ATTACHMENT 1

VOLUME 9

DAVIS-BESSE IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS SECTION 3.4 REACTOR COOLANT SYSTEM (RCS)

Revision 0

LIST OF ATTACHMENTS

1. ITS 3.4.1
2. ITS 3.4.2
3. ITS 3.4.3
4. ITS 3.4.4
5. ITS 3.4.5
6. ITS 3.4.6
7. ITS 3.4.7
8. ITS 3.4.8
9. ITS 3.4.9
10. ITS 3.4.10
11. ITS 3.4.11
12. ITS 3.4.12
13. ITS 3.4.13
14. ITS 3.4.14
15. ITS 3.4.15
16. ITS 3.4.16
17. ITS 3.4.17
18. Relocated Current Technical Specifications

ATTACHMENT 1

ITS 3.4.1, RCS PRESSURE, TEMPERATURE, AND FLOW DEPARTURE FROM NUCLEATE BOILING (DNB) LIMITS

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

A01

ITS

POWER DISTRIBUTION LIMITS

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

LCO 3.4.1 3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-2.

a. Reactor Coolant Hot Leg Temperature

b. Reactor Coolant Pressure

c. Reactor Coolant Flow Rate

APPLICABILITY: MODE 1

ACTION:

ACTION A — If any parameter above exceeds its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than or equal to 5% of RATED THERMAL
ACTION B — POWER within the next ~~4~~ 6 hours.

A02

L01

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1, SR 3.4.1.2, SR 3.4.1.3 4.2.5.1 Each of the parameters of Table 3.2-2 shall be verified to be within their limits at least once per 12 hours.

SR 3.4.1.4 4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

-----------------------------------NOTE-----------------------------------
Not required to be performed until 7 days after stable thermal conditions are established at ≥ 70% RTP.

L02

ITS 3.4.1

ITS

A01

TABLE 3.2-2

DNB MARGIN

Parameter	LCO 3.4.1.a Required Measured Parameters with Four Reactor Coolant Pumps Operating	LCO 3.4.1.b Required Measured Parameters with Three Reactor Coolant Pumps Operating
Reactor Coolant Hot Leg Temperature T_H °F	≤610	≤610 (1)
Reactor Coolant Pressure, psig. (2)	≥2062.7 2064.8	≥2058.7 (1) 2060.8
Reactor Coolant Flow Rate, gpm (3)	≥389,500	≥290,957

M01

LCO 3.4.1

NOTE to SR 3.4.1.1 and SR 3.4.1.2

(1) Applicable to the loop with 2 Reactor Coolant Pumps Operating.

LCO 3.4.1 Applicability NOTE

(2) Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

(3) These ~~minimum required measured flows include a flow rate uncertainty of 2.5%,~~ and are ~~based on a minimum of 52 lumped burnable poison rod assemblies in place in the core.~~

LA01

A03

DAVIS-BESSE, UNIT 1

3/4 2-14

Amendment No. 11, 16, 33, 97, 123, 135

Page 2 of 2

DISCUSSION OF CHANGES
ITS 3.4.1, RCS PRESSURE, TEMPERATURE, AND FLOW DEPARTURE FROM NUCLEATE BOILING (DNB) LIMITS

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 The CTS 3.2.5 Action requires the unit to reduce THERMAL POWER to "less than" 5% of RATED THERMAL POWER (RTP) within the next 4 hours if the DNB parameters are not restored to within limit in 2 hours. ITS 3.4.1 ACTION B requires the power reduction to MODE 2, which is less than or equal to 5% RTP, within the next 6 hours if the DNB parameters are not restored to within limit in 2 hours. This changes the CTS by allowing the unit be at 5% RTP instead of < 5% RTP. The change in the time period to reach 5% RTP is discussed in DOC L01.

This change is acceptable because it results in no technical change to the Technical Specifications. CTS 3.2.5 is applicable in MODE 1, which is greater than 5% RTP. CTS 3.0.1 (and ITS LCO 3.0.1) states that Actions are applicable during the MODES or other conditions specified for the Specification. Therefore, the CTS 3.2.5 Action to be less than 5% RTP ceases to be applicable once the unit enters MODE 2, i.e., at 5% RTP, and the Action is exited. As a result, changing the ACTION to be in "MODE 2" results in no operational difference from the CTS Action. This change is designated as administrative as it results in no technical change to the CTS.

A03 CTS 3.2.5, Table 3.2-2, Note (3) states that the minimum required Table 3.2-2 measured RCS flow rates include a flow rate uncertainty of 2.5%, "and are based on a minimum of 52 lumped burnable poison rod assemblies in place in the core." ITS 3.4.1 does not include the reason for the values of the measured RCS flow rate limits. This changes the CTS by deleting the specific reason for the measured RCS flow rate limit values. The change that moves the uncertainty value (2.5%) to the Bases is discussed in DOC LA01.

License Amendment 91 reduced the minimum RCS flow rate limits and as part of this amendment, added in the reason for the flow rate limits change (the limits were based on having a minimum of 64 lumped burnable poison rod assemblies). The reason was updated as part of License Amendment 135 and changed the number of assemblies to 52. For the current fuel cycle, while the flow rate limits have not been changed, Davis-Besse does not use lumped burnable poison rod assemblies. Therefore, the reason for the measured RCS flow rate limit values currently in the CTS is not correct. However, the basis of the RCS flow rate limits values is not needed to be included in the Technical Specifications to properly control the values - it is only information as to why the specific values were chosen. The ITS 3.4.1 Bases provides sufficient detail to explain the reason for the RCS flow rate limits. Therefore, not including the reason for the RCS flow rate limits has no impact of the technical requirements

DISCUSSION OF CHANGES
ITS 3.4.1, RCS PRESSURE, TEMPERATURE, AND FLOW DEPARTURE FROM NUCLEATE BOILING (DNB) LIMITS

and therefore, the change is acceptable. This change is designated as administrative as it results in no technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.2.5 requires that departure from nucleate boiling (DNB) parameters specified in CTS Table 3.2-2, including reactor coolant pressure, be maintained within specified limits. CTS Table 3.2-2 requires the measured reactor coolant system pressure to be ≥ 2062.7 psig for four reactor coolant pump operation and ≥ 2058.7 psig for three reactor coolant pump operation. ITS LCO 3.4.1.a requires RCS loop pressure be ≥ 2064.8 psig for four reactor coolant pump operation and ITS LCO 3.4.1.b requires RCS loop pressure be ≥ 2060.8 psig for three reactor coolant pump operation. These values are also provided in ITS SR 3.4.1.1. This changes the CTS by increasing the DNB reactor coolant pressure parameter limits.

