used.

12 A Completion Time of 1 hour is adopted for ITS 3.2.1, Required Action C.1. This Completion Time is consistent with CTS 3.1.3.5. This time period is reasonable since it is still relatively short, there is low probability of an accident occurring in this short period of time and it also limits the potential xenon redistribution.

13 The current licensing basis (CLB) permits 12 hours to place a unit in Hot Shutdown when an LCO is not met. To maintain consistency with current procedures, training and staffing requirements, the 12 hours permitted to place a unit in MODE 3 is retained in the ITS.

1 14 Not used.

1 15 Not used.

1 16 Not used.

17 NUREG LCO 3.2.4 Applicability for Quadrant Power Tilt Limits is MODE 1 with THERMAL POWER > 20% RTP. CTS 3.5.2.4.a Applicability for Quadrant Power Tilt Limits is during power operation above 15%. ITS LCO 3.2.3 for Quadrant Power Tilt Limits adopts the NUREG 3.2.4 Applicability which is slightly less restrictive than the CTS Applicability (Refer to the Discussion of Changes for this section for the justification for this less restrictive change).

(4) <Except as marked>

BASES

ACTIONS (continued)

C.2.2

The SDM and ejected rod worth limit can also be restored by reducing the THERMAL POWER to a value allowed by the regulating rod insertion limits in the COLR. The required Completion Time of 2 hours is sufficient to allow the operator to complete the power reduction in an orderly manner and without challenging the plant systems. Operation for up to 2 hours more in the restricted region shown in the COLR is acceptable, based on the low probability of an event occurring simultaneously with the limit out of specification in this relatively short time period. In addition, it precludes long term depletion with abnormal group insertions or configurations and limits the potential for an adverse xenon redistribution.

(5)

D.1

Required Action and associated Completion Time of Condition A, B, or C are not met,

(5)

If the regulating rods cannot be restored to within the acceptable operating limits for the original THERMAL POWER or if the power reduction cannot be completed within the required Completion Time, then the reactor ~~is~~ must be placed in a MODE MODE 3, in which this LCO does not apply. This Action ensures that the reactor does not continue operating in violation of the peaking limits, the ejected rod worth, the reactivity insertion rate assumed as initial conditions in the accident analyses, or the required minimum SDM assumed in the accident analyses. The required Completion Time of 12 ~~6~~ hours is reasonable, based on operating experience regarding the amount of time required to reach MODE 3 from RTP without challenging ~~plant~~ unit systems.

(2) (5) (2)

SURVEILLANCE REQUIREMENTS

Supp. 1

SR 3.2.1.1

TSTF-110, R2

(12)

This Surveillance ensures that the sequence and overlap limits are not violated. A Surveillance Frequency of 12 hours ~~or 4 hours, depending on whether the CONTROL ROD drive sequence alarm is OPERABLE or not,~~ is acceptable because little rod motion occurs in ~~4~~ hours ~~due to fuel burnup, and the probability of a deviation occurring simultaneously with an inoperable sequence monitor in this relatively short time frame is low.~~ Also, the Frequency

(3)

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

takes into account other information available in the control room for monitoring the status of the regulating rods.

SR 3.2.1.2

whether

groups

~~With an OPERABLE regulating rod insertion limit alarm, verification of the regulating rod insertion limits as specified in the COLR at a Frequency of 12 hours is sufficient to ensure the OPERABILITY of the regulating rod insertion limit alarm and to detect regulating rod banks that may be approaching the group insertion limits, because little rod motion due to fuel burnup occurs in 12 hours.~~ If the insertion limit alarm becomes inoperable, verification of the regulating rod group position at a Frequency of 4 hours is sufficient to detect whether the regulating rod groups may be approaching or exceeding their group insertion limits, although more frequent surveillance is prudent if the regulating rod insertion limit alarm is not OPERABLE. Also, the Frequency takes into account other information available in the control room for monitoring the status of the regulating rods.

position

Supp. 1

5

TSTF 110, R2

SR 3.2.1.3

Prior to achieving criticality, an estimated critical position for the CONTROL RODS is determined. Verification that SDM meets the minimum requirements ensures that sufficient SDM capability exists with the CONTROL RODS at the estimated critical position if it is necessary to shut down or trip the reactor after criticality. The Frequency of 4 hours prior to criticality provides sufficient time to verify SDM capability and establish the estimated critical position.

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 10 and GDC 26~~ UFSAR, Section 3.1

2. 10 CFR 50.46.

(continued)

4 <Except as marked>

BASES

SURVEILLANCE REQUIREMENTS (continued)

Incore Detector System consists of OPERABLE detectors configured as follows:

a. Nine detectors shall be arranged such that there are three detectors in each of three strings and there are three detectors lying in the same axial plane, with one plane at the core midplane and one plane in each axial core half;

b. The axial planes in each core half shall be symmetrical about the core midplane; and

c. The detector strings shall not have radial symmetry.

2

Backup

Figure B 3.2.~~3~~-1 (~~Minimum~~ Incore Detector System for AXIAL POWER IMBALANCE Measurement) depicts an example of this configuration. This arrangement is chosen to reduce the uncertainty in the measurement of the AXIAL POWER IMBALANCE by the ~~Minimum~~ Incore Detector System. For example, the requirement for placing one detector of each of the three strings at the core midplane puts three detectors in the central region of the core where the neutron flux tends to be higher. It also helps prevent measuring an AXIAL POWER IMBALANCE that is excessively large when the reactor is operating at low THERMAL POWER levels. The third requirement for placement of detectors (i.e., radial asymmetry) reduces uncertainty by measuring the neutron flux at core locations that are not radially symmetric.

INSERT B 3.2-23A

5

SR 3.2.~~3~~ 2.1

Supp 1

TSTF-110, R2

~~If the plant computer becomes inoperable, then the Excore System or Minimum Incore Detector System may be used to monitor the AXIAL POWER IMBALANCE. Although these systems do not provide a direct calculation and display of the AXIAL POWER IMBALANCE, a 1 hour Frequency provides reasonable time between calculations for detecting any trends in the AXIAL POWER IMBALANCE that may exceed its alarm setpoint and for undertaking corrective action.~~

~~When the Full Incore Detector System is OPERABLE, the operator receives an alarm if the AXIAL POWER IMBALANCE increases to its alarm setpoint. When the AXIAL POWER IMBALANCE is less then the alarm setpoint, verification of~~ the AXIAL POWER IMBALANCE indication every 12 hours ensures

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.~~3~~2 -5 .1 (continued)

that the AXIAL POWER IMBALANCE limits are not violated and ~~verifies that the alarm system is OPERABLE~~ takes into account other information and alarms available to the operator in the Control Room. This Surveillance Frequency is acceptable because the mechanisms that can cause AXIAL POWER IMBALANCE, such as xenon redistribution or CONTROL ROD drive mechanism malfunctions that cause slow AXIAL POWER IMBALANCE increases, can be discovered by the operator before the specified limits are violated.

Supp. 1

TSTF-110, R2

REFERENCES

1. 10 CFR 50.46.
2. ~~FSAR, Chapter [15].~~ 10 CFR 50.36. -4

BASES

SURVEILLANCE REQUIREMENTS (continued)

differences in the errors applicable for these systems. For QPT measurements using the Incore Detector System, the ~~Minimum~~ Incore Detector System consists of OPERABLE detectors configured as follows:

4 — Backup

a. Two sets of four detectors shall lie in each core half. Each set of detectors shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane.

b. Detectors in the same plane shall have quarter core radial symmetry.

Figure B 3.2 ~~4-1~~ 3-1 (~~Minimum~~ Incore Detector System for QPT Measurement) depicts an example of this configuration. ~~The symmetric incore system for QPT uses the Incore Detector System as described above and is configured such that at least 75% of the detectors in each core quadrant are OPERABLE.~~ 2

4 INSERT B 3.2-35A

SR 3.2 ~~4~~ 3 .1

Supp. 1 TSTF 110, R2

~~Should the plant computer become inoperable, then the Excore System or Minimum Incore Detector System may be used to monitor the QPT. Because these systems do not provide a direct calculation and display of the QPT, performing the calculations at a 12 hour Frequency is sufficient to follow any changes in the QPT that may approach the setpoint because with the exception of CONTROL ROD related effects detected by other systems, QPT changes are slow. This Frequency also provides operators sufficient time to undertake corrective actions if QPT approaches the setpoints.~~

~~When the full symmetrical Incore Detector System is in use, the operator receives an alarm, if QPT increases to the alarm setpoint. When QPT is less than the alarm setpoint,~~ ~~c~~ Checking the QPT indication every 7 days ensures that the operator can determine whether the plant computer software and Incore Detector System inputs for monitoring QPT are functioning properly, and ~~that the monitoring and alarm system remains OPERABLE~~ takes into account other information and alarms available to the operator in the Control Room. This procedure allows the QPT mechanisms, such as xenon redistribution, burnup gradients, and CONTROL ROD drive mechanism malfunctions, which can cause slow development of a QPT, to be detected. Operating

TSTF-110, R2

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4 3.1 (continued)

Supp.1

experience has confirmed the acceptability of a Surveillance Frequency of 7 days.

Following restoration of the QPT to within the steady state limit, operation at ≥ 95% RTP may proceed provided the QPT is determined to remain within the steady state limit at the increased THERMAL POWER level. In case QPT exceeds the steady state limit for more than 24 hours or exceeds the transient limit (Condition A, B, or D), the potential for xenon redistribution is greater. Therefore, the QPT is monitored for 12 consecutive hourly intervals to determine whether the period of any oscillation due to xenon redistribution causes the QPT to exceed the steady state limit again.

REFERENCES

1. 10 CFR 50.46.

2. ~~F~~SAR, Section [ ]. 4

3. ANSI N18.2-1973, American National Standards Institute, August 6, 1973. 12

2 4. BAW 10122A, "Normal Operating Controls," Rev. 1, May 1984. 2

3. 10 CFR 50.36. 4

ITS Section 3.2

ID 45

Subject Incorporate revision to ONS-013 (BWOG-70) and replace JFD annotations with TSTF 216 annotations on NUREG Markup. Revises LCO Note wording.

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Regulating Rod Position Limits

LCO 3.2.1 Regulating rod groups shall be within the physical position, sequence, and overlap limits specified in the COLR.

---------------------------NOTE---------------------------
1 Not required for any regulating rod positioned to perform SR 3.1.4.2.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Regulating rod groups sequence or overlap requirements not met.	A.1	Restore regulating rod groups to within limits.	2 hours
B. Regulating rod groups inserted in restricted or unacceptable region.	B.1	Restore regulating rod groups to within acceptable region.	2 hours
	<u>OR</u>		
	B.2	Reduce THERMAL POWER to less than or equal to THERMAL POWER allowed by regulating rod group position limits.	2 hours
C. Regulating rod groups inserted in unacceptable region.	C.1	Initiate boration to restore SDM to within the limits specified in the COLR.	1 hour

(continued)

CTS

3.2 POWER DISTRIBUTION LIMITS 4

3.2.1 Regulating Rod ~~Insertion~~ Position Limits

LCO 3.2.1 Regulating rod groups shall be within the physical ~~insertion~~ position, sequence, and overlap limits specified in the COLR. 4 — 3.1.3.5 3.5.2.5.b 3.5.2.5.c

APPLICABILITY: MODES 1 and 2. — 3.1.3.5 3.5.2 Applic 3.1.11

Supp. 1

-----NOTE-----
~~This LCO is not applicable while performing~~ SR 3.1.4.2.
Not required for any regulating rod ~~inserted~~ positioned to perform

TSTF 216

ACTIONS

CONDITION	REQUIRED ACTION 7	COMPLETION TIME	
~~A~~ B. Regulating rod groups inserted in restricted ~~operational~~ region, or unacceptable ~~, or sequence or overlap, or any combination~~ not met. 4 9	~~A~~ B.1 Perform SR 3.2.5.1. AND A.2 Restore regulating rod groups to within ~~limits~~ acceptable region. OR	Once per 2 hours 7 24 hours from discovery of failure to meet the LCO	3.5.2.5.c
B. Required Action and associated Completion Time of Condition A not met.	~~B.1~~ B.2 Reduce THERMAL POWER to less than or equal to THERMAL POWER allowed by regulating rod group ~~insertion~~ position limits. 4	2 hours	3.5.2.5.c
A. Regulating rod groups sequence or overlap requirements not met.	A.1 Restore regulating rod groups to within limits.	2 hours 9	3.5.2.5.b

(continued)

2 hours consistent with CTS 3.5.2.5.c and 3.5.2.6, respectively. Due to the deletion of NUREG 3.2.1 Required Action A.1 and the modification of the Completion Time for NUREG 3.2.1 Required Action A.2 to eliminate "...from discovery of failure to meet the LCO" (to retain the CTS requirements for regulating rods not within position limits), NUREG 3.2.1 Condition A was modified to include the condition where regulating rods are inserted in the unacceptable region as well as those inserted in the restricted region. It was necessary to combine the two conditions (NUREG Conditions A and C) because the deletion of "...from discovery of failure to meet the LCO" would allow rotating between the two conditions for an indefinite period of time. NUREG 3.2.1 Required Actions C.2.1 and C.2.2 were deleted since ITS provides similar Required Actions for the combined conditions.

Due to the deletion of Required Action A.1.1 of Specification 3.2.4, the second Completion Time of Required Action A.1.2.1 is no longer necessary and is deleted. NUREG Required Action A.1 for Specifications 3.2.1 and 3.2.3 allows additional time to restore the LCO limits provided power peaking factors are verified within limits (SR 3.2.5.1) once per 2 hours following entry into one of the LCO Conditions for these Specifications. NUREG Required Action A.1.1 for Specification 3.2.4 allows power peaking verification as an alternative to reducing THERMAL POWER and reducing the nuclear overpower trip setpoint. The NUREG provisions that allow use of the power peaking verification are considered appropriate since the LCO for Regulating Rod Insertion Limits, API operating limits, and QPT are used to ensure the core operates within the power peaking limits. However, ONS does not currently have an approved methodology for correlating the power peaking factors to these operating limits. The removal of this requirement maintains action requirements which are consistent with the CTS for similar situations. The Required Actions are renumbered appropriately.

Since the power peaking verification option is currently not a viable alternative at ONS, the adoption of NUREG 3.2.5 would be inappropriate and is not included in the ONS ITS.

1 8 Not used.

9 NUREG 3.2.1 Condition A is entered when the regulating rod groups are inserted into the restricted region, or sequence or overlap requirements are not met. ITS 3.2.1 provides separate ACTIONS for rods inserted into the restricted or unacceptable region and regulating rod groups not meeting sequence or overlap requirements. ITS 3.2.1 Condition B addresses regulating rod groups inserted in the restricted region or unacceptable region. ITS 3.2.1 Condition A addresses regulating rod groups sequence and overlap requirements not met. This was necessary to eliminate any confusion regarding what restoration to within limits means when applied to regulating rods discovered in the restricted region or the unacceptable region. The ITS Required Action B.1 provides a clear and concise requirement that the regulating rods must be

4 <Except as marked>

BASES

LCO (continued)

The overlap between regulating groups provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during regulating rod motion.

Error adjusted maximum allowable setpoints for regulating rod insertion are provided in the COLR. The setpoints are derived by an adjustment of the measurement system independent limits to allow for THERMAL POWER level uncertainty and rod position errors.

position

Actual alarm setpoints implemented in the unit may be more restrictive than the maximum allowable setpoint values to provide additional conservatism between the actual alarm setpoint and the measurement system independent limit. 6

APPLICABILITY

position

The regulating rod sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits maintain the validity of the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions used in the safety analyses. Applicability in MODES 3, 4, and 5 is not required, because neither the power distribution nor ejected rod worth assumptions are exceeded in these MODES. SDM in MODES 3, 4, and 5 is governed by LCO 3.1.1, "SHUTDOWN MARGIN (SDM)."

Supp. 1

TSTF 216

LCO 3.1.1 has been modified by a Note that suspends the LCO requirement ~~during the performance of~~ SR 3.1.4.2, which verifies the freedom of the rods to move. This SR requires the regulating rods to move below the LCO limit, which ~~normally~~ violates the LCO.

for those regulating rods not within the limits of the COLR solely due to testing in accordance with

may

would otherwise

ACTIONS

limits

2

The regulating rod insertion ~~alarm setpoints~~ provided in the COLR are based on both the initial conditions assumed in the accident analyses and on the SDM. Specifically, separate insertion limits are specified to determine whether the unit is operating in violation of the initial conditions (e.g., the range of power distributions) assumed in the accident analyses or whether the unit is in violation of the SDM or ejected rod worth limits. Separate insertion limits are provided because different Required Actions and Completion Times apply, depending on which insertion limit has been

position

position

(continued)

BASES

NOTE: The first five justifications for these changes from NUREG-1430 were generically used throughout the individual Bases section markups. Not all generic justifications are used in each section.

1 The brackets are removed and the proper plant specific information or value is provided.

2 Editorial changes are made for preference, clarity or consistency with the Improved Technical Specifications (ITS) Writer's Guide.

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

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

5 This change reflects changes made to the technical specifications.

6 At multiple locations in the Bases for Section 3.2, paragraphs stating that the actual alarm setpoints may be more conservative than the maximum allowable setpoints are deleted to remove any possible misinterpretation that this is not an acceptable practice in all other situations. Generally, alarm setpoints are conservative with respect to the allowable setpoint. The presence of this paragraph implies that this is not an acceptable practice in other circumstances.

1 7 Not used.

8 Text providing reference to an allowance for movement through the specified Applicability conditions as an exception to ITS LCO 3.0.3 is removed from the Bases because it is unnecessary. The ITS LCO 3.2.3 Required Actions direct the necessary remedial measures. Other Condition statements provide the Required Actions should those remedial measures not be satisfied (i.e. Required Action or associated Completion Time not met). No circumstances should exist that require entry into ITS LCO 3.0.3 and no exceptions should be necessary should entry into ITS LCO 3.0.3 be required. Further, the most limiting Required Action requires that the THERMAL POWER of the unit be reduced to less than 20% RTP. This places the unit in a condition outside of the Applicability of the Specification and simultaneously satisfies the requirements of ITS LCO 3.0.3.

9 ONS was designed and licensed to the proposed Appendix A to 10 CFR 50, which was published in the Federal Register on July 11, 1967 (FR 32FR10213). Appendix A to 10 CFR 50 effective in 1971 and subsequently amended, is somewhat different from the proposed 1967

criteria. UFSAR section 3.1 includes an evaluation of ONS with respect to the proposed 1967 criteria. The NUREG statement concerning the GDC criteria is modified in the ITS to reference the current licensing basis description in the UFSAR.

10 The Bases of NUREG Specifications 3.2.3 and 3.2.4 are revised to indicate the following changes:

1) LCO Bases discussion, which is more appropriate for the Bases discussion for the Action section and which essentially duplicated information in the Bases discussion for the Action section, is removed.

2) Bases discussion of PHYSICS TEST exceptions is removed from 3.2.3 and 3.2.4 Applicability Bases section since the ONS ITS does not allow PHYSICS TEST exceptions in MODE 1.

11 The Bases discussion for NUREG 3.2.4 Required Action F.1 is revised to delete the discussion relating to the basis for the maximum limit and why it is acceptable to operate at maximum quadrant power tilt when less than 20%. This information is provided in the Applicability discussion and need not be repeated.

12 In the Applicable Safety Analysis section of the Bases for NUREG LCO 3.2.4, reference is made to ANSI N18.2-1973 as establishing the requirement that the peak cladding temperature not exceed 2200°F. Similar statements in the NUREG reference 10 CFR 50.46 and the FSAR as the basis for this requirement. The ITS 3.2.3 Bases is revised to cite 10 CFR 50.46 as the appropriate reference.

13 The NUREG Bases discussion for Action A of 3.2.3 defines operation in violation of the AXIAL POWER IMBALANCE limits specified in the COLR as the restricted region of operation. This definition within the Action has been deleted and reference to "the restricted region of operation" has been change to "outside the limits specified in the COLR."

14 The NUREG Bases discussion for the Applicability and Required Action D.1 of 3.2.4 is revised to delete specifying the actual value of maximum percent quadrant power tilt allowed in favor of referring to the maximum limit specified in the COLR.

15 The Applicable Safety Analyses for NUREG B 3.2.1, B 3.2.3 and B 3.2.4 states that fuel cladding damage does not normally occur when the core is operated outside the conditions of these LCOs during normal operation. This sentence is incorrect and is not included in the ONS ITS Bases. Obviously, as the core is operated farther and farther away from the LCOs, the cladding will eventually fail.

16 The Background discussions for NUREG B 3.2.1, B 3.2.3 and B 3.2.4 are modified to define $F_Q(Z)$ and $F^N_{\Delta H}$. These terms are not used in the ITS

since ONS did not adopt the option of verifying power peaking (Refer to JFD 7 of Attachment 5). As a result these terms are not defined in the Definitions section of the ITS. The Bases Background discussion of $F_Q(Z)$ and $F^N_{\Delta H}$, as modified, remains valid and these terms were defined to aid in the discussion.

ITS Section 3.3

ID 22

Subject Revise JFD 29 of Att. 6 to remove incorrect reference to ONS 003.

additional guidance and clarification on the proper usage of the Required Actions without changing the intent of the ACTIONS.

26 The last paragraph of the Applicable Safety Analyses discussion for the Nuclear Overpower-High Setpoint function (on Base page B 3.3-12) is deleted since it is confusing and not entirely true. At ONS, the Variable Low Pressure and Nuclear Overpower Flux/Flow Imbalance trips are credited in all safety analyses regardless of the transient power excursion.

1 27 Not used.

1 28 Not used.

29 ITS SR 3.3.3.1 retains the current test frequency (CTS Table 4.1-1, Item 1) for Reactor Trip Modules. Therefore, the NUREG SR 3.3.3.1 Bases is modified to reflect the current licensing bases and reference to the
1 BAW topical report is removed.

30 The third paragraph of the LCO Bases for NUREG Specification 3.3.3 is revised to provide a discussion in terms of Reactor Trip Modules for consistency with the LCO requirement rather than in general terms of RPS channels.

ITS Section 3.3

ID 24

Subject Revise NUREG Bases Markup for 3.3 to refer to ESPS rather than ESFAS, 2nd paragraph, page B 3.3-48

4 <Except as marked> Analog

BASES

BACKGROUND (continued)

The ESPS ~~ESFAS,~~ in conjunction with the actuated equipment, provides protective functions necessary to mitigate ~~Design Basis~~ main Accidents ~~(DBAs)~~, specifically the loss of coolant accident (LOCA) and steam line break (SLB) events. The ESPS ~~ESFAS~~ relies on the OPERABILITY of the automatic actuation logic for each component to perform the actuation of the selected systems of LCO 3.3.7. M

Engineered ~~Safety Feature Actuation~~ Safeguards Protective System Bypasses

No provisions are made for maintenance bypass of ESPS instrumentation channels. Operational bypass of certain channels is necessary to allow accident recovery actions to continue and, for some channels, to allow ~~reactor~~ unit shutdown without spurious ESPS actuation. P — 4 2

Supp 1

The ESPS RCS pressure instrumentation channels include permissive bistables that allow manual bypass when reactor pressure is below the point at which the low and low low pressure trips are required to be OPERABLE. Once permissive conditions are sensed, the RCS pressure trips may be manually bypassed. Bypasses are automatically removed when bypass permissive conditions are exceeded.

2 INSERT B 3.3-48A

~~Each High RB Pressure channel may be manually bypassed after the other two channels in the Parameter have tripped. The manual bypass allows operators to take manual control of ESF Functions after initiation to allow recovery actions. The bypass may be manually removed and is automatically removed when RB pressure returns to below the trip setpoint.~~ 8

Reactor Coolant System Pressure

The RCS pressure is monitored by three independent pressure transmitters located in the RB. These transmitters are separate from the transmitters that feed the Reactor Protect~~on~~ion System (RPS). Each of the pressure signals generated by these transmitters is monitored by four bistables to provide two trip signals, at ≥ 1500 psig and ≥ 500 psig, and two bypass permissive signals, at ≥ 17~~00~~50 psig and ≥ 900 psig. 2 — ≥ 5

The outputs of the three bistables, associated with the low RCS pressure, ≥ 1500 psig, trip drive relays in two sets

(continued)

ITS Section 3.3

ID 42

Subject Incorporate revision to ONS-021 (BWOG-85) - Deletes SR 3.3.10.3, clarifies Bases of 3.3.9 & 10 to clarify Channel Check requires one decade of overlap.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.10.2	-------------------NOTE-------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	 18 months

1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1 (continued)

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. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that 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.

1 The agreement criteria includes an expectation of one decade
1 of overlap when transitioning between neutron flux
1 instrumentation. During a power reduction near the bottom
1 of the scale for the wide range monitors, a source range
1 monitor reading is expected with at least one decade of
1 overlap.

The Frequency, equivalent to 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 potentially more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels. When operating in Required Action A.1, CHANNEL CHECK is still required. However, in this condition, a redundant source range may not be available for comparison. CHANNEL CHECK may still be performed via comparison with wide range detectors, if available,

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.9.2

and verification that the OPERABLE source range channel is energized and indicating a value consistent with current unit status.

For source range neutron flux channels, CHANNEL CALIBRATION is a complete check and readjustment of the channels from the preamplifier input to the indicators. This test verifies the channel responds to measured parameters within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests.

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult. The detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output.

The Frequency of 18 months is based on demonstrated instrument CHANNEL CALIBRATION reliability over an 18 month interval, such that the instrument is not adversely affected by drift.

REFERENCES

1. 10 CFR 50.36.

BASES

ACTIONS

B.1 and B.2 (continued)

operating experience and allow the operators sufficient time to manually insert the CONTROL RODS prior to opening the CRD breakers.

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1

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 the two 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; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that 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.

1 The agreement criteria includes an expectation of one decade
1 of overlap when transitioning between neutron flux
1 instrumentation. During a power increase near the top of
1 the scale for the source range monitors, a wide range
1 monitor reading is expected with at least one decade of
1 overlap. Without such overlap, the wide range monitors are
1 considered inoperable unless it is clear that a source range
1 monitor inoperability is responsible for the lack of the
1 expected overlap.

The Frequency, equivalent to 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,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1 (continued)

the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but potentially more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels.

When operating in Required Action A.1, CHANNEL CHECK is still required. However, in this condition, a redundant wide range may not be available for comparison. CHANNEL CHECK may still be performed via comparison with power or source range detectors, if available, and verification that the OPERABLE wide range channel is energized and indicates a value consistent with current unit status.

SR 3.3.10.2

For wide range neutron flux channels, CHANNEL CALIBRATION is a complete check and readjustment of the channels, from the preamplifier input to the indicators. This test verifies the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests.

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult. In addition, the detectors are of simple construction, and any failures in the detectors will be apparent as a change in channel output. The Frequency is based on operating experience and consistency with the typical industry refueling cycle and is justified by demonstrated instrument reliability over an 18 month interval such that the instrument is not adversely affected by drift.

1
1
1

(continued)

BASES (continued)

REFERENCES 1. 10 CFR 50.36.

M16

MODE 2, MODES 3, 4 + 5 with any CRD trip breaker in the closed position and the CRD System is capable of withdrawal

3.5 INSTRUMENTATION SYSTEMS

A1

3.5.1 Operation Safety Instrumentation

Applicability

Applies to unit instrumentation and control systems.

Objective

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

L13

LCO + Applic

3.5.1.1 The reactor shall not be in a startup mode or in a critical state unless the requirements of Table 3.5.1-1, Column C are met, with the exception of Items 20, 21, and 22. For Items 20, 21, and 22, the requirements are specified in Specification 3.5.7.

A1

3.5.1.2 In the event that the number of protective channels operable falls below the limit given under Table 3.5.1-1, Column C; operation shall be limited as specified in Column D.

A1

3.5.1.3 For on-line testing or in the event of a protective instrument or channel failure, a key-operated channel bypass switch associated with each reactor protective channel may be used to lock the channel trip relay in the untripped state. Status of the untripped state shall be indicated by a light. Only one channel bypass key shall be accessible for use in the control room. Only one channel shall be locked in this untripped state or contain a dummy bistable at any one time.

3.5.1.4 For on-line testing or maintenance during reactor power operation, a key-operated shutdown bypass switch associated with each reactor protective channel may be used in conjunction with a key-operated channel bypass switch as limited by 3.5.1.3. Status of the shutdown bypass switch shall be indicated by a light.

(SEE 3.3.1)

Supp. 1

SR 3.3.10.1

RA B.1
RA B.2

3.5.1.5 During startup when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall not be greater than that readable on the source range instruments until the one decade overlap is achieved.

suspend positive reactivity additions + open CRD trip breakers within 1 hour

M15

TSC 95-03

the displays associated with the required channels. This less restrictive change is consistent with the NUREG.

L27 CTS Table 4.1-1, items 54 and 57, requires a monthly functional test of the containment high range radiation monitor and containment hydrogen monitor instrument channels. This monthly functional test is not included in ITS. Such a test is typically required when the instrumentation provides a safety related automatic actuation function. This instrument channel provides information only, and as such, a CHANNEL FUNCTIONAL TEST is not appropriate, nor required. This change is also consistent with the NUREG.

L28 CTS Table 3.5.1-1, Column D requires the unit be placed in hot shutdown within 12 hours when less than two source range channels are OPERABLE and rated power is $\leq$ 10% as shown on the power range channels and $\leq 4 \times 10^{-4}$ % rated power as shown on the wide range channels. Comparable ITS Required Actions do not require the unit be placed in hot shutdown. Therefore, the proposed ITS Required Actions are less restrictive in this aspect. However, other more appropriate required actions are added to replace the CTS required action (Refer to DOC M13). This change is also consistent with the NUREG.

L29 Not used.

1 L30 Not used.

L31 CTS Table 3.5.1-1, Column D requires the unit be placed in hot shutdown within 12 hours when less than two wide range channels are OPERABLE. ITS 3.3.10, Required Action A.1 only requires that power be reduced to $< 4 \times 10^{-4}$% RTP when one channel is inoperable. The proposed change is less restrictive since the CTS defines Hot Shutdown as the reactor having a K_{eff} of ≤ 0.99 and the reactor could have a K_{eff} of > 0.99 with power reduced below 4×10^{-4}% RTP as allowed by Required Action A.1. The proposed change is consistent with the NUREG.

L32 CTS Table 4.1-1, Column "Test," Items 5 and 6 require a functional test be performed on the source range and wide range channels prior to startup. This requirement is not retained in the ITS. Consistent with the NUREG, a CHANNEL CALIBRATION of the source range and wide range instruments is added (Refer to DOC M14). Because the calibration by definition encompasses the functional test, performance of the calibrations will ensure that testing is consistent with CTS requirements. The frequency of this testing is now based strictly on the time since its last performance and not dependent upon whether or not the unit is in startup. This change is acceptable, based on operating experience which demonstrates the source and wide range instruments are highly reliable.

L33 CTS 3.7.5.1 requires performance of SR 3.7.1.11 (Keowee emergency start) and SR 3.7.1.14 (EPSL automatic transfer). SR 3.7.1.11 verifies that each Keowee Hydro Unit (KHU) can emergency start from each control room,

attain rated speed and voltage within 23 seconds of an emergency start initiate, and be synchronized to the grid and loaded. The test is performed by manually starting one KHU from the Unit 1 and 2 Control Room and the other KHU from the Unit 3 Control Room. The accident analyses do not take credit for a manual Keowee start during operation above Cold Shutdown. Therefore, the requirement to test this function during operation above Cold Shutdown is not retained. This function is required to be OPERABLE during MODES 5 and 6 and during movement of irradiated fuel assemblies by ITS 3.3.22, "EPSL Manual Keowee Emergency Start Function."

L34 CTS 3.5.1.1 Applicability for the TSV Closure instrumentation channels is while in the startup mode or when the reactor is in a critical state. This is considered encompassed by ITS MODES 1 and 2. ITS 3.3.15 Applicability is in MODES 1, 2, and 3 except when all TSVs are closed. The exception of "when all TSVs are closed" is a less restrictive change and is consistent with comparable NUREG requirements (Table 3.3.11-1, Note c). The exception is appropriate since the TSVs are already performing their safety function when they are closed.

L35 CTS 3.8.10 requires the radiation monitor associated with the purge system valve isolation to be tested and verified OPERABLE immediately prior to refueling operations. CTS Table 4.1-2, Item 4, requires this functional test be performed "Prior to Refueling." ITS 3.3.16 Applicability is during CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment. ITS SR 3.3.16.2 requires the testing be performed once each refueling outage prior to CORE ALTERATIONS or beginning movement of irradiated fuel assemblies within containment. Permitting the specified testing to be conducted prior to beginning movement of irradiated fuel assemblies within containment in lieu of immediately prior to refueling operations is a less restrictive requirement upon unit operation (and is more stringent than the NUREG). Requiring performance of SR 3.3.16.2 once each refueling outage prior to CORE ALTERATIONS or prior to beginning movement of irradiated fuel assemblies within containment represents a reasonable relaxation of the CTS surveillance frequency. This continues to ensure that this function is verified prior to irradiated fuel assembly handling within containment.

L36 CTS Table 3.5.1-1, Column D requires the unit to be in hot shutdown within 24 hours when one or more TSV Closure Instrumentation channels is inoperable and Note (e) to the Table requires the unit be placed in Cold Shutdown within the following 72 hours if the minimum conditions are not met. ITS 3.3.15 ACTION A is added to require the TSVs to be declared inoperable within 1 hour (also, see DOC L19). ITS 3.7.2, Turbine Stop Valves, then dictates the required action for inoperable TSVs. With one or more TSVs inoperable in MODE 1, Required Action A.1 requires the TSVs be restored to OPERABLE status within 8 hours or Required Action B.1 requires the unit be in MODE 2 in 6 hours. Therefore, this portion of ITS is more restrictive since the unit must be in MODE 2 within 15 hours of an inoperable TSV Closure instrumentation channel where CTS required

the unit be in hot shutdown (equivalent to ITS MODE 3) within 24 hours. ITS 3.7.2 Action C allows 8 additional hours to close an inoperable TSV when in MODE 2 or 3 (total of 23 hours). In addition, if it were not closed, then an additional 12 hours (on top of the eight hours) is allowed to place the unit in MODE 3 and 18 hours to place the unit in MODE 4. This results in allowing a total of 35 hours to be in MODE 3 and 41 hours to be in MODE 4 from initial discovery of it being inoperable in MODE 1. This compares to the CTS time allowed to place the unit in Hot Shutdown (MODE 3) of 24 hours. Therefore, an additional 11 hours is allowed to place the unit in MODE 3. The additional time is reasonable considering the low probability of an accident occurring during this time period that would require closure of the TSVs. The more restrictive aspects of this change are addressed in DOC M22. The proposed less restrictive ITS Shutdown Times requirements are consistent with ITS 3.7.2, which is consistent with the NUREG.

L37 CTS 3.4.3.b requires a flow path with no OPERABLE emergency feedwater flow indicators to be restored to OPERABLE status within 72 hours. ITS 3.3.8 Required Action C.1 allows 7 days to restore an inoperable flow indicator when both are inoperable. Required Action C.1 allows an additional 4 days for restoration of a single channel when no channels are OPERABLE. The additional time to restore at least one channel allowed by Required Action C.1 is considered appropriate based on the relatively low probability of an event requiring PAM instrumentation and the availability of alternate means to obtain required information. This less restrictive change is consistent with the NUREG.

LESS RESTRICTIVE CHANGE L30

1 Not used.

Intentionally blank

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.9.3	Verify at least one decade overlap with intermediate range neutron flux channels.	Once each reactor startup prior to source range counts exceeding 10^5 cps if not performed within the previous 7 days

Supp. 1

TSTF 264

3.3 INSTRUMENTATION

3.3.10 ~~Intermediate~~ wide Range Neutron Flux 4

LCO 3.3.10 Two ~~intermediate~~ wide range neutron flux channels shall be OPERABLE. 3.5.1.1 T 3.5.1-1, Col C Item 1

APPLICABILITY: MODE 2, MODES 3, 4, and 5 ~~When~~ with any ~~CONTROL ROD~~ drive (CRD) trip breaker ~~is~~ in the closed position and the CRD System ~~is~~ capable of rod withdrawal 19 3.5.1.1 T 3.5.1-1, Note (6)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One required 4 channel inoperable.	A.1 Reduce THERMAL POWER to < ~~1E-10 amp~~ 4E-4% RTP. 22	2 hours	T3.5.1-1 Col D
B. Two required 4 channels inoperable.	B.1 Suspend operations involving positive reactivity changes. AND B.2 Open CRD trip breakers.	Immediately 1 hour	3.5.1.5 3.5.1.5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY	
SR 3.3.10.1 Perform CHANNEL CHECK.	12 hours	4.1.1 T4.1-1 Col "Check" Item 5 + 3.5.1.5

(continued)

Supp. 1

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.10.2	-------------------NOTE------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	9 ~~[18]~~ months 1
SR 3.3.10.3	Verify at least one decade overlap with power range neutron flux channels.	Once each reactor startup prior to intermediate range indication exceeding 1E-6 amp if not performed within the previous 7 days

DOC M14

Supp. 1

TSTF 264

7 NUREG SR 3.3.1.7, SR 3.3.5.4 and SR 3.3.11.4 are not adopted. Consistent with current licensing basis, response time testing of RPS, ESPS, and EFIC (ONS equivalent) circuitry is not performed. Plant equipment does not readily lend itself to such testing.

1 8 TSTF-xxx clarifies that agreement criteria include one decade of overlap
1 during a power increase when transitioning between the intermediate
1 range monitors and the power range monitors. ONS uses wide range
1 monitors which have an upper range up to 200% power, therefore, there is
1 no need to transition between monitors. As such, this portion of TSTF
1 is not included.

9 The unit specific design of the ONS ESPS provides for three analog channels of instrumentation for each of the monitored parameters. These three channels provide the required input to each of the eight automatic actuation logic channels. Contrary to the system design depicted in the requirements of the NUREG, these three instrument channels provide input to both trains of automatic actuation logic channels. This unit specific design difference required the deletion of the phrase "in each ESFAS train" from NUREG LCO 3.3.5 as well as appropriate changes to the Bases.

1 10 Not used.

11 NUREG Table 3.3.5-1 lists equipment actuated by each of the ESPS Parameters. This list is incomplete and has not been included in the ITS in favor of the more complete list provided in the Bases. Also, the term "Setpoint" is removed from the Parameter title in the Table and from Required Actions B.2.1, B.2.2 and B.2.3 since the setpoint is not a parameter. Referring to a setpoint as a parameter is inconsistent with the identification of other parameters and functions throughout the NUREG. Removal of this information represents no actual change in requirements.

12 In the conversion to ITS, NUREG Table 3.3.17-1, Post Accident Monitoring Instrumentation, is modified by Note c which indicates that the containment isolation valve position indication requirements apply only to containment isolation valves that are electrically controlled. This is consistent with ONS Regulatory Guide 1.97 response for CIV position indication and the NRC's Safety Evaluation Report for this response.

1 13 Not used.

14 The Functions specified in NUREG LCO 3.3.6 are modified to match the Functions as presented in the CTS, UFSAR, and other design basis documents. NUREG Specification 3.3.7 has also been modified to include ONS unit specific terminology. These changes were made to provide requirements consistent with the design of the ONS and consistent with the specific terminology and names associated with the ONS ESPS.

15 The Frequency of SR 3.3.7.1 has been changed to 31 days. The NUREG Frequency of 31 days on a STAGGERED TEST BASIS is not consistent with the CTS. The CTS requires this testing monthly, which is considered administratively equivalent to the proposed 31 day Frequency.

1 16 Not used.

1 17 Not used.

18 At ONS, the source range detectors are not turned off at power because the wide range instrument channels use one of the same fission chambers that supplies the source range instrument channels. To monitor the source range, two fission chambers are used and the outputs are added together. Only one fission chamber is used for the wide range output.

19 NUREG 3.3.10 Applicability is changed to specify that the wide range instrument channel is required in MODE 2 and in MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal. The addition of "MODES 3, 4, and 5" to the second statement of the Applicability was made to maintain the CTS allowance provided by Table 3.5.1-1 Note (b). This Note defines the upper limit of the applicable MODES for the required wide range instrument channel as being 10% indicated neutron power. Without the addition of the appropriate MODES, to the second statement of the Applicability for ITS 3.3.10, a wide range channel would be required at all times in MODE 1. This requirement is inconsistent with the RPS design requirement of the wide range instrument channels which is to provide indication of neutron power while operating at low power levels (MODE 2). The required indication of neutron power level is provided by the power range instruments while in MODE 1.

1 20 Not used.

1 21 Not used.

22 The value in NUREG Specifications 3.3.9 and 3.3.10 for THERMAL POWER level as indicated on the wide range neutron flux channels is changed to reflect the appropriate ONS plant specific value.

23 NUREG SR 3.3.1.2 is modified to require a comparison of calorimetric heat balance to power range channel output and adjustment when calorimetric exceeds power range level output by ≥ 2%. This is consistent with the description of SR 3.3.1.2 in the NUREG Bases. Current NUREG wording requires a verification that they are within 2% but provides no action if acceptance criteria is not met.

1 24 Not used.

25 NUREG 3.3.4, Condition A is modified to specifically state it applies to a diverse trip function being inoperable rather than an undervoltage or

BASES (continued)

SURVEILLANCE REQUIREMENTS

The SRs for each RPS Function are identified by the SRs column of Table 3.3.1-1 for that Function. Most ~~Functions~~ are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION ~~and RPS RESPONSE TIME~~ testing. 5

The SRs are modified by a Note. The ~~(first)~~ Note directs 2 the reader to Table 3.3.1-1 to determine the correct SRs to perform for each RPS Function.

13

~~Reviewer's Note: The CHANNEL FUNCTIONAL TEST Frequencies are based on approved topical reports. For a licensee to use these times, the licensee must justify the Frequencies as required by the NRC Staff SER for the topical report.~~

SR 3.3.1.1

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 two 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; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined ~~by the unit staff~~ 2 based on a combination of the channel instrument uncertainties, including isolation, 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 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. Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

8
~~INSERT TSTF 264~~
Supp. 1

The Frequency, ~~about once~~ equivalent to 2 every shift, is based on operating experience that demonstrates channel failure is rare. Since

(continued)

BASES

ACTIONS

C.1 (continued)

required (5)

operation is allowed with one or more source range neutron flux channels inoperable. The ability to continue operation is justified because the instrumentation does not provide a safety function during high power operation. However, actions are initiated within 1 hour to restore the channel(s) to OPERABLE status for future availability. The Completion Time of 1 hour is sufficient to initiate the action. The action must continue until channels are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1

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 the two 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; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

(2)

Agreement criteria are determined ~~by the unit staff~~, based on a combination of the channel instrument uncertainties, including isolation, 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 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. Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale. (17)

Supp. 1 INSERT B 3.3-83A

TSTF-264

equivalent to

The Frequency, ~~about once~~ every shift, is based on operating experience that demonstrates channel failure is rare. Since (2)

(continued)

INSERT B3.3-83A

The agreement criteria includes an expectation of one decade of overlap when transitioning between neutron flux instrumentation. During a power reduction near the bottom of the scale for the ~~intermediate~~wide range monitors, a source range monitor reading is expected with at least one decade of overlap.

4

BASES

ACTIONS

B.1 and B.2 (continued)

4 wide

reactor in the next lowest condition for which the ~~intermediate~~ range instrumentation is not required. This involves providing power level indication on the source range instrumentation by immediately suspending operations involving positive reactivity changes and, within 1 hour, placing the reactor in the tripped condition with the CRD trip breakers open. The Completion Times are based on unit operating experience and allow the operators sufficient time to manually insert the CONTROL RODS prior to opening the CRD breakers.

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1

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 the two 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; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

2

Agreement criteria are ~~determined by the unit staff~~ based on a combination of the channel instrument uncertainties, including isolation, 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 limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. 17

Supp. 1

INSERT B 3.3-88A

TSTF 264

equivalent to 2

The Frequency, ~~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

(continued)

INSERT B3.3-88A

The agreement criteria includes an expectation of one decade of overlap when transitioning between neutron flux instrumentation. During a power increase near the top of the scale for the source range monitors, a wide range ~~an intermediate range~~ monitor reading is expected with at least one decade of overlap. Without such overlap, the wide~~intermediate~~ range monitors are considered inoperable unless it is clear that a source range monitor inoperability is responsible for the lack of the expected overlap. ~~Further during a power reduction near the bottom of the scale for the power range monitors, an intermediate range monitor reading is expected with at least one decade overlap. Without such overlap, the intermediate range monitors are considered inoperable unless it is clear that a power range channel inoperability is responsible for the lack of expected overlap.~~

4

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1 (continued)

failure of redundant channels. potentially 2 The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels.

When operating in Required Action A.1, CHANNEL CHECK is still required. However, in this condition, a redundant 4 wide ~~intermediate~~ range ~~is not~~ may not be 2 available for comparison. CHANNEL CHECK may still be performed via comparison with power or source range detectors, if available, and verification that the OPERABLE ~~intermediate~~ wide 4 range channel is energized and indicates a value consistent with current unit status.

SR 3.3.10.2

For ~~intermediate~~ wide range neutron flux channels, CHANNEL CALIBRATION is a complete check and readjustment of the channels, from the preamplifier input to the indicators. This test verifies the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests.

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult. In addition, the detectors are of simple construction, and any failures in the detectors will be apparent as a change in channel output. The Frequency is based on operating experience and consistency with the typical industry refueling cycle and is justified by demonstrated instrument reliability over an [18] month interval such that the instrument is not adversely affected by drift. 1

Supp. 1

SR 3.3.10.3

SR 3.3.10.3 is the verification within ~~7 days~~ prior to reactor start~~up~~ of one decade of overlap with the power range neutron flux instrumentation prior to intermediate range indication exceeding 1E-6 amp. This ensures a 5

(continued)

BASES

Supp. 1

SURVEILLANCE REQUIREMENTS	SR 3.3.10.3 (continued) ~~continuous source of power indication during the approach to criticality. Failure to perform this Surveillance leaves the unit in a condition where the intermediate range channels provide adequate protection until the verification can be made.~~ ~~The test may be omitted if performed within the previous 7 days based on operating experience, which shows that intermediate range instrument overlap does not change appreciably within this test interval.~~ (5)
REFERENCES	~~None.~~ 10 CFR 50.36, (2)

ITS Section 3.3

ID 55

Subject Incorporate revision to ONS-001 (BWOG-060) - Removes separate SR for calibrating power range channel and consolidates with SR 3.3.1.3

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.2	-------------------NOTE------------------- Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP. -- Compare results of calorimetric heat balance calculation to the power range channel output and adjust power range channel output if calorimetric exceeds power range channel output by ≥ 2% RTP.	24 hours
SR 3.3.1.3	-------------------NOTE------------------- Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP. -- Compare out of core measured AXIAL POWER 1 IMBALANCE (API_0) to incore measured AXIAL 1 POWER IMBALANCE (API_1) as follows: 1 1 $(RTP/TP)(API_0 - API_1)$ = imbalance error 1 Adjust power range channel output if the absolute difference between the power range and incore measurements is ≥ 2% RTP.	31 days
1 SR 3.3.1.4	Perform CHANNEL FUNCTIONAL TEST.	45 days on a STAGGERED TEST BASIS

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
1	SR 3.3.1.5	-------------------NOTE------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	18 months

Table 3.3.1-1 (page 1 of 1)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION B.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Nuclear Overpower –				
a. High Setpoint	1,2(a),3(d)	C	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.4 SR 3.3.1.5	≤ 105.5% RTP
b. Low Setpoint	2(b),3(b) 4(b),5(b)	D	SR 3.3.1.1 SR 3.3.1.5	≤ 5% RTP
2. RCS High Outlet Temperature	1,2	C	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 618°F
3. RCS High Pressure	1,2(a),3(d)	C	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 2355 psig
4. RCS Low Pressure	1,2(a)	C	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≥ 1800 psig
5. RCS Variable Low Pressure	1,2(a)	C	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	As specified in the COLR
6. Reactor Building High Pressure	1,2,3(c)	C	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 4 psig
7. Reactor Coolant Pump to Power	1,2(a)	C	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	>2% RTP with ≤ 2 pumps operating
8. Nuclear Overpower Flux/Flow Imbalance	1,2(a)	C	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5	As specified in the COLR
9. Main Turbine Trip (Hydraulic Fluid Pressure)	≥ 30% RTP	E	SR 3.3.1.4 SR 3.3.1.5	≥ 800 psig
10. Loss of Main Feedwater Pumps (Hydraulic Oil Pressure)	≥ 2% RTP	F	SR 3.3.1.4 SR 3.3.1.5	≥ 75 psig
11. Shutdown Bypass RCS High Pressure	2(b),3(b) 4(b),5(b)	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 1720 psig

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal.

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) When not in shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal.

BASES

ACTIONS

E.1 (continued)

OPERABLE. To achieve this status, THERMAL POWER must be reduced < 30% RTP. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach 30% RTP from full power conditions in an orderly manner without challenging unit systems.

F.1

If the Required Action and associated Completion Time of Condition A are not met and Table 3.3.1-1 directs entry into Condition F, the unit must be brought to a MODE in which the specified RPS trip Function is not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced < 2% RTP. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach 2% RTP from full power conditions in an orderly manner without challenging unit systems.

SURVEILLANCE REQUIREMENTS

The SRs for each RPS Function are identified by the SRs column of Table 3.3.1-1 for that Function. Most Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION testing.

The SRs are modified by a Note. The Note directs the reader to Table 3.3.1-1 to determine the correct SRs to perform for each RPS Function.

SR 3.3.1.1

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 two instrument channels could be an indication of

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1 (continued)

excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined based on a combination of the channel instrument uncertainties, including isolation, 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 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. Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

The Frequency, equivalent to 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 the displays associated with the LCO's required channels.

For Functions that trip on a combination of several measurements, such as the Nuclear Overpower Flux/Flow Imbalance Function, the CHANNEL CHECK must be performed on each input.

SR 3.3.1.2

This SR is the performance of a heat balance calibration for the power range channels every 24 hours when reactor power is > 15% RTP. The heat balance calibration consists

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued)

of a comparison of the results of the calorimetric with the power range channel output. The outputs of the power range channels are normalized to the calorimetric. If the calorimetric exceeds the Nuclear Instrumentation System (NIS) channel output by ≥ 2% RTP, the NIS is not declared inoperable but must be adjusted. If the NIS channel cannot be properly adjusted, the channel is declared inoperable. A Note clarifies that this Surveillance is required to be performed only if reactor power is ≥ 15% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels, calorimetric data are less accurate.

The power range channel's output shall be adjusted consistent with the calorimetric results if the calorimetric exceeds the power range channel's output by ≥ 2% RTP. The value of 2% is adequate because this value is assumed in the safety analyses of UFSAR, Chapter 15 (Ref. 2). These checks and, if necessary, the adjustment of the power range channels ensure that channel accuracy is maintained within the analyzed error margins. The 24 hour Frequency is adequate, based on unit operating experience, which demonstrates the change in the difference between the power range indication and the calorimetric results rarely exceeds a small fraction of 2% in any 24 hour period. Furthermore, the control room operators monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

A comparison of power range nuclear instrumentation
channels against incore detectors shall be performed at a
31 day Frequency when reactor power is ≥ 15% RTP. A Note
clarifies that 24 hours is allowed for performing the first
Surveillance after reaching 15% RTP. If the absolute
difference between the power range and incore measurements
1 is ≥ 2% RTP, the power range channel is not inoperable, but
1 an adjustment of the measured imbalance to agree with the
incore measurements is necessary. If the power range

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.3 (continued)

channel cannot be properly recalibrated, the channel is declared inoperable. The calculation of the Allowable Value envelope assumes a difference in out of core to incore measurements of 2.5%. Additional inaccuracies beyond those that are measured are also included in the setpoint envelope calculation. The 31 day Frequency is adequate, considering that long term drift of the excore linear amplifiers is small and burnup of the detectors is slow. Also, the excore readings are a strong function of the power produced in the peripheral fuel bundles, and do not represent an integrated reading across the core. The slow changes in neutron flux during the fuel cycle can also be detected at this interval.

1 SR 3.3.1.4

A CHANNEL FUNCTIONAL TEST is performed on each required RPS channel to ensure that the entire channel will perform the intended function. Setpoints must be found within the Allowable Values specified in Table 3.3.1-1. 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 BAW-10167 (Ref. 7).

The Frequency of 45 days on a STAGGERED TEST BASIS is consistent with the calculations of Reference 7 that indicate the RPS retains a high level of reliability for this test interval.

1 SR 3.3.1.5

A Note to the Surveillance indicates that neutron detectors are excluded from CHANNEL CALIBRATION. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure virtually instantaneous response.

(continued)

BASES

1 SURVEILLANCE REQUIREMENTS

SR 3.3.1.5 (continued)

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature detectors (RTD)sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

The Frequency is justified by the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. UFSAR, Chapter 7.
2. UFSAR, Chapter 15.
3. 10 CFR 50.49.
4. EDM-102, "Instrument Setpoint/Uncertainty Calculations."
5. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1979.
6. BAW-1893, "Basis for Raising Arming Threshold for Anticipating Reactor Trip on Turbine Trip," October 1985.

(continued)

BASES

REFERENCES (continued)	7. BAW-10167, May 1986.
	8. 10 CFR 50.36.