The limits on the DNB related parameters specified in CTS 3.2.5 assure that each of the parameters is maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The proposed ITS limits are consistent with the UFSAR initial assumptions and have been analytically demonstrated adequate to maintain a minimum DNB ratio greater than the minimum allowable DNB ratio throughout each analyzed transient. For the current and previous operating cycles, in order to offset the slight non-conservatism for the reactor coolant pressure parameter in the CTS, a DNB penalty has been assessed against the retained DNB margin in the reload licensing analyses. With implementation of the proposed values in the ITS, this offset will no longer be necessary for future core reload analyses. The proposed change is acceptable because it replaces the current CTS values with corrected values that are more conservative. This change is designated as more restrictive because more limiting DNB RCS loop pressure limits are required in the ITS than are required in the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Table 3.2-2 Note (3) states, in part, that "These minimum required measured flows include a flow rate uncertainty of 2.5%." ITS 3.4.1 does not include this specific detail. The details of the Note are moved to the Bases of the applicable Surveillance, ITS SR 3.4.1.4. This changes the CTS by moving the details in CTS Table 3.2-2 Note (3) to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate

DISCUSSION OF CHANGES
ITS 3.4.1, RCS PRESSURE, TEMPERATURE, AND FLOW DEPARTURE FROM NUCLEATE BOILING (DNB) LIMITS

protection of public health and safety. The ITS still retains the information and is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 3 – Relaxation of Completion Time)* The CTS 3.2.5 Action requires the unit to reduce THERMAL POWER to < 5% of RTP within the next 4 hours if the DNB parameters are not restored to within limit in 2 hours. ITS 3.4.1 ACTION B requires the power reduction to ≤ 5% RTP (MODE 2) within the next 6 hours if the DNB parameters are not restored to within limit in 2 hours. This changes the CTS by extending the time for the unit to be placed outside the Applicability of the Specification. The change in the THERMAL POWER value is discussed in DOC A02.

The purpose of the CTS 3.2.5 Action is to limit the time the unit can be outside of the DNB parameter limits and remain within the Applicability of the Specification. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA or transient occurring during the allowed Completion Time. The change extends the time from 4 hours to 6 hours that the unit is allowed to be outside the DNB parameter limits and be in the Applicability of the Specification. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

L02 *(Category 7 – Relaxation of Surveillance Frequency - Non-24 Month Type Change)* CTS 4.2.5.2 requires RCS total flow rate be determined to be within limits once per 18 months. ITS SR 3.4.1.4 requires the same Surveillance, but includes a Note to allow the performance to be delayed for up to 7 days after stable thermal conditions are established at ≥ 70% RTP. This changes the CTS by delaying performance of the Surveillance until adequate conditions exist to perform the Surveillance.

The purpose of CTS 4.2.5.2 is to ensure the RCS total flow rate instrumentation is properly calibrated using a precision calorimetric heat balance. The change is acceptable because the new Surveillance Frequency continues to ensure a precision calorimetric heat balance is performed. This change delays the performance of the precision calorimetric heat balance for up to 7 days after stable thermal conditions are established at ≥ 70% RTP. This change is necessary since a precision heat balance necessary to perform the proper calibration is not obtainable at low power conditions when thermal power is not stable (i.e., power or flow are changing). At low power conditions, the ΔT across

DISCUSSION OF CHANGES
ITS 3.4.1, RCS PRESSURE, TEMPERATURE, AND FLOW DEPARTURE FROM NUCLEATE BOILING (DNB) LIMITS

the core will be too small to provide valid results. Furthermore, during this additional time period the RCS total flow is still required to be monitored by ITS SR 3.4.1.3, and the instrumentation used to perform this verification has been previously calibrated by the last performance of ITS SR 3.4.1.4. This change is designated as less restrictive because Surveillances can be performed less frequently under the ITS than in the CTS.

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

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

3.2.5 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:

Table 3.2-2 a. With four reactor coolant pumps (RCPs) operating:

2064.8

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 1

389.500 gpm 610 ; 2

b. With three RCPs operating:

2060.8

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. 1

610 290.957 gpm

APPLICABILITY: MODE 1.

-----NOTES----- 3

Table 3.2-2 Note (2) RCS loop pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute, or ; 2

b. THERMAL POWER step > 10% RTP.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
3.2.5 Action	A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
3.2.5 Action	B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

CTS

SURVEILLANCE REQUIREMENTS

CTS		SURVEILLANCE	FREQUENCY	
Table 3.2-2 Note (1)	SR 3.4.1.1	-----NOTE----- With three RCPs operating, the limits are applied to the loop with two RCPs in operation. -----		
4.2.5.1		Verify RCS loop pressure ≥ [2061.6] 2064.8 psig with four RCPs operating or ≥ [2057.2] 2060.8 psig with three RCPs operating.	12 hours	1 1
Table 3.2-2 Note (1)	SR 3.4.1.2	-----NOTE----- With three RCPs operating, the limits are applied to the loop with two RCPs in operation. -----		
4.2.5.1		Verify RCS hot leg temperature ≤ [604.6] 610 °F.	12 hours	1
4.2.5.1	SR 3.4.1.3	Verify RCS total flow ≥ [139.7 E6] lb/hr 389,500 gpm with four RCPs operating or ≥ [104.4 E6] lb/hr 290,957 gpm with three RCPs operating.	12 hours	1 1 3
4.2.5.2	SR 3.4.1.4	-----NOTE----- ~~Only required to be performed when stable thermal conditions are established in the higher power range of MODE 1.~~ Not required to be performed until 7 days after stable thermal conditions are established at ≥ 70% RTP. -----		4
		Verify RCS total flow rate is within limit by measurement.	[18] months	1

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.1, RCS PRESSURE, TEMPERATURE, AND FLOW DEPARTURE FROM NUCLEATE BOILING (DNB) LIMITS

1. Brackets have been removed and the proper plant specific information/value has been provided.

2. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01, Section 5.1.3.

3. Typographical error corrected.

4. The ISTS SR 3.4.1.4 Note currently requires performance of the SR immediately upon establishing stable conditions in the higher power range. The proposed change removes the ambiguity of "higher power range" by using a specific power level requirement. Also, as described in ISTS Section 1.4, Example 1.4-5, the wording of the Note regarding stable thermal conditions means that it must be completed when stable conditions are established. No time is provided after the establishment of stable conditions. The Note has been 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. Therefore, the Note is revised to allow 7 days after stable thermal conditions are established at ≥ 70% RTP. This is consistent with the current manner in which Davis-Besse performs the Surveillance, since it provides the necessary time to allow test procedure completion and calculation verifications.

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

All changes are 1
unless otherwise noted

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on DNB related parameters ensure that these parameters will not be less conservative than were assumed in the analyses and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed.

corresponds to

The LCO for minimum RCS pressure is consistent with operation within the nominal operating envelope and is above that used as the initial pressure in the analyses. A pressure greater than the minimum specified will produce a higher minimum DNBR. A pressure lower than the minimum specified will cause the plant to approach the DNB limit.

The LCO for maximum RCS coolant hot leg temperature is consistent with full power operation within the nominal operating envelope and is lower than the initial hot leg temperature in the analyses. A hot leg temperature lower than that specified will produce a higher minimum DNBR. A temperature higher than that specified will cause the plant to approach the DNB limit.

The RCS flow rate is not expected to vary during operation with all pumps running. The LCO for the minimum RCS flow rate corresponds to that assumed for the DNBR analyses. A higher RCS flow rate will produce a higher DNBR. A lower RCS flow will cause the plant to approach the DNB limit.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR criterion of ≥ [1.3]. This is the acceptance limit for the RCS DNBR parameters. Changes to the facility that could impact these parameters must be assessed for their impact on the DNBR 2

for the current reload cycle (Ref. 2)

All changes are 1 unless otherwise noted

BASES

APPLICABLE SAFETY ANALYSES (continued)

criterion. The transients analyzed for include loss of coolant flow events and dropped or stuck control rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE OPERATING LIMITS," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)." 3

nominal

The core outlet pressure assumed in the safety analyses is 2135 psia. 2200
The minimum pressure specified in LCO 3.4.1 is the limit value in the reactor coolant loop as measured at the hot leg pressure tap. corresponding 5

582

The safety analyses are performed with an assumed RCS coolant average temperature of 581°F (579°F plus 2°F allowance for calculational uncertainty). The corresponding hot leg temperature of 604.6°F is calculated by assuming an RCS core outlet pressure of 2135 psia and an RCS flow rate of 374,880 gpm. The maximum temperature specified is the limit value at the hot leg resistance temperature detector. 610

to limit the range of allowable, steady state operation, consistent with the initial conditions assumed in the DNB-related accident analyses.

380,000

The safety analyses are performed with an assumed RCS flow rate of 374,880 gpm. The minimum flow rate specified in LCO 3.4.1 is the minimum mass flow rate. including a 2.5% uncertainty

equivalent

Analyses have been performed to establish the pressure, temperature, and flow rate requirements for three pump and four pump operation. The flow limits for three pump operation are substantially lower than for four pump operation. To meet the DNBR criterion, a corresponding maximum power limit is required (see Bases for LCO 3.4.4, "RCS Loops - MODES 1 and 2").

The RCS DNB limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables: RCS loop (hot leg) pressure, RCS hot leg temperature, and RCS total flow rate to ensure that the core 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 and temperature limits are to be applied to the loop with two reactor coolant pumps (RCPs) running for the three RCPs operating condition.

All changes are 1 unless otherwise noted

BASES

LCO (continued)

measured values and are

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.~~

APPLICABILITY

In MODE 1, the limits on RCS pressure, RCS hot leg 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.

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.

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.

ACTIONS

A.1

Loop pressure and hot leg 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.

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, determine the cause for the off normal condition, and restore the readings within limits. The Completion Time is based on plant operating experience.

All changes are 1 unless otherwise noted

BASES

ACTIONS (continued)

B.1

If the Required Action A.1 is not met within the Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds.

The 6 hour Completion Time is reasonable, based on operating experience, to reduce power in an orderly manner in conjunction with even control of steam generator heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for loop (hot leg) pressure is sufficient to ensure that the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The RCS pressure value specified is dependent on the number of pumps in operation and has been adjusted to account for the pressure loss difference between the core exit and the measurement location. The value used in the plant safety analysis is ~~2135~~ 2200 psi~~a~~ (nominal). The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation is within safety analysis assumptions.

A Note has been added to indicate the pressure limits are to be applied to the loop with two pumps in operation for the three pump operating condition.

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for hot leg temperature is sufficient to ensure that the RCS coolant temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify that operation is within safety analysis assumptions.

All changes are 1 unless otherwise noted

BASES

SURVEILLANCE REQUIREMENTS (continued)

A Note has been added to indicate the temperature limits are to be applied to the loop with two pumps in operation for the three pump operating condition.

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify that operation is within safety analysis assumptions.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance once every [18] months allows the installed RCS flow instrumentation to be calibrated and verifies that the actual RCS flow is greater than or equal to the minimum required RCS flow rate. INSERT 1 2

INSERT 2 The Frequency of [18] months ~~reflects the importance of verifying flow after a refueling outage when the core has been altered or RCS flow characteristics may have been modified, which may have caused change of flow.~~ 2

INSERT 3

The Surveillance is modified by a Note that indicates the SR ~~does not need to be performed until stable thermal conditions are established at higher power levels.~~ The Note is necessary to allow measurement of the flow rate at normal operating conditions at power in MODE 1. The Surveillance cannot be performed at low power or in MODE 2 or below because at low power the ΔT across the core will be too small to provide valid results. 4

REFERENCES U 1. FSAR, ~~Chapter~~ Section [15]. 1 2

2. UFSAR, Appendix 4B. 1

1 **INSERT 1**

These minimum required measured flows include a flow rate uncertainty of 2.5%.

1 **INSERT 2**

is considered adequate for ensuring accurate RCS flow measurement instrumentation and has been shown by operating experience to be acceptable.

4 **INSERT 3**

is not required to be performed until 7 days after stable thermal conditions are established at ≥ 70% RTP.

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.1 BASES, RCS PRESSURE, TEMPERATURE, AND FLOW DEPARTURE FROM NUCLEATE BOILING (DNB) LIMITS

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.
2. The brackets have been removed and the proper plant specific information/value has been provided.
3. Typographical error corrected.
4. Changes made to be consistent with changes made to the Specification.
5. Editorial change made for clarity.

Attachment 1, Volume 9, Rev. 0, Page 22 of 415

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.1, RCS PRESSURE, TEMPERATURE, AND FLOW DEPARTURE FROM NUCLEATE BOILING (DNB) LIMITS

There are no specific NSHC discussions for this Specification.

Attachment 1, Volume 9, Rev. 0, Page 24 of 415

ATTACHMENT 2

ITS 3.4.2, RCS MINIMUM TEMPERATURE FOR CRITICALITY

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

ITS A01 ITS 3.4.2

REACTIVITY CONTROL SYSTEMS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

LCO 3.4.2 3.1.1.4 The Reactor Coolant System lowest loop temperature (T_{avg}) shall be ≥ 525°F.

APPLICABILITY: MODES 1 and 2*.

ACTION:

ACTION A With a Reactor Coolant System loop temperature ~~(T_{avg})~~ < 525°F, ~~restore T_{avg} to within its limit within 15 minutes or~~ be in HOT STANDBY MODE 2 with keff < 1.0 within the next 15 30 minutes.

A02

A03

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1 4.1.1.4 The RCS temperature (T_{avg}) shall be determined to be ≥ 525°F:

~~a. Within 15 minutes prior to achieving reactor criticality, and~~

~~b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 530°F.~~

every 12 hours

L01

Applicability *With K_{eff} ≥ 1.0.

DISCUSSION OF CHANGES
ITS 3.4.2, RCS MINIMUM TEMPERATURE FOR CRITICALITY

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 The CTS 3.1.1.4 Action states that with a Reactor Coolant System (RCS) operating loop temperature (T_{avg}) < 525°F, to "restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes." ITS 3.4.2 ACTION A states that with T_{avg} in one or more RCS loops not within limit, be in MODE 2 with k_{eff} < 1.0 within 30 minutes. This changes the CTS by eliminating the redundant and unnecessary requirement to restore T_{avg} to within its limit within 15 minutes. The change associated with entering MODE 2 with k_{eff} < 1.0 instead of HOT STANDBY is discussed in DOC A03.

This change is acceptable because it results in no technical change to the Technical Specifications. Although the CTS 3.1.1.4 Action allows only 15 minutes to restore the parameter to within the limit, it actually allows the entire 30 minutes to either restore the parameter or to be in HOT STANDBY (essentially outside the Applicability of CTS 3.1.1.4). In addition, the CTS 3.1.1.4 Action only requires actual steps to begin reducing reactor power at the beginning of the last 15 minutes of the 30-minute time period. However, CTS 3.0.2 states that "In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, completion of the ACTION Statement is not required." Therefore, for this specific case, if the parameter is restored between 15 minutes and 30 minutes after the Limiting Condition for Operation (LCO) parameter is not met, completion of the CTS 3.1.1.4 Action to be in HOT STANDBY is not required. Thus, 30 minutes is essentially allowed for either the parameter to be restored to within limit or the unit to be in HOT STANDBY (i.e., only one of the two CTS Actions must be met within 30 minutes). The CTS 3.0.2 requirement is retained in ITS LCO 3.0.2. Therefore, this change does not expand the total time interval allowed to restore the parameter, as a 30-minute time period is already essentially allowed by the CTS. This change is designated as administrative as it results in no technical change to the CTS.