A1 <Except as marked>

Table 4.1-1

INSTRUMENT SURVEILLANCE REQUIREMENTS

~~Channel Description~~ Function	Check	Test	Calibrate		Remarks
1. Protective Channel Coincidence Logic in the Reactor Trip Modules	NA	MO	NA		
2. Control Rod Drive Trip Breaker, SCR Control Relays E and F	NA	MO(1)	NA	(1)	This test shall independently confirm the operability of the shunt trip device and the undervoltage device. <SEE 3.3.3 + 4>
1.c ~~3.~~ Power Range Amplifier SR 3.3.1.2	~~BS(1)~~ 24 hrs L9	NA	(1) For Funct. 8 only	(1)	Heat balance check ~~each shift~~ 24 hrs L9. Heat balance calibration whenever indicated core thermal power exceeds neutron power by more than 2 percent.
1.a, 1.b, 8 ~~4.~~ ~~Power Range~~ Nuclear Overpower, Nuclear Overpower Flux/Flow Imbalance A5	SR 3.3.1.1 ~~BS~~ 12 hrs A9	SR 3.3.1.4 45 Days STB For Funct. 1.a + 8 only	SR 3.3.1.3 ~~MO~~(1)(2) 31 days A9	(1) (2)	Using incore instrumentation. SR 3.3.1.3. Axial offset upper and lower chambers after each startup if not done previous week. L8
5. Wide Range	BS(1)	PS	NA	(1)	When in service. <SEE 3.3.8, 9 + 10>
6. Source Range	BS(1)	PS	NA	(1)	When in service. <See 3.3.8, 9, 10, 3.9.2>
2 ~~7.~~ Reactor Coolant Temperature	SR 3.3.1.1 ~~BS~~	SR 3.3.1.4 45 Days STB	SR 3.3.1.5 ~~RF~~		Add SR 3.3.1.2 Note L14
3, 11 ~~8.~~ High Reactor Coolant Pressure	~~BS~~	45 Days STB	~~RF~~		Add SR 3.3.1.3 Note L15
4 ~~9.~~ Low Reactor Coolant Pressure	~~BS~~ 12 hrs	45 Days STB A9	~~RF~~ 18 months A9		
8 ~~10.~~ Flux-Reactor Coolant Flow Comparator	~~BS~~	45 Days STB A9	~~RF~~		Add SR 3.3.1.5 Note for Functions 2, 3, 4, 5, 6, 7, 8, 9, 10 + 11 L20
5 ~~11.~~ Reactor Coolant Pressure Temperature Comparator	~~BS~~	45 Days STB	~~RF~~		Add SR 3.3.1.5 for Functions 1.a + 1.b M35

Supp.1

Supp.1 Supp.1 Supp.1

A1 <Except as marked>

Supp. 1

Table ~~4.1-1~~ 3.3.1-1 (CONTINUED)

~~Channel Description~~ Function	SR 3.3.1.1 Check	SR 3.3.1.4 Test	SR 3.3.1.5 Calibrate	Remarks
7. ~~12.~~ Pump-Flux Comparator	~~ES~~ 12 hours	45 Days STB	~~RF~~ 18 months	A4
6. 13. High Reactor Building Pressure	M17 ~~DA~~ 12 hours	A4 45 Days STB	~~RF~~ 18 months	
14. High Pressure Injection & Reactor Building Isolation Logic (Non-essential systems)	NA	MO	NA	Includes Reactor Building Isolation of non-essential systems
15. High Pressure Injection Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	RF	
b. Reactor Building Pressure (4 psig)	ES	MO	RF	
16. Low Pressure Injection Logic	NA	MO	NA	
17. Low Pressure Injection Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	RF	
b. Reactor Building Pressure (4 psig)	ES	MO	RF	
18. Reactor Building Emergency Cooling and Isolation System Logic (Essential Systems)	NA	MO	NA	Reactor Building isolation includes essential systems
19. Reactor Building Emergency Cooling and Isolation System Analog Channel Reactor Building Pressure (4 psig)	ES	MO	RF	

<SEE 3.3.5 + 7>

3.3.1-1 A1

Table ~~4.1-1~~ (CONTINUED)

Function ~~Channel Description~~		~~Check~~	~~Test~~	~~Calibrate~~	~~Remarks~~	
~~49.~~	~~Emergency Feedwater Flow Indicators~~	~~MO~~	~~NA~~	~~RF~~		<SEE 3.3.8>
~~50.~~	~~PORV and Safety Valve Position Indicators~~	~~MO~~	~~NA~~	~~RF~~		R1
			SR 3.3.1.4	SR 3.3.1.5		
Supp.1 9. ~~51.~~	RPS Anticipatory Reactor Trip System Loss of Turbine Emergency Trip System Pressure Switches	NA	45 Days STB	~~RF~~ 18 months		A9
10. ~~52.~~	RPS Anticipatory Reactor Trip System Loss of Main Feedwater		A9			
	a) Control Oil Pressure Switches	NA	45 Days STB	~~RF~~ 18 months		A9
~~53.~~	~~Emergency Feedwater Initiation Circuits~~					
	~~a) Control Oil Pressure Switches~~	~~NA~~	~~MO~~	~~RF~~		<SEE 3.3.14>
~~54.~~	~~Containment High Range Radiation Monitor (RIA-57, 58)~~	~~NA~~	~~MO~~	~~RF~~	~~TMI Item II F.1.3~~	<SEE 3.3.8>

Page 10 of 10

Specification 3.3.1

specify that only the EPSL voltage sensing circuit(s) associated with required AC power source(s) are required to be OPERABLE. ITS 3.3.18 ACTION C is added to require the affected AC Source to be declared inoperable when the Required Action and associated Completion Time is not met or when two or more channels of a required circuit(s) are inoperable in MODES 5 and 6. ITS 3.3.18 ACTION D is added to require suspending movement of irradiated fuel assemblies when the Required Action and associated Completion Time is not met during movement of irradiated fuel assemblies. Since there are currently no EPSL requirements during cold shutdown and refueling shutdown, the proposed change is more restrictive since it is an additional restriction on operation.

M35 CTS Table 4.1-1 provides no specific CHANNEL CALIBRATION requirement for the Nuclear Overpower High and Low Setpoints. The High and Low Setpoints are calibrated administratively during reactor shutdowns and
1 reactor startups. ITS SR 3.3.1.5 is added for comparable ITS Functions 1.a and 1.b to provide an explicit 18 month CHANNEL CALIBRATION for these functions. This test ensures the channel responds to the measured parameter within the necessary range and accuracy and leaves the channels adjusted to account for instrument drift to ensure that the instrument channel will remain operational between successive tests. A Note to the SR specifically excludes neutron detectors from this CHANNEL
1 CALIBRATION. The addition of SR 3.3.1.5 is considered an appropriate restriction on unit operation since the accident analyses takes credit for these reactor trip functions. The addition of this requirement represents a more restrictive change and is consistent with the NUREG.

M36 CTS 3.4.1 does not allow the reactor to be heated above 250°F unless each emergency feedwater flow (EFW) path has at least one flow indicator operable (CTS 3.4.1.b). ITS 3.3.8, Item 21, Emergency Feedwater Flow, is added to require two channels of EFW flow to be operable in MODES 1, 2 and 3, along with associated Required Actions A, B, and G, the Table entry, and Notes that are applicable for this PAM function. This is consistent with the ONS Regulatory Guide 1.97 Safety Evaluation Report which identifies these indicators as a Category 1 variable and is appropriate to ensure its availability post accident since EFW flow is the primary indication used by the operator to verify that the EFW System is delivering the correct flow to each steam generator. ITS 3.3.8 ACTION A limits the time that one of the two channels can be inoperable to 30 days. If the channel is not restored within 30 days then ITS 3.3.8 ACTION B requires that a written report be submitted to the NRC which identifies the proposed restorative actions and discusses the root cause evaluation. ITS 3.3.8 ACTION G is a pointer to further appropriate action (dependent upon the PAM function) when the Required Action and associated Completion Times for inoperable channels are not met. The addition of these requirements represents a more restrictive change and is consistent with the NUREG.

M37 CTS 3.4.3.b requires the unit be in hot shutdown (equivalent to ITS MODE 3) within 12 hours when no EFW flow indicators in a flow path are

TECHNICAL CHANGE-LESS RESTRICTIVE

L1 CTS Table 3.5.1-1 Note (a) allows the minimum of three OPERABLE channels to be maintained during channel testing, calibration, or maintenance by placing one of the four available channels in bypass and one of the four available channels in the tripped condition leaving an effective one out of two trip logic. ITS 3.3.1 Action A also allows this configuration but does not limit its application (i.e., allowed for any reason, not just for channel testing, calibration, or maintenance) to only the CTS reasons. ITS requires the required action be completed within one hour where CTS required this action immediately. The proposed change is acceptable since the RPS can still perform its safety function in this configuration in the presence of a random failure of any single channel. The proposed change is consistent with the NUREG.

L2 CTS 3.5.1.1 requires the RPS functions of Table 3.5.1-1 to be OPERABLE when the reactor is in a startup mode or in a critical state. A critical state is considered encompassed by ITS MODES 1 and 2, which are defined as MODES where the reactivity condition is $\geq 0.99\ k_{eff}$. The CTS defines the startup mode to be when the shutdown margin is reduced with the intent of going critical. This is considered equivalent to ITS MODE 2 as described in the associated DOCs for Section 1.0. The applicability for ITS 3.3.1 Function 9, Main Turbine Trip (Hydraulic Fluid Pressure) and Function 10, Loss of Main Feedwater Pumps (Hydraulic Oil Pressure) is less restrictive since these functions are only required to be OPERABLE during MODE 1 (above 30%) and during MODE 1 and MODE 2 (above 2% RTP), respectively. Analyses presented in BAW 1893 show that for operation below these power levels, these trips are not necessary to minimize challenges to the PORV as required by NUREG-0737. Duke Energy has performed a plant specific analysis which concludes that the Oconee RPS System is consistent with the BAW analyses.

As a result of the change in applicability, the CTS Table 3.5.1-1 Column (D) required action of placing the unit in hot shutdown is modified to require reducing power to a level less than the applicability of either function in a time period appropriate for reaching that power level from full power conditions. The proposed change is consistent with the NUREG.

L3 ITS 3.3.2 Action A is added to allow one hour to restore an inoperable manual reactor trip function to OPERABLE status. This is acceptable since the automatic functions and various alternative manual trip methods, such as removing power to the RTMs, are still available. The 1 hour provides a limited time to affect repairs and avoid an unnecessary unit shutdown. This less restrictive change is consistent with the NUREG.

1 L4 Not used.

L5 CTS Table 3.5.1-1 does not include an allowance that allows placing an inoperable ESPS channel in the tripped condition and continuing

operation for an indefinite period. ITS 3.3.5 ACTION A is added to allow continued reactor operation for an indefinite period when one of three ESPS channels is inoperable provided the inoperable channel is placed in a tripped condition for actuation within one hour. This less restrictive provision is acceptable since this action leaves the system in a one-out-of-two condition for actuation. Thus, if another channel fails, the ESPS instrumentation can still perform its actuation functions. This less restrictive change is consistent with the NUREG.

L6 CTS does not include an allowance that allows delay of entry into actions when a channel is made inoperable for testing. The Note for ITS SR 3.3.5.2 allows a delay of up to 8 hours in the entry into the associated Condition and Required Action for the performance of this CHANNEL FUNCTIONAL TEST provided the remaining two instrumentation channels are OPERABLE or tripped. This Note provides a relaxation of ACTION requirements which is less restrictive than the application of CTS requirements. This Note provides a reasonable amount of time to perform the required testing while still allowing the channel being tested to remain in an untripped state. Additionally, the design of the ESPS system will not allow complete testing of an instrument channel with the channel in a tripped state. Therefore, placing the channel in a tripped state, as required by ITS 3.3.5 Required Action A.1, prevents the completion of the testing required by SR 3.3.5.2. This change is consistent with the NUREG.

L7 CTS Table 3.5.1-1 requires a unit shutdown in 12 hours when one or more ESPS channels in one or more functions are inoperable. ITS 3.3.6 Required Action A.1 and its associated Completion Time provide a 72 hour time period in which the unit may continue operation, with one or more ESPS Functions having one channel of the manual initiation feature inoperable, prior to entering an ACTION which results in the unit entering MODE 3. This change is made to provide ACTION requirements consistent with the safety function of the system, considering the allowed outage time for the actuated system. Therefore the less restrictive change is considered appropriate. This change is consistent with the NUREG.

L8 CTS Table 4.1-1 Item 4 requires a calibration of the power range instruments against the incore instruments monthly. This calibration is also required to be performed within some unspecified period of time after each startup if not performed within the previous week. These CTS requirements are replaced by ITS SR 3.3.1.3 and SR 3.3.1.4. The ITS SRs with their specified 31 day Frequencies represents less restrictive requirements in that the calibration is no longer required following each startup if not performed within the previous week. Removal of the required calibration following each startup, is acceptable because deviation between the AXIAL POWER IMBALANCE indicated by the power range instruments and that indicated by the incore instruments generally occurs slowly. The 31 day Frequency is consistent with NUREG.

L9 CTS Table 4.1-1, item 3 requires a heat balance check of the power range channels each shift. ITS SR 3.3.1.2 requires the heat balance check be performed every 24 hours. The 24 hour Frequency is adequate, based on unit operating experience, which demonstrates the change in the difference between the power range indication and the calorimetric results rarely exceeds a small fraction of 2% in any 24 hour period. Furthermore, the control room operators monitor redundant indications and alarms to detect deviations in channel outputs. The change is consistent with the NUREG.

L10 CTS Table 3.5.1-1 Note (i)1 & 2 requires that the power supplied to the CRDMs through the failed CRD be removed within one hour or allows 48 hours to place the breaker in trip if it has diverse features inoperable (undervoltage or shunt trip devices). ITS 3.3.4, Required Action A.2 is added to allow the option of removing power from the CRD trip breaker. This additional allowance is less restrictive in that it provides additional flexibility in dealing with trip breakers with an inoperable diverse trip function. The allowance for removing power from a trip breaker as an alternative to opening the breaker is currently allowed by CTS Table 3.5.1-1 Note (i)1 but is not specifically applicable to the inoperability of the diverse trip function for the trip breakers. The addition of ITS 3.3.4 Required Action A.2 provides consistent ACTION requirements to compensate for inoperable CRD trip breakers whether or not the inoperability is due to failure of a diverse trip function. The Completion Times in ITS remain, as they are in CTS, significantly different for a CRD trip breaker with an inoperable diverse trip function, as opposed to one which is inoperable for any other reason. ITS Required Action B.1 is added to provide the option of tripping the CRD trip breakers that do not have diverse features inoperable. This additional allowance is less restrictive in that it provides additional flexibility in dealing with an inoperable CRD trip breakers. This is appropriate since tripping the inoperable CRD trip breaker has the same effect as removing power to the CRDMs that are powered through the inoperable CRD trip breakers. The proposed changes are consistent with the NUREG.

L11 CTS Table 3.5.1-1 Note j does not include an allowance that allows placing a CONTROL ROD group with an inoperable ETA rely to be placed on a power supply which has OPERABLE ETA relays. ITS 3.3.4 Required Action C.1 is added and provides an alternative to the CTS requirements for inoperable SCR or electronic trip assembly (ETA) relays (CTS Table 3.5.1-1 Note (j)). Required Action C.1 specifically allows for a CONTROL ROD group with an inoperable ETA relay to be placed on a power supply which has OPERABLE ETA relays. This allowance provides new flexibility which is not currently allowed by CTS. Required Action C.1 is an acceptable alternative to opening an inoperable ETA relay because it places the affected CONTROL ROD group on a power supply that ensures the rods are de-energized upon a reactor trip. The change is consistent with the NUREG.

L12 CTS Table 3.5.1-1 Column D and Note (e) require the unit be placed in Hot Shutdown within 24 hours and in Cold Shutdown in the following 72 hours when the minimum ES digital actuation logic channels are not OPERABLE. ITS Required Actions Required Action A.1 and A.2 provide two options. One option is to place the associated component(s) in their ES configuration in one hour. The other is to declare them inoperable (and enter into their associated required actions) within one hour. ITS 3.3.7 Required Action A.1 is equivalent to the automatic actuation logic channel performing its safety function ahead of time. Required Action A.2, which requires entry into the Required Action of the affected supported systems, is appropriate since the net result of the automatic actuation logic failure is inoperability of the supported system. The one hour Completion Time reflects the urgency associated with the inoperability of a actuation logic channel which affects multiple safety system components. ITS 3.3 7 is considered less restrictive since declaring the supported systems inoperable associated with one digital channel would result in only one train of the supported system being declared inoperable. At least a 72 hour Completion Time (Refer to ITS 3.5.2 or 3.6.5 Required Actions and Completion Times) is provided to restore one train of ES actuated components to OPERABLE status prior to requiring a unit shutdown. In addition, where practicable, starting the supported system allows continued operation with no further restrictions. The proposed change is consistent with the NUREG.

L13 CTS 3.5.1.1 requires the source range and wide range instruments to be OPERABLE in a startup mode or in a critical state. CTS Table 3.5.1-1 Note b indirectly provides a qualification to this statement of Applicability. This note provides a relaxation of action requirements when "2 of 4 power range instrument channels are greater than 10% rated power." The Applicability of ITS 3.3.9 and 3.3.10 does not require either the source range instrument channel or the wide range instrument channel to be maintained OPERABLE above MODE 2. This represents a relaxation of requirements, by removing the requirement to take actions in the event that either the required source range instrument channel or the required wide range instrument channel is inoperable, when above 5% RTP (ITS) but less than or equal to 10% rated power, as indicated on the power range instruments (CTS). This is acceptable since the power range channels provide all assumed reactor protection above 5% RTP. This change is being made to provide clear statements of Applicability for these specifications which are consistent with the requirements of the NUREG.

L14 CTS Table 4.1-1 requires a calorimetric heat balance check and adjustment every shift. ITS SR 3.3.1.2, which requires the verification every 24 hours, is modified by a Note that allows this check be delayed as much as 24 hours after THERMAL POWER is ≥ 15% RTP. The ITS recognizes the difficulty in performing the heat balance check and the limitations of the calorimetric while operating at very low power levels. No specific allowance is provided in the CTS which removes the requirement to perform this calibration when in a critical state at low power levels. Below 15% RTP, ONS calculates heat balance power level

based totally upon the primary system parameters. Above 15% RTP, the secondary system parameters are also considered since they are generally more accurate at higher power levels. By allowing the delay in performance of this calibration until RTP is above 15%, a generally more accurate calorimetric (one including secondary system parameters) is available. The proposed change is consistent with the NUREG.

L15 CTS Table 4.1-1 requires a comparison of the out of core measured AXIAL POWER IMBALANCE to incore measured AXIAL POWER IMBALANCE every 31 days. ITS SR 3.3.1.3, which provides an equivalent requirement, is modified by a Note that allows a delay in performance of this SR until the unit is above 15% RTP. This allowance is appropriate due to the usable range of the incore nuclear instruments which are required for the performance of this SR. Below about 15% the incore nuclear instruments are not capable of providing reliable accurate indication of AXIAL POWER IMBALANCE. Adoption of this Note provides a specific relaxation of requirements where none existed in CTS. This change is consistent with the NUREG.

L16 CTS 3.4.2, which requires the automatic initiation circuitry associated with loss of main feedwater pumps to be OPERABLE prior to criticality, provides no allowed outage time when one of two loss of main feedwater instrumentation channels are inoperable. ITS 3.3.14, ACTION A is added to allow continued reactor operation for an indefinite period when one of two EFW System loss of main feedwater instrumentation channels in an EFW pump automatic initiation circuit is inoperable provided the inoperable channel is placed in a tripped condition for initiation within one hour. ITS 3.3.14 ACTIONS Note is added to allow separate condition entry for each EFW pump initiation circuit. This allows one hour to place the channel in trip for each function when Condition A is entered. This less restrictive provision is acceptable since this leaves the function in a one-out-of-one logic configuration for initiation versus the normal two-out-of-two logic configuration. This maintains at least equivalent reliability for EFW initiation. EFW is maintained single failure proof by the separate initiation circuits for each the three EFW pumps. This less restrictive change is consistent with NUREG Specification 3.3.11, ACTION A.

L17 CTS 3.7.6 and 3.7.7 both require an inoperable voltage sensing relay to be restored within 72 hours (Required Action A.1). ITS 3.3.19 Required Action A.1 and 3.3.20 Required Action A.1 require the inoperable channel to be placed in trip within 72 hours. This less restrictive change allows operation to continue indefinitely when the channel is placed in trip and continues to allow 72 hours to restore an inoperable channel that cannot be placed in trip. The actuation logic for DGVP is two-out-of-three. Placing the inoperable channel in the tripped condition fulfills the function of the channel (and places the function in a one-out-of-two configuration). Indefinite operation in this configuration is acceptable since the degraded grid voltage function is capable of performing its function in the presence of a single failure. This change is consistent with comparable NUREG 3.3.8 requirements.

L18 CTS 3.5.7 Applicability for the Main Steam Line Break and Feedwater Isolation Circuity is when main steam header pressure is greater than 700 psig. ITS 3.3.11, 12, & 13 Applicability is MODES 1 and 2, and MODE 3 with main steam header pressure greater than 700 psig except when all MFCVs and SFCVs are closed. The exception of "when all MFCVs and SFCVs are closed" is a less restrictive change and is consistent with comparable NUREG requirements (Table 3.3.11-1, Note d). The exception is appropriate since the MFCVs and SFCVs are already performing their safety function when they are closed.

Required Action B.2.2 of ITS 3.3.11, 12 and 13 is added to provide the option of closing the MFCVs and SFCVs in lieu of reducing main steam header pressure to less than 700 psig. This optional allowance is consistent with the applicability since closure of the MFCVs and SFCVs removes the unit from the Applicability of the LCO.

L19 CTS Table 3.5.1-1 requires a unit shutdown within 24 hours when one or more turbine stop valve closure channels is inoperable. ITS 3.3.15 ACTION A is added to allow one hour to declare the associated TSVs inoperable. The additional hour allowed to restore the instrumentation channel(s) prior to requiring further action is consistent with similar NUREG Required Actions that require supported equipment to be declared inoperable (e.g., NUREG Specification 3.3.7, Required Action A.2). The one hour Completion Time is considered sufficient to correct minor problems. Even with the 1 hour, the unit gets to subcriticality sooner (13 hours) than that time allowed by CTS Table 3.5.1-1, Column D for Item 16 (24 hours).

L20 CTS Table 4.1-1 calibration requirements for RPS functions that receive input from neutron detectors do not specifically exclude the detectors
1 from the calibration of that function. ITS SR 3.3.1.5, which provides comparable CHANNEL CALIBRATION requirements for RPS functions, includes a note that specifically excludes the neutron detectors from the CHANNEL CALIBRATION. This exclusion is appropriate because of the passive design of the detectors, the extreme difficulty in both accessing the detectors and in generating an appropriate input signal to the detectors, the fact that no specific adjustments can be made to the detectors, and the principles of detector operation that ensure a virtually instantaneous response. The proposed change is consistent with the NUREG.

L21 CTS Table 3.5.6-1 Action 3 for the Reactor Vessel Head Level and the Reactor Vessel Level (ITS 3.3.8 PAM #3 and #5) allows, if repairs are feasible, 7 days for restoration of a single inoperable instrument channel when one or both instrument channels are inoperable. Operation may continue with one inoperable channel, provided a report is submitted within the next 30 days outlining the cause of the inoperability and the plans and schedule for restoring the channel to OPERABLE status. When both are inoperable, if at least one instrument channel is not restored, the unit is then required to be in hot shutdown within 12 hours. ITS ACTION A allows 30 days for restoration of a single channel, and ITS

LESS RESTRICTIVE CHANGE L4

1 Not used.

Intentionally blank

Intentionally blank

LESS RESTRICTIVE CHANGE L20

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

CTS Table 4.1-1 calibration requirements for RPS functions that receive input from neutron detectors do not specifically exclude the detectors from the calibration of that function. ITS SR
1 3.3.1.5, which provides comparable CHANNEL CALIBRATION requirements for RPS functions, includes a note that specifically excludes the neutron detectors from the CHANNEL CALIBRATION. This exclusion is appropriate because of the passive design of the detectors, the extreme difficulty in both accessing the detectors and in generating an appropriate input signal to the detectors, the fact that no specific adjustments can be made to the detectors, and the principles of detector operation that ensure a virtually instantaneous response. The proposed change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change excludes the neutron detectors from the CHANNEL CALIBRATION requirements for RPS functions that receive input from the detectors. The probability of an accident is not increased by these changes because the proposed change does not involve any physical changes to plant systems, structures, or components (SSC), or the manner in which these SSC are operated, maintained, or modified. The consequences of an accident will not be increased because the change will not affect the ability of the power range detectors to monitor and respond to core conditions. Changes in neutron detector sensitivity are compensated for by performance of the 24 hour heat balance check and adjustment of SR 3.3.1.2. The neutron detectors are excluded from the CHANNEL CALIBRATIONS because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

 CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.3.1.2	-------------------NOTE------------------- Not required to be performed until [24] hours after THERMAL POWER is ≥ 15% RTP. (1) -- Compare results of ~~Verify~~ calorimetric heat balance to the ~~is~~ calculation (1) ~~≤ [2]% RTP greater than~~ power range channel output and Adjust power range channel output if calorimetric exceeds power range channel output by ≥ [2]% RTP. (1)	24 hours	DOC L14 (23) T 4.1-1 Item 3 Col. "Check"
SR 3.3.1.3	-------------------NOTE------------------- Not required to be performed until (1) [24] hours after THERMAL POWER is (2) ~~15% RTP~~. -- Compare out of core measured AXIAL POWER IMBALANCE (API_0) to incore measured AXIAL POWER IMBALANCE (API_1) as follows: (RTP/TP)(API_0 - API_1) = imbalance error. Adjust the power range channel imbalance output ~~Perform CHANNEL CALIBRATION~~ if the absolute value of the imbalance error is ≥ [2]% RTP.	31 days (1)	DOC L15 T 4.1-1 Item 4 Col. "Calibrate"
SR 3.3.1.4	Perform CHANNEL FUNCTIONAL TEST.	[45] days on a STAGGERED TEST BASIS (1)	T 4.1-1 Column "Test"
~~SR 3.3.1.5~~	~~-------------------NOTE------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.~~	~~[92] days~~	

TSTF-259

Supp. 1

TSTF 259

(continued)

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY	
Supp.1	SR 3.3.1.~~6~~ 5 TSTF-259 ----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ------------------------------ Perform CHANNEL CALIBRATION.	~~[~~18~~]~~ months	DOC L20 T 4.1-1 Column "Calibrate"
	~~SR 3.3.1.7 ----------NOTE---------- Neutron detectors are excluded from RPS RESPONSE TIME testing. ------------------------------ Verify that RPS RESPONSE TIME is within limits.~~	~~[18] months on a STAGGERED TEST BASIS~~	7

CTS → | 3.5.1.1 | T3.5.1-1 Col. D | T4.1-1 | T 2.3-1 + DOC M9 — CTS

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

Supp. 1

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION ~~A~~.1 (B) (6)	SURVEILLANCE REQUIREMENTS (4)	ALLOWABLE VALUE (1)
1. Nuclear Overpower –				
a. High Setpoint	1,2(a), 3(d) TSTF 218	~~A~~ C (6)	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.5 ~~SR 3.3.1.7~~ (7)	≤ ~~[104.9]%~~ 105.5 RTP SR 3.3.1.4 TSTF 259
b. Low Setpoint	2(b),3(b) 4(b),5(b)	~~B~~ D	SR 3.3.1.1 ~~SR 3.3.1.5~~ ~~SR 3.3.1.7~~ (7)	≤ 5% RTP SR 3.3.1.5
2. RCS High Outlet Temperature	1,2	C	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.~~6~~5	≤ [618]°F
3. RCS High Pressure	1,2(a), 3(d)	C	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.~~6~~5 ~~SR 3.3.1.7~~ (7)	≤ [2355] psig
4. RCS Low Pressure	1,2(a)	C	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.~~6~~5 ~~SR 3.3.1.7~~ (7)	≥ [1800] psig As specified in the COLR (39)
5. RCS Variable Low Pressure	1,2(a)	C	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.~~6~~5	≥ ([11.59] T_{out} – [5037.8]) psig
6. Reactor Building High Pressure	1,2,3(c)	C	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.~~6~~5	≤ [4] psig TSTF 259
7. Reactor Coolant Pump to Power	1,2(a)	C	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.~~6~~5 ~~SR 3.3.1.7~~ (7)	≥2 ~~[5]~~% RTP with ≤ 2 pumps operating SR 3.3.1.4 As specified (2)
8. Nuclear Overpower ~~RCS Flow and Measured AXIAL POWER IMBALANCE~~ Flux/Flow Imbalance (4)	1,2(a)	C TSTF 259	SR 3.3.1.1 SR 3.3.1.3 ~~SR 3.3.1.5~~ SR 3.3.1.~~6~~5 ~~SR 3.3.1.7~~ (7)	Nuclear Overpower RCS Flow and AXIAL POWER IMBALANCE setpoint envelope in the COLR
9. Main Turbine Trip (~~Control Oil~~ Hydraulic Fluid Pressure) (4)	≥ ~~[45]%~~ 30% RTP (46)	E	~~SR 3.3.1.1~~ SR 3.3.1.4 SR 3.3.1.~~6~~5	≥ ~~[45]~~ 800 psig (47)
10. Loss of Main Feedwater Pumps (~~Control~~ Hydraulic Oil Pressure) (4)	≥ ~~[15]~~ 2% RTP	F	~~SR 3.3.1.1~~ SR 3.3.1.4 SR 3.3.1.~~6~~5	≥ ~~[55]~~ 75 psig
11. Shutdown Bypass RCS High Pressure	2(b),3(b) 4(b),5(b)	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.~~6~~5	≤ [1720] psig TSTF-259

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal. — DOC M2, T 2.3-1 Col. "Shutdown Bypass"

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal. — DOC M2

(d) When not in shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal. — TSTF 218, DOC M2

15 The Frequency of SR 3.3.7.1 has been changed to 31 days. The NUREG Frequency of 31 days on a STAGGERED TEST BASIS is not consistent with the CTS. The CTS requires this testing monthly, which is considered administratively equivalent to the proposed 31 day Frequency.

1 16 Not used.

1 17 Not used.

18 At ONS, the source range detectors are not turned off at power because the wide range instrument channels use one of the same fission chambers that supplies the source range instrument channels. To monitor the source range, two fission chambers are used and the outputs are added together. Only one fission chamber is used for the wide range output.

19 NUREG 3.3.10 Applicability is changed to specify that the wide range instrument channel is required in MODE 2 and in MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal. The addition of "MODES 3, 4, and 5" to the second statement of the Applicability was made to maintain the CTS allowance provided by Table 3.5.1-1 Note (b). This Note defines the upper limit of the applicable MODES for the required wide range instrument channel as being 10% indicated neutron power. Without the addition of the appropriate MODES, to the second statement of the Applicability for ITS 3.3.10, a wide range channel would be required at all times in MODE 1. This requirement is inconsistent with the RPS design requirement of the wide range instrument channels which is to provide indication of neutron power while operating at low power levels (MODE 2). The required indication of neutron power level is provided by the power range instruments while in MODE 1.

1 20 Not used.

1 21 Not used.

22 The value in NUREG Specifications 3.3.9 and 3.3.10 for THERMAL POWER level as indicated on the wide range neutron flux channels is changed to reflect the appropriate ONS plant specific value.

23 NUREG SR 3.3.1.2 is modified to require a comparison of calorimetric heat balance to power range channel output and adjustment when calorimetric exceeds power range level output by $\geq 2\%$. This is consistent with the description of SR 3.3.1.2 in the NUREG Bases. Current NUREG wording requires a verification that they are within 2% but provides no action if acceptance criteria is not met.

1 24 Not used.

25 NUREG 3.3.4, Condition A is modified to specifically state it applies to a diverse trip function being inoperable rather than an undervoltage or

As a result of the modified applicability, NUREG 3.3.15 ACTIONS A and B are deleted since they no longer apply. These changes maintain the CTS requirements.

37 CTS do not specify an allowable value for the reactor building purge isolation radiation monitor. The UFSAR does not take credit for isolating the purge valves during a refueling accident. The isolation function serves only to minimize radioactive releases but is not required to maintain releases within 10 CFR 100 limits. As such, the specific wording related to the setpoint allowable value in NUREG SR 3.3.15.3 is not included in the ONS ITS.

38 NUREG LCO 3.3.18, "Remote Shutdown System," is not adopted. The ONS CTS does not include any requirements related to shutdown from outside the control room. This function is adequately controlled administratively. In addition, the proposed ONS ITS includes requirements related to the Standby Shutdown Facility, which is designed to mitigate the consequences of postulated fire or flooding incidents, or acts of industrial sabotage to one or more of the three units at Oconee. The SSF is in addition to and supplements the current shutdown capability described in the UFSAR.

39 The allowable value for the RCS Variable Low Pressure Function is located in the Core Operating Limits Report for Oconee. Therefore, the equation with bracketed values provided in NUREG Table 3.3.1-1 for Item 5, RCS Variable Low Pressure, is replaced with: "As specified in the COLR."

40 NUREG SR 3.3.1.11 requires a channel check of the EFW initiation function. A comparable channel check requirement is not included for ITS 3.3.14, EFW Pump Initiation Circuitry. The current test requirements, which do not include a channel check, were adopted and are considered adequate based on operating experience to ensure instrument channel operability.

1 41 Not used.

1 42 Not used.

1 43 Not used.

44 CTS Table 4.1-1, Item 1 specifies a monthly functional test for the Reactor Trip Modules. In the conversion to ITS, this Frequency is retained in ITS SR 3.3.3.1. This change is made to provide requirements consistent with CTS for this testing. No new requirements are added by this change and no existing requirements are removed.

45 NUREG Specification 3.3.17 is modified to incorporate plant specific requirements for HPI, LPI and RBS flow instrument channels and BWST water level instrument channels. At ONS there is only one flow

additional guidance and clarification on the proper usage of the Required Actions without changing the intent of the ACTIONS.

26 The last paragraph of the Applicable Safety Analyses discussion for the Nuclear Overpower-High Setpoint function (on Base page B 3.3-12) is deleted since it is confusing and not entirely true. At ONS, the Variable Low Pressure and Nuclear Overpower Flux/Flow Imbalance trips are credited in all safety analyses regardless of the transient power excursion.

1 27 Not used.

1 28 Not used.

29 ITS SR 3.3.3.1 retains the current test frequency (CTS Table 4.1-1, Item 1) for Reactor Trip Modules. Therefore, the NUREG SR 3.3.3.1 Bases is modified to reflect the current licensing bases and reference to the
1 BAW topical report is removed.

30 The third paragraph of the LCO Bases for NUREG Specification 3.3.3 is revised to provide a discussion in terms of Reactor Trip Modules for consistency with the LCO requirement rather than in general terms of RPS channels.

ITS Section 3.3

ID 56

Subject Remove JFDs associated with approved ONS generic changes ONS-002, 003, 011, 019, 020 and annotate NUREG Markup with TSTF numbers 211, 212, 215, and 217, respectively

CTS

3.3 INSTRUMENTATION 4

3.3.3 Reactor Protecti~~on~~ve System (RPS) – Reactor Trip Module (RTM)

LCO 3.3.3 Four RTMs shall be OPERABLE. 3.5.1.1 T3.5.1-1, Funct 17 Col (C)

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with any ~~CONTROL ROD~~ 2 drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal. 3.5.1.1 Doc M3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One RTM inoperable.	A.1.1 Trip the associated CRD trip breaker.	1 hour	T3.5.1-1, Col D+ Note f.1
	OR		
	A.1.2 Remove power from the associated CRD trip breaker.	1 hour	T3.5.1-1, Col D+ Note f.2
	AND		
	A.2 Physically remove the inoperable RTM.	1 hour	Doc M4
B. Required Action and associated Completion Time not met in MODE 1, 2, or 3. TSTF 217 Two or more RTM's inoperable in MODE 1, 2, or 3. OR	B.1 Be in MODE 3.	12 ~~6~~ hours	Doc M5
	AND		
	B.2.1 Open all CRD trip breakers.	12 ~~6~~ hours	5
	OR		
	B.2.2 Remove all power from ~~to the~~ CRD ~~System~~ trip breakers.	12 ~~6~~ hours	TSTF-211

Supp. 1

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME	
C. Required Action and associated Completion Time not met in MODE 4 or 5.	C.1	Open all CRD trip breakers.	6 hours	Doc M3
	OR			
Two or more RTMs inoperable in MODE 4 or 5. OR (TSTF-217)	C.2	Remove all power ~~to the~~ from CRD ~~System~~ trip breakers.	6 hours (TSTF-211)	Doc M3

Supp. 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.3.3.1	~~-----NOTE----- When an RTM is placed in an inoperable status solely for performance of this Surveillance, entry into associated Conditions and Required Actions may be delayed for up to 8 hours, provided at least two RTM channels are OPERABLE.~~ (TSTF-212) Perform CHANNEL FUNCTIONAL TEST.	31 ~~45~~ days ~~on a STAGGERED TEST BASIS~~ (4)	4.1.1, T4.1-1 Item 1 Column "Test"

Supp. 1

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
	C. One or more ETA relays inoperable.	C.1 Transfer affected CONTROL ROD group to power supply with OPERABLE ETA relays.	1 hour	DOC L11
		OR		
		C.2 Trip corresponding AC CRD trip breaker. (5) — 2	1 hour	T3.5.1-1, Note (j)
	D. Required Action and associated Completion Time not met in MODE 1, 2, or 3.	D.1 Be in MODE 3.	12 ~~6~~ hours	DOC M5
		AND		
		D.2.1 Open all CRD trip breakers.	12 ~~6~~ hours 5	DOC M5
		OR		
Supp. 1		D.2.2 Remove all power from ~~to~~ ~~the~~ CRD trip breakers ~~System~~.	12 ~~6~~ hours	DOC M5 TSTF-211
	E. Required Action and associated Completion Time not met in MODE 4 or 5.	E.1 Open all CRD trip breakers.	6 hours	DOC M3
		OR		
Supp. 1		E.2 Remove all power from ~~to~~ ~~the~~ CRD trip breakers ~~System~~.	6 hours	DOC M3 TSTF-211

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.3.4.1	Perform CHANNEL FUNCTIONAL TEST.	31 days	4.1.1 + T4.1-1, Item 2 Column, "Test"

CTS

3.3 INSTRUMENTATION

3.3.5 Engineered Safeguards Protective 4 ~~Safety Feature Actuation~~ System ESPS ~~(ESFAS)~~ Instrumentation

LCO 3.3.5 Three channels of ESPS 4 ~~ESFAS~~ analog instrumentation for each Parameter in Table 3.3.5-1 shall be OPERABLE ~~in each ESFAS train~~. 9

3.5.1.1
T3.5.1-1, Col C
Funct. Units
12a+b, 13a+b,
14a + 15a

APPLICABILITY: According to Table 3.3.5-1.

3.5.1.1
3.5.3 Note (1)+(2)

ACTIONS

---------------------------------------NOTE---------------------------------------
Separate Condition entry is allowed for each Parameter.

DOC A6

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Parameters with one channel inoperable.	A.1 Place channel in trip.	1 hour
B. One or more Parameters with two or more channels inoperable, <u>OR</u> Required Action and associated Completion Time not met. (TSTF 217)	B.1 Be in MODE 3. <u>AND</u> B.2.1 --------NOTE--------- Only required for RCS Pressure - Low ~~setpoint~~. 11 ---------------------- Reduce RCS pressure < ~~[1800]~~ 1750 psig. 1 <u>AND</u>	12 ~~6~~ hours 5 36 hours (continued)

DOC L5

T3.5.1-1, COL D for Items 12.a, 12.b, 13.a, 13.b, 14.a + 15.a

DOC A8

Supp. 1

4 ESPS

3.3 INSTRUMENTATION

3.3.6 Engineered ~~Safety Feature Actuation~~ Safeguards Protective System ~~(ESFAS)~~ Manual Initiation

LCO 3.3.6 Two manual initiation channels of each one of the ~~ESFAS~~ Functions below shall be OPERABLE: 3.5.1.1

14

a. High Pressure Injection; (ES Channels 1 and 2) — , Reactor Building (RB) Non-Essential Isolation, Keowee Start, Load Shed and Standby Breaker Input, and Keowee Standby Bus Feeder Breaker Input — T 3.5.1-1, Col C Funct. Unit 12

b. Low Pressure Injection; (ES Channels 3 and 4) — T 3.5.1-1, Funct. Unit 13

~~c.~~ ~~Reactor Building (RB)~~ Cooling; (ES Channels 5 and 6) — and RB Essential Isolation and Penetration Room Ventilation — T 3.5.1-1 Funct. Unit 14

~~d.~~ RB Spray~~s~~. (ES Channels 7 and 8) — and RB Essential Isolation — T 3.5.1-1 Funct. Unit 15

~~e. RB Isolation; and~~

~~[f. Control Room Isolation.]~~

Supp. 1 APPLICABILITY: MODES 1, 2, ~~and 3~~, TSTF 215
MODES 3 and 4 when associated engineered safeguard equipment is required to be OPERABLE. 3.5.1.1

ACTIONS

NOTE
Separate Condition entry is allowed for each Function. DOC A6

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ~~ESFAS~~ ESPS 4 Functions with one channel inoperable.	A.1 Restore channel to OPERABLE status.	72 hours DOC L7
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND	~~6~~ 12 5 hours T 3.5.1-1 COL D for items 12.c, 13.c, 14.b & 15.b

(continued)

3.3 INSTRUMENTATION

3.3.7 Engineered ~~Safety Feature Actuation~~ Safeguard Protective System (~~ESFAS~~) (ESPS) 4 Digital Automatic Actuation Logic Channels

LCO 3.3.7 ~~All the~~ Eight ~~ESFAS~~ Digital 4 automatic actuation logic ~~matrices~~ Channels 14 shall be OPERABLE. 3.5.1.1 T3.5.1-1, Col C 12d, 13d, 14c, 15c

APPLICABILITY: MODES 1, 2, ~~and 3~~, MODES 3 and 4 when associated engineered safeguard (ES) equipment is required to be OPERABLE. 3.5.1.1

TSTF-215

Supp. 1

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each automatic actuation logic ~~matrix.~~ Channel 14. DOC A6

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more digital automatic actuation logic ~~matrices~~ Channels 14 inoperable.	A.1 Place associated component(s) in ~~engineered safeguard~~ ES configuration. OR A.2 Declare the associated component(s) inoperable.	1 hour 1 hour

T3.5.1-1 Col D 12d, 13d, 14c, 15c — DOC L12

Supp.1 TSTF 217

28 <Except as marked>

MSLB Detection and MFW Isolation

CTS

One or more MFW Isolation Functions with two or more channels inoperable
OR

B.1 Be in MODE 3.
AND

12 hours 5

T3.5.1-1 Item 20, Col D

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B ~~F~~. Required Action and associated Completion Time not met ~~for Functions 1.c, 2, 3 or 4~~.	B.2.1 ~~F.1~~ Reduce ~~once through steam generator~~ main steam header pressure to < ~~750~~ 700 psig. OR	18 ~~12~~ hours 5
	B.2.2 Close all MFCVs and SFCVs.	18 hours

Doc L18

SURVEILLANCE REQUIREMENTS

---NOTE---
~~Refer to Table 3.3.11-1 to determine which SRs shall be performed for each EFIC Function.~~

SURVEILLANCE	FREQUENCY
SR 3.3.11.1 Perform CHANNEL CHECK.	12 hours
~~SR 3.3.11.2 Perform CHANNEL FUNCTIONAL TEST.~~	~~31 days~~
SR 3.3.11.~~3~~ 2 Perform CHANNEL CALIBRATION.	~~[18]~~ 9 months
~~SR 3.3.11.4 Verify EFIC RESPONSE TIME is within limits.~~	~~[18] months on a STAGGERED TEST BASIS~~

4.1.1 T 4.1-1 Item 62 Col "Check"

31

4.1.1 T 4.1-1 Item 62 Col "Calibrate"

7

BASES

Supp. 1

ACTIONS

B.1, B.2.1, and B.2.2 (continued) trip breakers TSTF 211

from all

which the LCO does not apply. This is done by placing the unit in at least MODE 3 with all CRD trip breakers open or with all power to the CRD System removed within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

12 5

5 if two or more RTMs are inoperable or

and associated

C.1 and C.2

Condition C applies if the Required Actions of Condition A are not met ~~within the required~~ Completion Time in MODE 4 or 5. In this case, the unit must be placed in a MODE in which the LCO does not apply. This is done by opening all CRD trip breakers or removing all power to the CRD System. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers or remove all power to the CRD System without challenging unit systems. 2

trip breakers 28 from all

Supp. 1

from all trip breakers

TSTF-211

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1

5

~~The Note defines a channel as being OPERABLE for up to 8 hours while bypassed for Surveillance testing. The Note allows channel bypass for testing without defining it as inoperable although during this time period it cannot actuate a reactor trip. This allowance is based on the assumption of the RPS reliability analysis in BAW-10167 (Ref. 2) that 8 hours is the average time required to perform channel Surveillance. The analysis demonstrated that the 8 hour testing allowance does not significantly reduce the probability that the RPS will trip when necessary. It is not acceptable to routinely remove channels from service for more than 8 hours to perform required Surveillance testing. Such a practice would be contrary to the assumptions of the reliability analysis that justified the LCO's Completion Times.~~

29

~~[Reviewer's Note: The CHANNEL FUNCTIONAL TEST Frequency is based on an approved topical report. For a licensee to use~~

(continued)

2 <Except as marked>

BASES

ACTIONS

C.1 and C.2 (continued)

The 1 hour Completion Time is sufficient to perform the Required Action.

D.1, D.2.1, and D.2.2

and associated Completion Time is

Supp. 1

~~If~~ with the Required Actions & of Condition A, B, or C are not met ~~within the required Completion Time~~ in MODE 1, 2, or 3, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to ~~at least~~ MODE 3, with all CRD trip breakers open or with all power to the CRD System removed within ~~6~~ hours. The allowed Completion Time of ~~6~~ hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

TSTF 211 — from all — trip breakers — 12 — 5 — TSTF 211

E.1 and E.2

~~If~~ with the Required Actions & of Condition A, B, or C are not met ~~within the required Completion Time~~ in MODE 4 or 5, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, all CRD trip breakers must be opened or all power to the CRD System removed within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers or remove all power to the CRD system without challenging unit systems.

trip breakers — from all — from all — trip breakers — TSTF 211

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1

SR 3.3.4.1 is to perform a CHANNEL FUNCTIONAL TEST every 31 days. This test verifies the OPERABILITY of the trip devices by actuation of the end devices. Also, this test independently verifies the undervoltage and shunt trip mechanisms of the AC breakers. The Frequency of 31 days is based on operating experience, which has demonstrated that failure of more than one channel of a given function in any 31 day interval is a rare event.

trip

REFERENCES

1. U FSAR, Chapter ~~[7]~~ 9.2 — 2 — 1
2. 10 CFR 50.36. — 2

TECHNICAL SPECIFICATIONS

NOTE: The first four justifications for these changes from NUREG-1430 were generically used throughout the individual LCO section markups. Not all generic justifications are used in each section.

1 The brackets are removed and the proper plant specific information or value is provided.

2 Editorial changes made for clarity, preference or consistency with the Improved Technical Specifications (ITS) Writer's Guide.

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

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

5 The current licensing basis (CLB) permits 12 hours to place a unit in Hot Shutdown (equivalent to ITS MODE 3) when an LCO is not met. To maintain consistency with current procedures, training and staffing requirements, the 12 hours permitted to place a unit in MODE 3 is retained in the ITS.

As a result of modifying the Completion Time to place a unit in MODE 3, Completion Times for concurrent Required Actions within an ACTION were extended from 6 hours to 12 hours. NUREG 3.3.1 Required Action C.2, NUREG 3.3.2 Required Action B.2, NUREG 3.3.3 Required Action B.2.1, and NUREG 3.3.4 Required Action D.2.1 are modified to allow 12 hours to open all control rod drive (CRD) trip breakers consistent with the concurrent Completion Time to place the unit in MODE 3. NUREG 3.3.3 Required Action B.2.2 and NUREG 3.3.4 Required Action D.2.2 are modified to allow 12 hours to remove power from all CRD trip breakers consistent with the concurrent Completion Time to place the unit in MODE 3. Also, NUREG 3.3.1 Required Action G.1 Completion Time is modified to allow 12 hours to reduce power to < 2%. This is necessary since the ONS ITS power reduction is to < 2% versus the NUREG power reduction to < 15%. A power level of < 2% is a condition very close to MODE 3.

6 CTS Table 3.5.1-1 Column C permits unlimited operation with three of the four available RPS instrumentation channels OPERABLE. NUREG LCO 3.3.1 and associated Actions are modified to reflect the CTS provision. NUREG LCO 3.3.1, ACTION A is deleted (and subsequent ACTIONS relettered) since the Required Action only applies when 4 channels are required OPERABLE. Condition B is modified to apply when one required channel is inoperable (one of the three required versus two of the four available) and Required Action B.2 is deleted since this applies to the 4 channel configuration.

7 NUREG SR 3.3.1.7, SR 3.3.5.4 and SR 3.3.11.4 are not adopted. Consistent with current licensing basis, response time testing of RPS, ESPS, and EFIC (ONS equivalent) circuitry is not performed. Plant equipment does not readily lend itself to such testing.

1 8 TSTF-xxx clarifies that agreement criteria include one decade of overlap
1 during a power increase when transitioning between the intermediate
1 range monitors and the power range monitors. ONS uses wide range
1 monitors which have an upper range up to 200% power, therefore, there is
1 no need to transition between monitors. As such, this portion of TSTF
1 is not included.

9 The unit specific design of the ONS ESPS provides for three analog channels of instrumentation for each of the monitored parameters. These three channels provide the required input to each of the eight automatic actuation logic channels. Contrary to the system design depicted in the requirements of the NUREG, these three instrument channels provide input to both trains of automatic actuation logic channels. This unit specific design difference required the deletion of the phrase "in each ESFAS train" from NUREG LCO 3.3.5 as well as appropriate changes to the Bases.

1 10 Not used.

11 NUREG Table 3.3.5-1 lists equipment actuated by each of the ESPS Parameters. This list is incomplete and has not been included in the ITS in favor of the more complete list provided in the Bases. Also, the term "Setpoint" is removed from the Parameter title in the Table and from Required Actions B.2.1, B.2.2 and B.2.3 since the setpoint is not a parameter. Referring to a setpoint as a parameter is inconsistent with the identification of other parameters and functions throughout the NUREG. Removal of this information represents no actual change in requirements.

12 In the conversion to ITS, NUREG Table 3.3.17-1, Post Accident Monitoring Instrumentation, is modified by Note c which indicates that the containment isolation valve position indication requirements apply only to containment isolation valves that are electrically controlled. This is consistent with ONS Regulatory Guide 1.97 response for CIV position indication and the NRC's Safety Evaluation Report for this response.

1 13 Not used.

14 The Functions specified in NUREG LCO 3.3.6 are modified to match the Functions as presented in the CTS, UFSAR, and other design basis documents. NUREG Specification 3.3.7 has also been modified to include ONS unit specific terminology. These changes were made to provide requirements consistent with the design of the ONS and consistent with the specific terminology and names associated with the ONS ESPS.

15 The Frequency of SR 3.3.7.1 has been changed to 31 days. The NUREG Frequency of 31 days on a STAGGERED TEST BASIS is not consistent with the CTS. The CTS requires this testing monthly, which is considered administratively equivalent to the proposed 31 day Frequency.

1 16 Not used.

1 17 Not used.

18 At ONS, the source range detectors are not turned off at power because the wide range instrument channels use one of the same fission chambers that supplies the source range instrument channels. To monitor the source range, two fission chambers are used and the outputs are added together. Only one fission chamber is used for the wide range output.

19 NUREG 3.3.10 Applicability is changed to specify that the wide range instrument channel is required in MODE 2 and in MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal. The addition of "MODES 3, 4, and 5" to the second statement of the Applicability was made to maintain the CTS allowance provided by Table 3.5.1-1 Note (b). This Note defines the upper limit of the applicable MODES for the required wide range instrument channel as being 10% indicated neutron power. Without the addition of the appropriate MODES, to the second statement of the Applicability for ITS 3.3.10, a wide range channel would be required at all times in MODE 1. This requirement is inconsistent with the RPS design requirement of the wide range instrument channels which is to provide indication of neutron power while operating at low power levels (MODE 2). The required indication of neutron power level is provided by the power range instruments while in MODE 1.

1 20 Not used.

1 21 Not used.

22 The value in NUREG Specifications 3.3.9 and 3.3.10 for THERMAL POWER level as indicated on the wide range neutron flux channels is changed to reflect the appropriate ONS plant specific value.

23 NUREG SR 3.3.1.2 is modified to require a comparison of calorimetric heat balance to power range channel output and adjustment when calorimetric exceeds power range level output by ≥ 2%. This is consistent with the description of SR 3.3.1.2 in the NUREG Bases. Current NUREG wording requires a verification that they are within 2% but provides no action if acceptance criteria is not met.

1 24 Not used.

25 NUREG 3.3.4, Condition A is modified to specifically state it applies to a diverse trip function being inoperable rather than an undervoltage or

shunt trip function being inoperable. This is consistent with the NUREG Bases discussion and the CTS requirements.

1 26 Not used.

27 NUREG LCO 3.3.14, "EFIC-EFW-Vector Valve Logic," is not adopted since it is not applicable to ONS. ONS design does not include vector valve logic.

28 NUREG Specification 3.3.11, Emergency Feedwater Initiation and Control (EFIC) System Instrumentation; NUREG Specification 3.3.12, EFIC Manual Initiation; and NUREG Specification 3.3.13, EFIC logic, are modified to address Main Steam Line Break Detection and MFW Isolation Circuitry only. ITS Specifications 3.3.14 and 3.3.15 are added to address Emergency Feedwater System Initiation Circuitry and Main Steam Line Break and Main Feedwater Isolation instrumentation separately. The NUREG Specification combines the EFW System Initiation, MSL Isolation and MFW Isolation functions into one Specification apparently due to common instrumentation and similar initiation circuitry. ONS does not have common instrumentation and similar initiation circuitry for these functions. Consistent with CTS, the ITS addresses these requirements by separate Specifications. The Specification titles, LCOs, ACTIONS, and Surveillance Requirements are appropriately modified to reflect ONS specific terminology and design requirements. Where appropriate, ITS Required Actions are based on similar NUREG Required Actions. For example, the Completion Time of one hour for ITS 3.3.15, Required Action A.1 is consistent with NUREG Specification 3.3.7, Required Action A.2, which allows one hour to declare an affected component inoperable when the actuation logic is inoperable.

29 SR 3.3.8.2 is added to the Post Accident Monitoring (PAM) SR Table to capture the 12 month calibration frequency of the containment pressure and hydrogen concentration functions. ITS SR 3.3.8.2 is modified by a note indicating that the SR is only applicable to these two functions. NUREG SR 3.3.17.2 (ONS SR 3.3.8.3) is modified by a note that indicates that the 18 month calibration is not applicable to these two functions. This change is necessary to accommodate the different frequencies for CHANNEL CALIBRATION.

30 NUREG Table 3.3.17-1 (ONS Table 3.3.8-1) is modified to list the Regulatory Guide 1.97 Type A and the Regulatory Guide 1.97 non-Type A instruments and their associated requirements as documented in the NRC Safety Evaluation Report for Regulatory Guide 1.97 related to Oconee. The "NOTE" at the bottom of the NUREG Table is deleted since it does not apply plant specific.

31 NUREG SR 3.3.11.2, as it relates to the MSLB Detection and MFW Isolation Circuitry, is deleted since the CTS (Table 4.1-1 Item 62) specifies the CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST on the same frequency. Both the CTS and ITS definitions specify that the required calibration

includes the CHANNEL FUNCTIONAL TEST. Therefore, the specific requirement to perform the CHANNEL FUNCTIONAL TESTS is not retained in the ITS.