A03 The CTS 3.1.1.4 Action states that with a Reactor Coolant System operating loop temperature (T_{avg}) < 525°F, to restore T_{avg} to within its limit within 15 minutes or be in "HOT STANDBY" within the next 15 minutes. ITS 3.4.2 ACTION A states that with T_{avg} in one or more RCS loops not within limit, be in "MODE 2 with k_{eff} < 1.0" within 30 minutes. This changes the CTS by requiring entry into MODE 2 with k_{eff} < 1.0 instead of entry into HOT STANDBY (MODE 3). The change associated with the time to be in HOT STANDBY is discussed in DOC A02.

DISCUSSION OF CHANGES
ITS 3.4.2, RCS MINIMUM TEMPERATURE FOR CRITICALITY

This change is acceptable because it results in no technical change to the Technical Specifications. CTS 3.1.1.4 is applicable in MODE 1 and MODE 2 with $k_{eff} \geq 1.0$. CTS 3.0.1 (and ITS LCO 3.0.1) states that Actions are applicable during the MODES or other conditions specified for the Specification. Therefore, the CTS 3.1.1.4 Action to enter HOT STANDBY (MODE 3) ceases to be applicable once the unit enters MODE 2 with $k_{eff} < 1.0$. As a result, changing the ACTION to "be in MODE 2 with $k_{eff} < 1.0$" results in no operational difference from the CTS Action. This change is designated as administrative as it results in no technical change to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.1.1.4 states that the RCS T_{avg} shall be determined to be $\geq$ 525°F within 15 minutes prior to achieving reactor criticality, and every 30 minutes when the reactor is critical and the RCS T_{avg} < 530°F. ITS SR 3.4.2.1 requires RCS T_{avg} in each loop to be verified $\geq$ 525°F every 12 hours. This changes the CTS by deleting the within 15 minutes prior to achieving criticality Frequency and the Surveillance Frequencies based on the condition of the reactor (critical) and reactor coolant temperature (< 530°F), and replacing them with a periodic 12 hour Frequency.

The purpose of CTS 4.1.1.4 is to ensure RCS T_{avg} is within limit when the reactor is critical. The requirement is that RCS T_{avg} be $\geq$ 525°F, and it is required to be met when the unit is operating in MODE 2 with $k_{eff} \geq 1.0$ and MODE 1. Based on ITS SR 3.0.4, this would require the SR to be met within 12 hours prior to entry into MODE 2 with $k_{eff} \geq 1.0$ (i.e., before the reactor is critical). This change is acceptable because the new Surveillance Frequency provides an acceptable level of assurance that the RCS T_{avg} is within limit. The 12 hour Frequency is considered frequent enough to prevent inadvertent violation of the LCO. In the approach to criticality, with the required reactor coolant pumps running, the RCS is at normal operating pressure, so the conditions before and after criticality are similar. The approach to criticality is a carefully controlled evolution during which RCS temperature is closely monitored. Therefore, 12 hours is frequent enough

DISCUSSION OF CHANGES
ITS 3.4.2, RCS MINIMUM TEMPERATURE FOR CRITICALITY

for the Technical Specifications to require recording of T_{avg} prior to criticality given that it is being routinely monitored. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

Attachment 1, Volume 9, Rev. 0, Page 30 of 415

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

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.1.1.4 LCO 3.4.2 Each RCS loop average temperature (T_{avg}) shall be ≥ 525°F.

APPLICABILITY: MODE 1,
MODE 2 with k_{eff} ≥ 1.0.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
3.1.1.4 Action	A. T_{avg} in one or more RCS loops not within limit.	A.1 Be in MODE 2 with k_{eff} < 1.0.	30 minutes	1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
4.1.1.4	SR 3.4.2.1 Verify RCS T_{avg} in each loop ≥ 525°F.	12 hours

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.2, RCS MINIMUM TEMPERATURE FOR CRITICALITY

1. Typographical error corrected.

Attachment 1, Volume 9, Rev. 0, Page 33 of 415

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

All changes are 1 unless otherwise noted

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

Establishing the value for the minimum temperature for reactor criticality is based upon considerations for:

a. Operation within the existing instrumentation ranges and accuracies ; and 3

b. Operation with reactor vessel above its minimum nil ductility reference temperature when the reactor is critical.

The reactor coolant moderator temperature coefficient used in core operating and accident analysis is ~~typically~~ defined for the normal operating temperature range (532°F to ~~579~~ 582°F). The Reactor Protection System (RPS) receives inputs from the narrow range hot leg temperature detectors, which have a range of 520°F to 620°F. The integrated control system controls average temperature (T_{avg}) using inputs of the same range. Nominal T_{avg} for making the reactor critical is 532°F. Safety and operating analyses for lower temperatures have not been made.

APPLICABLE SAFETY ANALYSES

There are no accident analyses that dictate the minimum temperature for criticality~~, but all low power safety analyses assume initial temperatures near the 525°F limit (Ref. 1)~~.

INSERT 1

The RCS minimum temperature for criticality satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of the LCO is to prevent much criticality outside the minimum normal operating regime (532°F ~~to 579°F~~) and to prevent operation in an unanalyzed condition.

The LCO limit of 525°F has been selected to be within the instrument indicating range (520°F to 620°F). The limit is also set slightly below the lowest power range operating temperature (532°F).

APPLICABILITY

The reactor has been designed and analyzed to be critical in MODES 1 and 2 only and in accordance with this Specification. Criticality is not permitted in any other MODE. Therefore, this LCO is applicable in MODE 1 and MODE 2 when $k_{eff} \geq 1.0$.

1 **INSERT 1**

Compliance with the LCO ensures that the reactor will not be made or maintained critical at a temperature significantly less than the hot zero power (HZP) temperature, which is assumed in the safety analysis (Ref. 1). Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

BASES

ACTIONS	A.1 With T_{avg} below 525°F, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 2 with k_{eff} < 1.0 in 30 minutes. Rapid reactor shutdown can be readily and practically achieved in a 30 minute period. The Completion Time reflects the ability to perform this action and maintain the plant within the analyzed range. If T_{avg} can be restored within the 30 minute time period, shutdown is not required. 4
SURVEILLANCE REQUIREMENTS	SR 3.4.2.1 RCS loop average temperature is required to be verified at or above 525°F every 12 hours. The SR to verify RCS loop average temperatures every 12 hours takes into account indications and alarms that are continuously available to the operator in the control room and is consistent with other routine Surveillances which are typically performed once per shift. In addition, operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.
REFERENCES	U 1. FSAR, ~~Chapter [15]~~ Section 15.2.1 1 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.2 BASES, RCS MINIMUM TEMPERATURE FOR CRITICALITY

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

2. The brackets have been removed and the proper plant specific information/value has been provided.

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

4. Typographical error corrected.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.2, RCS MINIMUM TEMPERATURE FOR CRITICALITY

There are no specific NSHC discussions for this Specification.

ATTACHMENT 3

ITS 3.4.3, RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

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

ITS

A01

ITS 3.4.3

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.4.3 3.4.9.1 The Reactor Coolant system (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2, 3.4-3 and 3.4-4 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

LA01

LA02

A02

a. A maximum heatup of 50°F in any one hour period, and

b. A maximum cooldown of 100°F in any one hour period with cold leg temperature $\geq$ 270°F and a maximum cooldown of 50°F in any one hour period with cold leg temperature <270°F.