32 The frequency of 92 days for NUREG SR 3.3.15.2 (ITS SR 3.3.16.2) is modified to partially incorporate the CLB. DPC considers the NUREG frequency of 92 days to be inappropriate for ONS. CTS 3.8.10 requires the radiation monitor that initiates purge isolation to be verified operable immediately prior to beginning refueling operations. For consistency with ITS SR 3.9.3.2, which verifies that the reactor building purge supply and exhaust valve actuates to the correct position on an actual or simulated actuation signal once each refueling outage prior to beginning CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the same SR Frequency is adopted for ITS SR 3.3.16.2. This is appropriate since the safety function of the radiation monitor is to isolate the purge valves. Requiring performance of SR 3.3.16.2 at this Frequency represents a reasonable relaxation of the current requirement of "immediately prior to beginning refueling operations."

33 ITS Specifications 3.3.17 through 3.3.21 are added to capture current technical specification requirements for Emergency Power Switching Logic Functions. The EPSL is designed to assure that power is supplied to the unit main feeder buses and, hence to the unit's essential loads. Appropriate LCOs, ACTIONS, and Surveillance Requirements are added.

34 NUREG LCO 3.3.8, "Emergency Diesel Generator (EDG) Loss of Power Start (LOPS)," is not adopted since it is not applicable to ONS. ONS does not use EDGs for emergency power. Comparable ITS requirements related to the Keowee Hydro Units, which are used at Oconee for emergency power, are included in ITS 3.3.19.

35 NUREG Specification 3.3.16, "Control Room Isolation - High Radiation," is not included in the proposed ONS ITS. ONS does not have an automatic Control Room isolation. At ONS, a high radiation alarm is annunciated in the Control Room at which time the Control Room operator can energize the outside air booster fans and filter systems to minimize unfiltered air entering the control room. Adequate administrative controls are in place to ensure the operability of this function.

36 The NUREG applicability for LCO 3.3.15 (ONS ITS LCO 3.3.16) of MODES 1, 2, 3, and 4 is not adopted in the conversion. The ITS requires the reactor building purge isolation - high radiation monitor to be operable only during CORE ALTERATION and during movement of irradiated fuel assemblies within the containment. At ONS, the reactor building purge valves are required to be verified sealed closed in MODES 1, 2, 3, and 4 and (refer to SR 3.6.3.1). Since the function of the high radiation channel is to initiate closure of these valves on a high radiation signal, the channel need not be OPERABLE during these MODES since the valves are closed.

As a result of the modified applicability, NUREG 3.3.15 ACTIONS A and B are deleted since they no longer apply. These changes maintain the CTS requirements.

37 CTS do not specify an allowable value for the reactor building purge isolation radiation monitor. The UFSAR does not take credit for isolating the purge valves during a refueling accident. The isolation function serves only to minimize radioactive releases but is not required to maintain releases within 10 CFR 100 limits. As such, the specific wording related to the setpoint allowable value in NUREG SR 3.3.15.3 is not included in the ONS ITS.

38 NUREG LCO 3.3.18, "Remote Shutdown System," is not adopted. The ONS CTS does not include any requirements related to shutdown from outside the control room. This function is adequately controlled administratively. In addition, the proposed ONS ITS includes requirements related to the Standby Shutdown Facility, which is designed to mitigate the consequences of postulated fire or flooding incidents, or acts of industrial sabotage to one or more of the three units at Oconee. The SSF is in addition to and supplements the current shutdown capability described in the UFSAR.

39 The allowable value for the RCS Variable Low Pressure Function is located in the Core Operating Limits Report for Oconee. Therefore, the equation with bracketed values provided in NUREG Table 3.3.1-1 for Item 5, RCS Variable Low Pressure, is replaced with: "As specified in the COLR."

40 NUREG SR 3.3.1.11 requires a channel check of the EFW initiation function. A comparable channel check requirement is not included for ITS 3.3.14, EFW Pump Initiation Circuitry. The current test requirements, which do not include a channel check, were adopted and are considered adequate based on operating experience to ensure instrument channel operability.

1 41 Not used.

1 42 Not used.

1 43 Not used.

44 CTS Table 4.1-1, Item 1 specifies a monthly functional test for the Reactor Trip Modules. In the conversion to ITS, this Frequency is retained in ITS SR 3.3.3.1. This change is made to provide requirements consistent with CTS for this testing. No new requirements are added by this change and no existing requirements are removed.

45 NUREG Specification 3.3.17 is modified to incorporate plant specific requirements for HPI, LPI and RBS flow instrument channels and BWST water level instrument channels. At ONS there is only one flow

instrument channel per train. If the flow instrument is inoperable, ONS considers the associated train inoperable since without flow indication the operator has no means of precluding pump runout or loss of NPSH. Therefore, the appropriate action for an inoperable flow channel is to declare the affected train inoperable. ACTION F is added to address the condition where one or more of these required flow instrument channels is inoperable. Required Action F.1 requires the affected train be declared inoperable and the appropriate action entered for the affected system. Condition F has a note that indicates that the Condition only applies to the flow instrument channels (Function 18, 19, and 20). The addition of ACTION F made it necessary to modify Conditions A, C, and D to exclude or include Functions for which the conditions are applicable. Table 3.3.17-1 is modified to replace the Condition reference from the Required Action E.1 as being not applicable since the appropriate action only applies to these particular instruments and that action is contained within Required Action F.1.

ACTION E was added to address the condition where one of two BWST water level channels is inoperable. Continuous operation with one of the two required channels is not appropriate because alternate indications are not available. This indication is crucial in determining when the water source for ECCS should be swapped from the BWST to the reactor building sump. Therefore, 24 hours is allowed to restore the indication, consistent with CTS requirements. With both BWST water level channels inoperable, the appropriate action is to shut down.

46 NUREG 3.3.1 Required Action E.1, G.1 and Table 3.3.1-1 provide a bracketed value for the Applicability of the Main Turbine Trip and Loss of Main Feedwater Pumps RPS Functions. The ITS Applicability for each function is based on analyses presented in BAW 1893 that show for operation below certain power levels, the trips are not necessary to minimize challenges to the PORV as required by NUREG-0737. The CTS Applicability for these functions is when the reactor is in a startup mode or in a critical state. Duke Energy has performed a plant specific analysis which concludes that the Oconee RPS System is consistent with the BAW analyses and that the appropriate plant specific applicability for each function is 30% RTP for the Main Turbine Trip function and 2% RTP for the Loss of Main Feedwater Pumps trip function.

47 NUREG Table 3.3.1-1 provides a bracketed value for allowable value for the Main Turbine Trip (Hydraulic Fluid Pressure) function and the Loss of Main Feedwater Pumps (Hydraulic Oil Pressure) function. CTS does not specify allowable values for these functions. Appropriate plant specific values are added.

48 ITS Specification 3.3.22 is added to require the Manual Keowee Emergency Start Function to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies. This addition is necessitated by the addition of requirements for AC Source in MODES 5 and 6 and during movement of irradiated fuel assemblies (refer to Section 3.8). Required

Action A.1 requires both Keowee Hydro Units to be declared inoperable immediately when the required channel is inoperable. ITS SR 3.3.22.1 requires a CHANNEL FUNCTIONAL TEST of the Keowee manual emergency start function every 12 months. The EPSL is designed to assure that power is supplied to the unit main feeder buses and, hence to the unit's essential loads.

49 ONS design requires that the voltage sensing circuit associated with an AC power source be OPERABLE for the AC power source to be considered OPERABLE. Therefore, since requirements for AC Source in MODES 5 and 6 and during movement of irradiated fuel assemblies are added (refer to Section 3.8), requirements for EPSL voltage sensing circuits are included in the ITS. LCO Note 2 is added to specify that only the EPSL voltage sensing circuit(s) associated with required AC power source(s) are required to be OPERABLE. ITS 3.3.18 ACTION C is added to require the affected AC Source to be declared inoperable when the Required Action and associated Completion Time is not met or when two or more channels of a required circuit(s) are inoperable in MODES 5 and 6. ITS 3.3.18 ACTION D is added to require suspending movement of irradiated fuel assemblies when the Required Action and associated Completion Time is not met during movement of irradiated fuel assemblies.

additional guidance and clarification on the proper usage of the Required Actions without changing the intent of the ACTIONS.

26 The last paragraph of the Applicable Safety Analyses discussion for the Nuclear Overpower-High Setpoint function (on Base page B 3.3-12) is deleted since it is confusing and not entirely true. At ONS, the Variable Low Pressure and Nuclear Overpower Flux/Flow Imbalance trips are credited in all safety analyses regardless of the transient power excursion.

1 27 Not used.

1 28 Not used.

29 ITS SR 3.3.3.1 retains the current test frequency (CTS Table 4.1-1, Item 1) for Reactor Trip Modules. Therefore, the NUREG SR 3.3.3.1 Bases is modified to reflect the current licensing bases and reference to the
1 BAW topical report is removed.

30 The third paragraph of the LCO Bases for NUREG Specification 3.3.3 is revised to provide a discussion in terms of Reactor Trip Modules for consistency with the LCO requirement rather than in general terms of RPS channels.

3.4 Reactor Coolant System

ITS Section 3.4

ID 4

Subject Change relocation of 3.4 LA13 from SLC to QA Topical (testing after maintenance or modification).

acceptable. Changes to the UFSAR are controlled in accordance with 10 CFR 50.59.

LA13 CTS 4.5.1.2.2 requires leak testing specified PIVs after maintenance,
1 repair or replacement. The requirements for testing after maintenance
1 or modification are moved to the Quality Assurance Topical Report.
1 Post-maintenance or post-modification tests are not usually part of the
1 Technical Specifications. CTS 6.1.2.1 describes review and control
1 requirements for procedures, modifications, tests, experiments,
1 reportable events, special reviews and investigations, and unplanned
1 onsite releases. These requirements are also relocated to the Quality
1 Assurance Topical Report (Refer to DOC LA7 for Section 5). CTS 6.1.2.1
1 also requires records of the above activities be provided as necessary
1 for required reviews. These requirements are also located in ANSI
1 N18.7-1976. Any changes to the Quality Assurance Topical Report require
1 a 10 CFR 50.54 evaluation. These evaluations ensure that any changes to
1 these requirements are appropriately reviewed. This change is
1 consistent with the NUREG.

LA14 CTS 2.2.2 specifies that the setpoint of the pressurizer code safety valves be in accordance with ASME, Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1967. This detail is relocated to the Bases. This detail is not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for system OPERABILITY. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable. Changes to the Bases are controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

ITS Section 3.4

ID 5

Subject Capitalize "RCS" in 3.4 DOC M19.

MODE 3 with RCS temperature ≤ 325°F. These requirements are provided to ensure the pressurizer is available for control of RCS pressure. The requirement to place the unit in MODE 3 with RCS temperature ≤ 325°F is a more restrictive requirement upon unit operations and is consistent with the NUREG.

M16 CTS requirements comparable to ITS SR 3.4.9.1 and SR 3.4.9.2 do not exist. ITS SR 3.4.9.1 requires periodic verification that pressurizer water level is within limits. ITS SR 3.4.9.2 requires periodic verification that the capacity of the required pressurizer heaters and associated power supplies are ≥ 126 kW. This change is necessary to periodically confirm the LCO requirements are met and ensures timely identification when LCO requirements are not met. These are additional requirements upon unit operation and are consistent with the NUREG.

M17 Not used.

M18 Not used.

M19 CTS requirements comparable to ITS LCO 3.4.8, LCO Note 1 part a, LCO
Note 1 part c, LCO Note 2, LCO Note 3, Actions, SR 3.4.8.1 and SR
3.4.8.2 do not exist. ITS LCO 3.4.8 requires two DHR loops be OPERABLE
and one loop shall be in operation. ITS LCO Note 1 part a permits DHR
pumps to not be in operation provided no operations are permitted which
1 can cause a reduction in the RCS boron concentration. ITS LCO Note 2
permits one DHR loop to be inoperable for ≤ hours provided the other DHR
loop is OPERABLE and in operation. ITS LCO Note 3 a DHR loop to be
considered OPERABLE when aligned for low pressure injection if capable
of being manually realigned to the DHR mode of operation. ITS
Specification 3.4.8 has an Applicability of MODE 5 with RCS loops not
filled. ITS SR 3.4.8.1 and ITS SR 3.4.8.2 require periodic verification
that required DHR loops are OPERABLE and in operation. This change is
appropriate due to the importance of maintaining decay heat removal.
The addition of these requirements is a more restrictive requirement
upon unit operation and is consistent with the NUREG.

M20 CTS 3.1.1.c.1 requires pressurizer safety valves to be OPERABLE when the reactor is critical. ITS LCO 3.4.10 requires the pressurizer safety valves to be OPERABLE with an Applicability of MODE 1, 2, and 3 with RCS temperature > 325°F. The change ensures the capability to maintain RCS pressure within required limits is available when the RCS is not open or LTOP is not providing overpressure protection. The added Applicability of MODE 2 when K_{eff} <1.0 and MODE 3 when RCS temperature > 325°F is a more restrictive requirement upon unit operation and is consistent with the NUREG.

M21 Not used.

ITS Section 3.4

ID 6

Subject 3.4 DOC M32 (6th line) "exist to be" to "exists to be"

periodically confirm the LCO requirements are met and ensures timely identification when LCO requirements are not met. These changes are additional restrictions upon unit operation and are consistent with the NUREG.

M31 CTS requirements comparable to ITS 3.4.13 Action A and RA B.2 does not exist for CTS 3.1.6.2 which is the CTS requirement comparable to ITS LCO 3.4.13.b or for CTS 3.1.6.1 which is the CTS requirement comparable to ITS 3.4.13.c. With unidentified LEAKAGE exceeding 1 gpm, CTS 3.6.1.2 requires the reactor to be shutdown within 24 hours of detection. With total LEAKAGE exceeding 10 gpm, CTS 3.6.1.1 requires the reactor to be shutdown within 24 hours of detection. ITS 3.4.13 Action A permits 4 hours to reduce LEAKAGE to within limits. If the Required Action or associated Completion Time of Condition A is not met ITS 3.4.13 Action B requires the unit be in MODE 3 within 12 hours and MODE 5 within 36 hours. The requirements to be in MODE 3 within 16 hours (4 hours + 12 hours) versus the CTS required 24 hours is a more restrictive requirement upon operation and is consistent with the NUREG. These changes provide appropriate limits upon the period of time that RCS leakage is permitted to exceed specified limits while in the Applicability of the Specification. The ITS 3.4.13 RA B.2 requirement to place the unit in MODE 5 within 40 hours (4 hours + 36 hours) is a more restrictive requirement upon unit operation and is consistent with the NUREG. Placing the unit in MODE 5 is appropriate since places the unit outside the Applicability of the Specification and reduces the potential for further increases in RCS leakage rate.

M32 With any RCS LEAKAGE through a non-isolable fault in an RCS strength boundary, CTS 3.1.6.3 requires the reactor be shutdown and cooldown to cold shutdown initiated within 24 hours. The CTS requirement to begin the cooldown to cold shutdown within 24 hours implies the reactor be shutdown in 24 hours. Although CTS 3.1.6.3 requires initiation of a
1 cooldown to cold shutdown within 24 hours, no explicit limit exists to be in cold shutdown. ITS 3.4.13 Action B requires the unit be in MODE 3 within 12 hours and in MODE 5 within 36 hours. This change provides appropriate limits upon the period of time that RCS leakage is permitted to exceed specified limits while in the Applicability of the Specification. The requirements to be in MODE 3 within 12 hours and MODE 5 within 36 hours are more restrictive requirement upon unit operation and are consistent with the NUREG.

M33 If unidentified reactor coolant exceeds 1 gpm, CTS 3.1.6.2 requires the reactor be shutdown within 24 hours. If total reactor coolant leakage exceeds 10 gpm, CTS 3.1.6.1 requires the reactor be shutdown within 24 hours. Neither CTS 3.1.6.1 and 3.1.6.2 require placing the unit in Cold Shutdown. With RCS leakage exceeding limits for reasons other than pressure boundary leakage, ITS 3.4.13 RA A.1 requires reducing RCS leakage to within limits in 4 hours. If Required Action and associated Completion Time of Condition A is not met, ITS 3.4.13 RA B.1 requires the unit be place in MODE 3 within 12 hours and MODE 5 within 36 hours. These changes provide appropriate limits upon the period of time that

ITS Section 3.4

ID 7

Subject Revise ITS 3.4.13.e to incorporate Amendment 227 (changes primary to secondary leakage limit to 150 gallon per day per SG)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE;

1 d. 300 gallon per day total primary to secondary LEAKAGE through all steam generators (SGs); and

1 e. 150 gallon per day primary to secondary LEAKAGE through any one SG.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time not met. OR Pressure boundary LEAKAGE exists.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	12 hours 36 hours

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 RCS Specific Activity

BASES

BACKGROUND

The Code of Federal Regulations, 10 CFR 100 (Ref. 1), specifies the maximum dose to the whole body and the thyroid an individual at the site boundary can receive for 2 hours during an accident. The limits on specific activity ensure that the doses are held to within the 10 CFR 100 limits during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to within the 10 CFR 100 dose guideline limits.

Analysis shows the potential offsite dose levels for an SGTR accident are within the 10 CFR 100 dose guideline limits (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensure that the resulting 2 hour doses at the site boundary will not exceed the 10 CFR 100 dose guideline
1 limits following an SGTR or a steam line break (SLB)
1 accident. The SLB safety analysis (Ref. 2) assumptions
1 bound the specific activity of the reactor coolant at the
1 LCO limits and a total existing reactor coolant steam
1 generator (SG) tube leakage rate of 300 gpd. However, the
1 300 gpd leakage has a negligable effect on the consequences
1 of a SLB. The analysis also assumes a reactor trip and a
1 turbine trip as a result of the SLB event.

1 The analysis results for the SGTR accident are significantly
1 impacted by the acceptance limits for RCS specific activity.
Reference to this analysis is used to assess changes to the facility that could affect RCS specific activity as they relate to the acceptance limits.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During unit life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area are necessary. Separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur detrimental to the safety of the facility and the public.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analysis radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA). However, the ability to monitor leakage provides advance warning to permit unit shutdown before a LOCA occurs. This advantage has been shown by "leak before break" studies.

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of
1 such an event. The steam line break (SLB) and Loss of Load
1 Safety analyses assume total primary to secondary LEAKAGE
1 greater than 300 gallon per day as the initial condition.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a SLB accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid and can be released to the environment.

1 The safety analysis assumptions for the SLB accident bounds
1 300 gallon per day primary to secondary LEAKAGE in one
generator as an initial condition. The dose consequences
1 resulting from the SLB accident are within the limits
defined in 10 CFR 100.

RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36 (Ref.3).

LCO

RCS leakage includes leakage from connected systems up to and including the second normally closed valve for systems which do not penetrate containment and the outermost isolation valve for systems which penetrate containment.

RCS operational LEAKAGE shall be limited to:

a. <u>Pressure Boundary LEAKAGE</u>

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals, gaskets, and steam generator tubes is not pressure boundary LEAKAGE.

(continued)

BASES

LCO (continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS makeup system. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)

Total primary to secondary LEAKAGE amounting to 300 gallon per day through all SGs produces acceptable offsite doses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

e. Primary to Secondary LEAKAGE through Any One SG

The 150 gallon per day limit on one SG is equivalent to a total of 300 gallon per day primary to secondary LEAKAGE allocated equally between the two generators.

(continued)

Specification 3.4.13

A1 <except as marked>

Add LCO Part d — A4

3.1.6 Leakage

Add ACT A + RA B.2 — M31

Specification

identified — L12 — M33

LCO 3.4.13.c

~~3.1.6.1~~ If the ~~total reactor~~ coolant leakage rate exceeds 10 gpm, the reactor shall be ~~shutdown~~ MODE 3 within ~~24~~ 12 hours of detection.

RA B.1

LA6 — A19

~~3.1.6.2~~ If unidentified reactor coolant leakage ~~(excluding normal evaporative losses)~~ exceeds 1 gpm ~~or if any reactor coolant leakage is evaluated as unsafe~~, the reactor shall be ~~shutdown~~ MODE 3 within ~~24~~ 12 hours of detection.

LCO 3.4.13.b

RA B.1

M33

pressure

~~3.1.6.3~~ If any reactor coolant leakage exists through a ~~non-isolable fault in a RCS strength~~ pressure boundary ~~(such as the reactor vessel, piping, valve body, etc., except the steam generator tubes)~~, the reactor shall be ~~shutdown~~ and cooldown to the ~~cold shutdown~~ MODE 5 condition shall be initiated within 24 hours of detection. placed in MODE 3 in 12 hrs.

LCO 3.4.13.a — A18

ACT B

M32

Supp 1

~~3.1.6.4~~ If the total leakage through the tubes of any one steam generator equals or exceeds 150 gallons per day, a reactor shutdown shall be initiated within 4 hours and the reactor shall be in ~~a cold condition~~ MODE 5 within the ~~next~~ 36 hours.

LCO 3.4.13.e
ACT A
ACT B

M33

~~3.1.6.5 If reactor shutdown is required by Specification 3.1.6.1, 3.1.6.2 or 3.1.6.3, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case and justified in writing as soon thereafter as practicable.~~

M34

~~3.1.6.6 Action to evaluate the safety implication of reactor coolant leakage shall be initiated within 4 hours of detection. The nature, as well as the magnitude, of the leak shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the guidelines of 10 CFR 20.~~

LA6

~~3.1.6.7 If reactor shutdown is required per Specification 3.1.6.1, 3.1.6.2, 3.1.6.3 or 3.1.6.4, the reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.~~

A20

3.1.6.8 When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different operating principles shall be operable, with one of the two systems sensitive to radioactivity. The systems sensitive to radioactivity may be out-of-service for 48 hours provided two other means to detect leakage are operable.

<SEE 3.4.15>

A21

~~3.1.6.9 Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which vent to the gas vent header and from which coolant can be returned to the reactor coolant system shall not be considered as reactor coolant leakage and shall not be subject to the consideration of Specifications 3.1.6.1, 3.1.6.2, 3.1.6.3, 3.1.6.4, 3.1.6.5, 3.1.6.6 or 3.1.6.7~~ except that such losses when added to leakage shall not exceed 30 gpm.

R4

3.1.6.10

a. The maximum allowable leakage for valves CF-12, CF-14, LP-47 and LP-48 shall be as follows:

<SEE 3.4.14>

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering are in accordance with NUREG-1430, Revision 1. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable, by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1430, Revision 1. During Improved Technical Specification (ITS) development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1430, Revision 1. Since the design is already approved by the NRC, adding more detail does not result in a technical change.

A2 The ONS 1, 2, & 3 current technical specifications (CTS) Bases are completely replaced by revised bases that reflect the format and applicable content of proposed ITS Section 3.4. The revised Bases are shown in the proposed ITS Bases for Section 3.4.

A3 CTS 3.1.6.10.a requires PIV leakage to be ≤ 5.0 gpm for specified valves. PIV leakage > 1.0 gpm and ≤ 5.0 gpm requires verification that the change in leakage rate since the previous test does not reduce the margin from the previously measured leakage to the 5.0 gpm limit by ≥ 50%. PIV leakage ≤ 1.0 gpm is acceptable without further evaluation. ITS SR 3.4.14.1 requires leakage from each PIV to be ≤ 0.5 gpm per nominal inch of valve size. The limit upon leakage rate increase since the previous test is separately deleted per DOC L1. The CTS listed ONS PIVs are nominally ≥ 10 inches in diameter. Consequently, for the CTS 3.1.6.10 specified valves the 0.5 gpm per nominal inch of diameter is equivalent to the maximum ITS specified PIV leakage of 5.0 gpm. Therefore, this change is administrative and is consistent with the NUREG. Other PIVs added by ITS LCO 3.4.14 may have leakage limits less than 5.0 gpm based on the specific PIV's nominal size.

A4 Explicit CTS requirements comparable to ITS LCO 3.4.13 part d do not
1 exist. ITS LCO 3.4.13 part d establishes a 300 gallon per day limit for
1 total primary to secondary leakage. CTS 3.1.6.4 permitted up to 150
1 gallon per day primary to secondary leakage through either of the two
1 SGs. A primary to secondary leakage at the 150 gallon per day limit for
each of the two SGs is equivalent to a total primary to secondary
1 leakage of 300 gallon per day. Therefore, this change is administrative
and is consistent with the NUREG.

A5 Not used.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE; 3.1.6.3

4

b. 1 gpm unidentified LEAKAGE; 3.1.6.2

c. 10 gpm identified LEAKAGE; 3.1.6.1

300 gallons per day

d. ~~1 gpm~~ total primary to secondary LEAKAGE through all steam generators (SGs); and Doc A4 3.1.6.4

Supp 1

e. ~~[720]~~ 150 gallons per day primary to secondary LEAKAGE through any one SG. 1

APPLICABILITY: MODES 1, 2, 3, and 4. Table 4.1-2 Footnote (2)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours	3.1.6.4
B. Required Action and associated Completion Time ~~of Condition A~~ not met. (2) OR Pressure boundary LEAKAGE exists.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	(5) 12 ~~6~~ hours 36 hours	3.1.6.1 3.1.6.2 3.1.6.3 3.1.6.4 3.1.6.3 3.1.6.4

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.~~16~~ 11 RCS Specific Activity

BASES

BACKGROUND

The Code of Federal Regulations, 10 CFR 100 (Ref. 1), specifies the maximum dose to the whole body and the thyroid an individual at the site boundary can receive for 2 hours during an accident. The limits on specific activity ensure that the doses are held to ~~a small fraction of~~ within the 10 CFR 100 limits during analyzed transients and accidents. 4

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.

4 The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to ~~a small fraction of~~ within the 10 CFR 100 dose guideline limits. ~~The limits in the LCO are standardized based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.~~ 2

~~The parametric evaluations~~ Analysis showed (5) safe the potential offsite dose levels for an SGTR accident ~~were an appropriately small fraction of~~ within the 10 CFR 100 dose guideline limits (Ref. 1). ~~Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.~~ 2 4

However, the 300 gpd leakage has negligible effect on the consequences of a SGTR. 4

Supp. 1

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensure that the resulting 2 hour doses at the site boundary will not exceed ~~a small fraction of~~ the 10 CFR 100 dose guideline limits following an steam line break (SLB) ~~SGTR~~ accident. The ~~SGTR~~ SLB safety analysis (Ref. 2) ~~assumes~~ assumptions bound the specific activity of the reactor coolant at the LCO limits and ~~an~~ existing reactor coolant steam generator (SG) tube leakage rate of ~~1 gpm~~ 300 gpd total. The analysis also assumes a reactor trip and a turbine trip ~~at the same time as~~ as a result of the SGTR event. 4

The analysis for the SGTR accident ~~establishes the~~ are significantly impacted by the acceptance limits for RCS specific activity. Reference to

(continued)

BASES

BACKGROUND (continued)

UNIT

the ability to monitor leakage provides advance warning to permit ~~plant~~ shutdown before a LOCA occurs. This advantage has been shown by "leak before break" studies. 2

APPLICABLE SAFETY ANALYSES

2

total

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis ~~for an event resulting in steam discharge to the atmosphere~~ assumes ~~a 1 gpm~~ primary to secondary LEAKAGE as the initial condition.

Supp. 1

Steam line break (SLB) and Loss of Load

greater than 300 gallon per day

2

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

and can be released to the environment

4

~~The FSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to the condenser. The 1 gpm primary to secondary LEAKAGE is relatively inconsequential.~~

~~The SLB is more limiting for site radiation releases.~~ The safety analysis for the SLB accident ~~assumes 1 gpm~~ primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the SLB accident are ~~well~~ within the limits defined in 10 CFR 100.

bounds 300 gallon per day

assumptions

4

RCS operational LEAKAGE satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10CFR50.36(Ref 3).

LCO

Insert B3.4-69A

RCS operational LEAKAGE shall be limited to:

9

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued

(continued)

BASES

LCO (continued)

degradation of the RCPB. LEAKAGE past seals, ~~and~~ gaskets, and steam generator tubes is not pressure boundary LEAKAGE. 4

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of ~~identified~~ unidentified LEAKAGE and is well within the capability of the RCS makeup system. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system. TSTF 54,R1

d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)

supp 1

Total primary to secondary LEAKAGE amounting to ~~1 gpm~~ 300 gallons per day through all SGs produces acceptable offsite doses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE. 4

1

supp 1

~~e. Primary to Secondary LEAKAGE through Any One SG~~

~~The~~ 150 ~~(720)~~ ~~gallon per day limit on one SG~~ ~~allocates the~~ is equivalent to a ~~total~~ 300 gallon per day ~~(1 gpm allowed)~~ primary to secondary LEAKAGE allocated equally between the two generators. 2 5

2

(continued)

ITS Section 3.4

ID 28

Subject DOC LA12 for Section 3.4 should refer to CTS 4.3.2 not CTS 4.5.2

operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable. Changes to the UFSAR are controlled in accordance with 10 CFR 50.59.

LA10 The requirements of CTS 3.1.3.3 regarding the subcriticallity requirements when the reactor coolant temperature is < 525°F is relocated to UFSAR Chapter 16. These details are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for shutdown margin. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable. Changes to the UFSAR are controlled in accordance with 10 CFR 50.59.

LA11 CTS 3.1.4 includes a detail regarding the comparison of RCS total activity to the limit. CTS 3.1.4 specifically excludes radionuclides with half lives less than 30 minutes from the comparison. This detail is relocated to the Bases. These details are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for system OPERABILITY. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable. Changes to the Bases are controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

1 LA12 CTS 4.3.2 specifies requirements for leak testing the RCS following any opening. This detail is relocated to UFSAR Chapter 16. This detail is not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains appropriate requirements for RCS operational leakage and OPERABILITY for leak detection instrumentation. Additionally, RCS leak testing is specified in appropriate ASME Code Inservice Inspection requirements which are required to be met by 10 CFR 50.55a. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is

ITS Section 3.4

ID 46

Subject Incorporate revision to ONS-013 (BWOG-70) - Moves LCO Note back to Applicability Note.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for loop pressure, loop average temperature, and RCS total flow rate shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

1 ----------------------------NOTE----------------------------
1 RCS loop pressure limit does not apply during:
1
1 a. THERMAL POWER ramp > 5% RTP per minute; or
1
1 b. THERMAL POWER step > 10% RTP.
1 --

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Two pressurizer safety valves shall be OPERABLE with lift settings ≥ 2475 psig and ≤ 2525 psig.

APPLICABILITY: MODES 1 and 2,
MODE 3 with all RCS cold leg temperatures > 325°F.

1 ---------------------------NOTE---------------------------
1 The lift settings are not required to be within the LCO
1 limits for entry into the applicable portions of MODE 3 for
1 the purpose of setting the pressurizer safety valves under
1 ambient (hot) conditions. This exception is allowed for
1 36 hours following entry into the applicable portions of
1 MODE 3 provided a preliminary cold setting was made prior to
1 heatup.
1 --

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. <u>OR</u> Two pressurizer safety valves inoperable.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 3 with any RCS cold leg temperature ≤ 325°F.	12 hours 18 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12 An LTOP System shall be OPERABLE with high pressure injection (HPI) deactivated, and the core flood tanks (CFTs) deactivated and:

a. An OPERABLE power operated relief valve (PORV) with a lift setpoint of ≤ 480 psig; and

b. Administrative controls implemented that assure ≥ 10 minutes are available for operator action to mitigate an LTOP event.

APPLICABILITY: MODE 3 when any RCS cold leg temperature is ≤ 325°F,
MODES 4, 5, and 6 when an RCS vent path capable of mitigating the most limiting LTOP event is not open.

--------------------------NOTES--------------------------

1. CFT deactivation is only required when CFT pressure is greater than or equal to the maximum RCS pressure for the existing RCS temperature allowed by the pressure and temperature limit curves provided in Specification 3.4.3.

2. The PORV is not required to be OPERABLE when no HPI pumps are running and RCS pressure < 100 psig.

BASES

LCO (continued)

operates within the limits assumed for the plant safety analyses. Operating within these limits will result in meeting DNBR criteria in the event of a DNB limited transient.

The pressure limits are applied to the loop with the highest pressure. The temperature limits are applied to the loop with the lowest loop average temperature.

APPLICABILITY

In MODE 1, the limits on RCS pressure, RCS loop average temperature, and RCS flow rate must be maintained during steady state with four pump or three pump operation in order to ensure that DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES the power level is low enough so that DNB is not a concern.

1 The Note indicates the limit on RCS pressure may be exceeded
1 during short term operational transients such as a THERMAL
1 POWER ramp increase > 5% RTP per minute or a THERMAL POWER
1 step increase > 10% RTP. These conditions represent short
1 term perturbations where actions to control pressure
1 variations might be counterproductive. Also, since they
1 represent transients initiated from power levels < 100% RTP,
1 increased DNBR margin exists to offset the temporary
1 pressure variations.

ACTIONS

A.1

Loop pressure and loop average coolant temperature are controllable and measurable parameters. With one or both of these parameters not within the LCO limits, action must be taken to restore the parameters. RCS flow rate is not a controllable parameter and is not expected to vary during steady state four pump or three pump operation. However, if the flow rate is below the LCO limit, the parameter must be restored to within limits or power must be reduced as required in Required Action B.1, to restore DNBR margin and eliminate the potential for violation of the accident analysis bounds.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

All accident analyses in the UFSAR that require safety valve actuation assume operation of both pressurizer safety valves to limit increasing reactor coolant pressure. The overpressure protection analysis is also based on operation of both safety valves and assumes that the valves open at the high range of the setting (2500 psig system design pressure plus 1%). These valves must accommodate pressurizer insurges that could occur during a startup, rod withdrawal, ejected rod, or loss of main feedwater. The startup accident establishes the minimum safety valve capacity. The startup accident is assumed to occur at < 15% power. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this Specification is required to ensure that the accident analysis and design basis calculations remain valid.

Pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

The two pressurizer safety valves are set to open at the RCS design pressure (2500 psig) and within the ASME specified tolerance to avoid exceeding the maximum RCS design pressure SL, to maintain accident analysis assumptions and to comply with ASME Code requirements. The upper and lower pressure tolerance limits are based on the ±1% tolerance requirements (Ref. 1) for lifting pressures above 1000 psig. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or both valves could result in exceeding the SL if a transient were to occur.

The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

1
1
1
1

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and portions of MODE 3 above the LTOP cut in temperature, OPERABILITY of two valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. Portions of MODE 3 are conservatively included, although the listed accidents may not require both safety valves for protection.

The LCO is not applicable in MODE 3 when any RCS cold leg temperature is ≤ 325°F, MODE 4 and MODE 5 because LTOP protection is provided. Overpressure protection is not required in MODE 6 with the reactor vessel head detensioned.

1 The Note allows entry into MODE 3 with the lift settings
1 outside the LCO limits. This permits testing and
1 examination of the safety valves at high pressure and
1 temperature near their normal operating range, but only
1 after the valves have had a preliminary cold setting. The
1 cold setting gives assurance that the valves are OPERABLE
1 near their design condition. Only one valve at a time will
1 be removed from service for testing. The 36 hour exception
1 is based on an 18 hour outage time for each of the two
1 valves. The 18 hour period is derived from operating
1 experience that hot testing can be performed in this time
1 frame.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS overpressure protection system. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the RCPB.

B.1 and B.2

If the Required Action cannot be met within the required Completion Time or if both pressurizer safety valves are inoperable, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and to MODE 3 with any RCS cold leg temperature ≤ 325°F within

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

18 hours. The 12 hours allowed is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. Similarly, the 18 hours allowed is reasonable, based on operating experience, to reach MODE 3 with any RCS cold leg temperature ≤ 325°F without challenging unit systems. With any RCS cold leg temperature at or below 325°F, overpressure protection is provided by LTOP. Reducing the RCS temperature to ≤ 325°F reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by two pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 2), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. ASME, Boiler and Pressure Vessel Code, Section XI.
3. 10 CFR 50.36.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Administrative Controls Performance

Limiting RCS pressure when RCS temperature is < 220°F provides a minimum margin to the RCS P/T limit. Restricting RCS makeup flow capability and pressurizer level and controls on the use of high pressure nitrogen limit the pressurization rate during an LTOP event. Alarms ensure early operator recognition of the occurrence of an LTOP event. The combination of minimum margin to the limit, limited pressurization rate and OPERABLE alarms ensure ten minutes are available for operator action to mitigate an LTOP event.

The LTOP System satisfies Criterion 2 and Criterion 3 of 10 CFR 50.36 (Ref.6).

LCO

The LCO requires an LTOP System OPERABLE with a limited coolant input capability and a pressure relief capability. The LCO requires HPI and the CFTs to be deactivated. For pressure relief, it requires the pressurizer coolant at or below a maximum level and the PORV OPERABLE with a lift setting at the LTOP limit, with other specified administrative controls.

The pressurizer is OPERABLE with a coolant level limited such that ≥ 10 minutes are available for operator action to mitigate the consequences of an LTOP event.

The PORV is OPERABLE when its block valve is open, its lift setpoint is set at ≤ 480 psig and testing has proven its ability to open at that setpoint, and power is available to the two valves and their control circuits.

1

APPLICABILITY

This LCO is applicable in MODE 3 when any RCS cold leg temperature is ≤ 325°F, and in MODES 4, 5 and 6 when an RCS vent capable of mitigating the most limiting LTOP event is not open. The Applicability temperature of 325°F is established by fracture mechanics analyses. The pressurizer safety valves provide overpressure protection to meet LCO 3.4.3 P/T limits above 325°F. With the vessel head off, overpressurization is not possible. With an RCS vent capable of mitigating the most limiting LTOP event open, an

(continued)

BASES

APPLICABILITY (continued)

LTOP event (including HPI actuation or CFT discharge) is incapable of pressuring the RCS above the RCS P/T limits.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the pressurizer safety valves OPERABLE to provide overpressure protection during MODES 1, 2, and 3 above 325°F.

1 The Applicability is modified by two Notes. Note 1 states
1 that CFT deactivation is only required when the CFT pressure
1 is more than or equal to the maximum RCS pressure for the
1 existing RCS temperature, as allowed in LCO 3.4.3. This
1 Note permits the CFT discharge valve surveillance performed
1 only under these pressure and temperature conditions.
1
1 Note 2 permits the PORV to be inoperable when no HPI pumps
1 are running and RCS pressure is < 100 psig. PORV
1 operability is not required when RCS pressure is < 100 psig
1 and HPI pumps are not operating since credible LTOP events
1 progress relatively slowly, thus giving the operator ample
1 time to respond.

ACTIONS

A.1

With the HPI activated, immediate actions are required to deactivate HPI. Emphasis is on immediate deactivation because inadvertent injection with one or more HPI pump OPERABLE is the event of greatest significance, since these events cause the greatest pressure increase in the shortest time.

The immediate Completion Times reflect the urgency of quickly proceeding with the Required Actions.

B.1, C.1, and C.2

A non deactivated CFT requires deactivation within 1 hour only when the CFT pressure is at or more than the maximum RCS pressure for the existing temperature allowed in LCO 3.4.3.

If deactivation is needed and cannot be accomplished in 1 hour, Required Action C.1 and Required Action C.2 provide

(continued)

Specification 3.4.10

A1 <except as marked>

3.1 REACTOR COOLANT SYSTEM

~~Applicability~~

~~Applies to the operating status of the reactor coolant system.~~

~~Objective~~

~~To specify those limiting conditions for operation of the reactor coolant system components which must be met to ensure safe reactor operation.~~

Specification

3.1.1 Operational Components

a. Reactor Coolant Pumps

1. Whenever the reactor is critical, one and two pump operation shall be prohibited, single-loop operation shall be restricted to testing, and other pump combinations permissible for given power levels shall be as shown in Table 2.3-1.

<See 3.4.4>

2. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one low pressure injection pump is circulating reactor coolant.

b. Steam Generator

1. One steam generator shall be operable whenever the reactor coolant average temperature is above 250°F.

<See 3.4.4 3.4.5 3.4.6 3.4.7 3.4.8>

<See 3.4.5 & 3.4.4>

c. Pressurizer Safety Valves A30

MODES 1, 2 & MODES 3 >325°F

LCO Applicability

1. ~~All~~ Two pressurizer code safety valves shall be operable ~~whenever the reactor is critical.~~ M20

{with lift setting ≥2475 psig & ≤2525 psig}

2. ~~At least one pressurizer code safety valve shall be operable whenever all reactor coolant system openings are closed, except for hydrostatic tests in accordance with the ASME Section III Boiler and Pressure Vessel Code.~~ M47

L17

Supp. 1

Add LCO Applicability Note A31

Add Action A L6

Add Action B M22

Specification 3.4.12

A1 <except as marked>

~~3.1.2.7 Not used~~

~~3.1.2.8 Not used~~

A32

~~3.1.2.9~~ ~~1.~~ The requirements of 2 below shall be met when both of the following conditions apply:

Applicability

MODE 3 when

a) ~~The~~ temperature of one or more of the RCS cold legs is $\leq$ 325°F, and

MODES 4, 5 and 6 when

b) ~~An~~ RCS vent path capable of mitigating the most limiting LTOP event is not open.

LCO ~~2.~~ ~~a)~~ Two trains of the low temperature overpressure protection (LTOP) system shall be operable,

LCO b) HPI train A and B shall be deactivated, and

LCO c) Both core flood tanks shall be deactivated.

LCO a. ~~3.~~ One LTOP train is comprised of the PORV with a lift setting of $\leq$ 480 psig.

Supp.1 | LCO Applicability Note 2 ~~a)~~ The PORV is not required to be operable when no HPI pumps are running and RCS pressure is < 100 psig.

LCO b. ~~4.~~ The second LTOP train is comprised of the controls which assure that 10 minutes are available for operator action to mitigate an LTOP event. ~~The following controls comprise the second LTOP train:~~

~~a) RCS pressure is limited to $\leq$ 345 psig for an RCS temperature < 220°F.~~

~~b) Pressurizer level shall be controlled such that 10 minutes are available for operator action to mitigate an LTOP event.~~

~~c) Makeup flow shall be restricted such that 10 minutes are available for operator action to mitigate an LTOP event.~~

~~d) Alarms shall be provided such that 10 minutes are available for operator action to mitigate an LTOP event.~~

~~e) The high pressure nitrogen system shall be controlled such that 10 minutes are available for operator action to mitigate an LTOP event.~~

LA8

Supp. 1 | Add LCO Applicability Note 1 — A27

A25 A CTS provision comparable to the Note to ITS 3.4.15 Actions does not exist. The Note to ITS 3.4.15 Actions states that ITS 3.0.4 is not applicable. This Note therefore permits entry into MODES 1, 2, 3 and 4 when not meeting LCO 3.4.15. CTS does not preclude entry into an operational condition when all the requirements for being in the operational condition are not met. Therefore this change is administrative in nature and is consistent with the NUREG.

A26 CTS 3.1.3.1 requires the reactor coolant temperature to be > 525°F. ITS LCO 3.4.2 requires the reactor coolant temperature to be ≥ 525°F. This difference is so slight as to be imperceptible. Reactor criticality at precisely 525°F is within the applicable analysis. Therefore this, change is considered to be administrative and is consistent with the NUREG.

A27 CTS 3.1.2.9.2.c requires CFTs be deactivated. The Bases for CTS 3.1.2 (page 3.1-4b) states that the CFTs may be deactivated by reducing CFT pressure to less than the maximum permissible RCS pressure for existing
1 RCS temperature. ITS 3.4.12 LCO Applicability Note 1 specifies that CFT isolation is not required when CFT pressure is less than the maximum RCS pressure for the existing RCS temperature allowed by the pressure and temperature limit curves provided in ITS Specification 3.4.3. Therefore this change is a presentation preference which administrative in nature and is consistent with the NUREG.

A28 CTS 3.1.2.9.5.a requires HPI pumps be deactivated immediately. ITS 3.4.12 Actions A require action be initiated immediately to deactivate HPI pumps. The intent of the ITS is the same, however, the ITS wording recognizes the literal impossibility of completing physical actions as may be required by a strict interpretation of "immediately" while still requiring timely completion. An ITS specified Required Action with an immediate completion time is required by ITS 1.3, Completion Times, to be pursued without delay and in a controlled manner. Therefore, this change is considered to be an administrative change and is consistent with the NUREG.

A29 CTS figures 3.1.2-3A, 3.1.2-3B and 3.1.2-3C are titled, "Reactor Coolant System Inservice Leak and Hydrostatic Test Heatup and Cooldown Limitations . . . ." The equivalent ITS figures, 3.4.3-3, 3.4.3-6 and 3.4.3-9 are titled, "Reactor Coolant System Leak and Hydrostatic Test Heatup and Cooldown Limitations . . . ." The term "inservice" is not necessary since the tables are also valid for non-inservice leak tests. The elimination of the term "inservice" from the title is considered an administrative change which preserves the intent of the CTS.

A30 CTS 3.1.1.c.1 requires all pressurizer code safety valves to be OPERABLE. ITS 3.4.10 requires two pressurizer safety valves to be OPERABLE. Since the ONS design provides two pressurizer safety relief valves, this change is an administrative change and is consistent with the NUREG.

A31 CTS 3.1.1.c.1 requires pressurizer safety valves to be OPERABLE when the
1 reactor is critical. The Note to ITS LCO 3.4.10 Applicability permits entry into MODE 3 for up to 36 hours with pressurizer lift setting not met for the purpose of setting the lift settings under ambient (hot) conditions. Since CTS does not impose OPERABILITY requirements for the pressurizer safety valves in this operating condition, this change is administrative and is consistent with the NUREG.

A32 CTS 3.1.2.9 provides requirements for LTOP. CTS 3.1.2.9.1.a and 3.1.2.9.1.b specify an Applicability of when one or more RCS cold legs are ≤ 325°F and an RCS vent path capable of mitigating the most limiting LTOP event is not open. ITS 3.4.12 specifies an Applicability of MODE 3 when any RCS cold leg temperature is ≤ 325°F and also during MODES 4, 5, and 6 when an RCS vent path capable of mitigating the most limiting LTOP event is not open. The different presentation in the ITS recognizes that an RCS vent path cannot be open in MODE 3 when any RCS cold leg temperature is ≤ 325°F or in portions of MODE 4 when RCS temperature is ≥ 212°F. This change is administrative and is consistent with the NUREG.

A33 CTS 4.17.4 requires performance of specified SG tube inservice inspections within specified frequencies. ITS SR 3.4.13.2 requires verification of steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program at a Frequency of "In accordance with the Steam Generator Tube Surveillance Program. The details regarding the specified SG tube inservice inspections and specified frequencies are relocated unchanged to ITS 5.5.10, "Steam Generator (SG) Tube Surveillance Program." Therefore this is an administrative change and is consistent with the NUREG.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

loop average 4B

LCO 3.4.1 RCS DNB parameters for loop pressure, ~~hot leg~~ temperature, and RCS total flow rate shall be within the limits specified ~~below:~~ Doc M1

in the COLR. 19

~~a. With four reactor coolant pumps (RCPs) operating:~~

~~RCS loop pressure shall be ≥ [2061.6] psig, RCS hot leg temperature shall be ≤ [604.6]°F, and RCS total flow rate shall be ≥ [139.7 E6] lb/hr; and~~

~~b. With three RCPs operating:~~

~~RCS loop pressure shall be ≥ [2057.2] psig, RCS hot leg temperature shall be ≤ [604.6]°F, and RCS total flow rate shall be ≥ [104.4 E6] lb/hr.~~

APPLICABILITY: MODE 1. Doc M1

Supp. 1

-----------------------------NOTE-----------------------------
RCS loop pressure limit does not apply during: Doc M1

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours Doc M1
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	~~6~~ 12 hours 5 Doc M1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

3.1.1.c.1

LCO 3.4.10 Two pressurizer safety valves shall be OPERABLE with lift settings ≥ ~~[2475]~~ psig and ≤ [2525] psig. 1

24 and

APPLICABILITY: MODES 1, 2, ~~and 3~~
MODE ~~4~~ 3 with all RCS cold leg temperatures > ~~[283]~~ 325°F. 1 24

3.1.1.c.1

4 Supp. 1 The applicable portions of

NOTE

The lift settings are not required to be within the LCO limits for entry into MODE~~S~~ 3 ~~and 4~~ for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for ~~[36]~~ hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup. 24 DOC M21 1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1	Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two pressurizer safety valves inoperable.	B.1 AND B.2	Be in MODE 3. Be in MODE ~~4~~ 3 with any RCS cold leg temperature ≤ ~~[283]~~ 325 °F.	~~6~~ 12 hours 5 ~~12~~ 18 hours

DOC L6

DOC M22

DOC M22

24

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

51 3.1.2.9.2

LCO 3.4.12 An LTOP System shall be OPERABLE with ~~a maximum of [one] makeup pump capable of injecting into the RCS,~~ high pressure injection (HPI) deactivated, and the core flood tanks (CFTs) deactivated ~~isolated~~ and: 27

3.1.2.9.3

Insert 3.4-22A

~~a. Pressurizer level ≤ [220] inches and an OPERABLE power operated relief valve (PORV) with a lift setpoint of ≤ [555] psig; or~~ 27

~~b. The RCS depressurized and an RCS vent of ≥ [0.75] square inch.~~ 3.1.2.9.4

27 1

APPLICABILITY: MODE ~~4~~ 3 when any RCS cold leg temperature is ≤ ~~[283]~~ 325 °F, 3.1.2.9.1
MODE 5,
MODE 6 when ~~the reactor vessel head is on~~ 27 2 DOC A27

Supp.1

-----------------------------------NOTES-----------------------------------
1. CFT ~~isolation is~~ deactivation only required when CFT pressure is greater than or equal to the maximum RCS pressure for the existing RCS temperature allowed by the pressure and temperature limit curves provided in ~~the PTLR~~. Insert 3.4-22B 28 Specification 3.4.3 7

MODES 4, 5 and 6 when an RCS vent path capable of mitigating the most limiting LTOP event is not open. 27

To maintain consistency with the CLB, the requirements for makeup pumps and pressurizer level is encompassed in ITS LCO 3.4.12 requirements for Administrative Controls. ITS 3.4.12 Required Actions F and G are provided which include appropriate Required Actions if Administrative Control requirements are not in place. Conforming changes are made to NUREG 3.4.12 Action I, NUREG SR 3.4.12.4, and ITS SR 3.4.12.6 is added to maintain consistency with the changes made to LCO 3.4.12 and associated ITS Action F.

1 28 A second Note to LCO 3.4.12 Applicability is provided to retain consistency with the CLB. Note 2 permits the Power Operated Relief Valve to be inoperable when no HPI pumps are running and RCS pressure is < 100 psig. PORV operability is not required when RCS pressure is < 100 psig and HPI pumps are not operating since credible LTOP events progress relatively slowly, thus giving the operator ample time to respond.

29 The RCS temperature specified in NUREG 3.4.12 Action D.1 is changed from > 175°F to > 232°F to reflect a more restrictive plant specific limit based on the current P/T limit curves. The 232°F value is based on the current limit from the P/T curves at an RCS pressure equivalent to the required CFT upper limit on pressure.

30 NUREG 3.4.12 Required Actions H.1 and H.2 are modified to retain the CLB regarding the Required Actions with an inoperable PORV. The Required Actions specified require either exiting the Applicability for ITS 3.4.12 by raising RCS temperature to > 325°F or reducing RCS pressure to less than 100 psig. With RCS pressure < 100 psig, additional time is available for operator action to mitigate the consequences of an LTOP event. Increasing temperature to > 325°F places the unit outside the applicability of the specification where overpressure protection is provided by the pressurizer safety valves.

31 The Note to SR 3.4.1.4 is revised to allow some time after the "stable thermal conditions are established in the higher power range of MODE 1" to actually perform the measurement. Additionally, the Note is revised to clarify when the Surveillance is due. SR 3.4.1.4 provides only a confirmation of the reading accuracy from SR 3.4.1.3 which has already identified acceptable flow rate. Since this parameter does not normally change significantly, there is no need to hasten this confirmation during the initial operation at full power. Therefore, the Note is revised to allow 7 days after stable thermal conditions are established in the higher power range of MODE 1. This is consistent with approved allowances for performance of similar surveillances at another plant, i.e., Vogtle.

32 Not used.

1 33 Not used.

The pressure limits are applied to the loop with the highest pressure.

5

BASES

LCO (continued)

meeting DNBR criteria in the event of a DNB limited transient.

The ~~pressure and~~ temperature limits are ~~to be~~ applied to the loop with ~~two reactor coolant pumps (RCPs) running for the three RCPs operating condition.~~ 2

the lowest loop average temperature.

~~The LCO numerical values for pressure, temperature, and flow rate are given for the measurement location but have not been adjusted for instrument error. Plant specific limits of instrument error are established by the plant staff to meet the operational requirements of this LCO.~~ 5

Supp. 1 5

APPLICABILITY

In MODE 1, the limits on RCS pressure, RCS ~~hot leg~~ loop average temperature, and RCS flow rate must be maintained during steady state with four pump or three pump operation in order to ensure that DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES the power level is low enough so that DNB is not a concern. 5

Supp. 1

The Note indicates the limit on RCS pressure may be exceeded during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, increased DNBR margin exists to offset the temporary pressure variations.

2

Move to page 3.4.2-2

Another set of limits on DNBR related parameters is provided in Safety Limit (SL) 2.1.1, "Reactor Core SLs." Those limits are less restrictive than the limits of LCO 3.4.1, but violation of an SL merits a stricter, more severe Required Action. ~~Should a violation of LCO 3.4.1 occur, the operator must check whether an SL may have been exceeded.~~ 6

ACTIONS

A.1

Loop pressure and ~~hot leg~~ loop average coolant temperature are controllable and measurable parameters. With one or both of 5

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

All accident analyses in the (U)FSAR that require safety valve actuation assume operation of both pressurizer safety valves to limit increasing reactor coolant pressure. The overpressure protection analysis ~~(Ref. 1)~~ is also based on operation of both safety valves and assumes that the valves open at the high range of the setting (2500 psig system design pressure plus 1%). These valves must accommodate pressurizer insurges that could occur during a startup, rod withdrawal, ejected rod, or loss of main feedwater~~, or main feedwater line break accident~~. The startup accident establishes the minimum safety valve capacity. The startup accident is assumed to occur at < 15% power. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this Specification is required to ensure that the accident analysis and design basis calculations remain valid. ⊢(2) ⊢(2) ⊢(4)

Pressurizer safety valves satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10CFR50.36 (Ref 3). (2)

LCO

The two pressurizer safety valves are set to open at the RCS design pressure (2500 psig) and within the ASME specified tolerance to avoid exceeding the maximum RCS design pressure SL, to maintain accident analysis assumptions and to comply with ASME Code requirements. The upper and lower pressure tolerance limits are based on the ± 1% tolerance requirements (Ref. 1) for lifting pressures above 1000 psig. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or both valves could result in exceeding the SL if a transient were to occur.

The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

Supp. 1

APPLICABILITY

In MODES 1, 2, and ~~3, and~~ portions of MODE ~~4~~ 3 above the LTOP cut in temperature, OPERABILITY of two valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during ⊢(5)

(continued)

BASES

APPLICABILITY (continued)

certain accidents. ~~MODE 3 and~~ portions of MODE 4 are conservatively included, although the listed accidents may not require both safety valves for protection.

The LCO is not applicable in MODE 4 when any RCS cold leg temperature is ≤ [283]°F, and MODE 5 because LTOP protection is provided. Overpressure protection is not required in MODE 6 with the reactor vessel head detensioned.

Supp. 1

The Note allows entry into MODES 3 ~~and 4~~ with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The [36] hour exception is based on an 18 hour outage time for each of the two valves. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS overpressure protection system. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the RCPB.

B.1 and B.2

If the Required Action cannot be met within the required Completion Time or if both pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and ~~to MODE 4~~ with any RCS cold leg temperature ≤ [283]°F within 12 hours. The 6 hours allowed is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. Similarly, the 12 hours allowed is

(continued)

BASES

5

LCO (continued)

~~For the depressurized RCS, an RCS vent is OPERABLE when open with an area of at least [0.75] square inches.~~

Supp. 1

APPLICABILITY

This LCO is applicable ~~in~~ MODE ~~4~~ 3 when any RCS cold leg temperature is ≤ ~~[283]~~ 325 °F, ~~in MODE 5, and in MODE 6 when the reactor vessel head is on.~~ and in MODES 4, 5 + 6 when an RCS vent capable of mitigating the most limiting LTOP event is not open. The Applicability temperature of ~~[283]~~ 325 °F is established by fracture mechanics analyses. The pressurizer safety valves provide overpressure protection to meet LCO 3.4.3 P/T limits above ~~[283]~~ 325 °F. ~~With the vessel head off, overpressurization is not possible.~~ 5 1

Insert B3.4-62A

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the pressurizer safety valves OPERABLE to provide overpressure protection during MODES 1, 2, and 3 ~~and MODE 4~~ above ~~[283]~~ 325 °F. 2 1

4

Supp. 1

The Applicability is modified by ~~a~~ two Notes ~~Note~~ (1) stat~~ing~~es that CFT ~~isolation~~ deactivation is only required when the CFT pressure is more than or equal to the maximum RCS pressure for the existing RCS temperature, as allowed in LCO 3.4.3. This Note permits the CFT discharge valve surveillance performed only under these pressure and temperature conditions. 2 5 5

Insert B3.4-62B

5

ACTIONS

A.1 ~~and B.1~~

With ~~two or more makeup pumps capable of injecting into the RCS or if~~ the HPI ~~is~~ activated, immediate actions are required to ~~render the other pump(s) inoperable or to~~ deactivate HPI. Emphasis is on immediate deactivation because inadvertent injection with ~~one~~ * or more HPI pump OPERABLE is the event of greatest significance, since ~~it~~ these events cause~~s~~ the greatest pressure increase in the shortest time. ~~Also, the vent cannot mitigate overpressurization from the injection of even one HPI pump.~~ 4 2

The immediate Completion Times reflect the urgency of quickly proceeding with the Required Actions.

~~Required Action A.1 is modified by a Note that permits two pumps capable of RCS injection for ≤ 15 minutes to allow for pump swaps.~~ 4

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

acceptable considering the frequency and adequacy of the RCS water inventory balance required by Required Action A.1.

Insert on page B3.4-84

4 Normal

TSTF 60

The ~~Required~~ Action ~~A.1 and Required Action A.2~~ are modified by a Note indicating that the provisions of LCO 3.0.4 do not apply. As a result, a MODE change is allowed when the sump ~~monitor is~~ inoperable. This allowance is provided because other instrumentation is available to monitor RCS LEAKAGE.

and required radiation monitors are

level indication

5

B.1.1, B.1.2, and B.2

With required gaseous or particulate containment atmosphere radioactivity monitoring instrumentation channels inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. With a sample obtained and analyzed or a water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of at least one of the radioactivity monitors.

2 for SR 3.4.13.1

TSTF 116, R1

Insert B3.4-85A

The 24 hour interval provides periodic information that is adequate to detect leakage. The 30 day Completion Time recognizes at least one other form of leak detection is available.

TSTF 60

~~Required Actions B.1.1, B.1.2, and B.2 are modified by a Note indicating that the provisions of LCO 3.0.4 do not apply. As a result, a MODE change is allowed when the containment atmosphere radioactivity monitor is inoperable. This allowance is provided because other instrumentation is available to monitor RCS LEAKAGE.~~

C.1 and C.2

both required leakage detection instruments are inoperable or if

Supp 1

If a Required Action of Condition A or B cannot ~~be met~~ within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within ~~6~~ hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating

2 5

12

(continued)

ITS Section 3.4

ID 47

Subject Incorporate revision to ONS-023 (BWOG-78) - Changes the order of the info in Condition C.

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
	B. Required containment atmosphere radioactivity monitor inoperable.	B.1.1 Analyze grab samples of the containment atmosphere.	Once per 24 hours
		<u>OR</u>	
		B.1.2 --------NOTE--------- Not required until 12 hours after establishment of steady state operation. -------------------- Perform SR 3.4.13.1.	Once per 24 hours
		<u>AND</u>	
		B.2 Restore required containment atmosphere radioactivity monitor to OPERABLE status.	30 days
1 1 1	C. Both required instrument functions inoperable.	C.1 Be in MODE 3.	12 hours
	<u>OR</u>	<u>AND</u>	
1 1 1	Required Action and associated Completion Time not met.	C.2 Be in MODE 5.	36 hours

BASES

ACTIONS

B.1.1, B.1.2, and B.2 (continued)

be operated for up to 30 days to allow restoration of at least one of the radioactivity monitors.

The 24 hour interval for SR 3.4.13.1 provides periodic information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The 30 day Completion Time recognizes at least one other form of leak detection is available.

C.1 and C.2

1 If both required leakage detection instruments are
1 inoperable or if a Required Action of Condition A or B
cannot be met within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitor. The check gives reasonable confidence that each channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a CHANNEL FUNCTIONAL TEST of the required containment atmosphere radioactivity

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
B. (continued) ----Note---- Not required until 12 hours after establishment of steady state operations.	B.1.2	Perform SR 3.4.13.1.	Once per 24 hours	DOC M41
	AND			
	B.2	Restore required containment atmosphere radioactivity monitor to OPERABLE status.	30 days	3.1.6.8
			TSTF 116, R1	
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 ~~6~~ hours	5 DOC M43
	AND			
	C.2	Be in MODE 5.	36 hours	DOC M43
~~D.~~ Both required ~~monitors~~ instrument functions inoperable. OR	~~D.1~~	~~Enter LCO 3.0.3.~~	~~Immediately~~	TSTF 257

Supp 1

52

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.4.15.1	Perform CHANNEL CHECK of required containment atmosphere radioactivity monitor.	12 hours	Doc M42
SR 3.4.15.2	Perform CHANNEL FUNCTIONAL TEST of required containment atmosphere radioactivity monitor.	92 days	Doc M42

(continued)

34 Consistent with the current licensing basis, the DHR interlock functions in NUREG 3.4.14 are not adopted. NUREG 3.4.14 Action C, SR 3.4.14.2 and SR 3.4.14.3 are not adopted. The ONS design does not include the autoclosure function identified in NUREG SR 3.4.14.3. The interlock function identified in SR 3.4.14.2 is included in the ONS design for only one of the two valves in each unit's decay heat drop line. These valves are maintained closed and deenergized except when transitioning to or from the DHR mode of operation. It is undesirable to adopt NUREG SR 3.4.14.2 since it either requires challenging the interlock by attempting to open the valve when it is should be closed for plant operating conditions or closing the valve (to confirm it can't be opened) when it needs to be open for plant conditions. Maintaining the valves deenergized in the closed position except during the transition to or from the decay heat removal mode of operation substantially reduces the potential for inadvertent opening during normal unit operation.

35 Notes 2 and 3 to NUREG SR 3.4.14.1 are not adopted. Note 2 states the SR is not required to be performed for RCS PIVs in the DHR flow path when in the DHR mode of operation. Note 2 is unnecessary since Note 1 states that in MODES 3 and 4 the SR is not required to be performed. An ONS unit must be in MODE 4 to be in the DHR mode of operation. Note 3 states that RCS PIVs actuated during performance of the SR need not be tested more than once if a repetitive testing loop cannot be avoided. Note 3 is unnecessary since the NUREG SR 3.4.14.1 frequency requiring testing within 24 hours following valve actuation is not adopted.

The NUREG SR 3.4.14.1 frequency requiring testing within 24 hours following valve actuation is not adopted based on the Reviewers Note in the NUREG Bases for SR 3.4.14.1 since ONS was licensed prior to 1980. This Note states the "24 hour..." Frequency of performance for Surveillance Requirement 3.4.14.1 is not required for B&W Owner's Group plants licensed prior to 1980. These plants were licensed prior to the NRC establishing formal Technical Specification controls for pressure isolation valves. Subsequently, these plants had their licenses modified by NRC Order to require certain PIV testing Frequencies (excluding the "24 hour..." Frequency) be included in that plant's Technical Specifications. The content of those Orders is considered acceptable.

1 36 Not used.

37 Not used.

38 Not used.

39 Not used.

40 Not used.

ITS Section 3.4

ID 60

Subject Remove JFD associated with approved ANO generic changes ANO-1-46, 47, & 48 and annotate NUREG Markup with TSTF numbers . . ..

Not be in operation TSTF 153

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops — MODE 3

Supp.1

LCO 3.4.5 Two RCS loops shall be OPERABLE and ~~at least~~ one RCS loop shall be in operation. 3.1.1.6 TSTF-261

---NOTE---

All reactor coolant pumps (RCPs) may ~~be de-energized~~ for ≤ 8 hours per 24 hour period for the transition to or from the Decay Heat Removal System, and all RCPs may be de-energized for ≤ 1 hour per 8 hour period for any other reason, provided: DOC M48

a. No operations are permitted that would cause reduction of the RCS boron concentration; and 3.1.1.a.2

b. Core outlet temperature is maintained at least ~~X~~10~~X~~°F below saturation temperature. 1 DOC M12

Supp.1

APPLICABILITY: MODE 3. TSTF 263 3.1.1.6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ~~required~~ RCS loop inoperable.	A.1 Restore ~~required~~ RCS loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	12 hours

DOC M10
DOC M10

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Supp. 1 C. Two ~~No~~ RCS loop(s) inoperable ~~OPERABLE~~. OR Required ~~No~~ RCS loop Not in operation.	C.1 Suspend all operations involving a reduction of RCS boron concentration. AND C.2 Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately DOC A12 TSTF 263 Immediately DOC M11

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required RCS loop is in operation.	12 hours DOC M9
SR 3.4.5.2	Verify correct breaker alignment and indicated power available to the required pump that is not in operation.	7 days DOC M9

TSTF-153

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops – MODE 4

Not be in operation

Doc M13

Supp.1

LCO 3.4.6 Two loops consisting of any combination of RCS loops and decay heat removal (DHR) loops shall be OPERABLE and ~~at least~~ one loop shall be in operation.

TSTF-261

-----NOTE-----

(1) All reactor coolant pumps (RCPs) may ~~be de-energized~~ for ≤ 8 hours per 24 hour period for the transition to or from the DHR System, and all RCPs and DHR pumps may be de-energized for ≤ 1 hour per 8 hour period for any other reason, provided: Doc M13

a. No operations are permitted that would cause reduction of the RCS boron concentration; and 3.1.1.9.2

b. Core outlet temperature is maintained at least 10°F below saturation temperature. DocM13

Insert 3.4-9A

---------- DocM13

(17)

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required ~~RCS~~ loop inoperable. ~~AND~~ ~~Two DHR loops inoperable.~~	A.1	Initiate action to restore a second loop to OPERABLE status.	Immediately
	AND A.2	--- Note --- Only required if DHR loop is OPERABLE. ---------- Be in MODE 5.	24 hours

Doc M13

Supp.1

TSTF 263

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~B. One required DHR loop inoperable.~~ ~~AND~~ ~~Two required RCS loops inoperable.~~	~~B.1 Initiate action to restore a second loop to OPERABLE status.~~ ~~OR~~ ~~B.2 Be in MODE 5.~~	~~Immediately~~ ~~24 hours~~
B ~~C~~. Two ~~Required RCS or DHR~~ Required loops inoperable. OR Required ~~No RCS or DHR~~ loop Not in operation.	B ~~C~~.1 Suspend all operations involving a reduction in RCS boron concentration. AND B ~~C~~.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately Immediately

Supp.1 TSTF 263 Supp.2 Doc M13 Doc M13

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify required ~~one~~ DHR or RCS loop is in operation.	12 hours
SR 3.4.6.2	Verify correct breaker alignment and indicated power available to the required pump that is not in operation.	7 days

Supp.1 TSTF 263 Doc M13 Doc M13

Supp.1

TSTF-263

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required DHR loop inoperable. AND Any required SG with secondary side water level not within limits.	A.1 Initiate action to restore a second DHR loop to OPERABLE status. OR A.2 Initiate action to restore SG secondary side water levels to within limits.	Immediately Immediately
B. No Required DHR loops ~~inoperable~~. OR Required ~~No~~ DHR loop Not in operation.	B.1 Suspend all operations involving a reduction in RCS boron concentration. AND B.2 Initiate action to restore one DHR loop to OPERABLE status and operation.	Immediately Immediately

Doc M14

Doc M14

DocM14

Supp.1

TSTF 63

DocM14

TSTF 263

Supp.1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify ~~one~~ required DHR loop is in operation.	12 hours
SR 3.4.7.2	Verify required SG secondary side water levels are ≥ [50]%.	12 hours

DocM14

DocM14

(continued)

T

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops – MODE 5, Loops Not Filled

DOC M19

LCO 3.4.8 TSTF 153 — Two decay heat removal (DHR) loops shall be OPERABLE and one DHR loop shall be in operation.

or for testing — 49

NOTES

1. All DHR pumps may ~~be de-energized~~ not be in operation for ≤ 15 minutes when switching from one loop to another or for testing provided: Doc M19

   ~~a.~~ The maximum RCS temperature is ≤ ~~[160]~~ 140 °F; Doc M19

   b. No operations are permitted that would cause a reduction of the RCS boron concentration; and — 2 3.1.1.a.2

   c. No draining operations to further reduce the RCS water volume are permitted. — 2 DOC M19

2. One DHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other DHR loop is OPERABLE and in operation. DOC M19

Insert 3.4-14A TSTF-262

Supp. 1

APPLICABILITY: MODE 5 with RCS loops not filled. DOC M19

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required DHR loop inoperable.	A.1 Initiate action to restore DHR loop to OPERABLE status.	Immediately

DOC M19

(continued)

TSTF-263

Supp. 1

8 The Note to SR 3.4.3.1 is modified to eliminate the modifier term "inservice" from leak and hydrostatic testing. The CLB for ONS permits the use of the leak/hydrostatic test limitations when performing leak testing of connected systems required by License Condition H. Elimination of the term "inservice" clarifies this allowance is still applicable.

1 9 Not used.

10 Not used.

1 11 Not used.

1 12 Not used.

13 Not used.

1 14 Not used.

15 Not used.

16 Not used.

1 17 Not used.

18 Not used.

19 DNB limits are included in the COLR (rather than LCO 3.4.1 and the associated SRs) for each pump combination operating condition. The DNB limits and THERMAL POWER limits are currently controlled administratively. Since they are subject to change with fuel design changes, they are to be controlled in the COLR. Controlling RCS DNBR limits outside the Technical Specifications is consistent with the CLB.

20 Not used.

21 When RCS cold leg temperature is ≤ 325°F, Low Temperature Overpressure Protection (LTOP) is applicable (ITS LCO 3.4.12) and provides appropriate requirements for pressurizer level in the Bases. Since the ONS MODE 4 range is > 200 °F and < 250°F, the Note to NUREG LCO 3.4.9 which excludes requirements for the pressurizer heaters in MODE 4 is not necessary. The NUREG 3.4.9 Applicability is modified to eliminate MODE 4 and eliminate Applicability in MODE 3 when LCO 3.4.12 is Applicable. NUREG 3.4.9 RA B.2 and RA C.2 are modified to require placing the unit in a condition outside the modified Applicability.

22 The ONS design provides the capability to power any pressurizer heater from the onsite emergency sources. Therefore, consistent with the guidance in the NUREG Bases, SR 3.4.9.3 is not adopted. SR 3.4.9.2 is modified to eliminate the reference to an emergency power source. The

41 Not used.

42 Not used.

43 Not used.

1 44 Not used.

1 45 Not used.

46 NUREG Figure 3.4.11-1 is modified by shifting the limit curve down by 10 μCi/gm. This change is necessary to be consistent with the associated plant specific accident analysis which assumes a slightly reduced value for the transient value of Dose Equivalent I-131.

47 NUREG SR 3.4.10.1 is modified to eliminate the reference to "as left" lift settings being within 1%. The NUREG presentation is based on pressurizer lift setting with a 3% tolerance. The eliminated phrase is necessary for the NUREG case since the "as left" requirements for the SR are more restrictive than the actual lift setting limits. Since the ONS design provides pressurizer lift setting of 2500 psig with a 1% tolerance, the SR 3.4.10.1 "as left" setting are the same as the "as found" lift settings. Elimination of the reference to "as left" lift settings eliminates potential confusion which could result from the implications that "as left" and "as found" lift settings are different.

48 NUREG 3.4.1 is revised to reflect plant specific analytical methodology. LCO 3.4.1 and SR 3.4.1.2 are modified to specify a limit on loop average temperature instead of hot leg temperature. The ONS methodology regarding DNBR parameters utilizes a limit on loop average temperature instead of hot leg temperature.

The Note to NUREG SR 3.4.1.1 is modified to indicate that with 3 RC pumps operating, the pressure limit is applied to the loop with the highest RCS pressure. With 3 reactor coolant pumps operating, the ONS methodology regarding DNBR parameters utilizes a maximum value (high) limit on RCS pressure. Utilizing the loop with the highest pressure assures the limit is applied to the loop with the most limiting RCS pressure.

The Note to NUREG SR 3.4.1.2 is modified to indicate that with 3 reactor coolant pumps running, the temperature limit is applied to the loop with the lowest loop average temperature. ONS methodology regarding DNBR parameters applies the limit on loop average temperature to the loop with the lowest loop average temperature.

49 Note 1 to NUREG 3.4.8 is modified to include testing as reason for DHR pumps to not be in operation. Certain ONS emergency power testing is performed when LCO 3.4.8 is applicable. This testing briefly interrupts

ITS Section 3.4

ID 61

Subject Remove JFD associated with approved ONS generic change ONS-024 and annotate NUREG Markup with TSTF number 219.

Administrative controls that assure ≥10 minutes are available to mitigate the consequences of an LTOP event not implemented and PORV inoperable 27

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~I.~~ H ~~Pressurizer level > [220] inches. AND PORV inoperable.~~	I.1 ~~Restore LTOP System to OPERABLE status.~~	~~1 hour~~ TSTF 219
OR	~~OR~~	
LTOP System inoperable for any reason other than Condition A through Condition ~~H~~ G.	I.2 Depressurize RCS and establish RCS vent ~~of ≥ [0.75] square inch~~ path capable of mitigating the most limiting LTOP event. 27	12 hours DOC M29

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
~~SR 3.4.12.1 Verify a maximum of [one] makeup pump is capable of injecting into the RCS.~~ 27	~~12 hours~~
SR 3.4.12.~~2~~ 1 Verify HPI is deactivated.	12 hours DOC M30
SR 3.4.12.~~3~~ 2 Verify each CFT is ~~isolated~~ deactivated 27	12 hours DOC M30

(continued)

41 Not used.

42 Not used.

43 Not used.

1 44 Not used.

1 45 Not used.

46 NUREG Figure 3.4.11-1 is modified by shifting the limit curve down by 10 μCi/gm. This change is necessary to be consistent with the associated plant specific accident analysis which assumes a slightly reduced value for the transient value of Dose Equivalent I-131.

47 NUREG SR 3.4.10.1 is modified to eliminate the reference to "as left" lift settings being within 1%. The NUREG presentation is based on pressurizer lift setting with a 3% tolerance. The eliminated phrase is necessary for the NUREG case since the "as left" requirements for the SR are more restrictive than the actual lift setting limits. Since the ONS design provides pressurizer lift setting of 2500 psig with a 1% tolerance, the SR 3.4.10.1 "as left" setting are the same as the "as found" lift settings. Elimination of the reference to "as left" lift settings eliminates potential confusion which could result from the implications that "as left" and "as found" lift settings are different.

48 NUREG 3.4.1 is revised to reflect plant specific analytical methodology. LCO 3.4.1 and SR 3.4.1.2 are modified to specify a limit on loop average temperature instead of hot leg temperature. The ONS methodology regarding DNBR parameters utilizes a limit on loop average temperature instead of hot leg temperature.

The Note to NUREG SR 3.4.1.1 is modified to indicate that with 3 RC pumps operating, the pressure limit is applied to the loop with the highest RCS pressure. With 3 reactor coolant pumps operating, the ONS methodology regarding DNBR parameters utilizes a maximum value (high) limit on RCS pressure. Utilizing the loop with the highest pressure assures the limit is applied to the loop with the most limiting RCS pressure.

The Note to NUREG SR 3.4.1.2 is modified to indicate that with 3 reactor coolant pumps running, the temperature limit is applied to the loop with the lowest loop average temperature. ONS methodology regarding DNBR parameters applies the limit on loop average temperature to the loop with the lowest loop average temperature.

49 Note 1 to NUREG 3.4.8 is modified to include testing as reason for DHR pumps to not be in operation. Certain ONS emergency power testing is performed when LCO 3.4.8 is applicable. This testing briefly interrupts

3.5 ECCS

ITS Section 3.5

ID 20

Subject Add requirement for manual OPERABILITY of LP-9, 10, 12, 14, 15, 16, 17, 18 - Adds SRs.

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
	SR 3.5.2.6	Verify, by visual inspection, each HPI train reactor building sump suction inlet is not restricted by debris and suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion.	18 months
1 1	SR 3.5.2.7	Cycle each LPI discharge valve to the LPI-HPI flow path open manually.	18 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.3.2	-------------------NOTE-------------------- Not applicable to operating LPI pump(s). -- Vent each LPI pump casing.	31 days
SR 3.5.3.3	Verify each LPI pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.5.3.4	Verify each LPI automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.5.3.5	Verify each LPI pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.5.3.6	Verify, by visual inspection, each LPI train reactor building sump suction inlet is not restricted by debris and suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion.	18 months
SR 3.5.3.7	Cycle each LPI discharge header crossover 1 valve, LPI cooler outlet throttle valve, 1 and LPI header isolation valve open manually.	18 months

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.7

The function of the LPI discharge valve (LP-15, LP-16) to the LPI-HPI flow path is to open and allow a cross-connection from the discharge side of an LPI pump to the suction of the HPI pumps. Manually cycling each valve open demonstrates the ability to fulfill this function. This test is performed on an 18 month Frequency. Operating experience has shown that these components usually pass the Surveillance when performed at the this Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR, Section 15.14.3.3.6.
3. 10 CFR 50.36.
4. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
5. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400.

BASES

APPLICABLE SAFETY ANALYSES (continued)

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also helps ensure that reactor building temperature limits are met.

The LPI System is assumed to provide injection in the large break LOCA analysis at full power (Ref. 2). This analysis establishes a minimum required flow for the LPI pumps, as well as the minimum required response time for their actuation.

The large break LOCA event assumes a loss of offsite power and a single failure (loss of the CT-4 transformer). For analysis purposes, the loss of offsite power assumption may be conservatively inconsistent with the assumed operation of some equipment, such as reactor coolant pumps (Ref. 3). During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the reactor building. The nuclear reaction is terminated by moderator voiding during large breaks. Following depressurization, emergency cooling water is injected into the reactor vessel core flood nozzles, then flows into the downcomer, fills the lower plenum, and refloods the core.

In the event of a Core Flood line break which results in a
LOCA, with a concurrent single failure on the unaffected LPI
train opposite the Core Flood break, the LPI discharge
header crossover valves (LP-9 and LP-10) must be capable of
1 being manually opened. The LPI cooler outlet throttle
1 valves and LPI header isolation valves must be capable of
1 being manually opened to provide assurance that flow can be
1 established in a timely manner even if the capability to
1 operate them from the control room is lost. These manual
1 actions will allow cross-connection of the LPI pump
discharge to the intact LPI/Core Flood tank header to
provide abundant emergency core cooling.

The safety analyses show that an LPI train will deliver sufficient water to match decay heat boiloff rates for a large break LOCA.

In the LOCA analyses, LPI is not credited until 48 seconds after actuation of the ESPS signal. This is based on a loss

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

of offsite power and the associated time delays in Keowee Hydro Unit startup, valve opening and pump start. Further, LPI flow is not credited until RCS pressure drops below the pump's shutoff head. For a large break LOCA, HPI is not credited at all.

The LPI trains satisfy Criterion 3 of 10 CFR 50.36 (Ref. 4).

LCO

In MODES 1, 2, and 3, two independent (and redundant) LPI trains are required to ensure that at least one LPI train is available, assuming a single failure in the other train. Additionally, individual components within the LPI trains may be called upon to mitigate the consequences of other transients and accidents. Each LPI train includes the piping, instruments, pumps, valves, heat exchangers and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an ES signal and the capability to manually (remotely) transfer suction to the reactor building sump. The safety grade flow indicator associated with an LPI train is required to be OPERABLE to support LPI train OPERABILITY.

In MODE 4, one of the two LPI trains is required to ensure sufficient LPI flow is available to the core.

During an event requiring LPI injection, a flow path is required to provide an abundant supply of water from the BWST to the RCS, via the LPI pumps and their respective supply headers, to the reactor vessel. In the long term, this flow path may be switched to take its supply from the reactor building sump.

This LCO is modified by three Notes. Note 1 changes the LCO requirement when in MODE 4 for the number of OPERABLE trains from two to one. Note 2 allows an LPI train to be considered OPERABLE during alignment, when aligned or when operating for decay heat removal if capable of being manually (remotely) realigned to the LPI mode of operation. This provision is necessary because of the dual requirements of the components that comprise the LPI and decay heat removal modes of the LPI System. Note 3 requires the LPI discharge header crossover valves (LP-9 and LP-10) to be OPERABLE in MODES 1, 2, and 3.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.3.4 and SR 3.5.3.5

These SRs demonstrate that each automatic LPI valve actuates to the required position on an actual or simulated ESPS signal and that each LPI pump starts on receipt of an actual or simulated ESPS signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The test will be considered satisfactory if control board indication verifies that all components have responded to the ESPS actuation signal properly (all appropriate ESPS actuated pump breakers have opened or closed and all ESPS actuated valves have completed their travel). The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of the ESPS testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.3.6

Periodic inspections of the reactor building sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage, on the need to preserve access to the location, and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation and has been confirmed by operating experience.

SR 3.5.3.7

The function of the LPI discharge header crossover valves
(LP-9, LP-10) is to open and allow a cross-connection
1 between LPI trains. The LPI cooler outlet throttle valves
1 (LP-12, LP-14) and LPI header isolation valves (LP-17,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.7 (continued)

1 LP-18) must be capable of being manually opened to provide
1 assurance that flow can be established in a timely manner
1 even if the capability to operate them from the control room
1 is lost. Manually cycling each valve open demonstrates the
ability to fulfill this function. This test is performed on an 18 month Frequency. Operating experience has shown that these components usually pass the Surveillance when performed at the this Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50.46.

2. UFSAR, Section 15.14.3.3.6.

3. UFSAR, Section 15.14.3.3.5.

4. 10 CFR 50.36.

5. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

6. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400.

LA4

4.5.1.1.3 Core Flooding System

a. During each refueling outage, a system test shall be conducted to demonstrate proper operation of the system. During pressurization of the Reactor Coolant System, verification shall be made that the check and isolation valves in the core flooding tank discharge lines operate properly.

b. The test will be considered satisfactory if control board indication of core flood tank level verifies that all valves have opened.

4.5.1.2 Component Tests

Supp. 2

<SEE 3.5.2 + 3>

4.5.1.2.1 Valves - Power Operated

a. Valves LP-17, -18, shall only be tested every cold shutdown unless previously tested during the current quarter.

b. During each refueling outage the following LPI system valves shall be cycled manually to verify the manual operability of these power operated valves:

(1) LPI pump discharge (ES) LP-17,-18
(2) LPI discharge throttling LP-12,-14
(3) LPI discharge header crossover LP-9,-10
(4) LPI discharge to HPI/RBS LP-15,-16

<SEE 3.4>

4.5.1.2.2 Check Valves

Periodic individual leakage testing[a] of valves CF-12, CF-14, LP-47 and LP-48 shall be accomplished prior to power operation after every time the plant is placed in the cold shutdown condition for refueling, after each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed. Whenever integrity of these valves cannot be demonstrated, the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily. For the allowable leakage rates and limiting conditions for operation, see Technical Specification 3.1.6.10.

Bases

A2

The Emergency Core Cooling Systems are the principle reactor safety features in the event of loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

The High Pressure Injection System under normal operating conditions has one pump operating. The HPI system test required by Specification 4.5.1.1.1 verifies that the HPI system responds as required to actuation of ES channels 1 and 2.

<SEE 3.4>

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

<SEE 3.5.1>

Supp. 1

4.5.1.1.3 Core Flooding System

a. During each refueling outage, a system test shall be conducted to demonstrate proper operation of the system. During pressurization of the Reactor Coolant System, verification shall be made that the check and isolation valves in the core flooding tank discharge lines operate properly.

b. The test will be considered satisfactory if control board indication of core flood tank level verifies that all valves have opened.

4.5.1.2 Component Tests

<SEE 3.5.3>

4.5.1.2.1 Valves - Power Operated

A13

a. Valves LP-17, -18, shall only be tested every cold shutdown unless previously tested during the current quarter.

SR 3.5.2.7

b. During each refueling outage the following LPI system valves shall be cycled open manually to verify the manual operability of these power operated valves:

(1) LPI pump discharge (ES) LP-17,-18
(2) LPI discharge throttling LP-12,-14
(3) LPI discharge header crossover LP-9,-10
(4) LPI discharge to HPI/RBS LP-15,-16 LA6

4.5.1.2.2 Check Valves

<SEE 3.4>

Periodic individual leakage testing[a] of valves CF-12, CF-14, LP-47 and LP-48 shall be accomplished prior to power operation after every time the plant is placed in the cold shutdown condition for refueling, after each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed. Whenever integrity of these valves cannot be demonstrated, the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily. For the allowable leakage rates and limiting conditions for operation, see Technical Specification 3.1.6.10.

Bases

A2

The Emergency Core Cooling Systems are the principle reactor safety features in the event of loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

The High Pressure Injection System under normal operating conditions has one pump operating. The HPI system test required by Specification 4.5.1.1.1 verifies that the HPI system responds as required to actuation of ES channels 1 and 2.

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

<SEE 3.4>

<SEE 3.5.1>

4.5.1.1.3 Core Flooding System

a. During each refueling outage, a system test shall be conducted to demonstrate proper operation of the system. During pressurization of the Reactor Coolant System, verification shall be made that the check and isolation valves in the core flooding tank discharge lines operate properly.

b. The test will be considered satisfactory if control board indication of core flood tank level verifies that all valves have opened.

4.5.1.2 Component Tests

LA5

4.5.1.2.1 Valves - Power Operated

a. Valves LP-17, -18, shall only be tested every cold shutdown unless previously tested during the current quarter.

Supp.1 SR 3.5.3.7

b. During each refueling outage the following LPI system valves shall be cycled manually to verify the manual operability of these power operated valves:

Open A13

(1) LPI pump discharge (ES) LP-17,-18
(2) LPI discharge throttling LP-12,-14
(3) LPI discharge header crossover LP-9,-10 LA6
(4) ~~LPI discharge to HPI/RBS LP-15,-16~~ <See 3.5.2>

<SEE 3.4>

4.5.1.2.2 Check Valves

Periodic individual leakage testing[a] of valves CF-12, CF-14, LP-47 and LP-48 shall be accomplished prior to power operation after every time the plant is placed in the cold shutdown condition for refueling, after each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed. Whenever integrity of these valves cannot be demonstrated, the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily. For the allowable leakage rates and limiting conditions for operation, see Technical Specification 3.1.6.10.

Bases

A2

The Emergency Core Cooling Systems are the principle reactor safety features in the event of loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

The High Pressure Injection System under normal operating conditions has one pump operating. The HPI system test required by Specification 4.5.1.1.1 verifies that the HPI system responds as required to actuation of ES channels 1 and 2.

<SEE 3.4>

(a)
To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

1 A13 CTS 4.5.1.2.1.b requires verifying the manual OPERABILITY of LPI system
valves to provide assurance that flow can be established in a timely
manner even if the capability to operate the valves from the control
1 room is lost. ITS SR 3.5.2.7 and SR 3.5.3.7 require the LPI system
valves to be cycled open every 18 months. The CTS requirement for
manual OPERABILITY is that the valves must be capable of being manually
opened. Therefore, the ITS requirements are equivalent and the change
is administrative.

LA4 CTS 4.5.1.1.3 requires a Core Flooding System test be conducted to demonstrate proper operation of the system. This test verifies that the check and isolation valves in the CFT discharge lines operate properly. This requirement is relocated to UFSAR Chapter 16. This detail is not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for CFT OPERABILITY. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59.

1 LA5 CTS 4.5.1.2.1.a specifies that Valves LP-17, -18 shall only be tested
1 every cold shutdown unless previously tested during the current quarter.
1 These valves are boundary valves between high pressure and low pressure
1 design piping. This restriction on functional testing is intended to
1 eliminate the potential for overpressurizing the low pressure system.
This requirement is relocated to UFSAR Chapter 16. This detail is not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for LPI OPERABILITY. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59.

1 LA6 CTS 4.5.1.2.1.b provides valve numbers associated with LPI System valve
1 names. The name of the valve is retained in the Technical
1 Specifications while the detail is moved to the Bases for ITS 3.5.2 and
1 3.5.3. This information provides details of design or process which are
1 not directly pertinent to the actual requirement, i.e., Definition,
1 Limiting Condition for Operation or Surveillance Requirement, but rather
1 describe an acceptable method of compliance. Since these details are
1 not necessary to adequately describe the actual regulatory requirement,
1 they can be moved to a licensee controlled document without a
1 significant impact on safety. Placing these details in controlled
1 documents provides adequate assurance that they will be maintained. The
1 Bases will be controlled by the Bases Control Process in Chapter 5 of
1 the proposed Technical Specifications. This change is consistent with
1 the NUREG.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.7	Verify the correct settings of stops for the following HPI stop check valves: a. [MUV-2]; b. [MUV-6]; and c. [MUV-10].	[18] months
SR 3.5.2.8	Verify the flow controllers for the following LPI throttle valves operate properly: a. [DHV-110]; and b. [DHV-111].	[18] months
SR 3.5.2.~~9~~ 6	Verify, by visual inspection, each ~~ECCS~~ HPI (12) train ~~containment~~ reactor building (4) sump suction inlet is not restricted by debris and suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion.	~~[18]~~ 9 months

(14)

(7) DOC M7

Supp.1 INSERT 3.5-6A (12)

INSERT 3.5-6A

CTS

Supp. 1 SR 3.5.2.7 Cycle each LPI discharge valve to the LPI-HPI flow path open manually.	18 months	4.5.1.2.1.b(4)

INSERT 3.5-8B

CTS

Supp.1 SR 3.5.3.7	Cycle each LPI discharge header crossover valve, LPI cooler outlet throttle valve, and LPI header isolation valve open manually.	18 months 4.5.1.2.1.b(1), (2), + (3)

analysis requirements. ITS LCO 3.5.2 addresses only HPI System requirements. The ONS HPI System consists of two trains with three HPI pumps that can feed either train. At power levels > 75%, three pumps and the HPI crossover valves are required to be OPERABLE and the suction header is required to be cross-connected to meet safety analyses assumptions. When less than three pumps are OPERABLE, the HPI pump discharge headers must be hydraulically separated between trains to meet safety analyses assumptions.

ITS 3.5.2, ACTIONS A through E are added to appropriately consider unique design and analysis requirements. Proposed Condition A addresses the situation where the required HPI pump or HPI crossover valve(s) are inoperable or the suction header is not cross-connected when reactor power level is > 75% power. Required Action A.1 requires restoration in 72 hours. If not restored, ACTION B then requires a power reduction to ≤ 75% within 12 hours (Required Action B.1). Condition C addresses the situation where one HPI train is inoperable. Required Action C.2 requires restoration within 72 hours. Required Action C.1 addresses the situation where an HPI train is inoperable and not capable of being manually realigned when the power level is > 75%. C.1 requires that THERMAL POWER be reduced to ≤ 75% RTP within 3 hours. This action is necessary because one train of HPI is not capable of meeting safety analysis requirements for power levels in excess of 75% power. Condition D addresses the situation where one LPI-HPI flow path is inoperable. Required Action D.1 allows 72 hours to restore the flow path. Condition E addresses the situation where less than three HPI pumps are OPERABLE. Required Action E.1 allows 72 hours to hydraulically separate the discharge header.

NUREG SRs 3.5.2.2, 3.5.2.4,, 3.5.2.5, 3.5.2.6 and 3.5.2.9 are modified
1 to address only HPI requirements. ITS SR 3.5.2.7 is added to
1 demonstrate the ability to manually transfer HPI suction from the BWST
1 to the LPI discharge by manually cycling the LPI discharge valves to the
1 LPI-HPI flow path.

13 NUREG 3.5.2 Condition B is modified to add a second and third entry condition that is premised on the inoperability two HPI trains or two LPI-HPI flow paths. The change retains CTS 3.3.1 Condition F provisions. In this case, the safety function provided by the HPI is not capable of being satisfied and the unit is operating in a condition outside of the accident analyses. This condition would ordinarily require entry into LCO 3.0.3. However, LCO 3.0.3 does not provide an explicit Completion Time for placing the unit in MODE 3 with RCS temperature less than or equal to 350°F. By providing this explicit entry condition, Proposed LCO 3.5.2 Required Actions E.1 and E.2 direct the appropriate remedial measures and explicitly establish the Completion Times for the Required Actions.

14 NUREG SR 3.5.2.1, SR 3.5.2.7 and SR 3.5.2.8 are not included in the proposed ONS ITS since there are no comparable ONS CTS requirements.

NUREG SR 3.5.2.1 provides assurance that valves cannot change position as the result of an active failure. SR 3.5.2.1 is not included since the ONS ECCS trains contain no power operated valves that are required to remain de-energized or whose control circuits require key locking the control in the correct position. Information Notice 87-01, "RHR Valve Misalignment Causes Degradation of ECCS in PWRS" was reviewed for applicability. This information notice addresses valve misalignments during testing that could result in both trains of ECCS being rendered inoperable. ONS is not susceptible to the events described in the information notice.

NUREG SR 3.5.2.7 is not included since the HPI trains do not contain stop check valves whose purpose is to balance flow or prevent HPI pump runout. ONS uses flow orifices which are not susceptible to repositioning.

NUREG SR 3.5.2.8 is not included since ONS LPI trains do not contain automatic flow controlling throttle valves. Flow is controlled manually using installed flow indicators which are required OPERABLE by LCO 3.3.8, "Post Accident Monitoring Instrumentation."

15 NUREG LCO 3.5.2 and 3.5.3 are modified to incorporate ONS CTS requirements necessary to appropriately consider unique design and analysis requirements. ITS LCO 3.5.3 addresses only LPI System requirements. NUREG LCO 3.5.3 is modified to require two LPI trains to be OPERABLE in MODES 1, 2 and 3 and one LPI train to be OPERABLE in MODE 4 (as indicated by ITS LCO 3.5.3 Note 1).

NUREG LCO 3.5.3 Note 2, as renumbered by TSTF-090, Revision 1, is deleted since ITS 3.5.3 applies to LPI only. LCO 3.5.3 Note 1, as added by TSTF-090, Revision 1, is modified (and renumbered as ITS LCO 3.5.3 Note 2) to apply "during alignment, when aligned and when operating" to remove the confusion associated with whether the DHR/LPI train was OPERABLE during the swapover and whether the DHR/LPI pump had to be running in order to satisfy the requirement that the system be in operation. The annotation that the manual control can be accomplished either locally or remotely preserves current operational flexibility.

Appropriate SRs, similar to those adopted for the HPI System, are added for the LPI System (ITS SRs 3.5.3.1 - SR 3.5.3.6) to demonstrate compliance with LCO requirements.

16 NUREG LCO 3.5.3 Actions were altered, while retaining the original intent of the Required Actions, in order to properly reflect the corrective actions should the LCO not be met. NUREG Condition B is designated as ITS Condition A. Condition A is entered when one train of LPI is inoperable in MODES 1, 2 or 3. ITS Required Action A.1 allows 72 hours to restore the LPI train to OPERABLE status. This is consistent with the CTS 3.3.2.a(2) restoration time. The 72 hour Completion Time

is an acceptable allowance based on the fact that the redundant LPI train can still satisfy the required ECCS safety function for the specified LCO Applicability. Condition C is entered when the Required Action and associated Completion Time of Condition A are not met. ITS Required Action C.1 requires that the unit be in MODE 3 within 12 hours and MODE 4 within 60 hours. This Completion Time in conjunction with the Completion Time of ITS Required Action A.1 (72 hours) is in accordance with CTS 3.3.2(a) requirements for the restoration of operability or completion of compensatory measures for the LPI systems. Further, the combination of ITS Conditions A and C preserves the philosophy of removing the unit from the MODES or other specified conditions for Applicability.

NUREG Condition A is designated as ITS Condition E. Condition E is entered when the required LPI train is inoperable during MODE 4. ITS Required Action E.1 requires that action be immediately initiated to restore the decay heat removal (DHR) loop to an OPERABLE status. This Required Action and its associated Completion Time are premised on the recognition that an ECCS safety function has been lost. Further, this Required Action and its associated Completion Time are structured such that no requirement for a reduction in RCS temperature exists (i.e., LCO 3.0.3 is not entered). If both LPI trains are inoperable, the corrective action is to restore at least one LPI train to an OPERABLE status prior to cooling the unit down and into a MODE that requires operation of the DHR mode of the LPI System. Required Action E.2 is inserted to provide a Required Action to place the unit in MODE 5 if the DHR mode of one LPI train is available despite the inoperability of both of the LPI trains. ONS has a third LPI pump (non ES) that can be used for DHR. This Required Action is conditional based on a NOTE that directs that this action is required only if the DHR mode of one LPI train is OPERABLE. If the cause of the inoperability for both LPI trains also made the DHR mode inoperable, then no attempt to cool down the unit is required. Required Action E.2 is inserted to ensure that a cooldown to MODE 5 is initiated provided the required DHR capability exists. These changes are consistent with NUREG LCO 3.4.5 and LCO 3.4.6 Actions when a decay heat removal system is unavailable.

17 The NUREG SR 3.5.4.1 Note permitting BWST temperature to be verified only when ambient air temperature is beyond the limits is deleted. Since the upper temperature can be exceeded for reasons other than ambient air temperature (e.g., BWST recirculation), ONS considers it more appropriate to monitor BWST water temperature. In addition, at ONS the same effort is required to verify ambient air temperature as is necessary to verify BWST temperature. Therefore, the exception is not considered necessary.

18 In general, current licensing basis (CLB) permits 12 hours to place a unit in Hot Shutdown when an LCO is not met (refer to JFD 5). However, the CTS shutdown requirement for the BWST not being restored to OPERABLE status within one hour is to place the unit in Hot Shutdown (equivalent

to ITS MODE 3) within 6 hours and Cold Shutdown (equivalent to ITS MODE 5) within and additional 30 hours, when the concentrated boric acid tank is available. When the CBAST is not available, no time is allowed and CTS LCO 3.0 must be entered requiring the unit to be placed in Hot Shutdown in 12 hours and in Cold Shutdown within the following 24 hours. To maintain consistency with CTS LCO 3.0, the Completion Time for ITS 3.5.4 Required Action C.1 is changed to 12 hours.

19 NUREG SR 3.5.2.3 is modified to adopt the current method of minimizing the potential for water hammer and pump cavitation. NUREG SR 3.5.2.3 verifies that the ECCS piping is full of water. At ONS, the physical design of the HPI and LPI systems are such that the SR could not be applied to all portions of the piping because of the inability to perform venting operations to satisfy the SR due to the absence of vents, physical danger associated with the evolution or due to localized radiation levels. Therefore, the ITS requires HPI/LPI pump casings be vented every 31 days consistent with CTS Table 4.1-2, Item 10 requirements. Venting the HPI/LPI pump casings periodically reduces the potential that such voids and pockets of entrained gases can adversely affect operation of the HPI/LPI Systems.

1 20 Not used.

21 NUREG 3.5.3 is modified to incorporate ONS CTS requirements necessary to appropriately consider unique analysis requirements. In the event of a Core Flood line break (classified as a small break LOCA) concurrent with a single failure on the unaffected LPI train opposite the Core Flood line break, the LPI discharge header crossover valves must be capable of
1 being manually opened. This action, along with manually opening
1 associated LPI cooler outlet throttle valve and LPI header isolation
1 valve will allow cross-connection of the OPERABLE LPI pump discharge to the intact LPI/Core Flood tank header to provide abundant long term cooling. ITS 3.5.3 LCO Note 3 is added to explicitly require the LPI discharge header crossover valves to be OPERABLE to manually open in MODES 1, 2, and 3. ITS 3.5.3 ACTION B requires the LPI discharge header crossover valves to be restored to OPERABLE status with 72 hours of being discovered manually inoperable to open in MODE 1, 2, and 3. ITS 3.5.3 ACTION D requires LCO 3.0.3 be entered immediately when one LPI train is inoperable in MODES 1, 2, and 3 concurrent with the LPI discharge header crossover valves are inoperable to open manually in MODE 1, 2, and 3. The 72 hour Completion time of ITS 3.5.3 Required Action B.1 is based on NRC recommendations (NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975) that are based on risk evaluation and are reasonable time for repairs. This is considered appropriate since loss of crossover capability is equivalent to a loss of a single train of LPI as it relates to providing abundant long term cooling. ACTION D is added to preclude concurrent inoperability of an LPI train and the LPI discharge header crossover valves since this could result in the inability to provide abundant long term cooling to the reactor. ITS SR

1 3.5.3.7 is added to require each LPI discharge header crossover valve,
1 LPI cooler outlet throttle valve and LPI header isolation valve to be cycled open once every 18 months. This is consistent with CTS requirements for these valves and is considered appropriate based on operating experience.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.~~5~~ 4 and SR 3.5.2.~~6~~ 5 (continued) 5

actual or simulated ~~ESFAS~~ ESPS signal. This SR is not required 4 for valves that are locked, sealed, or otherwise secured in position under administrative controls. 13 INSERT B 3.5-18A The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a ~~plant~~ unit 2 outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of the ~~ESFAS~~ ESPS testing, and equipment performance is monitored as part of the Inservice Testing Program.

~~SR 3.5.2.7~~ 5

~~This Surveillance ensures that these valves are in the proper position to prevent the HPI pump from exceeding its runout limit. This 18 month Frequency is based on the same reasons as those stated for SR 3.5.2.5 and SR 3.5.2.6.~~

~~SR 3.5.2.8~~

~~This Surveillance ensures that the flow controllers for the LPI throttle valves will automatically control the LPI train flow rate in the desired range and prevent LPI pump runout as RCS pressure decreases after a LOCA. The 18 month Frequency is justified by the same reasons as those stated for SR 3.5.2.5 and SR 3.5.2.6.~~

SR 3.5.2.~~9~~ 6 5

Periodic inspections of the ~~containment~~ reactor building 4 sump suction inlet (for LPI-HPI flow path) ensure that it is unrestricted and stays in proper operating condition. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a ~~plant~~ unit 2 outage, on the need to preserve access to the location, and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation and has been confirmed by operating experience.

Supp. 1 INSERT B 3.5-18B 5

(continued)

INSERT B 3.5-18A

The test will be considered satisfactory if control board indication verifies that all components have responded to the ESPS actuation signal properly (all appropriate ESPS actuated pump breakers have opened or closed and all ESPS actuated valves have completed their travel).

INSERT B 3.5-18B

SR 3.5.2.7

Supp. 1

The function of the LPI discharge valve (LP-15, LP-16) to the LPI-HPI flow path is to open and allow a cross-connection from the discharge side of an LPI pump to the suction of the HPI pumps. Manually cycling each valve open demonstrates the ability to fulfill this function. This test is performed on an 18 month Frequency. Operating experience has shown that these components usually pass the Surveillance when performed at the this Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

INSERT B 3.5-20B

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 1), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also helps ensure that reactor building temperature limits are met.

The LPI System is assumed to provide injection in the large break LOCA analysis at full power (Ref. 2). This analysis establishes a minimum required flow for the LPI pumps, as well as the minimum required response time for their actuation.

The large break LOCA event assumes a loss of offsite power and a single failure (loss of the CT-4 transformer). For analysis purposes, the loss of offsite power assumption may be conservatively inconsistent with the assumed operation of some equipment, such as reactor coolant pumps (Ref. 3). During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the reactor building. The nuclear reaction is terminated by moderator voiding during large breaks. Following depressurization, emergency cooling water is injected into the reactor vessel core flood nozzles, then flows into the downcomer, fills the lower plenum, and refloods the core.

Supp. 1

In the event of a Core Flood line break which results in a LOCA, with a concurrent single failure on the unaffected LPI train opposite the Core Flood break, the LPI discharge header crossover valves (LP-9 and LP-10) must be capable of being manually opened. The LPI cooler outlet throttle valves and LPI header isolation valves must be capable of being manually opened to provide assurance that flow can be established in a timely manner even if the capability to operate them from the control room is lost. These manual actions will allow cross-connection of the LPI pump discharge to the intact LPI/Core Flood tank header to provide abundant emergency core cooling.

INSERT B 3.5-20B (continued)

The safety analyses show that an LPI train will deliver sufficient water to match decay heat boiloff rates for a large break LOCA.

In the LOCA analyses, LPI is not credited until 48 seconds after actuation of the ESPS signal. This is based on a loss of offsite power and the associated time delays in Keowee Hydro Unit startup, valve opening and pump start. Further, LPI flow is not credited until RCS pressure drops below the pump's shutoff head. For a large break LOCA, HPI is not credited at all.

The LPI trains satisfy Criterion 3 of 10 CFR 50.36 (Ref. 4).

1

INSERT B 3.5-20C

In MODES 1, 2, and 3, two independent (and redundant) LPI trains are required to ensure that at least one LPI train is available, assuming a single failure in the other train. Additionally, individual components within the LPI trains may be called upon to mitigate the consequences of other transients and accidents. Each LPI train includes the piping, instruments, pumps, valves, heat exchangers and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an ES signal and the capability to manually (remotely) transfer suction to the reactor building sump. The safety grade flow indicator associated with an LPI train is required to be OPERABLE to support LPI train OPERABILITY. In MODE 4, one of the two LPI trains is required to ensure sufficient LPI flow is available to the core.

During an event requiring LPI injection, a flow path is required to provide an abundant supply of water from the BWST to the RCS, via the LPI pumps and their respective supply headers, to the reactor vessel. In the long term, this flow path may be switched to take its supply from the reactor building sump.

This LCO is modified by two Notes. Note 1 changes the LCO requirement when in MODE 4 for the number of OPERABLE trains from two to one. Note 2 allows an LPI train to be considered OPERABLE during alignment, when aligned or when operating for decay heat removal if capable of being manually (remotely) realigned to the LPI mode of operation. This provision is necessary because of the dual requirements of the components that comprise the LPI and decay heat removal modes of the LPI System.

The flow path for each train must maintain its designed independence to ensure that no single failure can disable both LPI trains.

INSERT B 3.5-23A (page 3 of 3)

SR 3.5.3.7

Supp. 1

The function of the LPI discharge header crossover valves (LP-9, LP-10) is to open and allow a cross-connection between LPI trains. The LPI cooler outlet throttle valves (LP-12, LP-14) and LPI header isolation valves (LP-17, LP-18) must be capable of being manually opened to provide assurance that flow can be established in a timely manner even if the capability to operate them from the control room is lost. Manually cycling each valve open demonstrates the ability to fulfill this function. This test is performed on an 18 month Frequency. Operating experience has shown that these components usually pass the Surveillance when performed at the this Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

ITS Section 3.5

ID 34

Subject Revise CTS Markup for ITS 3.5.3, page 4 of 6 to correctly refer to ITS SR 3.5.3.2 instead of "3.5.2.3."

Supp.1

SR 3.5.3.2 Note

(3) Operating pumps excluded.

<SEE 3.4+3.7>

(4) Number of safety valves to be tested each refueling shall be in accordance with ASME Codes Section XI, Article IWV-3511, such that each valve is tested at least once every 5 years.

(5) Applicable only to the interlocks associated with the Reactor Building Purge System. <SEE 3.3>

(6) Verification of the Emergency Condenser Circulating Water (ECCW) System function to supply siphon suction to the Low Pressure Service Water System shall be performed to ensure operability of the LPSW System.

<SEE 3.7>

ITS Section 3.5

ID 62

Subject Remove JFD associated with approved ONS generic change ONS-004 and annotate NUREG Markup with TSTF number ...

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
~~SR 3.5.2.1~~	~~Verify the following valves are in the listed position with power to the valve operator removed.~~ ~~Valve Number / Position / Function~~ ~~[ ] [ ] [ ]~~ ~~[ ] [ ] [ ]~~ ~~. . .~~ ~~[ ] [ ] [ ]~~	~~12 hours~~	(14)
Supp. 1 TSTF-260 SR 3.5.2.~~2~~ 1	Verify each ~~ECCS~~ HPI (12) manual, power operated, ~~or~~ ~~and automatic~~ non-automatic TSTF-260 valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days	DOC M6
~~SR 3.5.2.~~~~3~~ 2	~~Verify ECCS piping is full of water.~~	31 days	(19)
SR 3.5.2.~~4~~ 3	Verify each ~~ECCS~~ HPI (12) pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program	DOC A8
SR 3.5.2.~~5~~ 4	Verify each ~~ECCS~~ HPI (12) automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[~~18~~]~~ months	(1) 4.5.1.1.1.a
SR 3.5.2.~~6~~ 5	Verify each ~~ECCS~~ HPI (12) pump starts automatically on an actual or simulated actuation signal.	~~[~~18~~]~~ months	(1) 4.5.1.1.1.a

(continued)

INSERT 3.5-5A (19)

to ITS MODE 3) within 6 hours and Cold Shutdown (equivalent to ITS MODE 5) within and additional 30 hours, when the concentrated boric acid tank is available. When the CBAST is not available, no time is allowed and CTS LCO 3.0 must be entered requiring the unit to be placed in Hot Shutdown in 12 hours and in Cold Shutdown within the following 24 hours. To maintain consistency with CTS LCO 3.0, the Completion Time for ITS 3.5.4 Required Action C.1 is changed to 12 hours.