LA02

APPLICABILITY: At all times.

A03

ACTION:

Add proposed Conditions A and C Notes

A04

ACTIONS A and C With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at

ACTION B least HOT STANDBY within the next 6 hours and be in COLD SHUTDOWN within the following 30 hours.

LA03

Add proposed Required Action A.2 and C.2 Completion Times

M01

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1 4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens representative of the vessel materials shall be removed and examined, to determine changes in material properties, at the intervals defined in BAW 1543A. The results of these examinations shall be used to update Figures 3.4-2, 3.4-3 and 3.4-4.

A05

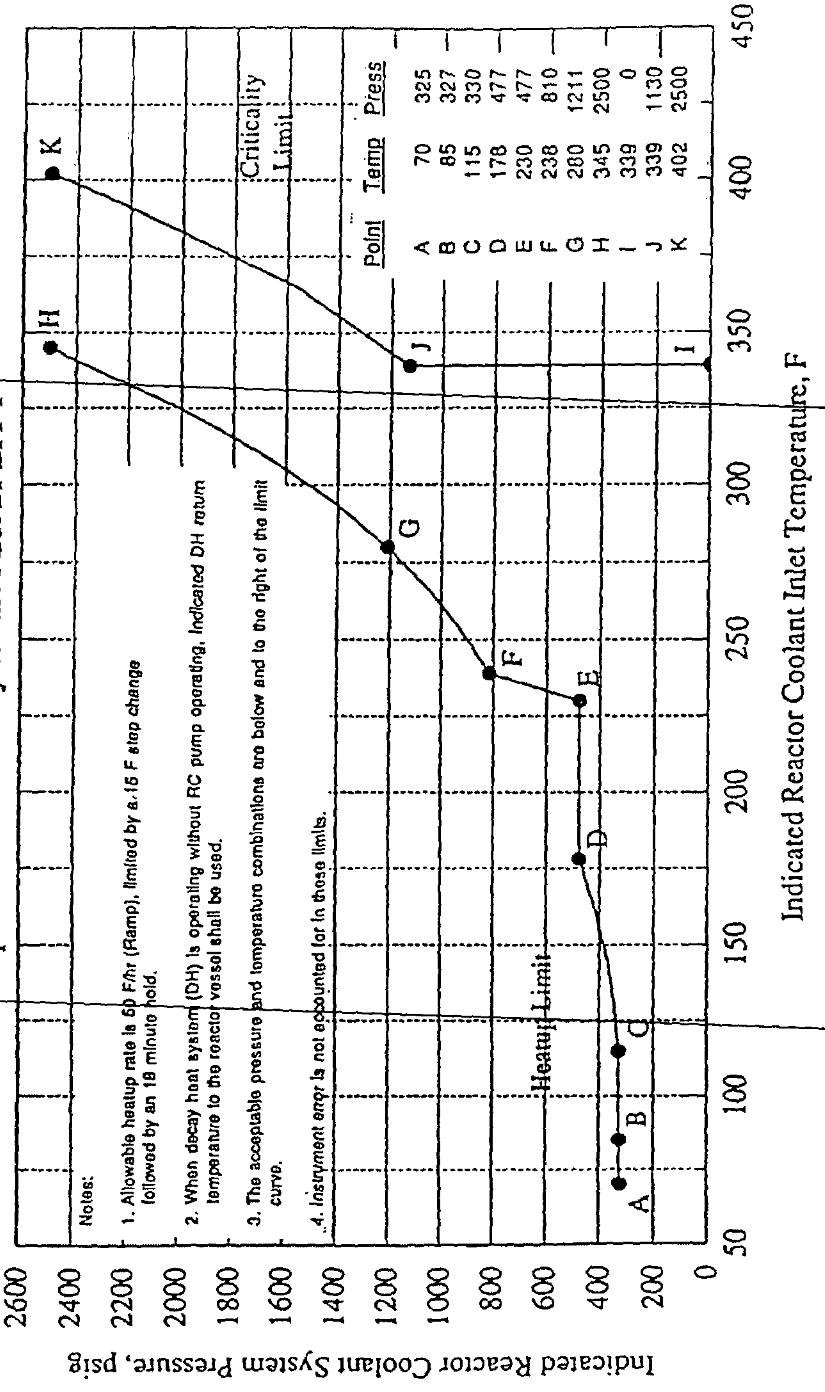
ITS 3.4.3

LA02

Page 2 of 5

Figure 3.4-2

Reactor Coolant System Pressure-Temperature Limits for Heatup and Core Criticality for the First 21 EFPY



Point	Temp	Press
A	70	325
B	85	327
C	115	330
D	178	477
E	230	477
F	238	810
G	280	1211
H	345	2500
I	339	0
J	339	1130
K	402	2500

A01

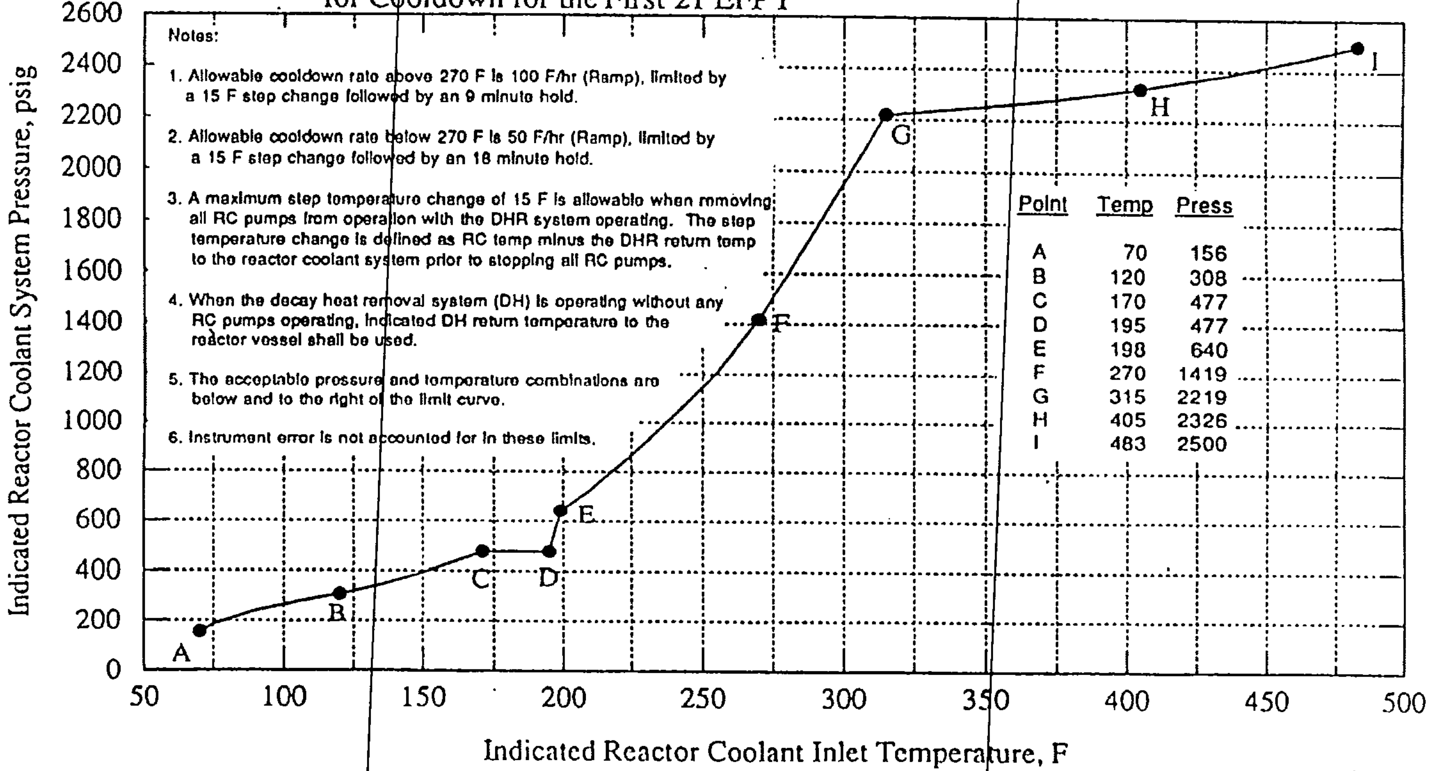
DAVIS-BESSE, UNIT 1 3/4 4-25 Amendment No. ~~116~~, 199