19 NUREG SR 3.5.2.3 is modified to adopt the current method of minimizing the potential for water hammer and pump cavitation. NUREG SR 3.5.2.3 verifies that the ECCS piping is full of water. At ONS, the physical design of the HPI and LPI systems are such that the SR could not be applied to all portions of the piping because of the inability to perform venting operations to satisfy the SR due to the absence of vents, physical danger associated with the evolution or due to localized radiation levels. Therefore, the ITS requires HPI/LPI pump casings be vented every 31 days consistent with CTS Table 4.1-2, Item 10 requirements. Venting the HPI/LPI pump casings periodically reduces the potential that such voids and pockets of entrained gases can adversely affect operation of the HPI/LPI Systems.

1 20 Not used.

21 NUREG 3.5.3 is modified to incorporate ONS CTS requirements necessary to appropriately consider unique analysis requirements. In the event of a Core Flood line break (classified as a small break LOCA) concurrent with a single failure on the unaffected LPI train opposite the Core Flood line break, the LPI discharge header crossover valves must be capable of
1 being manually opened. This action, along with manually opening
1 associated LPI cooler outlet throttle valve and LPI header isolation
1 valve will allow cross-connection of the OPERABLE LPI pump discharge to
the intact LPI/Core Flood tank header to provide abundant long term cooling. ITS 3.5.3 LCO Note 3 is added to explicitly require the LPI discharge header crossover valves to be OPERABLE to manually open in MODES 1, 2, and 3. ITS 3.5.3 ACTION B requires the LPI discharge header crossover valves to be restored to OPERABLE status with 72 hours of being discovered manually inoperable to open in MODE 1, 2, and 3. ITS 3.5.3 ACTION D requires LCO 3.0.3 be entered immediately when one LPI train is inoperable in MODES 1, 2, and 3 concurrent with the LPI discharge header crossover valves are inoperable to open manually in MODE 1, 2, and 3. The 72 hour Completion time of ITS 3.5.3 Required Action B.1 is based on NRC recommendations (NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975) that are based on risk evaluation and are reasonable time for repairs. This is considered appropriate since loss of crossover capability is equivalent to a loss of a single train of LPI as it relates to providing abundant long term cooling. ACTION D is added to preclude concurrent inoperability of an LPI train and the LPI discharge header crossover valves since this could result in the inability to provide abundant long term cooling to the reactor. ITS SR

ITS Section 3.6

ID 64

Subject Remove JFD associated with approved ONS generic change ONS-004 and annotate NUREG Markup with TSTF number ...

(4) reactor building

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
E ~~D~~. One ~~Two~~ ~~[required]~~ ~~containment~~ cooling trains inoperable ~~s~~ in MODE 3 or 4 (12)	E ~~D~~.1 Restore ~~one~~ ~~[required]~~ ~~containment~~ cooling train to OPERABLE status.	24 ~~72~~ hours (12)	3.3.5.a(2)
(12) INSERT 3.6-17A			
G ~~F~~. Required Action and associated Completion Time of Condition E ~~C~~ or F ~~D~~ not met.	~~E.1 Be in MODE 3.~~ ~~AND~~	~~6 hours~~ (12)	3.3.5.a(2) 3.3.6.a(2)(b)
	G.1 ~~E.2~~ Be in MODE 5.	36 hours	
H ~~F~~. Two ~~containment~~ reactor building spray trains inoperable in MODE 1 or 2. OR Any combination of three or more trains inoperable.	H ~~F~~.1 Enter LCO 3.0.3.	Immediately	DOC A16

(4)

INSERT 3.6-17B (12)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY	
SR 3.6.5 ~~6~~.1 (3) Verify each ~~containment~~ reactor building spray manual and non-automatic power operated ~~and automatic~~ valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days	DOC M18

TSTF-260

Supp. 1 INSERT 3.6-17C (12)

(continued)

TECHNICAL SPECIFICATIONS

NOTE: The first four justifications for these changes from NUREG-1430 were generically used throughout the individual LCO section markups. Not all generic justifications are used in each section.

1 The brackets are removed and the proper plant specific information or value is provided.

2 Editorial changes made for clarity, preference or consistency with the Improved Technical Specifications (ITS) Writer's Guide.

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

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

5 TSTF-052 was used as a model for making proposed changes to the NUREG. CTS was modified by Amendment Numbers 218, 218, and 215 for Units 1, 2, and 3, respectively, to specify leakage rate be determined for Type A testing using 10 CFR 50, Appendix J, Option B and for Type B and C testing suing 10 CFR 50, Appendix J, Option A. Added new ITS SR 3.6.1.1 to adopt 10 CFR 50, Appendix J, Option B for containment visual examinations and Type A leakage rate testing. Modified NUREG SR 3.6.1.1 (ITS SR 3.6.1.2) to limit its applicability to Type B and C leakage rate testing and to eliminate reference to visual examinations. This is necessary since ONS adopted Appendix J, Option B for Type A testing only. Visual examinations are associated with Type A testing.

6 SR 3.6.2.1 is modified to add a reference to Option A to reflect the current licensing basis for containment airlock testing. The specific airlock leakage acceptance criteria are not adopted because no requirement currently exists. The air lock's contribution to Type B and C containment leakage is limited such that the Type B and C containment leakage cannot exceed the applicable Type B and C containment leakage limits.

7 TSTF-030, Revision 2, which extends the Completion Time for a closed system with an inoperable isolation valve to 72 hours, is not incorporated into the ONS ITS. This extension is based on the closed system meeting the requirements of Standard Review Plan (SRP) 6.2.4. ONS closed systems do not meet these SRP requirements.

1 8 Not used.

9 The current licensing basis (CLB) permits 12 hours to place a unit in Hot Shutdown when an LCO is not met. To maintain consistency with

current procedures, training and staffing requirements, the 12 hours permitted to place a unit in MODE 3 is retained in the ITS.

10 NUREG SR 3.6.3.8 is not included in the proposed Oconee ITS since the CTS 3.6.3.b requires the reactor building (RB) purge valves to be closed and not operated whenever containment integrity is required (that is, above cold shutdown). Therefore the bracketed SR does not apply to ONS.

11 NUREG SR 3.6.3.2 is not included in the proposed Oconee ITS since Oconee does not have minipurge valves.

12 NUREG 3.6.6 is modified to retain the CTS 3.3.5 and 3.3.6 requirements for two trains of reactor building (RB) spray and three trains of RB cooling OPERABLE during MODES 1 and 2 and one train of RB spray and two trains of RB cooling OPERABLE during MODES 3 and 4. This is accomplished by the addition of an LCO Note and revision of the Conditions and Required Actions. During MODES 3 and 4, the potential energy of the high energy systems inside the reactor building is significantly less than during MODES 1 and 2. The result of this circumstance is that the impact of a loss of coolant accident (LOCA), main steam line break (MSLB), or main feedwater line break (MFLB) during MODES 3 or 4 is significantly less than during MODES 1 or 2 and the need for accident mitigation allows a reduced capacity of RB cooling. Requiring the availability of only one train of RB spray and two trains of RB cooling during MODES 3 and 4 is considered adequate given the reasons above and the limited time spent in these MODES.

NUREG SRs 3.6.6.4, 3.6.6.5, and 3.6.6.6 are modified to incorporate the requirements for required trains of reactor building spray during MODES 3 and 4.

13 NUREG SRs 3.6.6.3 is replaced by ITS SR 3.6.5.4, which requires the containment heat removal capability to be verified on a refueling outage frequency. At ONS, a specific surveillance of containment heat removal capability was added (Amendment Nos. 203, 203, and 200 for Oconee Units 1, 2, and 3 by letter dated January 13, 1994) to verify required systems are capable of performing their intended safety function of maintaining containment pressure and temperature below design limits following an accident. This amendment removed the LPSW flow and RBCU fan air flow requirements from which containment heat removal capability was inferred.

14 NUREG Specification 3.6.7, Spray Additive System, is deleted since such a system is not included in the design of Oconee Units 1, 2, and 3.

15 NUREG 3.6.8 is not included in the Oconee ITS since Oconee does not have permanently installed hydrogen recombiners. The Oconee Containment Hydrogen Recombiner System includes a portable hydrogen recombiner which will be moved to the affected unit following a LOCA, anchored to its foundation, and connected to piping penetrations. Also included is a

portable control panel, which will be locally mounted near the recombiner, anchored to its foundation and connected to its motor control center and the recombiner. Controls for the Hydrogen Recombiner System are relocated to UFSAR Chapter 16, Selected License Commitments. Refer to DOC LA3 in Attachment 3 for this section.

16 NUREG 3.6.5, Containment Air Temperature, is not included in the ITS. CTS does not specify a containment air temperature limit. ONS uses administrative controls to ensure maximum containment air temperatures are maintained below equipment qualification (EQ) temperatures and the initial temperature assumed in the design basis accident (DBA) analysis. A Surveillance is performed every shift to verify the average containment air temperature does not exceed the specified temperature limit. Any changes to the procedurally controlled temperature limit must be evaluated pursuant to 10 CFR 50.59. ONS has demonstrated, based on past performance and operating experience, that this is an effective method of controlling containment air temperature. Containment air temperature historical data indicates that the actual containment bulk air temperature is maintained at a conservative value with respect to the design basis bulk air temperature. As such, ONS prefers to continue controlling containment air temperature administratively.

17 The first Frequency of NUREG SR 3.6.6.8, which requires performance at the first refueling, is deleted since the first refueling of Oconee Units 1, 2, and 3 has already occurred.

18 The CTS requirements for reactor building (RB) purge valves will be maintained in the ITS by deleting the purge valve references from NUREG 3.6.3 Condition A and Condition B and by deleting NUREG 3.6.3 Condition D in its entirety. Since NUREG 3.6.3 Condition D was deleted, Condition E was renumbered to Condition D. In addition, NUREG SR 3.6.3.6, which requires leakage rate testing for purge valves with resilient seals, is not included in the proposed ITS. Since leakage rate testing is no longer included in Specification 3.6.3, ACTIONS Note 4 is not necessary and is deleted.

CTS 3.6.3.b requires that the reactor building (RB) purge valves be closed and not operated whenever RB integrity is required (that is, above cold shutdown) except during purge operation when RCS temperature is < 250°F and RCS pressure is < 350 psig or during testing or maintenance on one valve when in MODE 3 or 4. If these valves were to be stroked after they had been verified to meet the RB integrity requirement, then the Local Leak Rate Test (LLRT) performed to validate the purge valve leakage would be invalid. As a result of this requirement, for the duration of the operating cycle, the RB purge valves are not required to be leak tested above cold shutdown. The CTS requirements for the ONS RB purge isolation valves require closing these valves when RB integrity is required. Furthermore, operation of the

purge valves when RB integrity is required is specifically prohibited by CTS 3.6.3.b.3.

NUREG 3.6.3, in Conditions A, B, and D, addresses purge valves uniquely with respect to other RB isolation valves assuming that these valves are leak tested during operation with an acceptance criteria specific to the purge valves. NUREG SR 3.6.3.6 requires purge valve leakage rate testing every 184 days and within 92 days after opening the valve. While CTS 4.4.4.1 requires leakage rate testing be performed on these valves more frequently than other containment isolation valves, there are no purge valve leakage rate acceptance criteria and no actions required when the purge valves exceed their acceptance criteria (listed in plant procedures) provided that the combined Type B and C criteria continue to be met. The leakage from these valves is included in the combined Type B and C leakage test. Therefore, a separate leakage rate testing requirement is not included in the proposed ITS.

ITS Section 3.7

ID 8

Subject DOC M13 is missing and DOC A13 is described as M change. Changed A13 to M13. DOC A13 shown as not used. Revised CTS Markup to show M13 instead of A13.

Specification 3.7.10

A1 <except as marked>

3.15 Control Room Pressurization and Filtering System and Penetration Room Ventilation System

Applicability

Applies to the Unit 1 and 2, and Unit 3 control room pressurization and filtering systems and the penetration room ventilation system.

Objective

To define the conditions necessary to assure operability of the control room pressurization and filtering system and the immediate availability of the penetration room ventilation systems.

Specification

M21

MODES 1, 2, 3 + 4

3.15.1 Penetration Room Ventilation Systems

LCO Applicability RA B.1

a. Two trains of the penetration room ventilation systems shall be operable at all times ~~when containment integrity is required~~ or the reactor shall be shutdown within 12 hours with the following exception:

Action A

(1) If one of two trains of a penetration room ventilation system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days provided that all active components of the other train of the penetration room ventilation system shall be demonstrated to be operable within 24 hours and daily thereafter.

L1

3.15.2 Control Room Pressurization and Filtering Systems

a. With the reactor above hot shutdown conditions both outside air booster fans shall be operable.

(1) If one outside air booster fan is inoperable, restore the inoperable fan to operable status within 72 hours, or the unit shall be in hot shutdown within the next 12 hours.

(2) If both outside air booster fans are inoperable, restore at least one inoperable fan to operable status within 24 hours or the unit shall be in hot shutdown within the next 12 hours.

<See 3.7.9>

Add RA B.2

M13

Supp 1

1 of 3

are not separately retained. As a result, this change is a presentation preference and is administrative in nature.

A12 CTS 4.12.2 provides details regarding testing the testing the CRVS filters. These details are relocated to the Ventilation Filter Testing Program (VFTP) specified in ITS 5.5.12. ITS SR 3.7.9.2 requires performance of the testing in accordance with the VFTP. Consequently, this is an administrative change and is consistent with the NUREG.

1 A13 Not used.

A14 CTS 4.5.4.1.b.2, 4.5.4.1.c, 4.5.4.1.d and 4.5.4.1.e provides details regarding testing the testing the PRVS filters. These details are relocated to the Ventilation Filter Testing Program (VFTP) specified in ITS 5.5.12 (see DOC LA 21). ITS SR 3.7.10.2 is added to require performance of the testing in accordance with the VFTP. Consequently, this is an administrative change and is consistent with the NUREG.

A15 CTS 4.5.4.1.e requires the PRVS system be declared inoperable if the laboratory analysis of the carbon does not meet acceptance criteria. This requirement is not separately retained in the ITS. ITS SR 3.0.1 states that failure to meet SR requirements constitutes failure to meet the LCO. If ITS SR 3.7.10.2 is not met, LCO 3.7.10 is not met. Failure to meet the LCO is equivalent to declaring the system inoperable. Consequently, this is an administrative change and is consistent with the NUREG.

A16 A CTS provision comparable to the Note to ITS 3.7.12 RA A.1 does not exist. This Note states that the provisions of LCO 3.0.3 does not apply. Although no explicit comparable CTS provision exists, it is unreasonable to apply CTS 3.0 since a shutdown of the applicable unit(s) provides no cure for the associated condition involving the boron concentration in the spent fuel pool. Additionally, no realistic scenario is envisioned which would preclude promptly completing the CTS specified actions, and therefore potentially requiring entry into CTS 3.0. Consequently, this change is considered administrative and is consistent with the NUREG.

A17 Not used.

A18 A CTS provision comparable to the Note to ITS SR 3.7.2.1 and the Note to SR 3.7.2.2 does not exist. These Notes state that the SRs are only required to be performed in MODES 1 and 2. The Notes permit the valves to be considered OPERABLE in MODE 3 without performance of SR 3.7.2.1 or SR 3.7.2.2. CTS does not require performance of surveillances prior to entering the mode of applicability. Therefore, the ITS provision to not require testing the valves when in MODE 3 is consistent with the CTS, administrative and is consistent with the NUREG.

A19 Not used.

A20 A CTS provision comparable to the ITS Actions Note does not exist. This Action Note permits separate Condition Entry for each valve. CTS does not prohibit separate condition entry for each valve. Therefore this is an administrative change and is consistent with the NUREG.

A21 Not used.

A22 A CTS provision comparable to the Note to ITS 3.7.12 RA A.1 does not exist. This Note states that the provisions of LCO 3.0.3 do not apply. Although no explicit comparable CTS provision exists, it is unreasonable to apply CTS 3.0 since a shutdown of the applicable unit(s) provides no cure for the associated condition involving the boron concentration in the spent fuel pool. Additionally, no realistic scenario is envisioned which would preclude promptly completing the CTS specified actions, and therefore potentially requiring entry into CTS 3.0. Consequently, this change is considered administrative and is consistent with the NUREG.

A23 CTS 3.4.5 requires a specified volume of water be available in the UST, CST and HW. ITS 3.7.6 requires the UST, CST and HW be OPERABLE with OPERABILITY specified as containing the specified volume of water. ITS SR 3.7.6.1 specifies the required quantity of water consistent with the CTS 3.4.5 volume requirements. Therefore this change is an administrative change and is consistent with the NUREG.

A24 CTS Table 4.1-2, item number 3 requires testing main steam safety valves each refueling. The frequency of each refueling is modified by Note (4) which states that the number of safety valves to be tested each refueling shall be in accordance with ASME Codes Section XI, Article IWV-3511, such that each valve is tested at least once every 5 years. ITS SR 3.7.1.1 requires verification of each required lift setpoints in accordance with the Inservice Testing Program at a frequency which is in accordance with the Inservice Testing Program. The Inservice Test Program requires setpoint testing in accordance with ASME Section XI testing requirements. Therefore this is an administrative change.

A25 CTS 4.8 requires testing the main steam stop valves. ITS SR 3.7.2.1 and SR 3.7.2.2 requires testing the turbine stop valves. The change in name is a presentation preference only since the components referred to are the same. This is an administrative change associated with a plant preference regarding equipment terminology.

A26 Not used

A27 CTS 3.5.7.2 requires the Main Feedwater control valve to be OPERABLE. ITS LCO 3.7.3 requires two MFCVs and two SFCVs to be OPERABLE. The paragraph under the applicability section in CTS 3.5.7 defines the main feedwater controls valves as both the Main Feedwater System main control valves and Main Feedwater startup control valves. The ONS design provides two Main Feedwater System main control valves and two Main Feedwater startup control valves. This is an administrative requirement and is consistent with the NUREG.

A28 CTS 3.5.7.2.1.a permits a Main Feedwater control valve in one or more flow paths to be inoperable provided it is closed within 8 hours and verified to be once per 7 days. ITS Action A permits a MFCV to be inoperable provided it is closed or isolated within 8 hours and verified to be closed once per 7 days. ITS Action B permits a SFCV to be inoperable provided it is closed or isolated within 8 hours and verified to be once per 7 days. Since the MFCV and SFCV for each steam generator are in a different flow path, the CTS permits both a MFCV and a SFCV to be simultaneously inoperable for each steam generator. Therefore, this is an administrative change and is consistent with the NUREG.

A29 CTS 3.4.7 requires the main steam atmospheric dump valve flow path on each steam generator to be OPERABLE. ITS LCO 3.7.4 requires one ADV line per steam generator to be OPERABLE. The change from ADV flow path to ADV line is a presentation preference only. Therefore, this is an administrative requirement and is consistent with the NUREG.

A30 CTS 4.9.1 requires testing the EFW pumps in accordance with the requirements of CTS 4.0.4. CTS 4.0.4 states that Inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50 Section 50.55a(g)(4) to the extent practicable within the limitations of design, geometry and materials of construction of the components. ITS SR 3.7.5.2 requires verifying the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head. The SR 3.7.5.2 requirement to verify the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head is encompassed within the testing requirement required by ASME Section XI, subsection IWP. This change is an administrative change and is consistent with the NUREG.

A31 CTS 3.3.7.b states that tests or maintenance shall be allowed on any component of the LPSW system provided the redundant train of the LPSW system is operable. The entry condition for ITS Required Action A.1 is a required LPSW pump inoperable. With a required LPSW pump inoperable the redundant pump(s) and required flow path ensures the required safety function can be accomplished (albeit without an additional single failure). Although multiple flow paths exist from the discharge of the LPSW pumps to each required supplied component, some portions of the multiple flow paths are not totally redundant or independent. Consequently, the LPSW pumps and flow paths are not separated into trains.

A32 CTS Table 4.1-3, Note "*" clarifies that the SR Frequency is not applicable if reactor is in a cold shutdown condition for a period exceeding the sampling frequency. The provision of this Note regarding being in the cold shutdown condition for a period greater than the sampling frequency is deleted since it is unnecessary. Since requirements of Specification 3.7.14 are not Applicable in MODE 5

(equivalent to CTS Cold Shutdown condition) the LCO and associated SRs are not applicable.

A33 CTS 3.8.13.a, 3.8.13.b, 3.8.13.c and 3.8.13.d specify requirements associated with moving casks in the SFP. The CTS presentation for these requirements states "Prior to spent fuel cask movement . . . ." or "Prior to dry storage transfer cask movement . . . ." ITS 3.7.15 is Applicable during movement of either the spent fuel shipping cask or the dry storage transfer cask in the SFP area. Although the CTS presentation can be interpreted to apply at any time prior to cask movement in the SFP, the extreme extension of such an interpretation makes the requirements applicable at all times. This is clearly not the intent since the purpose of these requirements is to ensure the spent fuel has decayed sufficiently at the time of cask movement in the SFP. This is considered a change in presentation only. No change in interpretation or application is intended. Therefore, this change is an administrative change.

A34 An explicit CTS requirement comparable to ITS 3.7.15 Action A does not exist. With requirements of the LCO not met, ITS 3.7.15 Action A requires immediately suspending movement of cask in the SFP area. Since CTS does not permit cask movement under this condition, CTS requires, although not explicitly, the same action. Therefore this change is considered administrative.

A35 An explicit CTS requirement comparable to ITS SR 3.7.15.1 does not exist. ITS SR 3.7.15.1 requires verification by administrative means that the decay time for fuel assemblies is within limit. The Frequency for this SR is "Prior to cask movement in the spent fuel." Although an explicit requirement comparable to ITS SR 3.7.15.1 does not exist, the very existence of CTS 3.7.15.1 imposes a requirement for at least an administrative verification of the decay times in the specified rows at some time prior to movement of the cask in the spent fuel area. This change is therefore considered administrative.

CTS 3.19.1.a and 3.19.2.a requires ECCW siphon headers to be OPERABLE whenever the associated LPSW Systems are required to be OPERABLE. ITS 3.7.8 requires the ECCW siphon headers to be OPERABLE in MODES 1, 2, 3 and 4. As such, ITS 3.7.8 is slightly more restrictive in that operability is required when temperature is > 200°F. In MODES 1, 2, 3, and 4, the ECCW siphon headers are a normally operating system that is required to support the OPERABILITY of the LPSW System. Therefore, the ECCW siphon headers are required to be OPERABLE in these MODES.

M10 A CTS surveillance comparable to ITS SR 3.7.7.1 does not exist. This SR requires verification that valves in the LPSW flow path are in the correct position. Therefore this additional requirement is a more restrictive requirement upon unit operation and is consistent with the NUREG. Verifying the correct alignment for manual, and non-automatic power operated valves in the LPSW System flow path provides assurance that the proper flow paths exist for LPSW System operation.

M11 CTS 4.5.1.1.2.a.2 requires verification of the engineered safety features function of the LPSW water pumps. ITS SR 3.7.7.3 requires verification that each LPSW pump starts automatically on an actual or simulated actuation signal. The ITS requirements are more prescriptive and are therefore considered to be a more restrictive requirement upon unit operation. This change is consistent with the NUREG. The SR verifies proper automatic operation of the LPSW System pumps on an actual or simulated actuation signal.

M12 The footnote to CTS 4.5.1.1.2.a.2 specifies testing the Engineered Safeguard (ES) function of specified valves which provide coolant to the decay heat coolers. CTS 4.5.1.1.a.2 requires verification of the engineered safety feature function of the LPSW system which supplies coolant to the reactor building coolers. Each of these surveillance requirements is encompassed within ITS SR 3.7.7.2 which requires a verification that each LPSW system automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal. SR 3.7.7.2 verifies proper automatic operation of the LPSW System valves. The ITS requirements are more prescriptive and are therefore considered to be a more restrictive requirement upon unit operation. This change is consistent with the NUREG.

1 M13 A CTS requirement specific to the Penetration Room Ventilation System
1 (PRVS) specification and comparable to ITS 3.7.10 RA B.2 does not exist.
1 With one train of the PRVS inoperable, CTS 3.15.1.a requires the reactor
1 be shutdown within 12 hours but does not explicitly require the unit be
1 placed in a condition outside the applicability of the specification.
1 ITS 3.7.10 RA B.2 requires the unit be placed in MODE 5 within 36 hours.
1 Therefore, this is a more restrictive requirement upon unit operation
1 and is consistent with the NUREG. This is appropriate since in MODES 1,
1 2, 3, and 4, the PRVS is required to be OPERABLE consistent with the

1 OPERABILITY requirements of the containment. Placing the unit in MODE 5
1 places the unit in a MODE where the PRVS is not required to be OPERABLE.

M14 CTS 4.4.5.1.f requires verification the PRVS can maintain a negative pressure with respect to outside atmosphere. ITS SR 3.7.10.4 requires verification the PRVS can maintain at least 0.06 inches of water negative pressure with respect to atmospheric pressure during operation at a flow rate ≥ 900 CFM and ≤ 1100 CFM. The establishment of a minimum negative pressure requirement and minimum and maximum flow rates are more prescriptive requirements and are therefore more restrictive requirements upon unit operation. These additional requirement are consistent with the NUREG. The minimum value for the negative pressure provides reasonable assurance the PRVS can perform its safety function. The minimum and maximum flow limits ensure the PRVS is operating within a nominal range of its design flow.

M15 CTS Table 4.1-3, Item 4 requires sampling the spent fuel pool boron concentration monthly and after each makeup. No specific limit is placed on the time period to perform the surveillance after the makeup to the spent fuel pool. ITS SR 3.7.12.1 requires sampling the spent fuel pool within 12 hours after completing the makeup. Therefore this additional requirement is a more restrictive requirement upon unit operation. Sampling within 12 hours after the makeup provides reasonable assurance that boron concentration in the spent fuel pool is maintained consistent with the assumptions of the analysis.

1 M16 A CTS surveillance requirement comparable to ITS SR 3.7.5.5 does not
1 exist. This SR ensures that the EFW System is properly aligned by
1 verifying the flow paths to each steam generator prior to entering
1 MODE 2 after more than 30 days in MODE 5 or 6. OPERABILITY of EFW flow
1 paths must be demonstrated before sufficient core heat is generated that
1 would require the operation of the EFW System during a subsequent
1 shutdown. To further ensure EFW System alignment, flow path OPERABILITY
1 is verified, following extended outages to determine no misalignment of
1 valves has occurred. This SR is appropriate and is an acceptable
1 restriction upon operation since it ensures that the flow path from the
1 CST to the steam generator is properly aligned. This change is
1 consistent with the NUREG.

M17 Not used.

M18 If the control valves (MFCVs and SFCVs) are not restored to OPERABLE status within 72 hours CTS 3.5.7.2.1.b requires the unit be placed in Hot Shutdown within 12 hours and with RCS temperature less than 250°F within an additional 18 hours (total of 30 hours). ITS 3.7.3 RA C.1 requires the unit be placed in MODE 3 within 12 hours and MODE 4 within 18 hours. Requiring the unit be placed in MODE 4 within 18 hours is an additional restrictions upon unit operation. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

PRVS 4

~~EVS~~ CTS

3.7.~~12~~ 10

3.7 PLANT SYSTEMS

3.7.~~12~~ 10 ~~Emergency~~ Penetration Room Ventilation System ~~(EVS)~~ (PRVS) 4

LCO 3.7.~~12~~ 10 Two ~~EVS~~ PRVS trains shall be OPERABLE. 3.15.1.a

APPLICABILITY: MODES 1, 2, 3, and 4. 3.15.1.a

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One ~~EVS~~ PRVS train inoperable.	A.1 Restore ~~EVS~~ PRVS train to OPERABLE status.	7 days	4 3.15.1.a.1
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	~~6~~ 12 hours 22 36 hours	3.15.1.a Doc M13

Supp. 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.7.~~12~~ 10.1	Operate each ~~EVS~~ PRVS 4 train for ~~[≥ 10 continuous hours with the heaters operating or (for systems without heaters)]~~ ≥ 15 minutes~~]~~.	31 days 1	4.5.4.1.a
SR 3.7.~~12~~ 10.2	Perform required ~~EVS~~ PRVS 4 filter testing in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~.	In accordance with the ~~[~~VFTP~~]~~ 1	4.5.4.1.c 4.5.4.1.d 4.5.4.1.e

(continued)

ITS Section 3.7

ID 9

Subject Revise DOC LA8 to indicate relocation of marked CTS 4.12.1.a information to Bases for SR 3.7.9.1 & revise Bases for SR 3.7.9.1 to include relocated information.

BASES

ACTIONS

D.1 (continued)

apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every 92 days adequately checks this system. The trains need only be operated for ≥ one hour and all louvers verified to be OPERABLE to demonstrate the function
1 of the system. This test includes an external visual
1 inspection of the CRVS Booster Fan trains. The 92 day Frequency is based on the known reliability of the equipment.

SR 3.7.9.2

This SR verifies that the required CRVS Booster Fan train testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CRVS Booster Fan train filter test frequencies are in accordance with Regulatory Guide 1.52 (Ref. 4). The VFTP includes testing HEPA filter performance and carbon adsorber efficiency. Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.9.3

This SR verifies the integrity of the control room enclosure. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify that the CRVS Booster Fan trains are functioning properly. During the emergency mode of

(continued)

Specification 3.7.9

A1 <except as marked>

4.12 CONTROL ROOM PRESSURIZATION AND FILTERING SYSTEM

Applicability

Applies to control room pressurization and filtering system components

Objective

To verify that these systems and components will be able to perform their design functions.

Specification

4.12.1 Operating Tests

LA8

SR3.7.9.1 ~~a.~~ Control room outside air booster fan system tests shall be performed quarterly. These tests shall consist of an external visual inspection, a flow measurement for each unit and pressure drop measurements across each filter bank. Pressure drop across pre-filter shall not exceed 1 inch H_2O and pressure drop across HEPA shall not exceed 2 inches H_2O. Fan motors shall be operated continuously for at least one hour, and all louvers shall be proven operable.

Supp 1 <see 5.0>

SR3.7.9.1

LA11

18 Month

SR3.7.9.3 ~~b.~~ On a ~~refueling frequency~~, verify the system maintains the control room at a positive pressure with both outside air booster fans on during system operation.

18 Months

4.12.2 Filter Tests

A12

SR3.7.9.2 On a ~~refueling frequency~~, for the Unit 1 and 2 and the Unit 3 control room an in-place leakage test using DOP on HEPA units and Freon-112 (or equivalent) on carbon units shall be performed at design flow on each filter train. Removal of 99.5 percent DOP by each entire HEPA filter unit and removal of 99.0 percent Freon-112 (or equivalent) by each entire carbon adsorber unit shall constitute acceptance performance. These tests must also be performed after any maintenance which may affect the structural integrity of either the filtration system units or of the housing.

<see 5.0>

Bases

A2

The purpose of the control room pressurization filtering system is to protect the control room operators from the effects of accidental release of radioactive effluents or toxic gases in the Turbine Building or Auxiliary Building only. The system is designed with two 50 percent capacity filter trains each of which consists of a prefilter, high efficiency particulate filters, carbon filters, booster fans, air handling unit fans, and associated ductwork to pressurize the control room with outside air.

Since these systems are not normally operated, a periodic test is required to insure their operability when needed. Quarterly testing of this system will show that the system is available.

Refueling frequency testing of the installed carbon adsorber stage and absolute filters will verify the leak integrity of the cleanup system. Refueling frequency testing will also verify the ability of the system to maintain the control room at a positive pressure to minimize infiltration of hazardous effluents.

6/6/89

not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of testing. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. Changes to ITS Bases are controlled by the Technical Specification Bases Control Program included in ITS Section 5.5. This change is consistent with the NUREG.

LA6 Not used.

LA7 CTS 4.5.1.1.2.a.2 and 4.5.1.1.2.b provide details regarding testing for specified LPSW valves. This information is relocated to the UFSAR Chapter 16. This information provides details of testing which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of testing. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. Changes to UFSAR Chapter 16 are controlled by 10 CFR 50.59. Additionally, ITS 5.5.9 provide requirements regarding this IST program. This change is consistent with the NUREG.

LA8 CTS 4.12.1.a provides details regarding filter testing. This
1 information is relocated to the ITS Bases. This information provides
1 details of system operation which are not directly pertinent to the
1 actual requirement, i.e., Definition, Limiting Condition for Operation
1 or Surveillance Requirement, but rather describe additional unnecessary
1 details such as an acceptable method of testing. Since these details
1 are not necessary to adequately describe the actual regulatory
1 requirement, they can be moved to a licensee controlled document without
1 a significant impact on safety. Placing these details in the Bases
1 provides adequate assurance that they will be maintained. Changes to
1 ITS Bases are controlled by the Technical Specification Bases Control
1 Program included in ITS Section 5.5.

1 LA9 Not used.

LA10 CTS 3.19.1.a, 3.19.2.a, 3.19.1.b, 3.19.2.b, and 3.19.3 provide details regarding requirements for an OPERABLE ECCW siphon header. Footnote 6 to CTS Table 4.1-2 provides details regarding testing the Emergency Condenser Circulating Water System. This information is relocated to the ITS Bases. This information provides details of system operation which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of testing. Since these details are not necessary

Booster Fans

CREVS
B 3.7.9
5

BASES

ACTIONS

C.1, C.2.1, and C.2.2 (continued)

[An alternative to Required Action C.1 is to immediately suspend activities that could release radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.]

4

D.1

[[In MODE 5 or 6, or] during movement of irradiated fuel assemblies [, or during CORE ALTERATIONS], when two CREVS trains are inoperable, action must be taken immediately to suspend activities that could release radioactivity that could enter the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.]

E.1

If both CREVS trains are inoperable in MODE 1, 2, 3, or 4, the CREVS may not be capable of performing the intended function and the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

2

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every ~~month~~ 92 days adequately checks this system. ~~Monthly heater operations dry out any moisture that has accumulated in the charcoal because of humidity in the ambient air. [Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized.~~ ~~Systems without heaters~~ The trains need only be operated for ≥ ~~15 minutes~~ one hour to demonstrate the function of the system ~~.]~~ and all louvers verified to be OPERABLE. The ~~31~~ 92 day Frequency is based on the known reliability of the equipment ~~and the two train redundancy available~~.

5 4 5 21

7

Supp 1

This test includes an external visual inspection of the CRVS Booster fan trains.

(continued)

BASES

NOTE: The first five justifications for these changes from NUREG-1430 were generically used throughout the individual Bases section markups. Not all generic justifications are used in each section.

1 The brackets are removed and the proper plant specific information or value is provided.

2 Editorial changes are made for preference, clarity or consistency with the Improved Technical Specifications (ITS) Writer's Guide.

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

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

5 This change reflects changes made to the technical specifications.

6 The Condensate Storage Tank and Condenser hotwell are not safety grade.

1 7 Additional information is added to the Bases for ITS SR 3.7.9.1 to
1 describe the test as including an external visual inspection of the CRVS
1 Booster Fan trains consistent with current test requirements.

8 Closure of the TSVs do not provide complete isolation between the two main steam lines. Other steam lines upstream of the TSVs, in addition to the main steam relief valves MSRVs and Steam Supply for the Steam Driven Emergency Feedwater Pump, are required to isolate to provide full isolation.

9 The ONS design provides two turbine stop valves (TSVs) in parallel for each main steam line. The TSVs, located immediately adjacent to the main turbine, are not as close to containment as typical MSIVs. The steam lines for the turbine bypass valves are located upstream of the TSVs.

10 With the required LPSW pumps inoperable, an appropriate bases is provided for the extended time to achieve MODE 5 from MODE 3.

11 Not used.

12 Not used.

13 Since the TSVs safety function is not as a containment isolation valve, the bases for the Completion Time is modified to provide a more appropriate bases.

14 Partial stroke testing of the TSVs is not restricted during power operation.

15 The ONS design provides eight MSRVs for each main steam line and does not provide main steam isolation valves.

16 Not used.

17 Not used.

18 The term anticipated operational occurrence (AOO) is not used in the ONS current licensing bases. The Design Basis Accident (DBA) for ONS is a large break LOCA with a LOOP. UFSAR Chapter 15 includes analyses of other accidents and transients in addition to the DBA.

19. The Bases discussion for the Completion Time for NUREG 3.7.5 Action A.1 is corrected to indicate 7 days instead of 72 hours to be consistent with the associated Completion Time in the NUREG Specification.

20. The statement regarding short half lives in the Background section of the Bases for NUREG 3.7.17 is deleted since it is no longer germane with the deletion of the next sentence by TSTF-173.

21. The Bases discussion to NUREG SR 3.7.10.1 is modified to include verifying louvers are OPERABLE during the operation of the CRVS Booster Fan Trains. This change captures a testin detail which is relocated from CTS 4.12.1.a.

ITS Section 3.7

ID 26

Subject Incorporate changes to ONS-007 (BWOG-66) - Revises Bases for 3.7.13 to reflect appropriate nomenclature consistent with ITS Spec

B 3.7 PLANT SYSTEMS

B 3.7.13 Fuel Assembly Storage

BASES

BACKGROUND

The spent fuel pool is designed to store either new (nonirradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies in a vertical configuration underwater. The shared spent fuel pool between Unit 1 and Unit 2 is sized to store 1312 fuel assemblies. The Unit 1 and Unit 2 spent fuel storage cells are installed in parallel rows with center to center spacing of 10.65 inches. The Unit 3 storage pool is sized to store 822 fuel assemblies. The spent fuel storage cells are installed in parallel rows with center to center spacing of 10.60 inches. This spacing and construction, whereby the fuel assemblies are inserted into stainless steel cans with neutron absorbing Boraflex attached, is sufficient to maintain a k_{eff} of $\leq$ 0.95 for spent fuel of a maximum nominal initial enrichment of up to 5.0 wt % which have accumulated burnups $\geq$ the minimum qualifying burnups of Figure 3.7.13-1 for the spent fuel pool shared by Units 1 and 2 or Figure 3.7.13-2 for the Unit 3 spent fuel pool. Fuel which has not accumulated the minimum qualifying burnups is required to be stored in the specified pattern for restricted fuel.

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1 **APPLICABLE SAFETY ANALYSES**

The spent fuel pool is designed for noncriticality by use of adequate spacing, and "flux trap" construction whereby the fuel assemblies are inserted into stainless steel cans with neutron absorbing Boraflex attached.

1 The fuel assembly storage satisfies Criterion 2 of 10 CFR 50.36 (Ref. 1).

LCO

The restrictions on the placement of fuel assemblies within the fuel pool, according to the Figures in the accompanying LCO, ensure that the k_{eff} of the spent fuel pool will always remain $\leq$ 0.95 assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.~~16~~ 3 ~~Spent~~ Fuel Assembly Storage 4

BASES

BACKGROUND

Shared spent fuel pool between Unit 1 and Unit 2

The spent fuel ~~storage facility~~ pool 4 is designed to store either new (nonirradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies in a vertical configuration underwater. The ~~storage pool~~ is sized to store ~~[735]~~ 1312 1 ~~irradiated~~ fuel assemblies ~~, which includes storage for [16] failed fuel containers~~. Unit 1+2 4 The spent fuel storage cells are installed in parallel rows with center to center spacing of ~~[12 31/32]~~ 10.65 inches ~~in one direction, and [13 3/16] inches in the other orthogonal direction~~. This spacing and ~~"flux trap"~~ construction, whereby the fuel assemblies are inserted into with neutron absorbing ~~stainless steel cans~~ Boraflex attached 4, is sufficient to maintain a k_{eff} of $\leq$ 0.95 for spent fuel of a maximum nominal 2 original enrichment of up to ~~[3.3]~~ 5.0%. ~~However, as higher initial enrichment fuel assemblies are stored in the spent fuel pool, they must be stored in a checkerboard pattern taking into account fuel burnup to maintain a k_{eff} of 0.95 or less.~~ Insert B3.7-75A

4 Insert B 3.7-75 B 4

APPLICABLE SAFETY ANALYSES

Supp 1

The spent fuel ~~storage facility~~ pool 4 is designed for noncriticality by use of adequate spacing, and "flux trap" construction whereby the fuel assemblies ~~are inserted into~~ with 4 neutron absorbing ~~stainless steel cans~~ Boraflex attached.

Supp 1

The ~~spent~~ 4 fuel assembly storage satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10CFR50.36, (Ref. 1) 2.

LCO

The restrictions on the placement of fuel assemblies within the fuel pool, according ~~to~~ to the Figure ~~[3.7.16-1]~~ in the accompanying LCO, ensure that the k_{eff} of the spent fuel pool will always remain ~~<~~ $\leq$ 0.95 assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool ~~according to Figure [3.7.16-1]~~. ~~Fuel assemblies not meeting the criteria of Figure [3.7.16-1] shall be stored in accordance with Specification 4.3.1.1.~~ 2 4 2

(continued)

ITS Section 3.7

ID 33

Subject Revise ITS 3.7.14 Bases to incorporate Amendment 227. Revises Bases to say that 150 gpd per SG leakage limit is bound by analysis.

B 3.7 PLANT SYSTEMS

B 3.7.14 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube out-LEAKAGE from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicative of current conditions. During transients, I-131 spikes have been observed, as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products, in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated transients and accidents.

This limit bounds the activity value that might be expected
1 from 300 gallon per day tube leakage (LCO 3.4.13, "RCS Operational Leakage") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.11, "RCS Specific Activity"). The steam line failure, steam generator tube rupture and other design basis accidents or transients can result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant leakage.

Operating a unit at the allowable limits could result in a 2 hour EAB exposure that is within the 10 CFR 100 (Ref. 1) limits.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7 ~~17~~ 14 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube out-LEAKAGE from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicative of current conditions. During transients, I-131 spikes have been observed, as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products, in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated ~~operational occurrences~~ transients, and accidents. 18

This limit ~~is lower than~~ bounds the activity value that might be expected from ~~a 1 gpm~~ 300 gallon per day tube leak (LCO 3.4.13, "RCS ~~Operational Leakage~~") age of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.~~16~~ 11 "RCS Specific Activity"). The steam line failure ~~is assumed to~~ result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant leakage. ~~Most of the iodine isotopes have short half lives (i.e., < 20 hours). I-131, with a half life of 8.04 days, concentrates faster than it decays, but does not reach equilibrium because of blowdown and other losses.~~

Suppl 4 Suppl 20 TSTF 173

steam generator tube rupture and other design basis accidents or transients can 4

~~With the specified activity limit, the resultant 2 hour thyroid dose to a person at the exclusion area boundary (EAB) would be about 0.79 rem if the main steam safety valves (MSSVs) are open for the 2 hours following a trip from full power.~~ 4 2

Operating a unit at the allowable limits could result in a 2 hour EAB exposure ~~of a small fraction of~~ that is within the 10 CFR 100 (Ref. 1) limits ~~, or the limits established as the NRC staff approved licensing basis.~~

(continued)

ITS Section 3.7

ID 38

Subject Revise ITS to remove draft generic change ONS-025 due to being withdrawn from BWOG consideration on 11/6/97. This affects the wording of Condition D.

3.7 PLANT SYSTEMS

3.7.2 Turbine Stop Valves (TSVs)

LCO 3.7.2 Four TSVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3 except when all TSVs are closed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both TSVs for one main steam line inoperable in MODE 1.	A.1 Restore TSV(s) to OPERABLE status.	8 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2.	6 hours
C. ---------NOTE--------- Separate Condition entry is allowed for each TSV. ---------------------- One or more TSVs inoperable in MODE 2 or 3.	C.1 Close TSV. <u>AND</u> C.2 Verify TSV is closed.	8 hours Once per 7 days
1 D. Required Action and associated Completion Time of Condition B or C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	12 hours 18 hours

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS requirements comparable to ITS 3.7.2 LCO and ITS 3.7.2 Actions do not exist. ITS LCO 3.7.2 requires four TSVs to be OPERABLE. This LCO provides assurance that the TSVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 limits. With one or both TSVs for one steam line inoperable in MODE 1, ITS 3.7.2 RA A.1 requires the TSVs be restored to OPERABLE status within 8 hours. The 8 hour Completion Time is reasonable because the TSVs isolate a closed system which provides an additional barrier against releases. With the Required Action and associated Completion Time for Condition A not met, ITS 3.7.2 RA B.1 requires the unit be placed in MODE 2 within 6 hours. The Completion Time is reasonable, based on operating experience, to reach MODE 2. With one or more TSVs inoperable in MODE 2 or 3, ITS 3.7.2 RA C.1 requires the TSV to be closed within 8 hours and RA C.2 requires verification the TSV is closed once per 7 days. The 8 hour Completion Time is reasonable considering the low probability of an accident occurring during this time period that would require closure of the TSVs. Inoperable TSVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of TSV status indications available in the control room, and other administrative controls, to ensure these valves are in the closed position. With the
1 Required Action and associated Completion Time for Condition B or C not met, ITS 3.7.2 RA D.1 requires the unit be placed in MODE 3 within 12 hours and MODE 4 within 18 hours. If the TSV cannot be restored to OPERABLE status or closed in the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

The addition of this LCO with associated ACTIONS is a more restrictive requirement upon unit operation and is consistent with the NUREG. This LCO and associated Actions are necessary to ensure unit operation is consistent with assumptions in the applicable safety analysis. This more restrictive change provides reasonable assurance that the TSVs can perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 limits.

M2 CTS 3.4.1 requires the EFW pumps and flow paths to be OPERABLE when Reactor Coolant System (RCS) temperature is > 250°F. ITS 3.7.5 Applicability for EFW is when in MODES 1, 2, & 3 and in MODE 4 when a steam generator is relied upon for heat removal. ITS MODE 4 is when RCS temperature is < 250°F and > 200°F.

A CTS provision comparable to the Note to ITS LCO 3.7.5 does not exist. This Note requires only one MDEFW pump and one EFW flow path to be OPERABLE in MODE 4.

Requiring the EFW system to be OPERABLE in MODE 4 when the SGs are relied upon for decay heat removal is a more restrictive requirement upon unit operation and is consistent with the NUREG. Requiring EFW system to be OPERABLE when SGs are relied upon for decay heat removal ensures the steam generators are available to remove decay heat when necessary.

M3 Note 1 to CTS Table 4.1-2 provides the CTS applicability associated with the TSV. This applicability is specified to be when the reactor is critical. ITS 3.7.2 is applicable in MODE 1 and MODES 2 and 3 except when all TSVs are closed. The additional applicability associated with the TSVs in MODE 2 with Keff < 1.0 and MODE 3 is a more restrictive requirement upon unit operation and is consistent with the NUREG. The TSVs must be OPERABLE in MODES 1, 2 and 3 with any TSVs open. In these conditions when there is significant mass and energy in the RCS and steam generators, the TSVs must be OPERABLE or closed.

M4 CTS Table 4.1-2 Item 11 requires a "Functional" Test be performed for the EFW pump automatic start and automatic valve actuation features. ITS SR 3.7.5.3 requires verification that each EFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal. ITS SR 3.7.5.4 requires verification that each EFW pump starts automatically on an actual or simulated actuation signal. The ITS requirements are more prescriptive and are therefore considered to be more restrictive requirement upon unit operation. This change is consistent with the NUREG. SR 3.7.5.3 verifies that EFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an Emergency Feedwater System initiation signal by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. SR 3.7.5.3 is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is also acceptable based on operating experience and design reliability of the equipment. SR 3.7.5.4 verifies that each EFW pumps start on an actual or simulated initiation signal. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

M5 A CTS surveillance requirement comparable to ITS SR 3.7.5.1 does not exist. This SR requires verifying the correct alignment for valves in the EFW water and steam flow paths. This change is therefore a more

restrictive requirement upon unit operation. Verifying the correct alignment for manual, and power operated valves in the EFW water and steam supply flow paths provides assurance that the proper flow paths exist for EFW operation.

M6 An additional Completion Time is added to those in CTS 3.4.3 to not only require the MDEFW pump to be restored within 7 days from discovery of one inoperable motor driven pump (ITS Required Action A.1) or within 72 hours of the discovery of one inoperable turbine driven pump (ITS Required Action B.1), but also within 10 days from discovery of failure to meet any of the requirements of the LCO. The new Completion Time establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO. This is an additional restriction on unit operation consistent with the NUREG. The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which multiple Conditions exist concurrently.

M7 A CTS requirement comparable to ITS SR 3.7.13.1 does not exist. ITS SR 3.7.13.1 requires verification by administrative means that initial fuel enrichment and burnup of a fuel assembly is in accordance with applicable storage requirements prior to storing the fuel assembly in the spent fuel pool. Although this is already done without an explicit CTS surveillance requirement, the addition of this SR is an additional restriction upon unit operation and is consistent with the NUREG. The change ensures storage of fuel assemblies in the Spent Fuel Pool is consistent with assumptions in the analysis.

M8 A CTS surveillance comparable to ITS SR 3.7.6.1 does not exist. Although CTS 3.4.5 specifies the required water volume required to be maintained, no periodic verification is required. Requiring a periodic verification on a 12 hour frequency that the water volume is within the specified limits is a more restrictive requirement upon unit operation and is consistent with the NUREG. SR 3.7.6.1 verifies that the CST, UST, and HW contain the required volume of cooling water.

M9 CTS 3.3.7 requires LPSW pumps to be OPERABLE when the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F. ITS 3.7.7 requires these systems to be OPERABLE during MODES 1, 2, 3, and 4. MODE 4 is defined as subcritical with the average coolant temperature > 200°F and < 250°F. CTS criteria specified as 250°F is considered more limiting than the 350 psig criteria, since the saturation temperature of water at 350 psig is > 435°F. As such, ITS 3.7.7 is slightly more restrictive in that OPERABILITY is required when temperature is > 200°F. In MODES 1, 2, 3, and 4, the LPSW System is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the LPSW System. Therefore, the LPSW System is required to be OPERABLE in these MODES.

TSVs
~~MSIVs~~
3.7.2

CTS

3.7 PLANT SYSTEMS

3.7.2 Turbine Stop ~~Main Steam Isolation~~ Valves (TSVs) ~~(MSIVs)~~

18

Doc M1

LCO 3.7.2 Four TSVs ~~Two MSIVs~~ shall be OPERABLE.

APPLICABILITY: MODE 1. 3
~~MODES 2 and 3 except when all TSVs ~~MSIVs~~ are closed ~~and~~ ~~[deactivated]~~.~~

18

T4.1-2 Note 1

13

ACTIONS

19

18

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One or both TSVs ~~MSIV~~ inoperable in MODE 1. for one main steam line	A.1 Restore TSVs ~~MSIV~~ to OPERABLE status.	~~[8]~~ hours 51	Doc M1 19
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2.	6 hours	Doc M1
C. ---------NOTE--------- Separate Condition entry is allowed for each TSV ~~MSIV~~. ------------------------- One or more TSVs ~~MSIVs~~ inoperable in MODE 2 or 3.	C.1 Close TSV ~~MSIV~~. AND C.2 Verify TSV ~~MSIV~~ is closed.	~~[8]~~ hours 51 Once per 7 days	Doc M1 Doc M1
D. Required Action and associated Completion Time of Condition B or C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 ~~6~~ hours 18 ~~12~~ hours	22 Doc M1 DOC M1

Supp 1

feedwater control valves (MFCV) and both startup feedwater control valves (SFCV) is assumed in the accident analysis. Therefore, NUREG Specification 3.7.3 is modified to remove ACTIONS associated with the MFSVs (ACTION A. ACTION D is eliminated since the ONS current licensing basis does not require two valves in the same flow path to be OPERABLE.

35 Not used.

36 Not used.

37 The Note to NUREG 3.7.4 Action A is not adopted. This Note permits entry into a MODE or other specified condition of the LCO's Applicability when the LCO is not met. Since the ADVs are credited in certain small break LOCA analysis, it is considered inappropriate to permit changes in MODE or other specified conditions of the LCO's Applicability when the LCO is not met.

38 Not used.

39 TSTF-100, which modifies NUREG 3.7.4 Required Action B.1 to state, "Restore all but one ADV line to OPERABLE status." is not adopted. The change represented by TSTF-100 is potentially confusing and unnecessary, since ONS has only two ADV lines. The TSTF-100 modified Required Action implies more than two ADV lines are provided.

40 The Note to SR 3.7.5.2 is not adopted. The ONS design permits testing the turbine driven EFW pump using auxiliary steam supplied from another ONS unit or the plant auxiliary boiler.

1 41 Not used.

42 The Applicability to NUREG 3.7.14 is expanded to include "During movement of a cask over the spent fuel pool." Action A is modified to also require suspending movement of the cask over the spent fuel pool. The Note to 3.7.14 Required Action A.1 is moved to precede both Required Action A.1 and Required Action A.2. This is necessary to reflect the current licensing basis for Oconee. ONS is licensed to permit placing a shipping cask in the spent fuel pool. The water level in the spent fuel pool is an initial assumption for the cask drop accident.

1 43 Not used.

1 44 Not used.

45 NUREG SR 3.7.2.1 is modified to reflect the TSV closure time associated with closure channel A. A second surveillance, SR 3.7.2.2, is added to verify the TSV closure time associated with channel B. The ONS design provides two means of closing each TSV. Each closure mechanism results in different closure times. Closure channel A actuates the "disk dump" closure mechanism which closes the associated TSV in $\leq$ 1 second.

ITS Section 3.7

ID 39

Subject Revise ITS to remove draft generic change ONS-022 (BWOG-77) due to being rejected by the TSTF. Adds modifier "except when all MFCVs and SFCVs are closed and deactivated or isolated by a closed manual valve" back to Applicability.

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Control Valves (MFCVs), and Startup Feedwater Control Valves (SFCVs)

LCO 3.7.3 Two MFCVs and two SFCVs shall be OPERABLE.

1 APPLICABILITY: MODES 1, 2, and 3 except when all MFCVs and SFCVs are closed
1 and deactivated or isolated by a closed manual valve.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MFCV in one or more flow paths inoperable.	A.1 Close or isolate MFCV. <u>AND</u> A.2 Verify MFCV is closed or isolated.	8 hours Once per 7 days
B. One SFCV in one or more flow paths inoperable.	B.1 Close or isolate SFCV. <u>AND</u> B.2 Verify SFCV is closed or isolated.	8 hours Once per 7 days

(continued)

Specification 3.7.3

A1 <except as marked>

A27

3.5.7 Main Steam Line Break Detection and Feedwater Isolation

Applicability

<See 3.3>

two

App.

Supp.1

Applies to main steam line break (MSLB) detection and feedwater isolation ~~circuitry when main steam header pressure is greater than 700 psig~~ and to ~~the~~ Main Feedwater ~~main and startup control~~ ~~(Main Feedwater control)~~ ~~valves~~ when Reactor Coolant System ~~temperature is greater than 250 °F.~~

L9

MODES 1, 2 + 3, except when all MFCVs and SFCVs are closed and deactivated or isolated by a closed manual valve

Objective

~~To ensure availability of the MSLB detection and feedwater isolation circuitry and Main Feedwater control valves to protect against containment overpressurization during a MSLB inside containment.~~

Specifications

3.5.7.1 MSLB detection and feedwater isolation circuitry shall be operable per Table 3.5.1-1, Items 20, 21, and 22.

<see 3.3>

L7

LCO 3.5.7.2 The ~~Main Feedwater control~~ valves shall be operable.

3.5.7.2.1 The provisions of 3.5.7.2 may be modified as follows:

or isolated

RA A.1, A.2

RA B.1, B.2

a. ~~A~~ ONE Main Feedwater control valve in one or more flow paths may be inoperable provided the affected valve(s) are closed within 8 hours from discovery and verified closed once per 7 days.

A28

RA C.1, C.2

b. If the required actions and associated completion time of 3.5.7.2.1.a cannot be met, the reactor shall be placed in a ~~hot shutdown condition~~ MODE 3 within 12 hours, and be less than or equal to ~~an RCS temperature of 250 °F~~ MODE 4 in ~~an additional~~ 18 hours.

M18

Add Action Note — A20

Add SR 3.7.3.1 — M30

Bases

~~The operability requirements of the MSLB detection and feedwater isolation circuitry and Main Feedwater control valves ensure that containment overpressure protection is available during a MSLB accident inside containment. The specified completion times provide adequate time to take appropriate action to restore the operability of the MSLB detection and feedwater isolation circuitry and the Main Feedwater control valves, or, if necessary, sufficient time to reduce power in a controlled manner. The completion times are considered adequate given the low probability of a MSLB accident.~~

~~Analyses of the main steam line break accident have determined that the containment design pressure of 59 psig could be exceeded with continued feedwater flow into the reactor building. To prevent exceeding the containment design pressure, the MSLB detection and feedwater isolation circuitry is designed to trip both Main Feedwater pumps, isolate all main~~

A2

1 of 3

TSC 95-03 7/15/97

L8 CTS 4.5.4.1.a requires each train of the PRVS be run for 15 minutes at design flow ± 10% monthly. ITS SR 3.7.10.1 requires operating each PRVS train ≥ 15 minutes at a Frequency of 31 days. The elimination of the requirement that the fans operate within 10% of design flow during this SR is a less restrictive requirement upon unit operation and is consistent with the NUREG. The purpose of this 31 day fan operation is to demonstrate the function of the system. ITS SR 3.7.10.3 adequately verifies the flow capability of the PRVS trains, although at a reduced Frequency. ITS SR 3.7.10.3, which is performed at an 18 month on a STAGGERED TEST BASIS Frequency, adequately verifies system flow meets requirements. ITS SR 3.7.10.1 adequately demonstrates system function, which is the purpose of the 31 day test. This test has typically met its flow requirements when tested at the monthly frequency.

1 L9 CTS 3.5.7 requires the MFCVs and SFCVs to be OPERABLE when the Reactor
1 Coolant System is greater than 250°F. ITS 3.7.5 requires these valves
1 to be OPERABLE in MODES 1, 2, and 3 except when all MFCVs and SFCVs are
1 closed and deactivated or isolated by a closed manual valve. ITS MODES
1 1, 2, and 3 are equivalent to CTS "...greater than 250°F." The
1 exception "when all MFCVs and SFCVs are closed and deactivated or
1 isolated by a closed manual valve" is acceptable since the safety
1 function of these valves is being fulfilled when these valves are closed
1 or isolated by a closed manual valve. This change is a less restrictive
1 requirement upon unit operation and is consistent with the NUREG.

LESS RESTRICTIVE CHANGE L9

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS 3.5.7 requires the MFCVs and SFCVs to be OPERABLE when the Reactor Coolant System is greater than 250°F. ITS 3.7.5 requires these valves to be OPERABLE in MODES 1, 2, and 3 except when all MFCVs and SFCVs are closed and deactivated or isolated by a closed manual valve. ITS MODES 1, 2, and 3 are equivalent to CTS "...greater than 250°F." The exception "when all MFCVs and SFCVs are closed and deactivated or isolated by a closed manual valve" is acceptable since the safety function of these valves is being fulfilled when these valves are closed or isolated by a closed manual valve. This change is a less restrictive requirement upon unit operation and is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Power Company has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. The OPERABILITY status of the MFCVs or SFCVs are not assumed to be an initiator of any accident previously evaluated. Therefore, the probability of occurrence of an accident is not significantly increased. This change provides the option of not being in the Applicability of the LCO when all MFCVs and SFCVs are closed and deactivated or isolated by a closed manual valve. Since the exception is allowed when the safety function of the valves are being fulfilled, the consequences of a design basis accident are not affected by the change. Therefore, the proposed change does not involve a significant increase the consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters

1 governing normal plant operation. Isolation of the lines is still
1 required. Therefore, the possibility of a new or different kind
1 of accident from any accident previously evaluated is not created.
1
1 **3. Does this change involve a significant reduction in a margin of**
1 **safety?**
1
1 This change provides the option of not being in the Applicability
1 of the LCO when all MFCVs and SFCVs are closed and deactivated or
1 isolated by a closed manual valve. Since the change requires the
1 lines to be isolated when the unit is in MODE 3 or above and the
1 MFCVs and SFCVs are inoperable, the proposed change does not
1 involve a significant reduction in a margin of safety.

ENVIRONMENTAL ASSESSMENT

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

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

(i) the proposed change involves no Significant Hazards Consideration (refer to the No Significant Hazards Consideration section of this Technical Specification Change Request),

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

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

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

CTS

 (34)

3.7 PLANT SYSTEMS

3.7.3 ~~[Main Feedwater Stop Valves (MFSVs),]~~ Main Feedwater Control Valves (MFCVs), and ~~Associated~~ Startup Feedwater Control Valves (SFCVs) (34)

LCO 3.7.3 Two ~~[MFSVs],~~ MFCVs, ~~[or associated~~ and two SFCVs shall be OPERABLE. 3.5.7.2

Supp. 1 APPLICABILITY: MODES 1, 2, and 3 except when all ~~[MFSVs],~~ MFCVs, ~~[or~~ and ~~associated~~ SFCVs are closed and deactivated [or isolated by a closed manual valve. (1) (4) 3.5.7 APP

ACTIONS

DOC A20

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. One [MFSV] in one or more flow paths inoperable.~~	~~A.1 Close or isolate [MFSV].~~ ~~AND~~ ~~A.2 Verify [MFSV] is closed or isolated.~~	~~[8 or 72] hours~~ ~~Once per 7 days~~ (34)
A. ~~B.~~ One MFCV in one or more flow paths inoperable.	A ~~B~~.1 Close or isolate MFCV. AND A ~~B~~.2 Verify MFCV is closed or isolated.	3.5.7.2.1.a 8 ~~or 72~~ hours (1) 3.5.7.2.1.a Once per 7 days (1)

(continued)

TECHNICAL SPECIFICATIONS

NOTE: The first four justifications for these changes from NUREG-1430 were generically used throughout the individual LCO section markups. Not all generic justifications are used in each section.

1 The brackets are removed and the proper plant specific information or value is provided.

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

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

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

5 Not used.

1 6 Not used.

7 NUREG 3.7.18 is not adopted. The ONS analysis assumes a SG inventory of 55,000 lbm. as an initial condition for a main steam line break. The water level indication (operating range) that correlates to this inventory is ≈ 98%. A trip of the main feedwater pumps and main turbine occurs at ≈ 96%. Administrative controls are sufficient to assure compliance with steam generator level restrictions. and an upper steam generator level is not included in the CTS. Therefore, the controls for these values are proposed to be administrative and not incorporated in the ITS.

8 NUREG LCO 3.7.5 and the LCO Note are modified to reflect the ONS design and current licensing basis regarding EFW since the ONS design does not separate the EFW system into trains. Additionally, the ONS current licensing basis permits greater flexibility regarding multiple component inoperabilities in the EFW system.

NUREG 3.7.5 ACTION A is not adopted since the steam supply to the turbine driven EFW pump is not required to satisfy the single failure criteria. NUREG 3.7.5 Action B is changed to ITS Action A. ITS 3.7.5 Actions B, and C are added. These Actions reflect the current licensing basis regarding permitted conditions and associated allowable outage times (AOT) for inoperable Emergency Feedwater (EFW) equipment. NUREG Condition C is renumbered as Condition D and modified to include added Conditions B and C as well as to exclude the combination of an inoperable turbine driven EFW pump and an inoperable EFW flow path.

ITS Section 3.7

ID 40

Subject Revise ITS to remove TSTF-157 based on the high probability that the TSTF will be permanently withdrawn. Adds back NUREG SR 3.7.5.5 which requires alignment check of EFW flow paths.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.5.1 Verify each EFW manual, and non-automatic power operated valve in each water flow path and in the steam supply flow path to the turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.5.2 Verify the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.7.5.3 -------------------NOTE------------------- Not required to be met in MODES 3 and 4. -- Verify each EFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.7.5.4 ------------------NOTE------------------- Not required to be met in MODES 3 and 4. -- Verify each EFW pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.7.5.5 Verify proper alignment of the required EFW flow paths by verifying valve alignment from the condensate storage tank to each steam generator.	Prior to entering MODE 2 whenever unit has been in MODE 5 or 6 for > 30 days

B 3.7 PLANT SYSTEMS

B 3.7.5 Emergency Feedwater (EFW) System

BASES

BACKGROUND

The EFW System automatically supplies feedwater to the steam
generators to remove decay heat from the Reactor Coolant
System (RCS) upon the loss of normal feedwater supply. The
EFW pumps take suction through suction lines from the upper
1 surge tank (UST) and condenser Hotwell and pump to the steam
generator secondary side through the EFW nozzles. The steam
generators function as a heat sink for core decay heat. The
heat load is dissipated by releasing steam to the atmosphere
from the steam generators via the main steam relief valves
(MSRVs) (LCO 3.7.1, "Main Steam Relief Valves (MSRVs)"), or
atmospheric dump valves (ADVs) (LCO 3.7.4, "Atmospheric Dump
Valves (ADVs)"). If the main condenser is available, steam
may be released via the Turbine Bypass System and
recirculated to the condenser Hotwell.

The EFW System consists of two motor driven EFW pumps and
one turbine driven EFW pump, any one of which can provide
the required heat removal capability. Thus, the
requirements for diversity in motive power sources for the
EFW System are met. The steam turbine driven EFW pump
receives steam from either of the two main steam headers,
upstream of the main turbine stop valves (TSVs), or from the
Auxiliary Steam System which can be supplied from the other
two unit's Main Steam System. The EFW System supplies a
common header capable of feeding either or both steam
generators. The EFW System normally receives a supply of
1 water from the UST. The EFW System can also be aligned to
the condenser Hotwell. An additional source of water is the
condensate storage tank which can be pumped to the USTs.

The EFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

The three emergency feedwater pumps are started automatically upon a loss of both main feedwater pumps or a signal from the ATWS Mitigation System Actuation Circuitry (AMSAC). The two motor driven emergency feedwater pumps are also started automatically upon a low steam generator level which exists for at least 30 seconds.

(continued)

BASES

BACKGROUND (continued)

The EFW System is discussed in the UFSAR, Section 10.4.7, (Ref. 1).

APPLICABLE SAFETY ANALYSES

The EFW System mitigates the consequences of any event with a loss of normal feedwater.

The design basis of the EFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at 1060.5 psig for the MDEFW pump and 1100 psig for the TDEFW pump.

The limiting event for the EFW System is the loss of main feedwater with offsite power available.

The EFW System design is such that it can perform its function following a loss of the turbine driven main feedwater pumps combined with a loss of normal or emergency electric power.

The EFW System satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

This LCO provides assurance that the EFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. Three independent EFW pumps and two flow paths are required to be OPERABLE to ensure the availability of residual heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering one pump by a steam driven turbine supplied with steam from a source not isolated by the closure of the TSVs, and two pumps from a power source that, in the event of loss of offsite power, is supplied by the emergency power source.

The EFW System is considered to be OPERABLE when the components and flow paths required to provide EFW flow to the steam generators are OPERABLE. This requires that the turbine driven EFW pump be OPERABLE with a steam supply from either one of the main steam lines upstream of the TSVs or from the Auxiliary Steam System. The two motor driven EFW

(continued)

BASES

LCO (continued)

pump(s) are also required to be OPERABLE. The two required flow paths shall also be OPERABLE. The sources of water to the EFW System are required to be OPERABLE. The associated flow paths from the EFW System sources of water to all EFW pumps also are required to be OPERABLE. EFW automatic initiation instrumentation is not required to be OPERABLE in MODES 3 and 4 in accordance with LCO 3.3.14, "Emergency Feedwater (EFW) System Initiation Circuitry." In MODES 3 and 4 the EFW System is OPERABLE provided manual initiation capability is OPERABLE. Automatic initiation is not required in MODES 3 and 4 since additional time is available in these MODES for the operator to manually initiate the system if required.

The LCO is modified by a Note indicating that one motor driven EFW pump and EFW flow path, is required in MODE 4 when an SG is relied upon for heat removal. This is because of reduced heat removal requirements, the short duration of MODE 4 in which feedwater is required, and the insufficient steam supply available in MODE 4 to power the turbine driven EFW pump.

APPLICABILITY

In MODES 1, 2, and 3, the EFW System is required to be OPERABLE and to function in the event that the main feedwater is lost. In MODE 4, with RCS temperature above 212°F, the EFW System may be used for heat removal via the steam generators. In MODE 4, the steam generators are used for heat removal unless the DHR System is in operation. In MODE 4 steam generators are relied upon for heat removal whenever an RCS loop is required to be OPERABLE or operating to satisfy LCO 3.4.6, "RCS Loops - Mode 4."

In MODES 5 and 6, the steam generators are not used for DHR and the EFW System is not required.

ACTIONS

A.1

With one of the motor driven EFW pumps inoperable, action must be taken to restore the MDEFW pump to OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

(continued)

BASES

ACTIONS

A.1 (continued)

a. The redundant OPERABLE turbine driven EFW pump(s);

b. The availability of the redundant OPERABLE motor driven EFW pump; and

c. The low probability of an event occurring that would require the EFW System during the 7 day period.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B exist concurrently. The AND connector between 7 days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

B.1

When the turbine driven EFW pump or one EFW flow path is inoperable, action must be taken to restore the pump and flow path to OPERABLE status within 72 hours. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the EFW System, time needed for repairs, and the low probability of an accident occurring during this time period. The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B exist concurrently. The AND connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

(continued)

BASES

ACTIONS (continued)

C.1

With the two motor driven EFW pumps inoperable, action must be taken to restore at least one pump to OPERABLE status within 12 hours. The 12 hour Completion Time is reasonable, based on the redundant capabilities afforded by the turbine driven EFW pump, time needed for repairs, and the low probability of an accident occurring during this time period.

D.1 and D.2

When Required Action or Completion Time for Condition A, B or C is not met or when the turbine driven EFW pump and one EFW flow path are inoperable in MODE 1, 2, or 3, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 24 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

In MODE 4, with two EFW pumps and one flow path inoperable, operation is allowed to continue because only one motor driven EFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate DHR.

E.1

Required Action E.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until at least one EFW pump and one flow path are restored to OPERABLE status.

With all EFW pumps or flow paths inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore at least one EFW pump and flow path to OPERABLE status. LCO 3.0.3 is not

(continued)

BASES

ACTIONS

E.1 (continued)

applicable, as it could force the units into a less safe condition.

F.1

In MODE 4, either the steam generator loops or the DHR loops can be used to provide heat removal, which is addressed in LCO 3.4.6, "RCS Loops – MODE 4." With one required EFW pump or flow path inoperable, action must be taken to immediately restore the inoperable pump or flow path to OPERABLE status.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, and non-automatic power operated valves in the EFW water and steam supply flow paths provides assurance that the proper flow paths exist for EFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since those valves are verified to be in the correct position prior to locking, sealing, or securing.

This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

Verifying that each EFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that EFW pump performance has not degraded below the acceptance criteria during the cycle. Flow and differential head are normal indications of pump performance required by Section XI of the ASME Code

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.2 (continued)

(Ref. 3). Because it is undesirable to introduce cold EFW into the steam generators while they are operating, this test may be performed on a test flow path.

This test confirms OPERABILITY, trends performance, and detects incipient failures by indicating abnormal performance. Performance of inservice testing in the ASME Code, Section XI (Ref. 3), at 3 month intervals, satisfies this requirement.

SR 3.7.5.3

This SR verifies that EFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an Emergency Feedwater System initiation signal by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is also acceptable based on operating experience and design reliability of the equipment. This SR is modified by a Note which states that the SR is not required in MODES 3 and 4. In MODES 3 and 4, the heat removal requirements would be less, thereby providing more time for operator action to manually start the required EFW pump.

SR 3.7.5.4

This SR verifies that each EFW pump starts in the event of any accident or transient that generates an initiation signal. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This SR is modified by a Note which states that the SR is not required in MODES 3 and 4. In

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.4 (continued)

MODE 3 and 4, the heat removal requirements would be less, thereby providing more time for operator action to manually start the required EFW pump.

1 SR 3.7.5.5
1
1 This SR ensures that the EFW System is properly aligned by
1 verifying the flow paths to each steam generator prior to
1 entering MODE 2 after more than 30 days in MODE 5 or 6.
1 OPERABILITY of EFW flow paths must be demonstrated before
1 sufficient core heat is generated that would require the
1 operation of the EFW System during a subsequent shutdown.
1 The Frequency is reasonable, based on engineering judgment,
1 in view of other administrative controls to ensure that the
1 flow paths are OPERABLE. To further ensure EFW System
1 alignment, flow path OPERABILITY is verified, following
1 extended outages to determine no misalignment of valves has
1 occurred. This SR ensures that the flow path from the UST
1 to the steam generator is properly aligned.

REFERENCES

1. UFSAR, Section 10.4.7.
2. 10 CFR 50.36.
3. ASME, Boiler and Pressure Vessel Code, Section XI.

Specification 3.7.5

A1 <except as marked>

Table 4.1-2

MINIMUM EQUIPMENT TEST FREQUENCY

Item	Test	Frequency	
1. Control Rod Movement (1)	Movement of Each Rod	Monthly	<see 3.1>
2. Pressurizer Safety Valves	Setpoint	Each Refueling(4)	<see 3.4>
3. Main Steam Safety Valves	Setpoint	Each Refueling(4)	<see 3.7.1>
4. Refueling System Interlocks(5)	Functional	Prior to Refueling	<see 3.3>
5. Main Steam Stop Valves (1)	Movement of Each Stop Valve	Monthly	<see 3.7.2>
6. Reactor Coolant System (2) Leakage	Evaluate	Daily	<see 3.4>
7. Condenser Circulating Water(6) Flow Test	Functional	Each Refueling	<see 3.7.8>
8. High Pressure Service Water Pumps and Power Supplies	Functional	Monthly	<see 3.7.1>
9. Spent Fuel Cooling System	Functional	Prior to Refueling	<see 3.7.1>
10. High Pressure and Low (3) Pressure Injection System	Vent Pump Casings	Monthly and Prior to Testing	<see 3.5>
SR 3.7.5.4 SR 3.7.5.3 11. Emergency Feedwater Pump Automatic Start and Automatic Valve Actuation Feature	~~Functional~~ Insert C4.1-9A Insert C4.1-9B (M4)	~~Each Refueling~~ 18 Months	
12. Main Steam Atmospheric Dump Valves	Stroke Test	Each Refueling	<see 3.7.4>

(1) Applicable only when the reactor is critical. <see 3.7.2 + 3.1>

(2) Applicable only when the reactor coolant is above 200°F and at a steady-state temperature and pressure. See 3.4

Add SR 3.7.5.1 — M5

Add SR 3.7.5.5 — M16

6 of 7

TSC 96-10
3/31/97

Supp. 1

1 OPERABILITY requirements of the containment. Placing the unit in MODE 5
1 places the unit in a MODE where the PRVS is not required to be OPERABLE.

M14 CTS 4.4.5.1.f requires verification the PRVS can maintain a negative pressure with respect to outside atmosphere. ITS SR 3.7.10.4 requires verification the PRVS can maintain at least 0.06 inches of water negative pressure with respect to atmospheric pressure during operation at a flow rate ≥ 900 CFM and ≤ 1100 CFM. The establishment of a minimum negative pressure requirement and minimum and maximum flow rates are more prescriptive requirements and are therefore more restrictive requirements upon unit operation. These additional requirement are consistent with the NUREG. The minimum value for the negative pressure provides reasonable assurance the PRVS can perform its safety function. The minimum and maximum flow limits ensure the PRVS is operating within a nominal range of its design flow.

M15 CTS Table 4.1-3, Item 4 requires sampling the spent fuel pool boron concentration monthly and after each makeup. No specific limit is placed on the time period to perform the surveillance after the makeup to the spent fuel pool. ITS SR 3.7.12.1 requires sampling the spent fuel pool within 12 hours after completing the makeup. Therefore this additional requirement is a more restrictive requirement upon unit operation. Sampling within 12 hours after the makeup provides reasonable assurance that boron concentration in the spent fuel pool is maintained consistent with the assumptions of the analysis.

1 M16 A CTS surveillance requirement comparable to ITS SR 3.7.5.5 does not
1 exist. This SR ensures that the EFW System is properly aligned by
1 verifying the flow paths to each steam generator prior to entering
1 MODE 2 after more than 30 days in MODE 5 or 6. OPERABILITY of EFW flow
1 paths must be demonstrated before sufficient core heat is generated that
1 would require the operation of the EFW System during a subsequent
1 shutdown. To further ensure EFW System alignment, flow path OPERABILITY
1 is verified, following extended outages to determine no misalignment of
1 valves has occurred. This SR is appropriate and is an acceptable
1 restriction upon operation since it ensures that the flow path from the
1 CST to the steam generator is properly aligned. This change is
1 consistent with the NUREG.

M17 Not used.

M18 If the control valves (MFCVs and SFCVs) are not restored to OPERABLE status within 72 hours CTS 3.5.7.2.1.b requires the unit be placed in Hot Shutdown within 12 hours and with RCS temperature less than 250°F within an additional 18 hours (total of 30 hours). ITS 3.7.3 RA C.1 requires the unit be placed in MODE 3 within 12 hours and MODE 4 within 18 hours. Requiring the unit be placed in MODE 4 within 18 hours is an additional restrictions upon unit operation. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

M19 CTS 3.4.1 requires the CST, UST and HW to meet specified requirements when RCS temperature is > 250°F. ITS 3.7.6 APPLICABILITY requires the CST, UST and HW to meet specified requirements when in MODES 1, 2, & 3 and in MODE 4 when a steam generator is relied upon for heat removal. Requiring the CST, UST and HW to be OPERABLE in MODE 4 when SGs are relied upon for decay heat removal is a more restrictive requirement upon unit operation and is consistent with the NUREG. Requiring the UST, CST and HW to be OPERABLE in MODE 4, when a steam generator is relied upon for heat removal, ensures a water supply is available to the EFW system when the EFW system is required to be OPERABLE.

M20 The CTS 3.13 limit for secondary system activity is 1.4 μCi/cc I-131, which is roughly equivalent to 2.0 μCi/gm DOSE EQUIVALENT I-131. This limit has been determined to be inappropriately high. ITS 3.7.14 specifies a limit of 0.10 μCi/gm Dose Equivalent I-131. A value of 0.10 μCi/gm Dose Equivalent I-131 is assumed in the reanalysis of UFSAR Chapter 15 accidents and transients. A topical report regarding the UFSAR Chapter 15 reanalysis methodology is being submitted separately for NRC approval and is described in the transmittal letter associated with the ONS ITS conversion. Therefore this is a more restrictive requirement upon unit operation and is consistent with the NUREG.

M21 CTS 3.15.1 requires the PRVS to be OPERABLE whenever containment integrity is required. CTS 3.6.1 requires containment integrity whenever all three following conditions exist:

a.- reactor coolant pressure is ≥ 300 psig,
b.- reactor coolant temperature is ≥ 200°F, and
c.- nuclear fuel is in the core.

With these criteria, containment integrity is required sometime during ITS MODE 4 but not necessarily when this MODE was entered. The Applicabilities of ITS 3.7.10 for the PRVS include MODES 1, 2, 3, and 4. As such, the change represents an additional restriction on unit operation. This change is consistent with the NUREG. In MODES 1, 2, 3, and 4, the PRVS is required to be OPERABLE consistent with the OPERABILITY requirements of the containment.

M22 A CTS provision comparable to ITS 3.7.5 Action F does not exist since the CTS does not include operability requirements upon the EFW when the unit is less than 250°F. If the required EFW pump and required EFW flow path are not OPERABLE in MODE 4 when the steam generators are relied upon for decay heat removal, ITS 3.7.5 Action F requires action be initiated immediately to restore the required MDEFW pump and required EFW flow path to OPERABLE status. In this Condition, the Required Action is necessary to assure the steam generators can continue to be relied upon for decay heat removal. This change represents a more restrictive requirement upon unit operation.

M23 If one Atmospheric Dump Valve (ADV) is not restored to OPERABLE status within 7 days or if both ADVs are not restored to service within 24

hours, CTS 3.4.7.a and CTS 3.4.7.b require the unit be placed Hot Shutdown within 12 hours and below 350°F within another 24 hours. With Required Action and Associated Completion Time not met, ITS 3.7.4 Condition C requires the unit be placed in MODE 3 within 12 hours and in MODE 4 (> 200°F and < 250°F), without reliance upon a steam generator for heat removal, within 24 hours. The requirements to place the unit in a condition without reliance upon a steam generator for heat removal and to be in this condition within 24 hours (instead of 36 hours) are more restrictive requirements upon unit operation and are consistent with the NUREG. Placing the unit in a condition without reliance upon a steam generator for heat removal is necessary to insure heat removal capability if the heat removal capability provided by the condenser become unavailable. The 24 hour Completion Time is reasonable based upon operating experience to reach the condition in an orderly manner without challenging plant systems.

M24 A CTS requirement comparable to ITS SR 3.7.4.2 does not exist. ITS SR 3.7.4.2 requires stroking the ADV block valves through one complete cycle at an 18 month frequency. Cycling the valves open and closed demonstrates their ability to perform their function. This additional requirement is a more restrictive requirement upon unit operation and is consistent with the NUREG.

M25 CTS requires the Control Room booster fan trains to be OPERABLE above Hot Shutdown (i.e., when in ITS MODES 1 and 2). The Applicability of ITS LCO 3.7.9 is MODES 1, 2, 3 and 4. In MODES 1, 2, 3 and 4, the Control Room booster fan trains must be OPERABLE to reduce the radiation dose to personnel in the Control Room during and following an accident.

If the Control Room booster fan trains cannot be restored to OPERABLE status within the specified times, CTS 3.15.2.a and 3.15.2.b requires the unit be placed in Hot Shutdown within 12 hours. In this Condition ITS 3.7.9 RA D.2 requires the unit be placed in MODE 5 within 36 hours. Placing the unit in MODE 5 within 36 hours is necessary to place the unit in a condition outside the Applicability of ITS LCO 3.7.9. This change is a more restrictive requirement upon unit operation and is consistent with the NUREG.

M26 A specification comparable to ITS specification 3.7.11 does not exist. ITS specification 3.7.11 provides requirements for a minimum water level in the spent Fuel Pool. ITS LCO 3.7.11 requires the water level in the SFP to be 21.34 feet above the water level of the top of the irradiated fuel assemblies seated in the fuel storage racks. The minimum water level in the Spent Fuel Pool is consistent with the assumption of iodine decontamination factors following a fuel handling or cask drop accident. The Applicability for ITS Specification 3.7.11 is during movement of irradiated fuel assemblies in the spent fuel pool and during movement of the cask over the spent fuel pool. This LCO applies during movement of irradiated fuel assemblies in the Spent Fuel Pool or movement within the Spent Fuel Pool or movement of the cask over the Spent Fuel Pool since the potential for a release of fission products exists at these times.

With the SFP water level not within limits, ITS 3.7.11 RA A.1 requires immediately suspending movement of irradiated fuel assemblies in the SFP and the cask over the SFP. When the initial conditions for an accident cannot be met, immediate action must be taken to preclude the occurrence of an accident. With the Spent Fuel Pool at less than the required level, the movement of fuel assemblies in the Spent Fuel Pool is immediately suspended. This effectively precludes the occurrence of a fuel handling accident. ITS SR 3.7.11.1 requires periodic verification that the SFP water level is ≥ 21.34 feet above the top of irradiated fuel assemblies seated in the storage racks. This SR verifies that sufficient Spent Fuel Pool water is available in the event of a fuel handling or cask drop accident. The water level in the Spent Fuel Pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable.

The addition of this specification is a more restrictive requirement upon plant operation and is consistent with the NUREG. This specification ensures the water level in the Spent Fuel Pool is consistent with the assumptions of the safety analysis for fuel handling and cask drop accident.

M27 With less than the specified number of MSRVs OPERABLE, CTS Actions are specified by CTS 3.0. CTS 3.0 requires the unit to be in Hot Shutdown within 12 hours and cold shutdown within 36 hours. However, since the CTS applicability for the MSRVs is with RCS temperature > 250°F, CTS 3.0 only requires the RCS temperature be reduced to ≤ 250°F (equivalent to ITS MODE 4) at which point the CTS requirements for MSRVs is no longer applicable. Consequently, CTS 3.0 requires the unit be placed in MODE 4 within 36 hours. With less than the specified number of MSRVs OPERABLE, ITS 3.7.1 Action A requires the unit be placed in MODE 3 within 12 hours and MODE 4 within 18 hours. The requirement to be in MODE 4 within 18 hours instead of 36 hours is a more restrictive requirement upon unit operations and is consistent with the NUREG. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

M28 CTS requirements comparable to ITS SR 3.7.8.1, SR 3.7.8.2, SR 3.7.8.3 and SR 3.7.8.4 do not exist. SR 3.7.8.1 requires periodic verification that the required ESV pumps are in operation. This ensure that air from inleakage and degassing is removed to ensure the capability to establish siphon flow. SR 3.7.8.2 verifies periodic verification that Keowee Lake level is within limit, which ensures ECCW siphons can provide sufficient flow to ensure adequate NPSH is available for operating the LPSW pumps. SR 3.7.8.3 requires periodic verification that CCW inlet temperature is ≤ 90°F. This ensure inlet water to the LPSW system is consistent with assumptions. SR 3.7.8.4 provides periodic verification the correct alignment for manual, and non-automatic power operated valves in the ECCW siphon header flow paths, required ESV flow paths and required SSW flow paths. These SRs provides assurance that the proper

conditions (i.e., flowpath, pump status, lake level and water temperature) exist for ECCW siphon header operation.

M29 CTS Table 4.1-2, Item 7 requires performance of am Emergency Condenser Circulating Water System Functional Test. ITS SR 3.7.8.9 verifies upon a simulated or actual loss of power to the CCW pumps and ESV pumps that the rate of water level decrease in the CCW piping is within limits. This ensures air inleakage and degassing is within the removal capabilities of the ESV pumps. Limiting air inleakage is necessary to ensure the ECCW siphons headers can provide sufficient NPSH to the LPSW pumps.

CTS Table 4.1-2, Item 13 requires performance of an Essential Siphon Vacuum System Functional Test. ITS SR 3.7.8.5, SR 3.7.8.6, SR 3.7.8.7 and SR 3.7.8.8 provides requirements comparable to CTS Table 4.1-2, Item 13. SR 3.7.8.4 verifies that upon an actual or simulated actuation signal each ESV float valve actuates to the correct position.
SR 3.7.8.5 verifies that upon an actual or simulated actuation signal each required ESV and SSW valve actuates to the correct position.
SR 3.7.8.6 verifies that the developed suction head of each required ESV pump at the test point is ≥ the required developed suction head.
SR 3.7.8.7 verify each required ESV pump starts within specified time period upon an actual or simulated restoration of emergency power.
These SRs ensure the ESV and SSW Systems adequately support ECCW siphon header function. The ECCW siphon headers function to provide sufficient NPSH to the LPSW pumps.

The comparable ITS SR requirements are more prescriptive and therefore are more restrictive requirements upon unit operation.

M30 An explicit CTS requirement comparable to ITS SR 3.7.3.1 does not exist. ITS SR 3.7.3.1 requires verifying valve closure time is less than 25 seconds. A Note to SR 3.7.3.1 states the SR is only required to be performed in MODES 1 and 2. Although the stroke time for these valves is verified in accordance with the IST program, the inclusion of the stroke time in the ITS is a more restrictive requirement upon unit operation and is consistent with the NUREG. The inclusion of the SR Note is more restrictive since CTS does not require performance of SRs prior to entry into the applicability for the specification. The addition of the ITS SR Note is consistent with the NUREG. The inclusion of the stroke time is acceptable since it ensures equipment operation is consistent with the assumptions in the applicable safety analysis. The inclusion of the Note is acceptable since it permits testing the valves under operating conditions closer to those that would exist for an actual demand for the valves to function.

M31 An explicit CTS Applicability requirement for the ADVs lines does not exist. However, CTS 3.4.7.a and 3.4.7.b imply an Applicability for ADVs the ADV flow paths of RCS temperature ≥ 350°F since below this temperature, no further action is required for one or more inoperable ADV flow paths. ITS 3.7.4 has an applicability of MODE 1, 2, 3 and MODE

4 when SGs are relied upon for heat removal. The additional OPERABILITY requirements when in MODE 3 with RCS temperature less than 350°F and in MODE 4 when SGs are relied upon for heat removal is a more restrictive requirement upon unit operation and is consistent with the NUREG. This is appropriate since the analyses assumes ADVs are used to cool down the unit to MODE 4 for small break LOCAs.

M32 With an inoperable LPSW pump, CTS 3.3.7.b ultimately requires the unit RCS temperature be reduced to < 250°F and RCS pressure < 350 psig. The condition is comparable to ITS MODE 4. ITS 3.7.7 RA B.2 requires the unit be placed in MODE 5. This is a more restrictive requirement upon unit operation and is consistent with the NUREG. In MODES 1, 2, 3, and 4, the LPSW System is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the LPSW System. Therefore, the LPSW System is required to be OPERABLE in these MODES. Placing the unit in MODE 5 is necessary to exit the Applicability for the Specification.

M33 A CTS action comparable to ITS 3.7.6 Action A does not exist. With the water volume in the Condensate Storage Tank (CST), Upper Surge Tanks (UST) and condenser hotwell (HW) less than the specified limit, CTS actions are specified by CTS 3.0 which requires the unit be placed in Hot Shutdown within 12 hours and RCS temperature < 250°F within 36 hours. Since the CTS Applicability for this specification is with RCS temperature > 250°F, CTS 3.0 only requires the RCS temperature be less than 250°F within 36 hours instead of Cold Shutdown. ITS 3.7.6 RA A.2 requires the unit be in MODE 4 without reliance upon steam generators within 24 hours. The requirement to place the unit in MODE 4 without reliance upon steam generators within 24 hours is a more restrictive requirement upon unit operations and is consistent with the NUREG. Requiring the unit be placed in MODE 4 without reliance upon steam generators is appropriate since in this condition the EFW water sources are not required to be support the OPERABILITY of the EFW system.

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.7.5.4	----------NOTES---------- 1. Not required to be performed until [24] hours after reaching [800] psig in the steam generators. 2. Not ~~applicable~~ required to be met in ~~MODE 4~~ MODES 3 and 4. ------------------------- Verify each EFW pump starts automatically on an actual or simulated actuation signal.	~~[~~18~~]~~ months
SR 3.7.5.5	Verify proper alignment of the required EFW flow paths by verifying ~~[~~valve alignment ~~/flow]~~ from the condensate storage tank to each steam generator.	Prior to entering MODE 2 whenever plant has been in MODE 5 or 6 for > 30 days
~~SR 3.7.5.6~~	~~Perform a CHANNEL FUNCTIONAL TEST for the EFW pump suction pressure interlocks.~~	~~31 days~~
~~SR 3.7.5.7~~	~~Perform a CHANNEL CALIBRATION for the EFW pump suction pressure interlocks.~~	~~[18] months~~

10

3.4.3.e

14

1

T4.1-2
Item 11
4.9.3

Supp 1

1

12

NUREG Condition D is renumbered as Condition E and is modified to reflect the requirement to recover an EFW pump and EFW flow path. NUREG Condition E is renumbered as Condition F and is modified to reflect the requirement to recover a MDEFW pump and EFW flow path.

9 Not used.

10 Note 1 to NUREG SR 3.7.5.3 and SR 3.7.5.4 is not adopted. This testing is currently performed at low pressures to avoid either: a) making the system inoperable by tagging out the injection valves which would also open on the actuation signal, or b) injecting cold condensate into the steam generators. Valve and pump actuation can be demonstrated at low pressures, and along with full pressure, manual opening of the steam admission valves and pump flow testing, adequately demonstrates the capability of the system to perform these required safety functions. The wording of Note 2 is revised for consistency with Section 1.0, "Use and Application," Example 1.4-3, and other similar intent Notes within the NUREG. The "applicable" MODES are addressed only in the portion of the Specification entitled "APPLICABILITY" (with the exception of where applicable SRs of one specification are referenced by another specification, e.g., when a shutdown specification identifies the "applicable" SR from the operating specification rather that repeat each "required" SR). The remainder of the Specification addresses "requirements."

1 11 Not used.

12 The ONS unit design does not include EFW pump suction pressure interlocks. Therefore, the SRs are not adopted.

13 NUREG 3.7.2 Applicability is modified to MODES 1, 2, and 3 except when all TSVs are closed. This reflects that the TSVs are closed in MODE 1 up to approximately 15% RTP in addition to being closed in MODES 2 and 3.

14 NOTE 2 to NUREG 3.7.5.3 and SR 3.7.5.4 is modified to reflect the current licensing basis requirements regarding automatic initiation of EFW. CTS 3.4.2 requires the EFW automatic initiation circuitry to be OPERABLE prior to criticality (i.e., NUREG MODES 1 and 2).

15 NUREG LCO 3.7.7 - The ONS current licensing basis does not include the Component Cooling Water system for events which meet the criteria of 10 CFR 50.36. The safety related cooling water requirements are met by the Low Pressure Service Water (LPSW) System. Therefore, only the Low Pressure Service Water System is proposed to be incorporated in the ITS.

16 NUREG Specification 3.7.1 is modified to omit the table of specific lift setpoints. The specific lift setpoints are currently required to be tested by current Technical Specification (CTS) Table 4.1-2, item 3. However, the CTS does not contain the specific setpoints (including

associated tolerances). These setpoints are currently identified in the Updated Final Safety Analysis Report (UFSAR) and are adequately controlled therein under the design change and procedural control programs which include evaluations of changes in accordance with 10 CFR 50.59.

The NUREG figure for determining the allowable power level and trip settings with predetermined values is not adopted. Additionally, NUREG LCO 3.7.1 is revised to require that 7 MSRVs on each main steam line be OPERABLE regardless of power level. This value for OPERABLE MSRVs is consistent with current analysis. The capability provided by the NUREG to permit continued operation at a reduced power level with less than the required number of MSRVs is not adopted since no ONS specific analysis exists which supports this reduction or stipulates the appropriate reduced power level. NUREG Action A is modified to require unit shutdown if less than the required number of MSRVs are OPERABLE. NUREG Condition B is not adopted because it is no longer needed due to the modification to Action A. The NUREG Note to Actions, permitting separate Condition Entry for each MSRV, is not adopted because separate Condition Entry is not necessary when the Required Action for one required MSRV being inoperable is immediate unit shutdown.

17 Not used.

18 The ONS design does not provide Main Steam Isolation valves. The Turbine Stop Valves (TSV) provide comparable protection (isolation of the unaffected steam generator in the event of a steam line break). The ONS design provides two TSVs per main steam line.

19 Action A is modified to permit one or both TSVs for one main steam line to be inoperable in MODE 1. The ONS design provides two TSVs in parallel in each main steam line. Inoperability of one or both TSVs in one steam line prevents isolation of that steam line. Additionally, the basis for NUREG 3.7.2 Required Action A.1 is not predicated upon any temporary relaxation of the single failure criteria, but upon the limited time in this CONDITION and the low probability of an accident requiring actuation of the valves during the specified time interval. Since the basis for the Condition and associated allowable outage time (AOT) is not a temporary relaxation of the single failure criteria, more than one TSV may be permitted to be inoperable in MODE 1.

20 The ONS design does not provide sufficient inventory in the Condensate Storage Tank (CST) alone for the EFW water supply. The ONS design provides the EFW water supply in the combination of the Condensate Storage Tank (CST), Upper Surge Tank (UST) and Condenser Hotwell (HW). The NUREG references to the CST are modified to state CST, UST and the HW as appropriate. NUREG SR 3.7.6.1 is modified to reflect the current licensing basis requirements for inventory in these storage locations.

21 NUREG 3.7.6 Action A is not adopted since the current licensing basis does not provide for use of a backup water source as a temporary substitute for the inventory in the CST, UST, and HW.

22 The current licensing basis permits 12 hours to place a unit in Hot Shutdown (equivalent to ITS MODE 3) when an LCO is not met. To maintain consistency with current procedures, training and staffing requirements, the 12 hours permitted to place a unit in MODE 3 is retained in the ITS. The subsequent Completion Time to MODE 4 is modified accordingly (i.e., to allow an additional 6 hours).

23 The ONS design provides three pumps with common shared flow paths for the shared Units 1 and Unit 2 LPSW system. Two pumps with a common shared flow path are provided for the Unit 2 LPSW system. NUREG LCO 3.7.8 is modified to reflect the ONS specific design and current licensing basis regarding LPSW system OPERABILITY. An LCO Note is provided to address a plant specific allowance regarding LPSW system requirements when either Unit 1 or Unit 2 is defueled.

24 When in Hot Shutdown, the current licensing basis provides additional time to restore the LPSW system to OPERABLE status prior to requiring the plant to be placed in Cold Shutdown. NUREG 3.7.8 Actions are modified to reflect this allowance.

25 The ONS design utilizes hydroelectric units in lieu of diesel generators as the onsite emergency power source. A diesel generator is used in the Standby Shutdown Facility (SSF). Neither the hydroelectric units nor the SSF diesel generator are supported by the LPSW system. Additionally, The ONS design provides for the decay heat coolers to be supplied by multiple LPSW subsystems. As a result, loss of a single LPSW subsystem does not result in a loss of LPSW supply to a decay heat cooler.

26 Not used.

27 The ONS current licensing basis requires the Control Room Ventilation System Booster Fans, Filters, and associated ducts and dampers necessary to pressurize the control room to be OPERABLE. For clarity regarding the scope of this specification, the NUREG 3.7.10 term "trains" is modified in ITS 3.7.9 to state "Booster Fan trains."

Additionally, NUREG 3.7.10 is modified to reflect the current licensing basis for the Control Room Ventilation System Booster Fan Trains. The current licensing basis reflected in the CTS requirements regarding the Control Room Ventilation System Booster Fan trains is retained in the ITS 3.7.9 LCO, Actions and Surveillance Requirements.

28 NUREG SR 3.7.10.3 is not adopted. Only manual actuation capability for the Control Room Ventilation System Booster Fan Trains is provided in the ONS design.

29 NUREG 3.7.11 is not adopted. The ONS current licensing basis does not credit the Chilled Water System (CWS) in the accident analyses. Additionally, the CTS does not contain any requirements for the CWS. Therefore, controls for this system are administrative and not adopted in the ITS.

30 NUREG Specification 3.7.13, Fuel Storage Pool Ventilation System (FSPVS) is not adopted. Although the ONS design provides a comparable system, the Spent Fuel Pool Ventilation System (SFPVS), and the CTS includes comparable requirements, the NUREG specification is not adopted because the SFPVS has been reviewed against, and determined not to satisfy, the selection criteria for Technical Specifications provide in 10 CFR 50.36. The selection criteria were established to ensure that the Technical Specifications are reserved for those conditions or limitations on plant operation considered necessary to limit the possibility of an abnormal situation or event that could result in an immediate threat to the health and safety of the public. The rationale for relocation of each of the Specifications is provided in the report, "Application of Selection Criteria to the Oconee Nuclear Stations Unit 1, 2, and 3 Technical Specifications."

31 Not used.

32 The Applicability for NUREG 3.7.15 and the Frequency for SR 3.7.15.1 are modified to reflect the current licensing basis. Specifically, the NUREG provision regarding the modification of Applicability when a spent fuel pool (SFP) verification has been completed after the last movement of fuel assemblies in the SFP is not adopted. Related NUREG Required Action A.2.2 is also not adopted. The ONS current licensing basis includes consideration of other events such as dropping the shipping cask into the SFP which are not excluded by the SFP verification.

The current licensing basis regarding frequency of verifying the spent fuel pool boron concentration is retained. The periodic frequency of 31 days is considered adequate since the event specific frequency of " . . . once within 12 hours after completion of a makeup to the spent fuel pool . . . " provides reasonable assurance the required boron concentration is maintained for the only expected event that can result in decreasing the boron concentration in the spent fuel pool.

33 NUREG Specification 3.7.16 is modified to reflect the current licensing basis regarding the requirements applicable to spent fuel storage locations in both the spent fuel pool shared between the Unit 1 and Unit 2 and the Unit 3 spent fuel pool. This change is necessary to encompass the CTS requirements for the three ONS units and the two spent fuel pools.

34 The current licensing basis for main feedwater isolation valves does not include the main feedwater stop valves (MFSV). The MFSVs are not assumed to close in the accident analysis. Closure of both the main

4 <unless otherwise marked>

B 3.7 PLANT SYSTEMS

B 3.7.5 Emergency Feedwater (EFW) System

Supp. 1

R

upper surge tank (UST) + condenser Hotwell

BASES

BACKGROUND

The EFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System (RCS) upon the loss of normal feedwater supply. The EFW pumps take suction through ~~separate and independent~~ suction lines from the ~~condensate storage tank (CST) (LCO 3.7.6, "Condensate Storage Tank (CST)")~~, and pump to the steam generator secondary side through the EFW nozzles. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam ~~safety~~ relief valves (MSSVs) (LCO 3.7.1, "Main Steam ~~Safety~~ relief Valves (MSSVs)"), or atmospheric ~~vent~~ dump valves (ADVs) (LCO 3.7.4, "Atmospheric ~~Vent~~ dump Valves (ADVs)"). If the main condenser is available, steam may be released via the Turbine Bypass System and recirculated to the ~~CST~~ condenser Hotwell.

The EFW system can also be aligned to the condenser Hotwell

R D

Insert B3.7-23A

4

~~[The following system description is provided as an example. Actual system description should be provided by the specific unit.]~~ The EFW System consists of two ~~turbine~~ motor driven EFW pumps, ~~each of which provides a nominal 100% capacity,~~ and one ~~nonsafety grade motor~~ turbine driven EFW pump. The steam turbine driven EFW pumps receive steam from either of the two main steam headers, upstream of the main ~~steam isolation~~ turbine stop valves ~~(MSIVs)~~ TSVs. The EFW System supplies a common header capable of feeding either or both steam generators. ~~The 100% capacity is sufficient to remove decay heat and cool the unit to decay heat removal (DHR) entry conditions.~~ The EFW System normally receives a supply of water from the ~~CST~~ UST. ~~A safety grade source of water is also supplied by the Service Water System (SWS). Automatic valves on the supply piping open on low pressure in the supply piping to transfer the water supply from the CST to the SWS. A third source of water can be supplied by manually aligning the fire protection header to the EFW pump suction.~~ Thus, the requirements for diversity in motive power sources for the EFW System are met.

4

Supp. 1

An additional source of water is the condensate storage tank which can be pumped to the USTs.

2

The EFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

any one of which can provide the required heat removal capacity.

(continued)

4

BASES

SURVEILLANCE REQUIREMENTS (continued)

Supp. 1

SR 3.7.5.5

This SR ensures that the EFW System is properly aligned by verifying the flow paths to each steam generator prior to entering MODE 2 after more than 30 days in MODE 5 or 6. OPERABILITY of EFW flow paths must be demonstrated before sufficient core heat is generated that would require the operation of the EFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment, in view of other administrative controls to ensure that the flow paths are OPERABLE. To further ensure EFW System alignment, flow path OPERABILITY is verified, following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the ~~CST~~ UST to the steam generator is properly aligned. ~~(This SR is not required by those units that use EFW for normal startup and shutdown.)~~

5

~~SR 3.7.5.6 and SR 3.7.5.7~~

~~For this facility, the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION for the EFW pump suction pressure interlocks are as follows:~~

REFERENCES

1. U FSAR, Section ~~9.2.7~~ 10.4.7.

2. ~~FSAR, Section 9.2.8~~ 10CFR 50.36.

3. ASME, Boiler and Pressure Vessel Code, Section XI.

2

ITS Section 3.7

ID 54

Subject Remove DOC LA9 from Section 3.7 and show in CTS markup that CTS 4.5.4.1.b.1 is addressed by Section 5.0.

Specification 3.7.10

A1 <except as marked>

~~4.5.4 Penetration Room Ventilation System~~

~~Applicability~~

~~Applies to testing of the Penetration Room Ventilation System~~

~~Objective~~

~~To verify that the Penetration Room Ventilation System is operable.~~

Specification

4.5.4.1 Operational and Performance Testing

SR 3.7.10.1 ~~a.~~ ~~Monthly~~ 31 days, each train of the Penetration Room Ventilation System shall be operated for at least ~~15 minutes~~ ~~at design flow ±10%~~. L8

~~b.~~ During each ~~refueling outage~~ 18 months, ~~it~~ shall be demonstrated that:

Supp1

1. The Penetration Room Ventilation System fans operate at design flow (± 10%) when tested in accordance with ANSI N510-1975.

SR 3.7.10.2 2. The pressure drop across the combined HEPA filters and charcoal adsorber banks is less than six inches of water at the system design flow rate (± 10%). <See 5.0> A14

SR 3.7.10.3 ~~3.~~ Each ~~branch~~ Train of the Penetration Room Ventilation System is capable of automatic initiation.

SR 3.7.10.5 ~~4.~~ The bypass valve for filter cooling ~~is manually operable~~ can be opened.

SR 3.7.10.2

c. Leak tests using DOP or halogenated hydrocarbon, as appropriate shall be performed on the Penetration Room purge filters:

1. During each refueling outage;
2. After each complete or partial replacement of a HEPA filter bank or charcoal adsorber bank; <see 5.0>
3. After any structural maintenance on the system housing;
4. After painting, fire, or chemical release in any ventilation zone communicating with the system.

d. The results of the DOP and halogenated hydrocarbon tests on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal, respectively, when tested in accordance with ANSI N510-1975.

A14

2 of 3

not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of testing. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. Changes to ITS Bases are controlled by the Technical Specification Bases Control Program included in ITS Section 5.5. This change is consistent with the NUREG.

LA6 Not used.

LA7 CTS 4.5.1.1.2.a.2 and 4.5.1.1.2.b provide details regarding testing for specified LPSW valves. This information is relocated to the UFSAR Chapter 16. This information provides details of testing which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of testing. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. Changes to UFSAR Chapter 16 are controlled by 10 CFR 50.59. Additionally, ITS 5.5.9 provide requirements regarding this IST program. This change is consistent with the NUREG.

LA8 CTS 4.12.1.a provides details regarding filter testing. This
1 information is relocated to the ITS Bases. This information provides
1 details of system operation which are not directly pertinent to the
1 actual requirement, i.e., Definition, Limiting Condition for Operation
1 or Surveillance Requirement, but rather describe additional unnecessary
1 details such as an acceptable method of testing. Since these details
1 are not necessary to adequately describe the actual regulatory
1 requirement, they can be moved to a licensee controlled document without
1 a significant impact on safety. Placing these details in the Bases
1 provides adequate assurance that they will be maintained. Changes to
1 ITS Bases are controlled by the Technical Specification Bases Control
1 Program included in ITS Section 5.5.

1 LA9 Not used.

LA10 CTS 3.19.1.a, 3.19.2.a, 3.19.1.b, 3.19.2.b, and 3.19.3 provide details regarding requirements for an OPERABLE ECCW siphon header. Footnote 6 to CTS Table 4.1-2 provides details regarding testing the Emergency Condenser Circulating Water System. This information is relocated to the ITS Bases. This information provides details of system operation which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of testing. Since these details are not necessary

to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in the Bases provides adequate assurance that they will be maintained. Changes to ITS Bases are controlled by the Technical Specification Bases Control Program included in ITS Section 5.5.

LA11 CTS 4.12.1.a includes requirements for proving all louvers in the Control Room Booster Fan Trains are OPERABLE. This information is relocated to the ITS Bases. This information provides details of system operation which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of testing. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in the Bases provides adequate assurance that they will be maintained. Changes to ITS Bases are controlled by the Technical Specification Bases Control Program included in ITS Section 5.5.

LA12 CTS 3.14 and 4.18 specify requirements for snubbers. These requirements are relocated to UFSAR Chapter 16. This requirements are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of testing. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. Changes to UFSAR Chapter 16 are controlled by 10 CFR 50.59. This change is consistent with the NUREG.

LA13 CTS 4.9.2 requires testing automatic valves in the EFW System in accordance with CTS 4.0.4. CTS 4.0.4 requires inservice testing of ASME Code Class 1, 2 and 3 pumps and valves be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50 Section 50.55a(g)(4) to the extent practicable within the limitations of design, geometry and materials of construction of the components. This information is relocated to the Inservice Testing Program. This information provides details of testing requirements which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of testing. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The Inservice Testing Program is controlled by 10 CFR 50.55a. This change is consistent with the NUREG.

LA14 CTS Table 4.1-2, Item 12 and associated Note 7 provides details regarding equipment functional testing requirements for equipment actuated by the MSLB Feedwater isolation feature. These requirements **are relocated** to UFSAR Chapter 16. This requirements are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of testing. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. Changes to UFSAR Chapter 16 are controlled by 10 CFR 50.59. This change is consistent with the NUREG.

ITS Section 3.7

ID 65

Subject Remove JFD associated with approved ONS generic change ONS-004 and annotate NUREG Markup with TSTF number.