A01

LA02

Figure 3.4-3

Reactor Coolant System Pressure-Temperature Limits for Cooldown for the First 21 EFPY



Notes:

1. Allowable cooldown rate above 270 F is 100 F/hr (Ramp), limited by a 15 F step change followed by an 9 minute hold.

2. Allowable cooldown rate below 270 F is 50 F/hr (Ramp), limited by a 15 F step change followed by an 18 minute hold.

3. A maximum step temperature change of 15 F is allowable when removing all RC pumps from operation with the DHR system operating. The step temperature change is defined as RC temp minus the DHR return temp to the reactor coolant system prior to stopping all RC pumps.

4. When the decay heat removal system (DH) is operating without any RC pumps operating, indicated DH return temperature to the reactor vessel shall be used.

5. The acceptable pressure and temperature combinations are below and to the right of the limit curve.

6. Instrument error is not accounted for in these limits.

Point	Temp	Press
A	70	156
B	120	308
C	170	477
D	195	477
E	198	640
F	270	1419
G	315	2219
H	405	2326
I	483	2500

Attachment 1, Volume 9, Rev. 0, Page 45 of 415

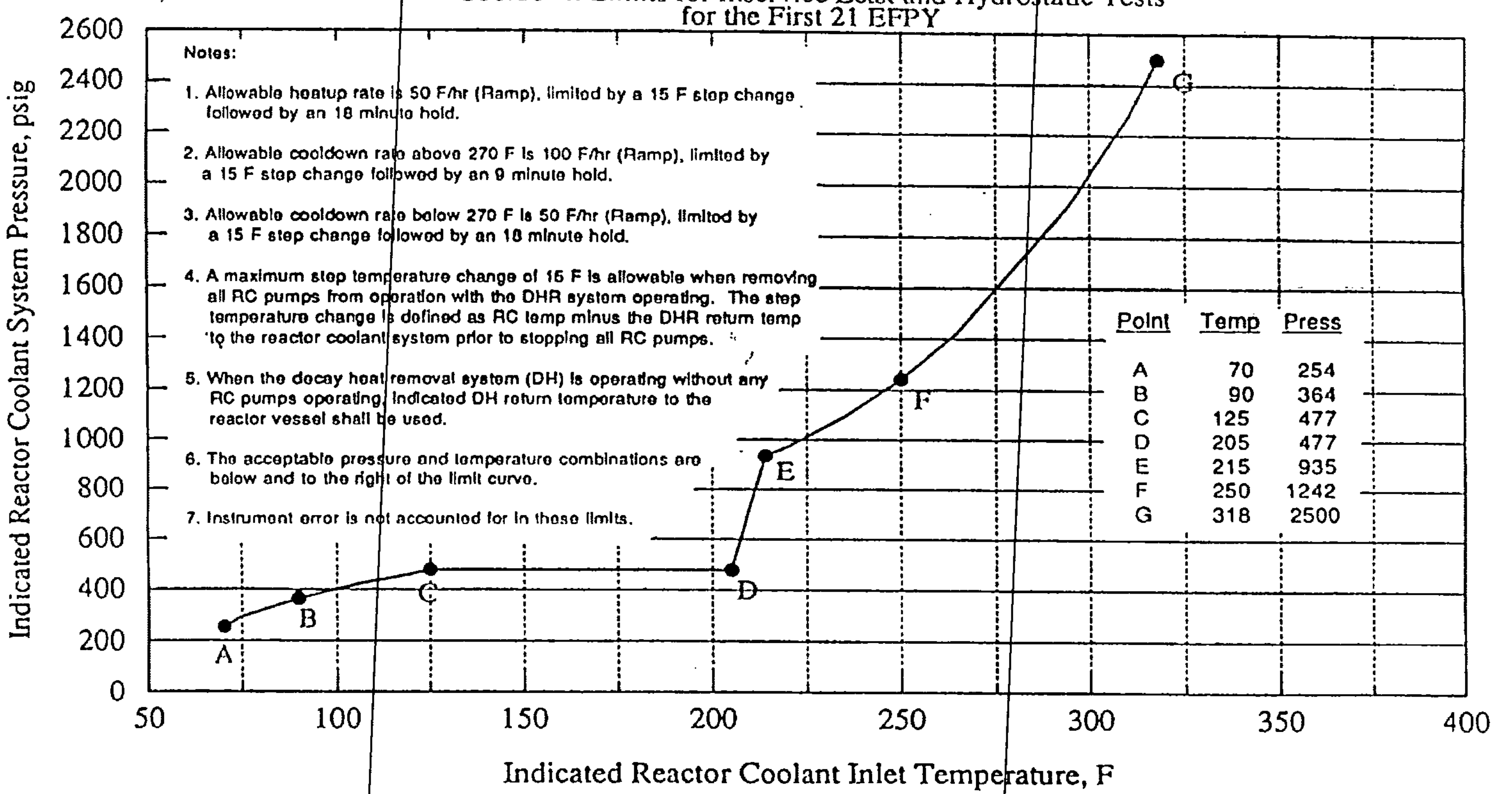
ITS 3.4.3

A01

LA02

Figure 3.4-4

Reactor Coolant System Pressure-Temperature Heatup and Cooldown Limits for Inservice Leak and Hydrostatic Tests for the First 21 EFPY



Notes:

1. Allowable heatup rate is 50 F/hr (Ramp), limited by a 15 F step change followed by an 18 minute hold.
2. Allowable cooldown rate above 270 F is 100 F/hr (Ramp), limited by a 15 F step change followed by an 9 minute hold.
3. Allowable cooldown rate below 270 F is 50 F/hr (Ramp), limited by a 15 F step change followed by an 18 minute hold.
4. A maximum step temperature change of 15 F is allowable when removing all RC pumps from operation with the DHR system operating. The step temperature change is defined as RC temp minus the DHR return temp to the reactor coolant system prior to stopping all RC pumps.
5. When the decay heat removal system (DH) is operating without any RC pumps operating, indicated DH return temperature to the reactor vessel shall be used.
6. The acceptable pressure and temperature combinations are below and to the right of the limit curve.
7. Instrument error is not accounted for in these limits.