TSTF-260

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.7.5.1	8 the non-automatic Verify each EFW manual, power operated, and ~~automatic~~ valve in each water flow path and in ~~both~~ steam supply flow paths to the ~~steam~~ turbine driven pumps, that is not locked, sealed, or otherwise secured in position, is in the correct position. 2	31 days 4	Doc M5
SR 3.7.5.2	40 ~~-----NOTE----- Not required to be performed for the turbine driven EFW pumps, until [24] hours after reaching [800] psig in the steam generators.~~ Verify the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program ~~[31] days on a STAGGERED TEST BASIS~~ TSTF 101	4.9.1
SR 3.7.5.3	-----NOTE 8----- 1. ~~Not required to be performed until [24] hours after reaching [800] psig in the steam generators.~~ 10 ~~2.~~ Not ~~applicable~~ required to be met in ~~MODE 4~~ MODES 3+4. 14 Verify each EFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[~~18~~]~~ months	3.4.3.e T4.1-2 Item 11 4.9.3

(continued)

LPSW System
~~SWS~~ CTS
3.7.~~8~~ 7

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	12 ~~6~~ hours (22) 3.3.7.b 60 ~~36~~ hours (24) 3.3.7.b

SURVEILLANCE REQUIREMENTS

Doc M10

Supp. 1

SURVEILLANCE		FREQUENCY
SR 3.7.~~8~~7.1	------------------NOTE------------------ Isolation of LP SW~~S~~ flow to individual components does not render the LP SW~~S~~ System inoperable. -- Verify each LP SW~~S~~ manual, power operated, ~~and~~ ~~automatic~~ non-automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days TSTF-260
SR 3.7.~~8~~7.2	Verify each LP SW~~S~~ automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[~~18~~]~~ months (1) 4.5.1.1.2.a.2 Note 1 to 4.5.1.1.2.a.2
SR 3.7.~~8~~7.3	Verify each LP SW~~S~~ pump starts automatically on an actual or simulated actuation signal.	~~[~~18~~]~~ months (1) 4.5.1.1.2.A.2

feedwater control valves (MFCV) and both startup feedwater control valves (SFCV) is assumed in the accident analysis. Therefore, NUREG Specification 3.7.3 is modified to remove ACTIONS associated with the MFSVs (ACTION A. ACTION D is eliminated since the ONS current licensing basis does not require two valves in the same flow path to be OPERABLE.

35 Not used.

36 Not used.

37 The Note to NUREG 3.7.4 Action A is not adopted. This Note permits entry into a MODE or other specified condition of the LCO's Applicability when the LCO is not met. Since the ADVs are credited in certain small break LOCA analysis, it is considered inappropriate to permit changes in MODE or other specified conditions of the LCO's Applicability when the LCO is not met.

38 Not used.

39 TSTF-100, which modifies NUREG 3.7.4 Required Action B.1 to state, "Restore all but one ADV line to OPERABLE status." is not adopted. The change represented by TSTF-100 is potentially confusing and unnecessary, since ONS has only two ADV lines. The TSTF-100 modified Required Action implies more than two ADV lines are provided.

40 The Note to SR 3.7.5.2 is not adopted. The ONS design permits testing the turbine driven EFW pump using auxiliary steam supplied from another ONS unit or the plant auxiliary boiler.

1 41 Not used.

42 The Applicability to NUREG 3.7.14 is expanded to include "During movement of a cask over the spent fuel pool." Action A is modified to also require suspending movement of the cask over the spent fuel pool. The Note to 3.7.14 Required Action A.1 is moved to precede both Required Action A.1 and Required Action A.2. This is necessary to reflect the current licensing basis for Oconee. ONS is licensed to permit placing a shipping cask in the spent fuel pool. The water level in the spent fuel pool is an initial assumption for the cask drop accident.

1 43 Not used.

1 44 Not used.

45 NUREG SR 3.7.2.1 is modified to reflect the TSV closure time associated with closure channel A. A second surveillance, SR 3.7.2.2, is added to verify the TSV closure time associated with channel B. The ONS design provides two means of closing each TSV. Each closure mechanism results in different closure times. Closure channel A actuates the "disk dump" closure mechanism which closes the associated TSV in ≤ 1 second.

ITS Section 3.7

ID 66

Subject Remove JFD associated with approved ONS generic change ONS-007 and annotate NUREG Markup with TSTF number 255.

Supp. 2

TSTF-255

3.7 PLANT SYSTEMS

3.7.~~16~~ 13 ~~Spent~~ Fuel Assembly Storage

New or

LCO 3.7.~~16~~ 13 The combination of initial enrichment and burnup of each spent fuel assembly stored in ~~[Region 2] shall be within the acceptable [burnup domain] of Figure 3.7.16-1 or in accordance with Specification 4.3.1.1.~~ Insert 3.7-36A 33 — 3.8.16.a, 3.8.16.b

APPLICABILITY: Whenever any fuel assembly is stored in ~~[Region 2] of~~ the spent fuel pool. 55 — 3.8.16.c

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ---------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Initiate action to move the noncomplying fuel assembly ~~from [Region 2].~~ to the correct location. 33	Immediately

Doc A22

3.8.17

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.~~16~~ 13.1 Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with ~~Figure 3.7.16-1 or Specification 4.3.1.1.~~ Insert 3.7-36B	Prior to storing the fuel assembly in ~~[Region 2]~~ the spent fuel pool.

Doc M7

33

INSERT 3.7.37A

Replace with CTS TABLE 3.8.1

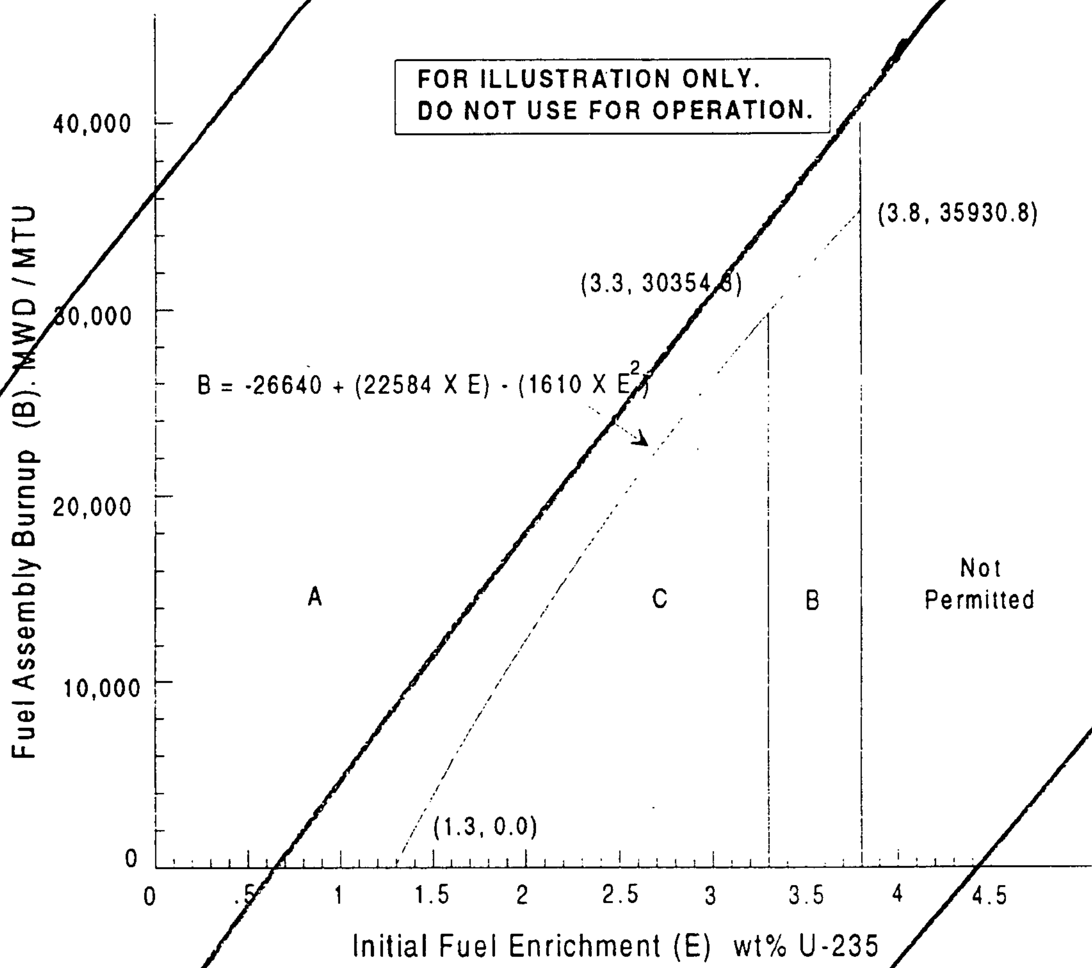
1

CTS

T3.8.1

Supp.1

TSTF-255



Category "A" Fuel - May be located anywhere within the storage racks.
Category "B" Fuel - Shall only be located adjacent to Category "A" Fuel or water holes within the storage racks.
Category "C" Fuel - Shall not be located adjacent to Category "B" Fuel.

33

Minimum Qualifying Figure 3.7.~~16~~ 13-1 (page 1 of 1) Initial
Burnup versus Enrichment Curve for Unrestricted Storage in the
Spent Fuel ~~Storage Racks~~ Pool

Units 1 and 2

2

feedwater control valves (MFCV) and both startup feedwater control valves (SFCV) is assumed in the accident analysis. Therefore, NUREG Specification 3.7.3 is modified to remove ACTIONS associated with the MFSVs (ACTION A. ACTION D is eliminated since the ONS current licensing basis does not require two valves in the same flow path to be OPERABLE.

35 Not used.

36 Not used.

37 The Note to NUREG 3.7.4 Action A is not adopted. This Note permits entry into a MODE or other specified condition of the LCO's Applicability when the LCO is not met. Since the ADVs are credited in certain small break LOCA analysis, it is considered inappropriate to permit changes in MODE or other specified conditions of the LCO's Applicability when the LCO is not met.

38 Not used.

39 TSTF-100, which modifies NUREG 3.7.4 Required Action B.1 to state, "Restore all but one ADV line to OPERABLE status." is not adopted. The change represented by TSTF-100 is potentially confusing and unnecessary, since ONS has only two ADV lines. The TSTF-100 modified Required Action implies more than two ADV lines are provided.

40 The Note to SR 3.7.5.2 is not adopted. The ONS design permits testing the turbine driven EFW pump using auxiliary steam supplied from another ONS unit or the plant auxiliary boiler.

1 41 Not used.

42 The Applicability to NUREG 3.7.14 is expanded to include "During movement of a cask over the spent fuel pool." Action A is modified to also require suspending movement of the cask over the spent fuel pool. The Note to 3.7.14 Required Action A.1 is moved to precede both Required Action A.1 and Required Action A.2. This is necessary to reflect the current licensing basis for Oconee. ONS is licensed to permit placing a shipping cask in the spent fuel pool. The water level in the spent fuel pool is an initial assumption for the cask drop accident.

1 43 Not used.

1 44 Not used.

45 NUREG SR 3.7.2.1 is modified to reflect the TSV closure time associated with closure channel A. A second surveillance, SR 3.7.2.2, is added to verify the TSV closure time associated with channel B. The ONS design provides two means of closing each TSV. Each closure mechanism results in different closure times. Closure channel A actuates the "disk dump" closure mechanism which closes the associated TSV in ≤ 1 second.

Closure channel B actuates the closure mechanism utilizing a restricting orifice and closes the associated TSV in ≤ 15 seconds. These closure times are consistent with the current licensing basis for TSV closure.

46 The ONS design uses a portion of the Condenser Circulating Water System to provide a flow path from Lake Keowee to the suction of the Low Pressure Service Water Systems by utilizing "siphon headers". Siphon header OPERABILITY requires associated vacuum pumps and seal water supply headers. NUREG Specification 3.7.9 is modified to encompass the CLB requirements for the ECCW siphon headers.

47 The current licensing basis permits 12 hours to place a unit in Hot Shutdown (equivalent to ITS MODE 3) and 60 hours to place the Unit in a condition outside the Applicability of the Specification when an LCO is not met. To maintain consistency with current procedures, training and staffing requirements, the 12 hours permitted to place a unit in MODE 3 and 60 hours to place the Unit in a condition outside the Applicability of the specification is retained in the ITS.

48 Consistent with the CLB, Lake Keowee water level limits are specified in UFSAR Chapter 16. NUREG SR 3.7.9.1 is modified to require verifying lake level within limits. The ITS Bases for this SR refers to UFSAR Chapter 16 for the actual limits. Lake level requirements are maintained in UFSAR Chapter 16 since the values are subject to change resulting from modifications and changes in operating practices, which may impact LPSW System flow requirements.

49 ITS SR 3.7.8.1 is added to require periodic verification that required ESV pumps are in operation. Periodic verification that required ESV pumps are operating is appropriate since operation of required ESV pumps is a requirement for OPERABILITY of an ECCW siphon header. ITS SR 3.7.8.4 is adopted to provide for periodic verification that non-automatic valves in the ECCW siphon header, required ESV and required SSW flow paths are in the correct position. Periodic verification that valves in the ECCW siphon header, required ESV and required SSW flow paths are in the correct position is appropriate since correct valve positions are a requirement for OPERABILITY of an ECCW siphon header. Consistent with the CLB, ITS SR 3.7.8.5, SR 3.7.8.6, SR 3.7.8.7, SR 3.7.8.8 and SR 3.7.8.9 are adopted to retain requirements encompassed in CTS Table 4.1-2, Items 7 and 13.

50 Bracketed NUREG SR 3.7.9.3 is not adopted. NUREG SR 3.7.9.3 requires operation of each cooling tower fan every 31 days. The ONS design does not include cooling towers. Consequently, there are no cooling tower fans.

51 The brackets are removed and the Completion Time for NUREG 3.7.2 Required Actions A.1 and C.1 of 8 hours is adopted. There is no CLB for these value since a similar CTS requirement does not exist. Eight hours is considered reasonable considering the low probability of an event in

this time period and considering the TSVs isolate a closed system which provides an additional barrier against releases.

52 **The brackets are removed and the plant specific value of one provided in** LCO 3.7.4. The ONS design provides one atmospheric dump valve line per steam generator.

53 Bracketed NUREG SR 3.7.10.5 is not adopted since there is no comparable CTS requirement. (NUREG-1430 provides no Bases information for this SR.)

1 54 Not used.

55 The Applicability for NUREG 3.7.16 is modified to eliminate the bracketed reference to Region 2. Although restrictions upon storage location exist in the spent fuel pool, the LCO is applicable to storage anywhere in the spent fuel pool.

58 The portion of NUREG SR 3.7.1.1 requiring MSRVs settings to be within 1% of the setpoint following testing is not adopted. This modification to the SR is made to reduce the potential for confusion. The current licensing basis for ONS requires the MSRVs setting to be within 1% of the setpoint at all times. Retention of the deleted wording could imply that the settings may be within some different tolerance band at other times.

59 The brackets are removed and the closure time specified in NUREG SR 3.7.3.1 is changed to 25 seconds. The value of 25 seconds for closure of the MFCVs and SFCVs is consistent with current plant analysis.

60 The brackets are removed and the Frequency for NUREG 3.7.4.2 is adopted. There is no CTS value for SR 3.7.4.2 since it is not a current requirement. The Frequency of 18 months for the surveillance of the ADV block valve is reasonable and is consistent with the Frequency for the ADV valves.

61 The second entry Condition for NUREG 3.7.9 Action B is not adopted. The current licensing basis for one inoperable ECCW siphon allows 60 hours to be in cold shutdown. This Completion Time is retained for ITS 3.7.8 Required Action B.2. The Completion Time of 60 hours is not considered to be appropriate for the second entry Condition in NUREG 3.7.9 Action B. There is no CTS provision for this condition. With the ECCW inoperable for reasons other than an inoperable ECCW siphon, CTS 3.0 applies requiring the unit be placed in Hot Shutdown within 12 hours and cold shutdown in 36 hours. For this Condition, the non-adoption of the second entry Condition for NUREG 3.7.9 Action B results in entry into LCO 3.0.3 which is comparable to CTS 3.0.

62 The brackets are removed and the bracketed NUREG value of 0.10 μCi/gm adopted. This is the value for secondary specific activity assumed in the UFSAR Chapter 15 reanalysis. (The methodology for this reanalysis has been submitted separately for NRC review. Refer to submittal letter for additional information.)

63 Specification 3.7.15 is added to the NUREG to capture CTS requirements regarding decay time restrictions in specified rows in the SFPs prior to movement of casks in the SFP. This change is necessary since the ONS design requires placing a cask in the SFP, in proximity to spent fuel stored in the SFP.

64 The brackets are removed and the bracketed NUREG value of 90°F adopted. This is the value for service water inlet temperature assumed in the accident and transient.

ITS Section 3.9

ID 35

Subject Revise ITS submittal to incorporate approved TSTF-96, R1 - No change to ITS since it replaces draft generic change ONS-009.

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2 a. ~~Two~~ ONE source range neutron flux monitor~~s~~ shall be OPERABLE, and 3.8.2
b. One additional source range neutron flux monitor shall be OPERABLE during CORE ALTERATIONS and 3.8.2
during positive reactivity additions. 3.8.2

6

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ~~[~~required~~]~~ source range neutron flux monitor inoperable during CORE ALTERATIONS or positive reactivity additions.	A.1 Suspend CORE ALTERATIONS. AND A.2 Suspend positive reactivity additions.	Immediately Immediately
B. ~~Two [required]~~ No OPERABLE source range neutron flux monitor~~s inoperable~~.	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status. AND B.2 Perform SR 3.9.1.1.	Immediately ~~4 hours~~ ~~AND~~ Once per 12 hours ~~thereafter~~

3.8.9 6 3.8.9

3.8.9

TSTF 96, R1

Doc M6

Supp. 1

condition is subject to change, i.e., during CORE ALTERATIONS and positive reactivity additions, two monitors are required to provide independent and redundant monitoring capability of the reactivity changes in the core.

1 7 Not used.

8 Not used.

9 Not used.

10 NUREG SR 3.9.4.1 was modified to remove reference to a minimum decay heat removal system flow rate. The CTS does not establish a minimum flow requirement. The actual minimum flow rate is administratively controlled in operating procedures. Operation of the system is sufficient to ensure adequate mixing of the coolant to prevent boron stratification. Adequate heat removal is a function of a number of system parameters in addition to a minimum volumetric flow rate. As such, the operator has direct indication of the adequacy of the decay heat removal system in removing decay heat and adjustments would be made based on the trended indications. A minimum volumetric flow rate is not necessary to support this determination or the possible corrective actions. Although not done for this reason, this change establishes consistency between this SR and numerous ITS Section 3.4 SRs requiring verification of DHR operation.

11 Not used.

12 The frequency of 18 months for NUREG SR 3.9.3.2 is modified to partially retain the CLB. DPC considers the NUREG frequency of 18 months to be inappropriate for ONS since the purge valves remain isolated for extended periods of time during unit operation. CTS 3.8.10 requires the reactor building purge isolation capability to be verified immediately prior to refueling operations. ITS SR 3.9.3.2 requires verification that each reactor building purge supply and exhaust valve actuates to the correct position on an actual or simulated actuation signal once each refueling outage prior to CORE ALTERATIONS or movement of irradiated fuel assemblies inside containment. Requiring performance of SR 3.9.3.2 prior to CORE ALTERATIONS or movement of irradiated fuel assemblies within containment represents a reasonable relaxation of the current requirement of " . . . immediately prior to . . . " surveillance frequency and remains within the NUREG specified frequency of 18 months.

13 LCO 3.9.1 is modified to eliminate the reference to a "refueling cavity." The ONS design does not provide a separate refueling cavity. The term "fuel transfer canal" is used to refer to what is typically referred to as the refueling cavity.

14 LCO 3.9.6 is modified to reflect the plant specific assumption regarding water level. The analysis for the fuel handling accident uses a value of 21.34 feet of water for a fuel handling accident. The Applicability for Specification 3.9.4 and 3.9.5, and 3.9.5 Required Action A.2 are modified accordingly to maintain consistency with the related change to LCO 3.9.6. Although the Applicability of Specification 3.9.4 and 3.9.5 are not based directly upon the fuel handling accident analysis the values are the same as is explained in the Bases to the Applicability for Specification 3.9.4. With a water level of 21.34 feet the volume of water still provides a substantial heat sink.

ITS Section 3.9

ID 67

Subject Remove JFD associated with approved ONS generic change ONS-008 and annotate NUREG Markup with TSTF number 214.

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 ~~a.~~ Boron concentrations of the Reactor Coolant System~~,~~ and the refueling canal ~~and the refueling cavity~~ shall be maintained within the limit specified in the COLR. (2) 3.8.4 (13)

Supp 1 ~~b. Boron concentration shall not be reduced unless reactor coolant is circulating.~~ TSTF 214

3.8.4

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately	3.8.9
	AND		
	A.2 Suspend positive reactivity additions.	Immediately	3.8.9
	AND		
	A.3 Initiate action to restore boron concentration to within limit.	Immediately	3.8.9

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	72 hours	T4.1-3 Item 1b

TECHNICAL SPECIFICATIONS

NOTE: The first four justifications for these changes from NUREG-1430 were generically used throughout the individual LCO section markups. Not all generic justifications are used in each section.

1 The brackets are removed and the proper plant specific information or value is provided.

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

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

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

1 5 Not used.

6 CTS 3.8.2 established the requirements for (source range) neutron flux monitoring in MODE 6. This Specification requires one OPERABLE monitor when "core geometry is not being changed," and two OPERABLE monitors "whenever core geometry is being changed." The NUREG is modified to reflect these CTS requirements. ITS LCO 3.9.2.a requires one source range neutron flux monitor be OPERABLE during the LCO Applicability (MODE 6). ITS LCO 3.9.2.b requires one additional source range neutron flux monitor be OPERABLE during CORE ALTERATIONS and during positive reactivity additions.

NUREG 3.9.2 Condition A is modified to establish a Condition that was entered when one of the required source neutron flux monitors is inoperable "during CORE ALTERATIONS or positive reactivity additions." NUREG Required Actions A.1 and A.2 replicate CTS 3.8.9 requirements for this Condition. With this change, RA A.1 and A.2 provide the appropriate Required Actions for this Condition.

NUREG 3.9.2 Condition B is modified to establish a Condition that is applicable when there are no OPERABLE source range neutron flux monitors. Required Action B.1, in addition to Required Actions A.1 and A.2 if during CORE ALTERATIONS or during positive reactivity additions, replicates the CTS 3.8.9 requirements for this condition.

These changes maintain the requirements of the CTS while providing adequate monitoring capability of changes in the core's neutron flux. When core reactivity conditions are stable, i.e., no CORE ALTERATIONS or positive reactivity additions in progress, one neutron source range monitor is adequate. During the conditions when the core's reactivity

3.10 Standby Shutdown Facility

ITS Section 3.10

ID 29

Subject Revise ITS as appropriate to consider the fact that the SSF Submersible Pump is not a QA grade pump. Adds separate SR for Submersible Pump.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.10.1.11	Verify for required SSF battery that the cell to cell and terminal connections are clean, tight and coated with anti-corrosion material.	12 months
SR 3.10.1.12	Verify battery capacity of required battery is adequate to supply, and maintain in OPERABLE status, the required maximum loads for the design duty cycle when subjected to a battery service test.	12 months
SR 3.10.1.13	Perform CHANNEL CALIBRATION for each required SSF instrument channel.	18 months
SR 3.10.1.14	Verify SSF valves operate through one complete cycle of full travel.	In accordance with the Inservice Testing Program
SR 3.10.1.15	-------------------NOTES------------------- Not applicable to the SSF submersible pump. --- Verify the developed head of each required SSF pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.10.1.16	Verify the developed head of the SSF submersible pump at the flow test point is greater than or equal to the required developed head.	2 years

BASES

SURVEILLANCE REQUIREMENTS

SR 3.10.1.12 (continued)

The Surveillance Frequency for this test is 12 months. This Frequency is considered acceptable based on operating experience.

SR 3.10.1.13

CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test 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 to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis. This Frequency is justified by the assumption of an 18 month calibration interval to determine the magnitude of equipment drift in the setpoint analysis.

SR 3.10.1.14

Cycling the SSF valves through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will operate when required. These valves are required to operate to ensure the required flow path.

The specified Frequency is in accordance with the IST Program requirements. Operating experience has shown that these components usually pass the SR when performed at the IST Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.10.1.15

This SR requires the SSF pumps to be tested in accordance
with the IST Program. The IST verifies the required flow
1 rate at a discharge pressure to verify OPERABILITY. The SR
1 is modified by a note indicating that it is not applicable
1 to the SSF submersible pump.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.10.1.15 (continued)

The specified Frequency is in accordance with the IST Program requirements. Operating experience has shown that these components usually pass the SR when performed at the IST Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

1 SR 3.10.1.16
1
1 This SR requires the SSF submersible pump to be tested on a
1 2 year Frequency and verifies the required flow rate at a
1 discharge pressure to verify OPERABILITY.
1
1 The specified Frequency is based on the pump being not QA
1 grade and on operating experience that has shown it usually
1 passes the SR when performed at the 2 year Frequency.

REFERENCES

1. UFSAR, Section 9.6.
2. Oconee Probabilistic Risk Assessment.
3. 10 CFR 50.36.
4. IEEE-450-1987.
5. Regulatory Guide 1.9, Rev. 0, December 1974.
6. ASTM-D975-1981.
7. NRC Letter from L. A. Wiens to J. W. Hampton, "Safety Evaluation for Station Blackout (10 CFR 50.63) - Oconee Nuclear Station, Units 1, 2, and 3," dated March 10, 1992.
8. NRC Letter from L. A. Wiens to J. W. Hampton, "Supplemental Safety Evaluation for Station Blackout (10 CFR 50.63) - Oconee Nuclear Station, Units 1, 2, and 3," dated December 10, 1992.
9. NRC Letter from L. A. Wiens to H. B. Tucker, "Safety Evaluation Report on Effect of Tornado Missiles on Oconee Emergency Feedwater System," dated July 28, 1989.

4.20 STANDBY SHUTDOWN FACILITY

A1

Applicability

Applies to the periodic surveillance testing requirements for the Standby Shutdown Facility (SSF) consisting of the SSF Auxiliary Service Water, SSF Portable Pumping System, SSF RC Makeup Systems, associated instrumentation, electrical generation and distribution, support systems, and the interfaces with normal in-plant systems.

Objective

To verify that the systems and components associated with the SSF are operable.

Specification

4.20.1 SSF Pumps and Valves

Supp 1

A6

a. Inservice testing of SSF ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50, §50.55a(g)(4) to the extent practicable within the limitations of design, geometry and materials of construction of the components with the exception as permitted by Specification 4.20.1.c.

A6 SR 3.10.1.14

A6 SR 3.10.1.15

b. In the event that a pump or valve is determined to be inoperable by the performance of a surveillance test, then actions shall be taken for the affected system as required by Specification 3.18. A5

Supp. 1 SR 3.10.1.16 c. Inservice testing of the submersible pump for the SSF Portable Pumping System will be performed on a two (2) year frequency and will consist of testing developed head and flow.

4.20.2 SSF Instrumentation

a. The frequency and type of surveillance required for SSF instrumentation shall be as stated in Table 4.20-1. A1

b. In the event that an instrument is determined to be inoperable by the performance of a surveillance test, then actions shall be taken for the affected system as required by Specification 3.18. A5

3.18.2 through 3.18.7. This provision is captured by the ITS 3.10.1 ACTIONS Note which states LCO 3.0.4 is not applicable. LCO 3.0.4 does not allow entry into a MODE or other specified condition in the Applicability when an LCO is not met. Therefore, the ACTIONS note, which takes exception to the LCO 3.0.4 requirement, captures the CTS provision. This change represents no actual change in requirements, only a change in presentation of requirements.

A5 CTS 4.20.1.b, CTS 4.20.2.b, CTS 4.20.3.a.5, CTS 4.20.3.b.5 states that when equipment is inoperable due to the performance of a surveillance test, then actions shall be taken as required by Specification 3.18. In the ITS the SRs must be met for the LCO to be considered met as required by ITS SR 3.0.1. Therefore, statements of this type are not necessary and are deleted. This change represents no actual change in requirements, only a change in presentation of requirements.

A6 CTS 4.20.1.a requires Inservice Testing (IST) of SSF ASME Code Class 1,
2, and 3 pumps and valves be performed in accordance with Section XI of
the ASME Boiler and Pressure Vessel Code as required by
1 10 CFR 50.55a(g)(4). ITS SR 3.10.1.14 requires verification that each
SSF valve cycles through one complete cycle of full travel at a
1 Frequency in accordance with the IST Program. ITS SR 3.10.1.15 requires
verification that the developed head of each SSF pump at the flow test
point be greater than or equal to the required developed head at a
Frequency in accordance with the IST Program. The inservice testing
1 required by CTS 4.20.1.a (and 10 CFR 50.55a(g)(4) includes requirements
to verify that each SSF pump's developed head at the flow test point is
greater than or equal to the required developed head and to verify that
each SSF valve cycles through one complete cycle of full travel. This
portion is retained as ITS SRs 3.10.1.14 and 15. Therefore, this
portion of the requirement is equivalent and the addition of these
1 explicit requirements are an administrative change. The general
1 requirement for testing of ASME Code Class 1, 2, and 3 pumps and valves
1 in accordance with Section 11 of the ASME Boiler and Pressure Vessel
1 Code and applicable addenda as required by 10 CFR 50.55a(g)(4) is not
1 retained in ITS since it duplicates applicable regulations. Therefore,
1 elimination of the CTS requirement is administrative and is consistent
with comparable NUREG requirements.

A7 CTS 4.20.3.a.2 requires the SSF DG to be operated for at least 60 minutes with a load greater than or equal to 3000 kw. The comparable ITS requirement (SR 3.10.1.9) specifically requires the DG be synchronized and loaded and operated. Since the DG must be synchronized and loaded regardless of whether currently specified, the proposed addition of the more prescriptive surveillance is considered an administrative change and is consistent with comparable NUREG SR 3.8.1.3 requirements.

A8 CTS 4.20.3.a.3 requires the diesel fuel oil from the day tank and underground tank be sampled and analyzed for viscosity, water and sediment in accordance with applicable ASTM Specifications for Diesel

Fuel Oil quarterly. ITS SR 3.10.1.8 specifies this testing be performed every 92 days in accordance with and maintained within limits of the Diesel Fuel Oil Testing Program. The details of sampling and analysis are moved to Section 5.0 and discussed in the discussion of changes for Section 5.0. This relocation to Section 5.0 is considered administrative. This change is consistent with comparable NUREG SR 3.8.3.3 requirements.

A9 CTS 4.20.3.a.2 requires the SSF DG to be operated for at least 60 minutes with a load greater than or equal to 3000 kw. It further states that this test may be preceded by an engine prelube period and/or other warm-up procedures recommended by the manufacturer. Note 1 is added to SR 3.10.1.9 to allow gradual loading. Note 2 to SR 3.10.1.9 is added to clarify that momentary transients outside the load range do not invalidate this Surveillance. These practices are currently being used, and have been specifically added for clarity. These changes are consistent with the NUREG SR 3.8.1.3 and are considered administrative in nature.

A10 CTS 4.20.3.a.2.b requires a test to verify that the fuel oil transfer pump starts and transfers fuel from the storage system to the day tank in accordance with CTS 4.20.1.a, which cites IST requirements. ITS SR 3.10.1.7 provides equivalent requirements at a Frequency in accordance with the IST Program. As such, the requirements are equivalent and the change is considered administrative. This change is consistent with comparable NUREG SR 3.8.1.6 requirements.

A11 CTS Table 4.20.1-1, Note 2 indicates that the Reactor Coolant System (RCS) Temperature instrument string is normally aligned through a transfer isolation device to each Unit control room and thus, the CHANNEL CHECK is performed in accordance with Specification 4.1, Table 4.1-1, Item 7, which requires a shiftly CHANNEL CHECK. Note 2 further clarifies that the RCS Temperature instrument string to the SSF Control Room will be calibrated each refueling outage. The comparable CHANNEL CHECK requirement in ITS is SR 3.3.1.1 for Function 2 of Specification 3.3.1. SR 3.10.1.1 excludes this instrument from CHANNEL CHECK requirements. As such, the CTS and ITS requirements are equivalent and the change is considered administrative.

A12 CTS 3.18.1.b specifies that the provisions of Specification 3.0 do not apply. Specification 3.0 is similar to ITS 3.0.3 and addresses situations which involve conditions in excess of those addressed by an LCO or associated action statements. This CTS provision exists to prevent requiring the unit be placed in cold shutdown (equivalent to ITS MODE 5) since the CTS 3.18 is only applicable above 250°F (equivalent to ITS MODE 3). Additionally, since ITS 3.10.1 Actions encompass the range of potential Conditions, elimination of CTS 3.18.1.b does not result in a technical change. This change is therefore administrative in nature and is consistent with the NUREG.

A13 CTS 4.20.3.b.3.a requires verification annually that the cells, end-cell plates and battery racks show no visual indication of structural damage or degradation. ITS SR 3.10.1.10 (comparable to NUREG SR 3.8.4.3) requires verification every 12 months that battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration that could degrade battery performance. The CTS acceptance criteria of "no visual indication of structural damage or degradation" is equivalent to the ITS acceptance criteria of "no visual indication of physical damage or abnormal deterioration that could degrade battery performance." The ITS wording of the surveillance does more clearly acknowledge that physical damage or abnormal deterioration has to be of the type that degrades battery performance. However, the CTS wording refers to structural damage or degradation and would allow such an assessment. As such, the CTS and ITS requirements are equivalent and the change is considered administrative. The change is consistent with NUREG SR 3.8.4.3 as modified by TSTF-038.

A14 CTS 4.20.3.b requires the SSF battery to be declared inoperable when compliance with periodic inspection requirements are not corrected. Comparable ITS 3.10.2 Required Action B.1 requires the SSF Power System to be declared inoperable since an inoperable required SSF battery makes the SSF Power System inoperable. ITS LCO 3.10.2, which requires the battery cell parameters for SSF Batteries to be within limit, is
1 applicable only when the SSF Power System battery is required to be OPERABLE. Declaring the SSF Power System inoperable forces entry into the appropriate action of ITS 3.10.1 without the need for evaluating and concluding that the SSF battery is required to support OPERABILITY of the SSF Power System. Since there are no changes in requirements, the proposed change is administrative.

A15 CTS 4.20.3.b requires the SSF battery to be declared inoperable when compliance with periodic inspection requirements are not corrected. When cell parameters do not meet the Category A or B limits but continue to meet the Category C limits the SSF battery remains OPERABLE. ITS 3.10.2 ACTIONS Note is added to specify that LCO 3.0.4 is not applicable. When the required SSF battery is OPERABLE, restricting MODE changes until Category A and B limits are met is unreasonable and not required by the CTS. In addition, CTS 3.18.8 provides a similar exception for the SSF Power System, whose OPERABILITY is supported by the SSF Battery (Refer to DOC A4 for this section). Since this Note is consistent with the CTS 4.20.3.b, the addition of the note is administrative.

detail is not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for DG OPERABILITY. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable. Changes to the Bases are controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.10.1.11	Verify for required SSF battery that the cell to cell and terminal connections are clean, tight and coated with anti-corrosion material.	12 months	4.20.3.b.3.b
SR 3.10.1.12	Verify battery capacity of required battery is adequate to supply, and maintain in OPERABLE status, the required maximum loads for the design duty cycle when subjected to a battery service test.	12 months	4.20.3.b.4
SR 3.10.1.13	Perform CHANNEL CALIBRATION for each required SSF instrument channel.	18 months	T 4.20-1 Items 1, 3, 4 & 5 COL "Calibrate"
SR 3.10.1.14	Verify SSF valves operate through one complete cycle of full travel.	In accordance with the Inservice Testing Program	DOC A6
Supp.1 SR 3.10.1.15	-------------------NOTES------------------- Not applicable to the SSF submersible pump. --- Verify the developed head of each required SSF pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program	DOC A6
Supp.1 SR 3.10.1.16	Verify the developed head of the SSF submersible pump at the flow test point is greater than or equal to the required developed head.	2 years	4.20.1.c

BASES

SURVEILLANCE REQUIREMENTS

SR 3.10.1.12 (continued)

The Surveillance Frequency for this test is 12 months. This Frequency is considered acceptable based on operating experience.

SR 3.10.1.13

CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test 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 to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis. This Frequency is justified by the assumption of an 18 month calibration interval to determine the magnitude of equipment drift in the setpoint analysis.

SR 3.10.1.14

Cycling the SSF valves through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will operate when required. These valves are required to operate to ensure the required flow path.

The specified Frequency is in accordance with the IST Program requirements. Operating experience has shown that these components usually pass the SR when performed at the IST Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.10.1.15

Supp. 1

This SR requires the SSF pumps to be tested in accordance with the IST Program. The IST verifies the required flow rate at a discharge pressure to verify OPERABILITY. The SR is modified by a note indicating that it is not applicable to the SSF submersible pump.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.10.1.15 (continued)

The specified Frequency is in accordance with the IST Program requirements. Operating experience has shown that these components usually pass the SR when performed at the IST Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Supp. 1

SR 3.10.1.16

This SR requires the SSF submersible pump to be tested on a 2 year Frequency and verifies the required flow rate at a discharge pressure to verify OPERABILITY.

The specified Frequency is based on the pump being not QA grade and on operating experience that has shown it usually passes the SR when performed at the 2 year Frequency.

REFERENCES

1. UFSAR, Section 9.6.
2. Oconee Probabilistic Risk Assessment.
3. 10 CFR 50.36.
4. IEEE-450-1987.
5. Regulatory Guide 1.9, Rev. 0, December 1974.
6. ASTM-D975-1981.
7. NRC Letter from L. A. Wiens to J. W. Hampton, "Safety Evaluation for Station Blackout (10 CFR 50.63) - Oconee Nuclear Station, Units 1, 2, and 3," dated March 10, 1992.
8. NRC Letter from L. A. Wiens to J. W. Hampton, "Supplemental Safety Evaluation for Station Blackout (10 CFR 50.63) - Oconee Nuclear Station, Units 1, 2, and 3," dated December 10, 1992.

(continued)

BASES

REFERENCES (continued)	9.	NRC Letter from L. A. Wiens to H. B. Tucker, "Safety Evaluation Report on Effect of Tornado Missiles on Oconee Emergency Feedwater System," dated July 28, 1989.

Table B 3.10.1-1 (page 1 of 1)
SSF Instrumentation

FUNCTION	REQUIRED CHANNELS PER UNIT
1. Reactor Coolant System Pressure	1
2. Reactor Coolant System Temperature (Tc)	1/Loop
3. Reactor Coolant System Temperature (Th)	1/Loop
4. Pressurizer Water Level	1
5. Steam Generator A & B Water Level	1/SG

ITS Section 4.0

ID 32

Subject Revise 4.0 DOC A3 to change "in" to "and" in the second sentence.

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering are in accordance with NUREG-1430. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable, by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1430. During Improved Technical Specification (ITS) development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1430. Since the design is already approved by the NRC, adding more detail does not result in a technical change.

A2 CTS 5.3.1.2 incorrectly refers to FSAR Figure 4.3-3 for the distribution CRAs and APSRs. There is no UFSAR Figure 4.4-3. The correct reference is UFSAR Figure 4.7. This change involves correcting the cross reference identification for a UFSAR figure and is therefore an administrative change. This information is part of details which are being relocated. Refer to LA3 for additional information.

A3 The statement in CTS 5.1.1 referring to Figure 2.1-4 of the Oconee FSAR as showing the plan of the site is deleted. This is merely a statement
1 of fact regarding the location of the plan of the site and need not be in Technical Specifications. Therefore, the removal of this reference is administrative. Additionally, since this information is already in the UFSAR, any changes will continue to be controlled in accordance with 10 CFR 50.59. This change is consistent with the NUREG.

ITS Section 4.0

ID 51

Subject Incorporate TSTF-123, R1 - minor change to wording of 4.2.2

4.0 DESIGN FEATURES

4.1 Site Location

The Oconee Nuclear Station is approximately eight miles northeast of Seneca, South Carolina. The minimum distance from the reactor center line to the boundary of the exclusion area and to the outer boundary of the low population zone, as defined in 10 CFR 100.3, shall be one mile and six miles respectively.

4.2 Reactor Core

4.2.1 <u>Fuel Assemblies</u>

The reactor shall contain 177 fuel assemblies. Each assembly shall consist of a matrix of zirconium alloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 <u>Control Assemblies</u>

The reactor core shall contain 61 full-length CONTROL ROD
1 Assemblies (CRA) and 8 APSR assemblies. The full-length CRA and
APSR assemblies shall conform to the design described in the UFSAR
or reload report.

4.3 Fuel Storage

4.3.1 <u>Criticality</u>

The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal U-235 enrichment of 5.0 weight percent;

(continued)

4.0 DESIGN FEATURES

5.1.1

4.1 Site Location ~~[Text description of site location.]~~ Insert 4.0-1A (1)

4.2 Reactor Core

Zirconium alloy (1)

4.2.1 Fuel Assemblies

5.3.1.1

The reactor shall contain ~~[177]~~ fuel assemblies. Each assembly shall consist of a matrix of ~~[Zircaloy or ZIRLO]~~ fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

clad (2)

TSTF 123, R1

lower case Assemblies full-length CONTROL ROD Assemblies (CRA) (4)

4.2.2 CONTROL RODS

5.3.1.2

TSTF-123, R1

Supp.1 APSR, assemblies

The reactor core shall contain [60] ~~safety and regulating~~ and ~~[8] axial power shaping CONTROL RODS. The [control] material shall be [silver indium cadmium, boron carbide, or hafnium metal] as approved by the NRC.~~ 61 (1)

TSTF-123, R1 Insert 4.0-1B

4.3 Fuel Storage

4.3.1 Criticality

(4)

~~4.3.1.1~~ The spent fuel storage racks are designed and shall be maintained with: 5.4.1 T3.8.3 T3.8.1

nominal

a. Fuel assemblies having a maximum U-235 enrichment of ~~[4.5]~~ weight percent; 5.0 (1)

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in ~~[~~Section 9.1 of the FSAR~~]~~. U (1) 5.4.1.1

(continued)

ITS Section 5.0

ID 10

Subject Remove prescriptive information regarding which organization fulfills the operating experience review requirement.

Specification 5.0

A1 <except as marked>

General Manager, Nuclear Services A27

[5.3.1] the ~~Nuclear Technical Services Manager~~ and as approved by the Site Vice President, Nuclear Operations.

The Operations Superintendent shall have a minimum of eight years of responsible nuclear or fossil station experience, of which a minimum of three years shall be nuclear station experience. A maximum of two years of the remaining five years of experience may be fulfilled by academic training, or related technical training, on a one-for-one time basis. The Operations Superintendent shall hold or have held a Senior Reactor Operator license. [5.2.2.F]

[5.3.1] The Shift Operations Manager shall have a minimum of eight years of responsible nuclear or fossil station experience, of which a minimum of three years shall be nuclear station experience. A maximum of two years of the remaining five years of experience may be fulfilled by academic training, or related technical training on a one-for-one time basis. The Shift Operations Manager shall hold a Senior Reactor Operator license. [5.2.2.F]

L3

Operations Superintendent or

[5.2.1.e] 6.1.1.5 The individuals who train the operating staff and those who carry out radiation protection and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.1.1.6 Minimum operating shift crew requirements shall be as specified in Table 6.1-1. A1

6.1.1.7 ~~Retraining and replacement of station personnel shall be in accordance with Section 5.5 of the ANSI/ANS-3.1-1978, "Selection and Training of Nuclear Power Plant Personnel."~~

6.1.1.8 ~~A training program for the fire brigade shall meet or exceed the requirements of Section 27 of the NFPA Code-1975, except that training sessions may be held quarterly.~~ LA5

6.1.1.9

LA17

[5.2.2.g] ~~The Shift Manager is an experienced SRO, who has been instructed in additional academic subjects, and will be assigned on-shift to provide the accident assessment capability.~~

Supp. 1

~~The operating experience assessment function will be provided by the Station Safety Review Group.~~ A5

Insert 5.2.2g from ITS M10

ADMINISTRATIVE CHANGES

A1 Formatting and renumbering are in accordance with NUREG-1430, Revision 1. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable, by plant operators as well as other users. The reformatting, renumbering. And rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1430, Revision 1. During Improved Technical Specification (ITS) development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additionally, information has been added to more fully describe each subsection. This wording is consistent with the NUREG. Since the wording is already approved by the NRC, adding more detail does not result in a technical change.

A2 The Oconee Nuclear Station (ONS) 1, 2 & 3 CTS Bases are being administratively deleted in their entirety in favor of the NUREG Bases.

A3 The portion of the Current Technical Specification (CTS) Definition 1.8.2 referencing CTS sections 6.4.6 and 6.4.7 is not retained in the ITS. Since the referenced sections are non-existent, this change is administrative and is consistent with the NUREG.

A4 The CTS 4.2.1 requirements are not retained in the ITS since they duplicate requirements contained in the regulations. Such duplication is unnecessary and results in additional administrative burden to revise the duplicate TS when these regulations are revised. Since removal of the duplication results in no actual change in the requirements, removal of the duplicative information is considered an administrative change. Further, changes to the requirements are controlled by the NRC. This change is consistent with the NUREG.

1 A5 TS 6.1.1.9 specifies that the operating experience assessment review
1 function be provided by the Station Safety Review Group. Duke Energy
1 currently implements its Operating Experience Program by a Nuclear
1 System Directive (NSD). The NSD specifies participants and
1 responsibilities. The specific requirement that the Station Safety
1 Review Group provide this function is not retained. Since the technical
1 requirement is to provide the function not who provides the function,
1 this deletion is considered administrative.

A6 Not used.

the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

LA4 CTS Table 6.1-1 specifies requirements for licensed and non-licensed personnel. This table specifies additional personnel than those required by 10 CFR 50.54(m) or ITS 5.2.2. The requirements for these additional personnel is relocated to UFSAR Chapter 16. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of the requirements to UFSAR Chapter 16 provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

LA5 CTS 6.1.1.7 and 6.1.1.8 contain requirements regarding training for station personnel and the fire brigade. These training requirements are relocated to UFSAR Chapter 16. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of the requirements to UFSAR Chapter 16 provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

1 LA6 Not used.

LA7 CTS 6.1.2.1, 6.4.1.o and 6.4.1.p, which delineate the PORC and Technical Review and Control requirements; CTS 6.4.2 which requires each procedure required by 6.4.1 to be reviewed by an appropriate manager or designee; and CTS 6.4.3 which addresses temporary changes to procedures; are relocated to the Quality Assurance Topical Report. CTS 6.1.2.1 describes review and control requirements for procedures, modifications, tests, experiments, reportable events, special reviews and investigations, and unplanned onsite releases. CTS 6.1.2.1 also requires records of the above activities be provided as necessary for required reviews. These requirements are also located in ANSI N18.7-1976. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of the requirements to the Quality Assurance Topical Report

provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the Quality Assurance Topical Report are controlled by the provisions of 10 CFR 50.54. This change is consistent with the NUREG.

LA8 CTS 6.1.3 delineates the requirements of the offsite review committee (Nuclear Safety Review Board (NSRB)). The CTS requirements specify the Function, Organization, Review, Audits, and Responsibilities and Authorities of the NSRB. The proposed change moves these requirements to the Quality Assurance Topical Report. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of these requirements to the Quality Assurance Topical Report provides reasonable assurance that these requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the QA Topical Report are controlled by the provisions of 10 CFR 50.54. This change is consistent with the NUREG.

LA9 CTS 6.2.1 and 6.2.2 details associated with certain licensee internal actions to be taken for reportable events is moved to the Quality Assurance Topical Report. The regulatory requirements for submittal of the reports are contained in 10 CFR 50.73. These details are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of these details to the Quality Assurance Topical Report provides reasonable assurance that the details are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these details are reduced. Therefore, relocation of these details is acceptable. Changes to the QA Topical Report are controlled by the provisions of 10 CFR 50.54. This change is consistent with the NUREG.

LA10 CTS 6.4.1.f requires procedures for a station survey following an earthquake. This requirement is relocated to UFSAR Chapter 16. This requirement is not required to be in the ITS to provide adequate protection of the public health and safety since relocation of the requirements to UFSAR Chapter 16 provides reasonable assurance that the requirements are implemented. This approach provides an effective level

of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of this detail is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

LA11 CTS 6.4.1.n requires providing procedures for implementing the Process Control Program (PCP). This change relocates the requirements of CTS 6.4.1.n to the Quality Assurance Topical Report. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of the requirements to the Quality Assurance Topical Report provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the QA Topical Report are controlled by the provisions of 10 CFR 50.54. This change is consistent with the NUREG.

LA12 CTS 6.4.1.i includes details regarding the content of procedures for controlling ph in recirculated coolant after a LOCA. CTS 6.4.1.k includes details regarding the content of procedures for remote or local operation of components necessary to establish high and low pressure injection within 15 minutes of a line break. These details are relocated to UFSAR Chapter 16. These details are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of these details to UFSAR Chapter 16 provides reasonable assurance that the details are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these details are reduced. Therefore, relocation of these details is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

LA13 CTS 6.4.4.a requirements for a respiratory protective program are relocated to UFSAR Chapter 16. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of these requirements to UFSAR Chapter 16 provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these

requirements is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

LA14 CTS 6.4.4.f, Radiological Environmental Monitoring Program, requirements are relocated to UFSAR Chapter 16. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of these requirements to UFSAR Chapter 16 provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

LA15 CTS 6.5 requirements for records retention are relocated to the Quality Assurance Topical Report. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of these requirements to UFSAR Chapter 16 provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

LA16 CTS 6.6.1.1 contains requirements for submitting a report following receipt of an operating license; installation of fuel that has a different design or has been manufactured by a different fuel supplier; modifications that may have altered the nuclear, thermal, or hydraulic performance of the unit; and amendments to the license involving planned increase in power operation. The change relocates these requirements to UFSAR Chapter 16. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of these requirements to UFSAR Chapter 16 provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

LA17 CTS 6.1.1.9 requires the Shift Manager, who provides the on-shift accident assessment capability, to be an experienced SRO. The

ITS Section 5.0

ID 12

Subject Correct ITS retyped pages for 5.5.12.a and 5.5.12.b to agree with NUREG Markup.

5.5 Programs and Manuals (continued)

5.5.11 Secondary Water Chemistry

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.12 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the testing of CRVS Booster Fan Trains at the frequencies specified in Regulatory Guide 1.52, Revision 2.

The VFTP is applicable to the Penetration Room Ventilation System (PRVS) and the Control Room Ventilation System (CRVS) Booster Fan Trains.

1 a. Demonstrate, for the PRVS, that a dioctyl phthalate (DOP)
1 test of the high efficiency particulate air (HEPA) filters
1 shows ≥ 99% removal when tested in accordance with ANSI
1 N510-1975 at the system design flow rate ± 10%.

1 b. Demonstrate, for the CRVS Booster Fan Trains, that a DOP
1 test of the HEPA filters shows ≥ 99.5% removal when tested
1 at in accordance with ANSI N510-1975 at the system design
1 flow rate ± 10%.

(continued)

5.5 Programs and Manuals

5.5.12 <u>Ventilation Filter Testing Program (VFTP)</u> (continued)

c. Demonstrate, for the PRVS, that a halogenated hydrocarbon test of the carbon adsorber shows ≥ 99% removal when tested
1 in accordance with ANSI N510-1975 at the system design flow
1 rate ± 10%.

d. Demonstrate, for the CRVS Booster Fan Trains, that a halogenated hydrocarbon test of the carbon adsorber shows
1 ≥ 99% removal when tested at in accordance with ANSI N510-
1 1975 at the system design flow rate ± 10%.

e. Demonstrate, for the CRVS Booster Fan Trains and PRVS, that a laboratory test of a sample of the carbon adsorber shows ≥ 90% radioactive methyl iodide removal when tested in accordance with ASTM D3803-1989 (30°C, 95% RH).

f. Demonstrate, for the PRVS, that the pressure drop across the combined HEPA filters and carbon adsorber banks is less than
1 6 in. of water at the system design flow rate ~~(± 10%)~~.

g. Demonstrate, for the CRVS Booster Fan Trains, that the pressure drop across the pre-filter is ≤ 1 in. of water and the pressure drop across the HEPA filters is ≤ 2 in. of
1 water at the system design flow rate ± 10%.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.13 <u>Explosive Gas and Storage Tank Radioactivity Monitoring Program</u>

This program provides controls for potentially explosive gas mixtures, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined. The liquid radwaste quantities shall be determined by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

(continued)

Specification 5.0

(A1) <except as marked>

~~4.5.4 Penetration Room Ventilation System~~

~~Applicability~~

~~Applies to testing of the Penetration Room Ventilation System~~

~~Objective~~

~~To verify that the Penetration Room Ventilation System is operable.~~

<see 3.7>

Specification

4.5.4.1 Operational and Performance Testing

a. Monthly, each train of the Penetration Room Ventilation System shall be operated for at least 15 minutes at design flow ±10%.

b. During each refueling outage, it shall be demonstrated that:

Supp.2 | 5.5.12.a 1. The Penetration Room Ventilation System fans operate at design flow (± 10%) when tested in accordance with ANSI N510-1975.

5.5.12.f 2. The pressure drop across the combined HEPA filters and charcoal adsorber banks is less than six inches of water at the system design flow rate (± 10%).

3. Each branch of the Penetration Room Ventilation System is capable of automatic initiation.

4. The bypass valve for filter cooling is manually operable.

<see 3.7>

~~c. Leak tests using DOP or halogenated hydrocarbon, as appropriate shall be performed on the Penetration Room purge filters:~~

~~1. During each refueling outage;~~

~~2. After each complete or partial replacement of a HEPA filter bank or charcoal adsorber bank;~~

~~3. After any structural maintenance on the system housing;~~

~~4. After painting, fire, or chemical release in any ventilation zone communicating with the system.~~

(LA21)

5.5.12.a d. The results of the DOP and halogenated hydrocarbon tests on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal, respectively, when tested in accordance with ANSI N510-1975.

5.5.12.c

The provisions of SR 3.0.2 & SR 3.0.3 are applicable

(A28)

(M5)

The provisions of SR 3.0.2 + SR 3.0.3 are applicable to the Ventilation Filter System Testing Program.

A1 <except as marked>

A28

M5

Specification 5.0

4.12 CONTROL ROOM PRESSURIZATION AND FILTERING SYSTEM

Applicability

Applies to control room pressurization and filtering system components

Objective

To verify that these systems and components will be able to perform their design functions.