Point	Temp	Press
A	70	254
B	90	364
C	125	477
D	205	477
E	215	935
F	250	1242
G	318	2500

Attachment 1, Volume 9, Rev. 0, Page 46 of 415

A01

~~Table 4.4-5~~

~~Reactor Vessel Material Irradiation Surveillance Schedule~~

~~DELETED~~

Attachment 1, Volume 9, Rev. 0, Page 47 of 415

DISCUSSION OF CHANGES
ITS 3.4.3, RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.4.9.1 states that the RCS temperature and pressure shall be limited "during heatup, cooldown, criticality, and inservice leak and hydrostatic testing." CTS 3.4.9.1 is applicable at all times. ITS 3.4.3 states that the RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained. ITS 3.4.3 is applicable at all times. This changes the CTS by eliminating the LCO requirements that the limits must be met only during heatup, cooldown, criticality, and inservice leak and hydrostatic testing.

This change is acceptable because the CTS and ITS limits, including heatup, cooldown, criticality, and inservice leak and hydrostatic testing, are applicable at all times. Stating that the limits are applicable during heatup, cooldown, criticality, and inservice leak and hydrostatic testing in the LCO presents an apparent conflict with the Applicability which states that the limits apply at all times. This change is designated as administrative as it is an editorial change to eliminate an apparent conflict in the CTS.

A03 CTS 3.4.9.1 Action states that with any of the P/T limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes, perform an engineering evaluation to determine the effects of the out-of-limit condition on the integrity of the Reactor Coolant System, and determine that the Reactor Coolant System remains acceptable for continued operation. ITS 3.4.3 Conditions A and C are modified by a Note that requires the determination that the RCS is acceptable for continued operation be performed whenever the Condition is entered. This changes the CTS by explicitly stating that a determination that the RCS is acceptable for continued operation must be performed whenever the Condition is entered.

This change is acceptable because it is the current understanding and application of the CTS Action. The CTS 3.4.9.1 Action is currently interpreted as requiring a determination that the RCS is acceptable for continued operation whenever the LCO is not met. This change is designated as editorial as it clarifies the current understanding of the CTS requirement.

A04 CTS 3.4.9.1 Action states, in part, that with any of the P/T limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes. ITS 3.4.3 ACTION C states that with the requirements of the LCO not met any time other than MODE 1, 2, 3, or 4, to immediately initiate action to restore the parameter(s) to within limits. This changes the CTS by requiring immediate action to restore P/T limits and continuing the action until complete, when the unit is in other than MODE 1, 2, 3, or 4.

DISCUSSION OF CHANGES
ITS 3.4.3, RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

This change is acceptable because this change reflects an enhanced presentation of the existing intent. The CTS 3.4.9.1 Action to "restore... within 30 minutes" is proposed to be revised to "initiate action to restore...Immediately" for conditions other than MODES 1, 2, 3, and 4. This existing Action would appear to provide a half hour in which pressure and temperature requirements could exceed the limits, even it capable of being returned to within limits. Also, if the parameters are incapable of being restored within the limits within 30 minutes, the existing Action would appear to result in the requirement of a Licensee Event Report, since no additional Actions apply (the unit is already in MODE 5 or below). The intent of the Action is believed to be more appropriately presented in ITS 3.4.3 Required Action C.1. This interpretation of the intent is supported by the Babcock and Wilcox Standard Technical Specifications, NUREG-1430, Rev 3.1. This change is designated as administrative as it reflects an enhanced presentation of the existing intent.

A05 CTS 4.4.9.1.2 states that the reactor vessel material irradiation surveillance specimens representative of the vessel materials shall be removed and examined to determine changes in material properties, at the intervals defined in BAW 1543A. The results of these examinations shall be used to update Figures 3.4-1, 3.4-3, and 3.4-4. ITS 3.4.3 does not contain this Surveillance nor the Table. This changes the CTS by deleting the reactor vessel material irradiation Surveillance Requirement.

This change is acceptable because the Surveillance is unnecessary and repetitive. The unit is required by applicable regulations to remove material irradiation surveillance specimens and generate P/T curves in accordance with 10 CFR 50, Appendix H. Therefore, the Surveillance serves no purpose and is removed. This change is designated as administrative as it eliminates a requirement that is duplicative of a regulatory requirement in the CFR.

MORE RESTRICTIVE CHANGES

M01 CTS 3.4.9.1 Action states that if the P/T limits are exceeded, an analysis must be performed and a determination made that the RCS remains acceptable for continued operation. No time limit is given for the performance of this analysis and determination. ITS 3.4.3 Required Action A.2 states that when the LCO is not met in MODES 1, 2, 3, or 4, determination is required that the RCS is acceptable for continued operation within 72 hours. ITS 3.4.3 Required Action C.2 states that when the LCO is not met any time other than in MODES 1, 2, 3, or 4, determination is required that the RCS is acceptable for continued operation prior to entering MODE 4. This changes the CTS by specifying a finite time to perform the determination.

This change is acceptable because it provides adequate time to evaluate exceeding the LCO requirements. The Completion Time of 72 hours is considered reasonable for operation in MODES 1, 2, 3, and 4 because P/T limits are based on very conservative flaw assumptions and large factors of safety. The Completion Time of "prior to entering MODE 4" during operations other than MODE 1, 2, 3, or 4 is considered reasonable since it would prevent entry into the operating MODES, and is consistent with LCO. 3.0.4. This change is designated

DISCUSSION OF CHANGES
ITS 3.4.3, RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

as more restrictive as it provides a limited time to perform an action for which the CTS provides not time limit.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.4.9.1 states that the RCS (except the pressurizer) temperature and pressure shall be limited. The LCO also contains limits on RCS heatup and cooldown rates. ITS 3.4.3 states that the RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within limits. This changes the CTS by moving the exclusion of the pressurizer from the LCO to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains P/T limits on the RCS. Neither the CTS or the ITS P/T limits apply to the pressurizer. It is the ITS convention to state this detail of the LCO in the ITS Bases. This detail of the LCO is not required to be in the Technical Specifications in order to provide adequate protection of the public health and safety. Also this change is acceptable because the removed information will be adequately controlled in the ITS Bases. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 4 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAPM, IST Program, PTLR, or IIP)* CTS 3.4.9.1 states, in part, that the Reactor Coolant system temperature and pressure shall be limited in accordance with the limits lines shown on Figures 3.4-2, 3.4-3, and 3.4-4. Additionally, CTS 3.4.9.1.a and 3.4.9.1.b specify the maximum heatup rate and the maximum cooldown rates, respectively. ITS 3.4.3 states that the RCS pressure, RCS temperature, and RCS heatup and cooldown rate shall be maintained within the limits specified in the PTLR. This changes the CTS by relocating the Figures and the maximum heatup and maximum cooldown rates to the PTLR.

The removal of these figures, heatup rate, and cooldown rate from the Technical Specification to the PTLR is acceptable because the PTLR is developed and utilized under NRC-approved methodologies, which will ensure that the RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates are met. This type of information is not necessary to be included in the Technical Specification to provide adequate protection of public health and safety. The ITS still retains the RCS P/T Limit requirements. The methodologies used to develop the parameters in the PTLR have obtained prior approval by the

DISCUSSION OF CHANGES
ITS 3.4.3, RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

NRC. Also, this change is acceptable because the removed information will be adequately controlled in the PTLR under the requirements provided in ITS 5.6.4, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)." ITS 5.6.4 ensures that the applicable RCS pressure and temperature limits are met. This change is designated as a less restrictive removal of detail change because the detailed P/T limits are being removed from the Technical Specifications.