Specification

<see 3.7>

4.12.1 Operating Tests

a. Control room outside air booster fan system tests shall be performed quarterly. These tests shall consist of an external visual inspection, a flow measurement for each unit and pressure drop measurements across each filter bank. Pressure drop across pre-filter shall not exceed 1 inch H_2O and pressure drop across HEPA shall not exceed 2 inches H_2O. Fan motors shall be operated continuously for at least one hour, and all louvers shall be proven operable.

Supp 1 5.5.12.g

b. On a refueling frequency, verify the system maintains the control room at a positive pressure with both outside air booster fans on during system operation.

<see 3.7>

5.5.12

4.12.2 Filter Tests

LA3

On a refueling frequency, for the Unit 1 and 2 and the Unit 3 control room an in-place leakage test using DOP on HEPA units and Freon-112 (or equivalent) on carbon units shall be performed at design flow on each filter train. Removal of 99.5 percent DOP by each entire HEPA filter unit and removal of 99.0 percent Freon-112 (or equivalent) by each entire carbon adsorber unit shall constitute acceptance performance. These tests must also be performed after any maintenance which may affect the structural integrity of either the filtration system units or of the housing.

5.5.12.b
5.5.12.d

LA3

Supp 1

Bases

Add 5.5.12.e

M9

A2

The purpose of the control room pressurization filtering system is to protect the control room operators from the effects of accidental release of radioactive effluents or toxic gases in the Turbine Building or Auxiliary Building only. The system is designed with two 50 percent capacity filter trains each of which consists of a prefilter, high efficiency particulate filters, carbon filters, booster fans, air handling unit fans, and associated ductwork to pressurize the control room with outside air.

Since these systems are not normally operated, a periodic test is required to insure their operability when needed. Quarterly testing of this system will show that the system is available.

Refueling frequency testing of the installed carbon adsorber stage and absolute filters will verify the leak integrity of the cleanup system. Refueling frequency testing will also verify the ability of the system to maintain the control room at a positive pressure to minimize infiltration of hazardous effluents.

INSERT 5.0-12A CTS

The VFTP is applicable to the Penetration Room Ventilation System (PRVS) and the Control Room Ventilation System (CRVS) Booster Fan Trains.

a. Demonstrate, for the PRVS, that a dioctyl phthalate (DOP) test of the high efficiency particulate air (HEPA) filters shows ≥ 99% removal when tested in accordance with ANSI N510-1975 at the system design flow rate ± 10%. 4.5.4.1.b.1 4.5.4.1.d Supp 1

b. Demonstrate, for the CRVS Booster Fan Trains, that a DOP test of the HEPA filters shows ≥ 99.5% removal when tested at in accordance with ANSI N510-1975 at the system design flow rate ± 10%. 4.12.2 Supp. 1

c. Demonstrate, for the PRVS, that a halogenated hydrocarbon test of the carbon adsorber shows ≥ 99% removal when tested in accordance with ANSI N510-1975 at the system design flow rate ± 10%. 4.5.4.1.d Supp. 1

d. Demonstrate, for the CRVS Booster Fan Trains, that a halogenated hydrocarbon test of the carbon adsorber shows ≥ 99% removal when tested at in accordance with ANSI N510-1975 at the system design flow rate ± 10%. 4.12.2 Supp. 1

e. Demonstrate, for the CRVS Booster Fan Trains and PRVS, that a laboratory test of a sample of the carbon adsorber shows ≥ 90% radioactive methyl iodide removal when tested in accordance with ASTM D3803-1989 (30°C, 95% RH). Doc M4 4.5.4.1.e

f. Demonstrate, for the PRVS, that the pressure drop across the combined HEPA filters and carbon adsorber banks is less than 6 in. of water at the system design flow rate ± 10%. 4.5.4.1.b.2

g. Demonstrate, for the CRVS Booster Fan Trains, that the pressure drop across the pre-filter is ≤ 1 in. of water and the pressure drop across the HEPA filters is ≤ 2 in. of water at the system design flow rate ± 10%. 4.12.1.a Supp. 1

ITS Section 5.0

ID 19

Subject Revise DOC A30 to correctly refer to ITS 5.5.7 and the Pre-Stressed Concrete Containment Tendon Testing Program.

construction. This requirement is not retained in ITS since it duplicates applicable regulations. Since the requirements are directly tied to the requirements of 10 CFR 50.55a, elimination of CTS 4.0.4 is a less restrictive change and is consistent with the NUREG.

A30 An explicit CTS requirement for a program to monitor containment tendon degradation does not exist. CTS 4.4.2 specifies details associated with
1 containment tendon testing. The program required in ITS 5.5.7 meets the
1 intent of CTS 4.4.2. The details of CTS 4.4.2 are relocated to the Pre-
1 stressed Concrete Containment Tendon Surveillance Program. The requirement to establish a program to accomplish existing requirements is an administrative requirement and is consistent with the NUREG.

A31 CTS 4.0.2 permits application of a frequency extension for specified frequencies. The frequencies for determining the quantity of radioactive material in the Liquid Holdup Tanks and Waste Gas Holdup Tanks are addressed by CTS 4.0.2. Accordingly, CTS permits frequency extensions for the determination of the quantity of radioactive material in the Liquid Holdup Tanks and Waste Gas Holdup Tanks. An explicit CTS requirement for the surveillance of the hydrogen concentration in the Waste Gas Holdup Tanks does not exist. ITS SR 3.0.2 permits extending SR Frequencies. Therefore, this change is an administrative change and is consistent with the NUREG.

A32 An explicit CTS requirement for a program to monitor explosive gas and radioactivity levels in storage tanks does not exist. CTS 3.9 and 3.10 specify details associated with explosive gas and radioactivity levels in storage tanks. ITS 5.5.13 requires establishment of a program which provides controls for explosive gas and storage tank radioactivity monitoring. The program required in ITS 5.5.13 meets the intent of CTS 3.9 and 3.10. The details of CTS 3.9 and 3.10 are relocated to the UFSAR. The requirement to establish a program to accomplish existing requirements is an administrative requirement and is consistent with the NUREG.

A33 CTS 3.9.2 and CTS 3.10.3 are not retained. CTS 3.9.2 and CTS 3.10.3 state that CTS 3.0 does not apply. CTS 3.0 requires the unit be shutdown if the LCO or associated Action statement cannot be met due to circumstances in excess of those addressed in the specification. The requirements of CTS 3.0 are comparable to ITS LCO 3.0.3. ITS LCO 3.0.3 is not applicable to the programs in ITS section 5.5 unless specifically provided for in the program description in ITS 5.5. The non-retention of CTS 3.9.2 and 3.10.3 is an administrative change and is consistent with the NUREG.

ITS Section 5.0

ID 25

Subject Revise Section 5.0 DOCs A15, A17 & A18 to change "duplicates" to "duplicate."

A14 CTS 6.6.3.d is not retained in the ITS. This specification requires submittal of a special report as required by CTS 4.2.1 and 4.2.4. Since neither of these specifications require submittal of a special report, elimination of the reports from the Administrative Controls section of the CTS is an administrative change and is consistent with the NUREG.

A15 CTS Table 6.1-1 requirements regarding SRO and RO minimum operating
shift, additional requirement 2 regarding licensed operator presence in
the control room during specified evolutions and additional requirement
4 regarding an SRO in charge of fuel handling are not retained in the
1 ITS since they duplicate requirements contained in the regulations.
Such duplication is unnecessary and results in additional administrative
burden to revise the duplicate TS when these regulations are revised.
Since removal of the duplication results in no actual change in the
requirements, removal of the duplicative information is considered an
administrative change. Further, changes to the requirements are
controlled by the NRC. This change is consistent with the NUREG.

A16 CTS 6.6.3.e requirements associated with Reactor Building Surveillance, Containment Leakage Tests, are not retained in the ITS since they duplicates requirements contained in the regulations. Such duplication is unnecessary and results in additional administrative burden to revise the duplicate TS when these regulations are revised. Since removal of the duplication results in no actual change in the requirements, removal of the duplicative information is considered an administrative change. Further, changes to the requirements are controlled by the NRC. This change is consistent with the NUREG.

A17 CTS 6.6.2.1 requirements associated with reportable events are not
1 retained in the ITS since they duplicate requirements contained in the
regulations. Such duplication is unnecessary and results in additional
administrative burden to revise the duplicate TS when these regulations
are revised. Since removal of the duplication results in no actual
change in the requirements, removal of the duplicative information is
considered an administrative change. Further, changes to the
requirements are controlled by the NRC. This change is consistent with
NUREG-1430.

A18 CTS 6.7 requirements associated with environmental qualification of
safety related electrical equipment are not retained in the ITS since
1 they duplicate requirements contained in the regulations. Such
duplication is unnecessary and results in additional administrative
burden to revise the duplicate TS when these regulations are revised.
Since removal of the duplication results in no actual change in the
requirements, removal of the duplicative information is considered an

ITS Section 5.0

ID 31

Subject Revise ITS 5.6.5.a to add item 12 requiring AXIAL POWER IMBALANCE and RCS Variable Low Pressure Protective limits to be in the COLR.

5.6 Reporting Requirements

5.6.5 <u>CORE OPERATING LIMITS REPORT (COLR)</u> (continued)

1. Shutdown Margin limit for Specification 3.1.1;

2. Moderator Temperature Coefficient limit for Specification 3.1.3;

3. Physical Position, Sequence and Overlap limits for Specification 3.2.1 Rod Insertion Limits;

4. AXIAL POWER IMBALANCE operating limits for Specification 3.2.2;

5. QUADRANT POWER TILT (QPT) limits for Specification 3.2.3;

6. Nuclear Overpower Flux/Flow/Imbalance and RCS Variable Low Pressure allowable value limits for Specification 3.3.1,

7. RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits for Specification 3.4.1

8. Core Flood Tanks Boron concentration limits for Specification 3.5.1;

9. Borated Water Storage Tank Boron concentration limits for Specification 3.5.4;

10. Spent Fuel Pool Boron concentration limits for Specification 3.7.12;

11. RCS and Transfer Canal boron concentration limits for
1 Specification 3.9.1; and
1
1 12. AXIAL POWER IMBALANCE protective limits and RCS
1 Variable Low Pressure protective limits for
1 Specification 2.1.1.

(continued)

(A1) <except as marked>

(A35) Add 3.1.1, SDM; 3.1.3, MTC; + 3.9.1, Boron Concentration

6.9 CORE OPERATING LIMITS REPORT

Specification

5.6.5.a 6.9.1 Core operating limits shall be established prior to each reload cycle, or prior to any remaining part of a reload cycle, for the following:

Supp.1 | 5.6.5.a.12 (1) Axial Power Imbalance Protective Limits and Variable Low RCS Pressure Protective Limits for Specification 2.1.

5.6.5.a.6 (2) Reactor Protective System Trip Setting Limits for the Flux/Flow/Imbalance and Variable Low Reactor Coolant System Pressure trip functions in Specification 2.3.

5.6.5.a.3 (3) Power Dependent Rod Insertion Limits for Specifications 3.1.3.5, 3.1.11, 3.5.2.1.b, 3.5.2.2.d.2.e, 3.5.2.3, and 3.5.2.5.c.

~~(4) Concentrated Boric Acid Storage Tank volume and boron concentration for Specification 3.2.2.~~ LA22

5.6.5.a.8 (5) Core Flood Tank boron concentration for Specification 3.3.3.

5.6.5.a.9 (6) Borated Water Storage Tank boron concentration for Specification 3.3.4.

5.6.5.a.10 (7) Spent Fuel Pool boron concentration for Specification 3.8.15.

5.6.5.a.5 (8) Quadrant Power Tilt Limits for Specification 3.5.2.4.a, 3.5.2.4.b, 3.5.2.4.d, 3.5.2.4.e, and 3.5.2.4.f. (M6)

5.6.5.a.4 (9) Power Imbalance Limits for Specification 3.5.2.6 Add 5.6.5.a.7

and shall be documented in the CORE OPERATING LIMITS REPORTS.

6.9.2 5.6.5.b ~~The approved methods used to determine the core operating limits given in Technical Specification 6.9.1 are specified in the CORE OPERATING LIMITS REPORT.~~ The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically:

(1) DPC-NE-1002A, Reload Design Methodology II, October 1985.

(2) NFS-1001A, Reload Design Methodology, April 1984.

(3) DPC-NE-2003A, Oconee Nuclear Station Core Thermal Hydraulic Methodology Using VIPRE-01, July 1989.

(4) DPC-NE-1004A, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, November 1992.

(5) BAW-10162P-A, TACO3 Fuel Pin Thermal Analysis Computer Code, B&W Fuel Company, November, 1989.

(6) BAW-10183P, Fuel Rod Gas Pressure Criterion, B&W Fuel Company, as approved by SER dated February, 1994.

requirement for this individual to be an experienced SRO is relocated to UFSAR Chapter 16. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of these requirements to UFSAR Chapter 16 provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

LA18 CTS 4.5.5 requires LPI System leakage rate testing and provides an associated maximum allowable leakage limit. Its purpose is to maintain a preventive leakage rate for the LPI System which will prevent significant offsite exposures following an accident. This is one of the programs required by CTS License Condition H, which is addressed by ITS 5.5.3, "Primary Coolant Sources Outside Containment." The specific requirements of the LPI System leakage rate testing program are relocated to UFSAR Chapter 16. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of these requirements to UFSAR Chapter 16 provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

1 LA19 Not used.

LA20 CTS 4.4.2 provides details regarding testing associated with containment structural integrity. These details are relocated to the Containment Tendon Testing Program. These details are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of these requirements to the Containment Tendon Testing Program provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these details are reduced. Therefore, relocation of these details is acceptable. This change is consistent with the NUREG.

LA21 CTS 4.5.4.1.c and 4.5.4.1.e provide details of the method of ventilation system filter testing which are not directly pertinent to the actual requirement. These details are relocated to UFSAR Chapter 16. These details are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of the details to UFSAR Chapter 16 provides reasonable assurance that the details are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these details are reduced. Therefore, relocation of these details is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

LA22 CTS 6.9.1.4 requires establishments of values and documentation in the COLR for limits associated with volume and boron concentration in the Concentrated Boric Acid Tank. These requirements are relocated to UFSAR Chapter 16. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of the requirements to UFSAR Chapter 16 provides reasonable assurance that the details are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

INSERT 5.0-20A

CTS

1. Shutdown Margin limit for Specification 3.1.1; DOC A35

2. Moderator Temperature Coefficient limit for Specification 3.1.3; DOC A35

3. Physical Position, Sequence and Overlap limits for Specification 3.2.1 Rod Insertion Limits, 6.9.1.3

4. AXIAL POWER IMBALANCE operating limits for Specification 3.2.2; 6.9.1.9

5. QUADRANT POWER TILT (QPT) limits for Specification 3.2.3; 6.9.1.8

6. Nuclear Overpower Flux/Flow/Imbalance and RCS Variable Low Pressure allowable value limits for Specification 3.3.1; 6.9.1.2

7. RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits for Specification 3.4.1; DOC M6

8. Core Flood Tanks Boron concentration limits for Specification 3.5.1; 6.9.1.5

9. Borated Water Storage Tank Boron concentration limits for Specification 3.5.4; 6.9.1.6

10. Spent Fuel Pool Boron concentration limits for Specification 3.7.12; 6.9.1.7

11. RCS and Transfer Canal boron concentration limits for Specification 3.9.1; and DOC A35

Supp.1 12. AXIAL POWER IMBALANCE protective limits and RCS Variable Low Pressure protective limits for Specification 2.1.1. 2.1

ITS Section 5.0

ID 50

Subject Remove TSTF-119 - Adds back specific requirement to have written procedures for Fire Protection program implementation

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

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

b. The emergency operating procedures involving potential or actual release of radioactivity;

1 c. Quality assurance for effluent and environmental monitoring;

1

1 d. Fire Protection Program Implementation; and

1 e. All programs specified in Specification 5.5.

Specification 5.0

A1 <except as marked>

5.4.1.a k. Long-term emergency core cooling systems. ~~Procedures shall include provision for remote or~~ local operation of system components necessary to establish high and low pressure injection within 15 minutes after a line break. A9 LA12

Supp.1 | 5.4.1.d l. Fire Protection Program implementation.

5.4.1.e m. Offsite Dose Calculation Manual implementation. A24

n. Process Control Program implementation. LA11

o. Technical Review and Control Program implementation.

p. Plant Operations Review Committee implementation.

6.4.2 Each procedure of specification 6.4.1 above, and changes thereto, shall be reviewed and approved by an appropriate division manager, superintendent/manager, or one of their designated direct reports prior to implementation and shall be reviewed periodically as set forth in administrative procedures. For procedures which implement offsite environmental, technical, and laboratory activities, the above review and approval may be performed by the General Manager, Environmental Services or designee. LA7

6.4.3 Temporary changes to procedures of Specification 6.4.1 above may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operators license on the affected unit; and

c. The change is approved by an appropriate division manager, superintendent/manager, or one of their designated direct reports within 14 days of implementation.

Supp.1 | Add 5.4.1.e M11

Add 5.4.1.c M4

verification of radioactive iodine removal capability provides reasonable assurance that the removal capability is consistent with the assumptions of the design analysis for the Control Room Ventilation System.

M10 CTS 6.1.1.9 states the shift manager is an experienced SRO, who has been instructed in additional academic subjects, and will be assigned on-shift to provide the accident assessment capability. ITS 5.2.2.g states the Shift Work Manager, whose functions include those of a Shift Technical Advisor (STA), shall provide advisory technical support to the Shift Supervisor (SS) in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the Shift Work Manager shall meet the qualifications for STA specified by the Commission Policy Statement on Engineering Expertise on Shift. The ITS requirements are more prescriptive and therefore a more restrictive requirement upon unit operations. The additional requirements are acceptable since they are reasonable and are consistent with existing practice.

M11 CTS Specification 6.4.1 requires written procedures for specified
1 activities and conditions. ITS 5.4.1.e requires that each program specified in ITS Section 5.5 have written procedures established, implemented and maintained. Specifically, those programs for which this is a new requirement are 1) Component Cyclic or Transient Limits, 2) Technical Specification Bases Control Program, and 3) Safety Function Determination Program. This change imposes new requirements which are more restrictive, and has no adverse impact on safety. The requirement to establish, implement and maintain procedures for these new programs is reasonable and is not a significant impact upon unit operation.

M12 Explicit CTS requirements regarding licensee initiated changes to the ODCM do not exist. ITS 5.5.1 establishes explicit requirements for such changes. Therefore, this change is a more restrictive requirement upon unit operations. The addition of these requirements is acceptable since it is reasonable, generally consistent with current practice and not a significant impact upon unit operations.

M13 Explicit CTS requirements regarding licensee initiated changes to the Radiological Effluent Controls of the UFSAR do not exist. ITS 5.5.5 establishes explicit requirements for such changes. Therefore, this change is a more restrictive requirement upon unit operations. The addition of these requirements is acceptable since it is reasonable, generally consistent with current practice and not a significant impact upon unit operations..

CVR ATT3

ID 75

Subject Att. 3 revised to remove JFD reference column and show draf generic change disposition - show xref. between ONS#/BWOG#/TSTF#.

ATTACHMENT 3
DRAFT GENERIC CHANGES INCORPORATED INTO THE ITS

ONS NO.	BWOG NO.	STATUS
ONS-001	BWOG-60	Replaced by TSTF-2xx
ONS-002	BWOG-61	Replaced by TSTF-211
ONS-003	BWOG-62	Replaced by TSTF-212
ONS-004	BWOG-63	Replaced by TSTF-2xx
ONS-005	BWOG-64	Replaced by TSTF-213
ONS-006	BWOG-65	Withdrawn
ONS-007	BWOG-66	Replaced by TSTF-255
ONS-008	BWOG-67	Replaced by TSTF-214
ONS-009		Replaced by TSTF-096 R1
ONS-011	BWOG-69	Replaced by TSTF-215
ONS-012	BWOG-70	Withdrawn
ONS-013	BWOG-71	Replaced by TSTF-216
ONS-016		Replaced by TSTF-159 R1
ONS-017	BWOG-84	Replaced by TSTF-220
ONS-018	BWOG-74	Replaced by TSTF-256
ONS-019	BWOG-75	Replaced by TSTF-217
ONS-020	BWOG-76	Replaced by TSTF-218
ONS-021	BWOG-85	Replaced by TSTF-2xx
ONS-022	BWOG-77	Withdrawn
ONS-023	BWOG-78	Replaced by TSTF-257
ONS-024	BWOG-79	Replaced by TSTF-219
ONS-025		Withdrawn
ANO-1-46, R1	BWOG-81	Replaced by TSTF-2xx
ANO-1-47, R1	BWOG-82	Replaced by TSTF-2xx
ANO-1-48, R1	BWOG-83	Replaced by TSTF-2xx

CVR ATT4

ID 76

Subject Att. 4 revised to show disposition of generic changes based on latest TSTF status and to include new TSTFs (including ONS draft generic changes).

ATTACHMENT 4
DISPOSITION OF GENERIC CHANGES TO NUREG-1430

TSTF NUMBER	NRC STATUS	ITS SPECs	STATUS[1]	NOTES	LCO JFD(s)	BASES JFD(s)
TSTF-1 R1	Rejected	3.0.5	NI	Not incorporated based on being rejected.	--	--
TSTF-2 R1	Pending	3.8.3	NI	Change not approved by NRC.	--	--
TSTF-3 R1	Rejected	1.1	NI	Not incorporated based on being rejected.	--	--
		3.4.16	NI	Not incorporated based on being rejected.	--	--
		3.7.17	NI	Not incorporated based on being rejected.	--	--
TSTF-4 R1	Rejected	1.1	NI	Station is not implementing a PTLR at this time.	20	--
		5.6.6	NI	Station is not implementing a PTLR at this time.	20	--
TSTF-5 R1	Approved	2.0	FE	N/A	--	6
TSTF-6 R1	Approved	3.0.1	F	N/A	--	5
TSTF-7 R1	Withdrawn	3.0.3	NI	Not incorporated based on being withdrawn.	--	--
TSTF-8 R2	Approved	3.8.1	NI	Not incorporated based on not adopting SRs related to this change.	6	5
		3.8.4	NI	Not incorporated based on not adopting SRs related to this change	6	5
		B 3.0.1	F	N/A	--	--
TSTF-9 R1	Approved	3.1	FE	N/A	29	26
		3.2.1	F	N/A	--	--
		3.3.9	FE	N/A	2	5
1 TSTF-16 R1	Rejected	3.8.9	NI	Not incorporated based on plant specific design	23	5
TSTF-17 R1	Approved	3.6.2	F	N/A	--	--

	TSTF NUMBER	NRC STATUS	ITS SPECs	STATUS[1]	NOTES	LCO JFD(s)	BASES JFD(s)
1	TSTF-19 R1	Pending	1.1	F	Incorporated based on assumption that change has high probability of being approved by NRC prior to being needed. (Reviewer recommends approval)	--	--
1			3.3.1	F	Incorporated based on assumption that change has high probability of being approved by NRC prior to being needed. (Reviewer recommends approval)	--	--
1			3.3.17	F	Incorporated based on assumption that change has high probability of being approved by NRC prior to being needed. (Reviewer recommends approval)	--	--
	TSTF-20	Approved	3.9.6	F	N/A	--	--
	TSTF-21	Approved	B 3.9.5	F	N/A	--	--
	TSTF-22	Rejected	3.9.4	NI	Not incorporated based on being rejected.	--	--
	TSTF-26	Approved	3.4.2	F	N/A	--	--
1	TSTF-27 R3	Pending	3.4.2	NI	Not incorporated based on not being approved by NRC.	--	--
	TSTF-28	Approved	3.4.16	F	N/A	--	--
1	TSTF-30 R2	Approved	3.6.3	NI	Not incorporated based on ONS closed systems not meeting SRP 6.2.4 requirements.	7	5
1	TSTF-36 R3	Pending	3.8	NI	Not incorporated based on not being approved by NRC.	--	--
			3.7.13	NI	Not incorporated based on not being approved by NRC.	--	--

	TSTF NUMBER	NRC STATUS	ITS SPECs	STATUS[1]	NOTES	LCO JFD(s)	BASES JFD(s)
1	TSTF-37 R1	Pending	3.3.17	NI	Not incorporated based on not being approved by NRC. Also, not applicable to ONS, since do not used EDGs for Emergency Power.	--	--
			3.8.1	NI	Not incorporated based on not being approved by NRC.	--	--
			5.6	NI	Not incorporated based on not being approved by NRC.	--	--
	TSTF-38	Approved	3.8.4	F	N/A	--	--
1 1 1	TSTF-39 R1	Withdrawn	1.1	NI	Not incorporated based on being withdrawn. (Note: Replaced by TSTF-205 which was incorporated in Supplement 1)	--	--
	TSTF-40	Approved	1.1	F	N/A	--	--
	TSTF-41	Rejected	3.0	NI	Not incorporated based on NRC rejecting.	--	--
	TSTF-42	Approved	SR 3.0.2	F	N/A	--	--
	TSTF-43	Approved	SR 3.0.2 Bases	F	N/A	--	--
	TSTF-44	Rejected	3.6.3	NI	Not incorporated based on being rejected.	--	--
	TSTF-45 R1	Approved	3.6.3	F	N/A	--	--
	TSTF-46 R1	Approved	3.6.3	F	N/A	--	--
	TSTF-51	Pending	Various	NI	Not incorporated based on not being approved.	--	--

ATTACHMENT 4
DISPOSITION OF GENERIC CHANGES TO NUREG-1430

TSTF NUMBER	NRC STATUS	ITS SPECs	STATUS[1]	NOTES	LCO JFD(s)	BASES JFD(s)
TSTF-52	Pending	1.1	FE	Referenced NRC model letter for Appendix B implementation.	27	--
		B 3.0.2	FE	Referenced NRC model letter for Appendix B implementation.	--	9
		3.6	FE	Did not incorporate reference to program in bypass leakage surveillance since it is not applicable. Incorporated as pertinent to CTS, i.e., Option B for Type A testing, and Option B for Types B and C testing. Minor editorial changes to Bases.	5, 6	5
		5.5.2	FE	Incorporated as pertinent to CTS, i.e., Option B for Type A testing, and Option B for Types B and C testing. Referenced NRC model letter for Appendix B implementation.	11	--
TSTF-54 R1	Approved	B 3.4.13	F	N/A	--	--
TSTF-56	Approved	B 3.4.11	NI	Not incorporated based on not including NUREG 3.4.11 in ONS ITS	--	--
TSTF-57	Approved	B 3.4.10	F	N/A	--	--
TSTF-60	Approved	3.4.15	F	N/A	--	--
TSTF-61	Approved	3.4.13	F	N/A	--	--
TSTF-63	Approved	3.4.7	F	Further modified by draft generic change ANO-1-48.	12	5
1 TSTF-64	Withdrawn	1.1	NI	Not incorporated based on being withdrawn.	--	--

ATTACHMENT 4
DISPOSITION OF GENERIC CHANGES TO NUREG-1430

	TSTF NUMBER	NRC STATUS	ITS SPECs	STATUS(1)	NOTES	LCO JFD(s)	BASES JFD(s)
	TSTF-65	Pending	2.0	NI	Not incorporated based on not being approved by NRC.	--	--
			5.0	NI	Not incorporated based on not being approved by NRC.	--	--
	TSTF-68 R1	Pending	3.9.3	NI	Requires plant specific calculation to support.	--	--
	TSTF-70 R1	Approved	3.7.15	NI	Not incorporated since associated NUREG RA not adopted.	--	--
1	TSTF-71 R1	Approved	B 3.0.6	F	N/A	--	--
	TSTF-86	Rejected	5.2	NI	Not incorporated based on being rejected.	--	--
1	TSTF-88	Withdrawn	T 1.1-1	NI	Not incorporated based on not being withdrawn.	--	--
	TSTF-90 R1	Approved	3.5.3	FE	Incorporated with minor editorial changes.	15	5
	TSTF-91 R1	Rejected	3.3.8	NI	Change not incorporated based on being rejected.	--	--
1	TSTF-92 R1	Pending	3.9.3	FE	Incorporated based on Rev. 0 being approved by NRC on 12/31/96.	--	--
1 1	TSTF-96 R1	Pending	3.9.2 3.9.3	F	Incorporated based on Rev. 0 being approved. Rev. 1 makes change applicable to B&W STS.	--	--
	TSTF-100	Approved	3.7.4	NI	Not incorporated because the change is potentially confusing and unnecessary, since ONS has only two ADV lines.	39	--
	TSTF-101	Approved	3.7.5	F	N/A	--	--
	TSTF-102	Rejected	3.7.2 3.7.3	NI	Change not incorporated based on being rejected.	--	--
	TSTF-104	Approved	3.0.4	F	N/A	--	--

ATTACHMENT 4
DISPOSITION OF GENERIC CHANGES TO NUREG-1430

	TSTF NUMBER	NRC STATUS	ITS SPECs	STATUS[1]	NOTES	LCO JFD(s)	BASES JFD(s)
1 1	TSTF-105 R1	Pending	3.4.1	NI	Not incorporated based on not being approved by the NRC.	--	--
	TSTF-106 R1	Approved	5.0	NI	Not incorporated since the CLB does not include testing the associated fuel oil properties. TSTF-106 clarifies requirements regarding testing requirements for new fuel following addition to the storage tanks.	35	--
1	TSTF-110 R2	Approved	3.1 3.2	F	N/A	--	--
1	TSTF-113 R4	Pending	3.4.11	NI	Change not incorporated based on the PORVs Specification not being included in ONS ITS.	--	--
	TSTF-115	Withdrawn	3.8 5.0	NI	Not incorporated based on being withdrawn.	--	--
1	TSTF-116 R2	Pending	3.4.13 3.4.15	F	Incorporated based on assumption that change has high probability of being approved by NRC prior to being needed.	--	--
	TSTF-118	Approved	5.5.9	NI	The CLB regarding Steam Generator Tube Surveillance testing intervals is retained without adopting.	21	--
			5.5.13	NI	Not appropriate since the CLB adopted for the Diesel Fuel Oil Testing Program does not specify a Frequency for fuel oil testing.	22	--
1 1	TSTF-119	Rejected	5.4.1	NI	Not incorporated based on being rejected. Removed in Supplement 1.	--	--
1 1	TSTF-120	Rejected	5.5.13	NI	Change not incorporated based on being rejected by NRC.	--	--
1 1	TSTF-121	Withdrawn	5.2.2	NI	Change not incorporated based on being withdrawn.	--	--

ATTACHMENT 4
DISPOSITION OF GENERIC CHANGES TO NUREG-1430

	TSTF NUMBER	NRC STATUS	ITS SPECs	STATUS(1)	NOTES	LCO JFD(s)	BASES JFD(s)
	TSTF-122	Approved	B LCO 3.0.2	F	N/A	--	--
1	TSTF-123 R1	Approved	4.2.2	FE	Incorporated with minor editorial changes.	--	--
1	TSTF-124	Approved	1.1	F	N/A	--	--
1	TSTF-125 R1	Pending	1.1	F	Incorporated based on NRC reviewer recommending approval of TSTF.	--	--
	TSTF-126	Approved	2.1.1	F	N/A	--	--
	TSTF-137	Approved	3.4.16	F	N/A	--	--
	TSTF-138	Rejected	3.4.13	NI	Change not incorporated based on being rejected.	--	--
	TSTF-139 R1	Approved	3.7.14	F	N/A	--	--
1	TSTF-140 R1	Approved	3.7.6	F	N/A	--	--
	TSTF-141	Approved	3.1.2	F	N/A	--	--
	TSTF-142	Approved	3.1.2	F	N/A	--	--
	TSTF-143	Approved	3.1.4 3.1.6	F	N/A	--	--
1	TSTF-145 R1	Withdrawn	3.6.3	NI	Not incorporated based on being withdrawn.	--	--
	TSTF-152	Approved	5.6.1	F	N/A	--	--
	TSTF-153	Approved	3.4.5, 6, 7 & 8	F	N/A	--	--
			3.9.4	F	N/A	--	--

ATTACHMENT 4
DISPOSITION OF GENERIC CHANGES TO NUREG-1430

	TSTF NUMBER	NRC STATUS	ITS SPECs	STATUS[1]	NOTES	LCO JFD(s)	BASES JFD(s)
1	TSTF-154 R2	Pending	3.1.8 & 9	NI	Change not incorporated based on not being approved by NRC.	--	--
	TSTF-155	Withdrawn	3.5.1	NI	Change not incorporated based on being withdrawn.	--	--
1	TSTF-156 R1	Pending	3.1.9	F	Change incorporated for plant specific reasons.	18	5
1 1	TSTF-157 R1	Pending	3.7.5	NI	Change not incorporated based on not being approved.	11	5
1	TSTF-158 R1	Pending	3.1.5	NI	Change not incorporated based on not being approved.	--	--
1	TSTF-159 R1	Pending	3.1.6	F	Change incorporated for plant specific reasons.	20	5
1 1 1	TSTF-160 R1	Pending	3.1.4 3.1.8 3.2.1 3.2.2 3.2.5	NI	Change not incorporated based on not being applicable to ONS based on plant specific reasons.	--	--
1	TSTF-163 R1	Pending	3.8.1	NI	Change not incorporated based on not being applicable to ONS.	6	5
	TSTF-165	Approved	3.0.5	F	N/A	--	--
	TSTF-166	Approved	3.0.6	F	N/A	--	--
	TSTF-167	Rejected	5.7.2	NI	Change not incorporated based on being rejected.	--	--
	TSTF-173	Approved	B 3.7.17	F	N/A	--	--
	TSTF-174	Approved	3.7.6	NI	Change not incorporated based on not including ACTION A for plant specific reasons.	21	5

ATTACHMENT 4
DISPOSITION OF GENERIC CHANGES TO NUREG-1430

	TSTF NUMBER	NRC STATUS	ITS SPECs	STATUS(1)	NOTES	LCO JFD(s)	BASES JFD(s)
1 1	TSTF-196	Rejected	3.6.3 3.9.3	NI	Change not incorporated based on not being approved by NRC.	--	--
1 1 1	TSTF-197	Pending	3.9.3 3.9.4 3.9.5	NI	Change not incorporated based on not being approved by NRC.	--	--
	TSTF-198	Pending	3.8.6	NI	Change not incorporated based on not being approved by NRC.	--	--
	TSTF-199	Pending	3.8.4 3.8.5	NI	Change not incorporated based on not being approved by NRC.	--	--
	TSTF-200	Pending	3.8.4	NI	Change not incorporated based on not being approved by NRC.	--	--
	TSTF-201	Pending	3.8.6	F	Change incorporated for plant specific reasons.	41	5
	TSTF-202	Pending	3.8.4 3.8.6	NI	Change not incorporated based on not being approved by NRC.	--	--
	TSTF-203	Pending	B 3.8.6	NI	Change not incorporated based on not being approved by NRC.	--	--
1 1	TSTF-204	Pending	3.8.5	NI	Change not incorporated based on not being approved by NRC.	--	--
1 1 1	TSTF-205	Pending	1.1	F	Change incorporated based on the high probability of being approved. Note: This change supersedes TSTF-39, R1.	--	--
1 1	TSTF-209	Pending		NI	Change not incorporated based on not being approved by NRC.	--	--
1 1	TSTF-210	Pending	3.7.16	NI	Change not incorporated based on not being approved by NRC.	--	--
1	TSTF-211	Pending	3.3.3	F	N/A	--	--

	TSTF NUMBER	NRC STATUS	ITS SPECs	STATUS[1]	NOTES	LCO JFD(s)	BASES JFD(s)
1	TSTF-212	Pending	3.3.3	F	N/A	--	--
1	TSTF-213	Pending	1.1	F	N/A	--	--
1	TSTF-214	Pending	3.9.1	F	N/A	--	--
1 1	TSTF-215	Pending	3.3.6 3.3.7	F	N/A	--	--
1 1	TSTF-216	Pending	3.1.5 3.2.1	F	N/A	--	--
1 1 1 1 1 1	TSTF-217	Pending	3.3.1 3.3.3 3.3.5 3.3.11 3.3.12 3.3.13	F	N/A	--	--
1	TSTF-218	Pending	3.3.1	F	N/A	--	--
1	TSTF-219	Pending	3.4.12	F	N/A	--	--
1	TSTF-220	Pending	3.1.6	F	N/A	--	--
1 1	TSTF-249	Pending	3.1.8 3.1.9	NI	Change not incorporated based on not being approved by NRC.	--	--
1 1	TSTF-250	Pending	4.3.1	NI	Change not incorporated based on not being approved by NRC.	--	--
1 1	TSTF-251	Pending	5.6.9	NI	Change not incorporated based on not being approved by NRC.	--	--
1 1	TSTF-252	Pending	5.6.10	NI	Change not incorporated based on not being approved by NRC.	--	--
1 1	TSTF-253	Pending		NI	Change not incorporated based on not being approved by NRC.	--	--

ATTACHMENT 4
DISPOSITION OF GENERIC CHANGES TO NUREG-1430

	TSTF NUMBER	NRC STATUS	ITS SPECs	STATUS[1]	NOTES	LCO JFD(s)	BASES JFD(s)
1 1	TSTF-254	Pending	3.8.3.5	NI	Change not incorporated based on not being approved by NRC.	--	--
1	TSTF-255	Pending	3.7.16	F	N/A	--	--
1	TSTF-256	Pending	3.1.9	F	N/A	--	--
1	TSTF-257	Pending	3.4.15	F	N/A	--	--
1 1	TSTF-2xx (BWOG-085)	Pending	3.3.9 3.3.10	F	N/A	--	--
1 1 1 1	TSTF-2xx (BWOG-083)	Pending	3.4.5 3.4.6 3.4.7 3.4.8	F	N/A	--	--
1 1 1	TSTF-2xx (BWOG-082)	Pending	3.4.6 3.4.7 3.4.8	F	N/A	--	--
1 1	TSTF-2xx (BWOG-081)	Pending	3.4.5 3.4.6	F	N/A	--	--
1 1 1 1	TSTF-2xx (BWOG-063)	Pending	3.5.2 3.6.6 3.7.5 3.7.8	F	N/A	--	--
1 1	TSTF-2xx (BWOG-060)	Pending	3.3.2	F	N/A	--	--

Table Notations:

(1) F - Fully Incorporated
FE - Fully incorporated with editorial enhancements
NI - Not incorporated

ITS Section CVR ATT7

ID 68

Subject Change Att. 7 of cover letter to reflect change to LA DOCS in Sections 2.0, 3.2, 3.4, 3.5, 3.7, 3.10, and 5.0

ATTACHMENT 7
CTS REQUIREMENTS RELOCATED TO LICENSEE CONTROLLED DOCUMENTS

	DOC	ISSUE RELOCATED	DOCUMENT	CONTROL
	ITS SECTION: 1.0 USE AND APPLICATION			
	LA1	Requirements related to the Process Control Program and definition for the Gaseous Radwaste Treatment System and Ventilation Exhaust Treatment System	SLC	10 CFR 50.59
	LA2	Descriptive information regarding instrumentation, Heat Balance Check and Heat Balance Calibration	Bases	Bases Control Program
	ITS SECTION: 2.0 SAFETY LIMITS			
1	LA1	Not used		
	LA2	Actions when a Safety Limit is violated	QA Topical Report	10 CFR 50.54
	ITS SECTION: 3.1 REACTIVITY CONTROL SYSTEM			
	LA1	Details regarding positioning remaining rods in the affected group	Bases	Bases Control Program
	LA2	Verification regarding loss of power to APSRs and calibration requirements for rod position indications	SLC	10 CFR 50.59
	LA3	Requirement regarding OPERABILITY when a control rod cannot be located by position indication instrumentation	Bases	Bases Control Program
1	LA4	Values for SDM	COLR	5.6.5
	LA5	Notification to NRC regarding cause of reactivity anomaly	SLC	10 CFR 50.59
	LA6	Requirements for verifying control rod inserting time after maintenance or modification	QA Topical Report	10 CFR 50.54
	LA7	Requirement that safety rod groups be fully withdrawn prior to any other reduction in shutdown margin	Bases	Bases Control Program
	R1	Control rod program verification	SLC	10 CFR 50.59

ATTACHMENT 7
CTS REQUIREMENTS RELOCATED TO LICENSEE CONTROLLED DOCUMENTS

DOC	ISSUE RELOCATED	DOCUMENT	CONTROL
ITS SECTION: 3.2 POWER DISTRIBUTION LIMITS			
LA1	Regulating rod group overlap value	COLR	ITS 5.6.5
LA2	Requirements for generation of a power distribution maps using the incore instrumentation system	SLC	10 CFR 50.59
1 LA3	Value for SDM	COLR	ITS 5.6.5
R1	Requirements associated with the control rod drive patch panels	SLC	10 CFR 50.59
R2	Requirements associated with the incore instrumentation	SLC	10 CFR 50.59
ITS SECTION: 3.3 INSTRUMENTATION			
LA1	Information regarding how the shutdown bypass setpoints are controlled or set	Bases	Bases Control Program
LA2	Details regarding functional test	Bases	Bases Control Program
LA3	Information regarding total channels available for each function and the number of channels necessary to trip each function	Bases	Bases Control Program
LA4	Details regarding the number of RCP monitor channels required to be OPERABLE for the RCP monitor logic to be considered OPERABLE	SLC	10 CFR 50.59
LA5	Information regarding total number of channels available and equipment identification nomenclature.	Base	Bases Control Program
LA6	Information regarding equipment actuated by an ESPS signal	Bases	Bases Control Program
LA7	Description of the method of sensing loss of main feedwater pumps	Bases	Bases Control Program
LA8	Information regarding how channels are bypassed and controls on the bypass key	Bases	Bases Control Program

ATTACHMENT 7
CTS REQUIREMENTS RELOCATED TO LICENSEE CONTROLLED DOCUMENTS

DOC	ISSUE RELOCATED	DOCUMENT	CONTROL
LA9	Information regarding how the minimum of three operable channels may be maintained	Bases	Bases Control Program
LA10	SCR (ETA) control relay letter designators included in the description of the function	Bases	Bases Control Program
LA11	Information regarding combinations of direct indications necessary for an OPERABLE subcooling margin monitor	Bases	Bases Control Program
LA12	Method of performance of surveillance	Bases	Bases Control Program
LA13	Method of performance of surveillance	Bases	Bases Control Program
LA14	Information regarding equipment nomenclature	Bases	Bases Control Program
R1	Surveillance requirements for instrumentation not retained in ITS	SLC	10 CFR 50.59
ITS SECTION 3.4: REACTOR COOLANT SYSTEM			
LA1	Surveillance of PORV	IST Program	10 CFR 50.55a
LA2	Requirements regarding establishment of a steam bubble prior to reducing shutdown margin < 1% $\Delta k/k$ and specifies a minimum level of 80 inches in the pressurizer	SLC	50.59
LA3	Not used.		
LA4	Not used.		
LA5	Requirements regarding inspection of the PORV	SLC	10 CFR 50.59
LA6	Administrative requirements associated with evaluation of the safety implications of RCS LEAKAGE	SLC	10 CFR 50.59

ATTACHMENT 7
CTS REQUIREMENTS RELOCATED TO LICENSEE CONTROLLED DOCUMENTS

DOC	ISSUE RELOCATED	DOCUMENT	CONTROL
LA7	Details regarding testing methods associated with testing the PIVs	Bases	Bases Control Program
LA8	Details regarding administrative controls which constitute the second LTOP subsystem.	Bases	Bases Control Program
LA9	Details regarding verification that required RCS vent paths are open.	SLC	10 CFR 50.59
LA10	Subcriticallity requirements when the reactor coolant temperature is < 525°F	SLC	10 CFR 50.59
LA11	Detail regarding the comparison of RCS total activity	Bases	Bases Control Program
LA12	Requirements for leak testing the RCS following opening	SLC	10 CFR 50.59
LA13	Leak testing for specified PIVs after maintenance, repair or replacement	QA Topical Report	10 CFR 50.54
LA14	Requirement that setpoint of the pressurizer code safety valves be in accordance with ASME, Boiler and Pressure Vessel Code.	Bases	Bases Control Program
R1	Restrictions regarding steam generator secondary side pressure	SLC	10 CFR 50.59
R2	Restrictions upon the pressurizer heatup and cooldown rates and use of the pressurizer spray	SLC	10 CFR 50.59
R3	Not used.		
R4	Requirements associated with RCS LEAKAGE which is returnable leakage	SLC	10 CFR 50.59
ITS SECTION: 3.5 EMERGENCY CORE COOLING SYSTEMS			
LA1	Details on what constitutes an OPERABLE train and valve numbers associated with HPI crossover valves	Bases	Bases Control Program

ATTACHMENT 7
CTS REQUIREMENTS RELOCATED TO LICENSEE CONTROLLED DOCUMENTS

	DOC	ISSUE RELOCATED	DOCUMENT	CONTROL
	LA2	Requirements regarding locking and tagging CFT electrically operated discharge valves and locking manual valve on discharge line from BWST.	SLC	10 CFR 50.59
	LA3	Details of the methods of performing HPI and LPI tests and what constitutes acceptable test	Bases	Bases Control Program
	LA4	Core Flooding System Test	SLC	10 CFR 50.59
	LA5	Verification of manual OPERABILITY of power operated valves	SLC	10 CFR 50.59
1 1	LA6	LPI valve idencification numbers	Bases	Bases Control Program
	R1	Requirements for the High Pressure Injection and Chemical Addition Systems	SLC	10 CFR 50.59
	ITS SECTION: 3.6 CONTAINMENT SYSTEMS			
	LA1	Details of methods of performing a test and what constitutes an acceptable test	Bases	Bases Control Program
	LA2	Leakage integrity testing for purge isolation valves	SLC	10 CFR 50.59
	LA3	Requirements regarding Containment Hydrogen Control Systems	SLC	10 CFR 50.59
	LA4	Details of what constitutes an OPERABLE system	Bases	Bases Control Program
	LA5	Determination of fouling rate for reactor building cooling units	SLC	10 CFR 50.59
	LA6	Requirements regarding abnormal degradation of reactor building structural integrity	Pre-Stressed Concrete Tendon Surveil-lance Program	ITS 5.5.7

ATTACHMENT 7
CTS REQUIREMENTS RELOCATED TO LICENSEE CONTROLLED DOCUMENTS

	DOC	ISSUE RELOCATED	DOCUMENT	CONTROL
	LA7	Requirements associated with inservice testing of ASME Code valves in accordance with Section XI.	IST Program	10 CFR 50.55a
	LA8	Requirement that the valve seals of the containment purge isolation valves be visually inspected and adjusted or replaced as appropriate.	SLC	10 CFR 50.59
	ITS SECTION: 3.7 PLANT SYSTEMS			
	LA1	Not used.		
	LA2	Requirements associated with Section XI requirements for relief valve testing	IST Program	10 CFR 50.55a
	LA3	Requirements for testing movement capability of TSVs	IST Program	10 CFR 50.55a
	LA4	Methodology for testing EFW system	SLC	10 CFR 50.59
	LA5	Details regarding the steam supply requirements for the turbine driven EFW pump	Bases	Bases Control Program
	LA6	Not used.		
	LA7	Details regarding testing for specified LPSW valves	SLC	10 CFR 50.59
1 1	LA8	Details regarding filter testing	Bases	Bases Control Program
1	LA9	Not used		
	LA10	Details regarding requirements for OPERABLE ECCW siphon header and testing of Emergency Condenser Circulating Water System	Bases	Bases Control Program
	LA11	Requirements for testing louvers in the Control Room Booster Fan Trains	Bases	Bases Control Program
	LA12	Requirements for snubbers	SLC	10 CFR 50.59
	LA13	Testing requirements for automatic valves in the EFW System	IST Program	10 CFR 50.55a

ATTACHMENT 7
CTS REQUIREMENTS RELOCATED TO LICENSEE CONTROLLED DOCUMENTS

DOC	ISSUE RELOCATED	DOCUMENT	CONTROL
LA14	Equipment functional testing requirements for equipment actuated by the MSLB Feedwater isolation feature	SLC	10 CFR 50.59
R1	Independence of controls of emergency feedwater system from integrated control system	SLC	10 CFR 50.59
R2	Testing spent fuel pool (SFP) cooling system	SLC	10 CFR 50.59
R3	Testing High Pressure Service Water Pumps and Power Supplies	SLC	10 CFR 50.59
R4	Requirements for Reactor Building Purge and Spent Fuel Pool Ventilation System	SLC	10 CFR 50.59
R5	Requirements for Reactor Building Polar Crane and Auxiliary Hoist	SLC	10 CFR 50.59
R6	Not used.		
R7	Testing requirements for Radioactive Material Sources	SLC	10 CFR 50.59
ITS SECTION: 3.8 ELECTRICAL POWER SYSTEMS			
LA1	Requirements regarding Keowee lake level and power restrictions for Keowee Hydro Units (KHU) during periods of commercial power generation and requirements regarding the location of these restrictions	SLC	10 CFR 50.59
LA2	Requirements for 100 kV transmission circuit when energizing the standby buses from an OPERABLE Lee combustion turbine	Bases	Bases Control Program
LA3	Verification of peak inverse voltage capability for each I&C auctioneering diodes	SLC	10 CFR 50.59
LA4	Requirements regarding alternate power source availability	Bases	Bases Control Program
CTS SECTION: 3.9 REFUELING OPERATIONS			

ATTACHMENT 7
CTS REQUIREMENTS RELOCATED TO LICENSEE CONTROLLED DOCUMENTS

DOC	ISSUE RELOCATED	DOCUMENT	CONTROL
LA1	Not used.		
LA2	Decay time prior to the movement of irradiated fuel assemblies from reactor	SLC	10 CFR 50.59
LA3	Indication available for source range neutron instrumentation	Bases	Bases Control Program
LA4	Restriction value of K_{eff}.	Bases	Bases Control Program
R1	Area radiation monitoring in the reactor building and spent fuel pool	SLC	10 CFR 50.59
R2	Communications between the control room and refueling personnel	SLC	10 CFR 50.59
R3	Minimum distance between fuel assemblies being handled simultaneously and handling of fuel assemblies with the Auxiliary Hoist	SLC	10 CFR 50.59
R4	Restrictions on maximum weight of suspended loads which can be transported over spent fuel in spent fuel pools	SLC	10 CFR 50.59
ITS SECTION: 3.10 STANDBY SHUTDOWN FACILITY			
LA1	Information regarding what constitutes an OPERABLE SSF System or Instrumentation	Bases	Bases Control Program
LA2	Requirements for instrumentation associated with the Reactor Coolant Makeup Pumps, the Auxiliary Service Water Pump, the Underground Fuel Oil Storage Tank Inventory, the DG Service Water Pump and DG Air Start System pressure	SLC	10 CFR 50.59
LA3	Annual inspection of the SSF DG	SLC	10 CFR 50.59
LA4	Verification that DG starts from standby condition and runs according to procedures and requirements recommended by manufacturer	Bases	Bases Control Program

ATTACHMENT 7
CTS REQUIREMENTS RELOCATED TO LICENSEE CONTROLLED DOCUMENTS

	DOC	ISSUE RELOCATED	DOCUMENT	CONTROL
	LA5	DG testing after maintenance or modification	QA Topical Report	10 CFR 50.54
	LA6	Duration of battery discharge test	Bases	Bases Control Program
	LA7	Specific gravity to be corrected to 77°F	Bases	Bases Control Program
1	LA8	Not used		
	ITS SECTION: 4.0 DESIGN FEATURES			
	LA1	Not used.		
	LA2	Descriptive details related to the Reactor Containment, Penetrations, and Containment Systems	UFSAR	10 CFR 50.59
	LA3	Descriptive details related to the Reactor Core and the Reactor Coolant System (RCS)	UFSAR	10 CFR 50.59
	LA4	Equivalency of Restricted area to the exclusion area for purposes of 10 CFR 20 and gaseous releases	UFSAR	10 CFR 50.59
	ITS SECTION: 5.0 ADMINISTRATIVE CONTROLS			
	LA1	Methods for implementing requirements for Liquid Holdup Tanks, Waste Gas Holdup Tanks, and Explosive Gas Mixtures	SLC	10 CFR 50.59
	LA2	Inspection and maintenance of structural integrity of reactor internals	SLC	10 CFR 50.59
	LA3	Method of ventilation system filter testing	SLC	10 CFR 50.59
	LA4	Additional requirements for licensed and non-licensed personnel.	SLC	10 CFR 50.59
	LA5	Requirements regarding training for station personnel and the fire brigade	SLC	10 CFR 50.59
1	LA6	Not used		

ATTACHMENT 7
CTS REQUIREMENTS RELOCATED TO LICENSEE CONTROLLED DOCUMENTS

DOC	ISSUE RELOCATED	DOCUMENT	CONTROL
LA7	Requirements regarding PORC and Technical Review and Control; temporary procedure changes; review and control for procedures, modifications, tests, experiments, reportable events, special reviews and investigations, and unplanned onsite releases; records of the above activities	QA Topical Report	10 CFR 50.54
LA8	Requirements for the offsite review committee	QA Topical Report	10 CFR 50.54
LA9	Details associated with internal actions required for reportable events	QA Topical Report	10 CFR 50.54
LA10	Requirements for procedures for a station survey following an earthquake	SLC	10 CFR 50.59
LA11	Requirements for procedures for implementing Process Control Program	QA Topical Report	10 CFR 50.54
LA12	Content of procedures for controlling ph in recirculated coolant after a LOCA and procedures for remote or local operation of components necessary to establish high and low pressure injection within 15 minutes of a line break	SLC	10 CFR 50.59
LA13	Requirements for a respiratory protective program	SLC	10 CFR 50.59
LA14	Requirements regarding Radiological Environmental Monitoring Program	SLC	10 CFR 50.59
LA15	Requirements for records retention	QA Topical Report	10 CFR 50.54

ATTACHMENT 7
CTS REQUIREMENTS RELOCATED TO LICENSEE CONTROLLED DOCUMENTS

DOC	ISSUE RELOCATED	DOCUMENT	CONTROL
LA16	Requirements for submitting a report following receipt of an operating license; installation of fuel that has a different design or has been manufactured by a different fuel supplier; modifications that may have altered the nuclear, thermal, or hydraulic performance of the unit; and amendments to the license involving planned increase in power operation	SLC	10 CFR 50.59
LA17	Requirement for Shift Manager (current title is Shift Work Manager) to be an experienced SRO	SLC	10 CFR 50.59
LA18	Requirements of the LPI System leakage rate testing	SLC	10 CFR 50.59
LA19	Not used		
LA20	Details regarding testing associated with containment structural integrity.	Pre-Stressed Concrete Contain-ment Tendon Testing Program	ITS 5.5.7
LA21	Method of ventilation system filter testing	SLC	10 CFR 50.59
LA22	Requirements regarding establishment of values and documentation in COLR for volume and boron concentration in Concentrated Boric Acid Tank	SLC	10 CFR 50.59