LA03 *(Type 3 – Removing Procedural Details for Meeting TS Requirements and Related Reporting Problems)* CTS 3.4.9.1 Action states that with any P/T limits exceeded, to perform an engineering evaluation to determine the effects of the out-of-limit condition on the integrity of the RCS. ITS 3.4.3 ACTIONS A and C, in part, state that with the requirements of the LCO not met, to determine the RCS is acceptable for continued operation. The specific requirement to perform an engineering evaluation is not included in ITS 3.4.3. This changes the CTS by moving the requirement to "perform an engineering evaluation" to determine the effects of the out-of-limit condition on the integrity of the RCS to the Bases.

The removal of these details for performing actions from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to determine that the RCS remains acceptable for continued operation and this detail of how the determination is made is not required to be in the Technical Specifications in order to provide adequate protection of the public health and safety. The requirement to perform an engineering evaluation to determine the effects of the out-of-limit condition on the integrity of the RCS is a step in determining that the RCS remains acceptable for continued operation. Therefore, this detail on how the determination is made is moved to the Bases, which provides additional detail on how to the determination should be made. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. This change is designated a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

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

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

3.4.9.1 LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
Action	A. ------------NOTE------------ Required Action A.2 shall be completed whenever this Condition is entered. -------------------------------- Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
Action	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.	6 hours 36 hours
Action	C. ------------NOTE------------ Required Action C.2 shall be completed whenever this Condition is entered. -------------------------------- Requirements of LCO not met in other than MODE 1, 2, 3, or 4.	C.1 Initiate action to restore parameter(s) to within limit. AND C.2 Determine RCS is acceptable for continued operation.	Immediately Prior to entering MODE 4

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.4.9.1.1	SR 3.4.3.1	-----NOTE----- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. -----	
		Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR.	30 minutes

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.3, RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

None

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The PTLR contains P/T limit curves for heatup, cooldown, criticality, and inservice 1
leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Appendix G (Ref. 3).

Linear elastic fracture mechanics (LEFM) methodology is used to determine the stresses and material toughness at locations within the RCPB. The LEFM methodology follows the guidance given by 10 CFR 50, Appendix G; ASME Code, Section III, Appendix G; and Regulatory Guide 1.99 (Ref. 4).

BASES

BACKGROUND (continued)

Material toughness properties of the ferritic materials of the reactor vessel are determined in accordance with the NRC Standard Review Plan (Ref. 5), ASTM E 185 (Ref. 6), and additional reactor vessel requirements. These properties are then evaluated in accordance with Reference 3.

The actual shift in the nil ductility reference temperature (RT_{NDT}) of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 6) and Appendix H of 10 CFR 50 (Ref. 7). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 3.

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The calculation to generate the ISLH testing curve uses different safety factors (per Ref. 3) than the heatup and cooldown curves. The ISLH testing curve also extends to the RCS design pressure of 2500 ~~psia~~. psig 1

The P/T limit curves and associated temperature rate of change limits are developed in conjunction with stress analyses for large numbers of operating cycles and provide conservative margins to nonductile failure. Although created to provide limits for these specific normal operations, the curves also can be used to determine if an evaluation is necessary for an abnormal transient.

BASES

BACKGROUND (continued)

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 8) provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. Since the P/T limits are not derived from any DBA analysis, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The two elements of this LCO are:

criticality, 1

a. The limit curves for heatup, cooldown, and ISLH testing and

; 2

b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follows:

BASES

LCO (continued)

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature ; 2

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced) ; and 2

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits Specification provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," LCO 3.4.2, "RCS Minimum Temperature for Criticality," and Safety Limit (SL) 2.1, "SLs," also provide operational restrictions for pressure and temperature and maximum pressure. MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

ACTIONS

A.1 and A.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

BASES

ACTIONS (continued)

Besides restoring operation to within limits, an engineering evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed ~~before continuing operation~~ within 72 hours. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components. The evaluation must be completed, documented, and approved in accordance with established plant procedures and administrative controls. 1 3

ASME Code, Section XI, Appendix E (Ref. 8) may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline. The evaluation must extend to all components of the RCPB.

The 72 hour Completion Time is ~~reasonable~~ provided to accomplish the evaluation. The evaluation for a mild violation ~~is~~ may be possible within this time, but more severe violations may require special, event specific stress analyses or inspections. ~~A favorable evaluation must be completed before continuing to operate.~~ 4 3

Condition A is modified by a Note requiring Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If ~~a~~ any Required Action and associated Completion Time of Condition A ~~are~~ is not met, the plant must be brought to a lower MODE because: (a) the RCS remained in an unacceptable pressure and temperature region for an extended period of increased stress, or (b) a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. With reduced pressure and temperature conditions, the possibility of propagation of undetected flaws is decreased. 3

If the required restoration activity cannot be accomplished within 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature.

BASES

ACTIONS (continued)

If the required evaluation for continued operation cannot be accomplished within 72 hours, or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Actions B.1 and B.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions. However, if the favorable evaluation is accomplished while reducing pressure and temperature conditions, a return to power s
operation may be considered without completing Required Action B.2. B.1 and 3

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified acceptable by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished within this time in a controlled manner.

In addition to restoring operation to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analysis, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 8), may also be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

BASES

ACTIONS (continued)

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone, per Required Action C.1, is insufficient because higher than analyzed stresses may have occurred and may have affected RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes.

This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that requires this SR to be performed only during system heatup, cooldown, and ISLH testing.

REFERENCES

1. BAW-10046A, Rev. 1, July 1977.
2. 10 CFR 50, Appendix G.
3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
4. Regulatory Guide 1.99, Revision 2, May 1988.
5. NUREG-0800, Section 5.3.1, Rev. 1, July 1981.
6. ASTM E 185-82, July 1982.
7. 10 CFR 50, Appendix H.
8. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.3 BASES, RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

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

2. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01, Section 5.1.3.

3. Changes are made to be consistent with the Specification.

4. Editorial change.

Attachment 1, Volume 9, Rev. 0, Page 64 of 415

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.3, RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

There are no specific NSHC discussions for this Specification.

Attachment 1, Volume 9, Rev. 0, Page 66 of 415

ATTACHMENT 4

ITS 3.4.4, RCS LOOPS - MODES 1 AND 2

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

ITS 3.4.4

A01

ITS

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

LCO 3.4.4.a

3.4.1.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: MODES 1 and 2*. A03

ACTION:

LCO 3.4.4.b

L01 10

ACTION A

a. With one reactor coolant pump not in operation, STARTUP and POWER OPERATION may be initiated and may proceed provided THERMAL POWER is restricted to less than 80.6% of RATED THERMAL POWER and within 4 hours the setpoints for the following trips have been reduced in accordance with Specification 2.2.1 for operation with three reactor coolant pumps operating:

LCO 3.4.4.b

1. High Flux

2. Flux-ΔFlux-Flow

Add proposed ACTION B M01

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

4.4.1.1.1 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours. LA01

ACTION A

4.4.1.1.2 The Reactor Protection System trip setpoints for the instrumentation channels specified in the ACTION statement above shall be verified to be in accordance with Specification 2.2.1 for the applicable number of reactor coolant pumps operating either: L02

10 L01

ACTION A

a. Within 4 hours after switching to a three pump combination if the switch is made while operating, or

b. Prior to reactor criticality if the switch is made while shutdown. A02

*See Special Test Exception 3.10.3. A03

DISCUSSION OF CHANGES
ITS 3.4.4, RCS LOOPS - MODES 1 AND 2

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 4.4.1.1.2.b requires a verification that the three reactor coolant pumps (RCPs) operating Reactor Protection System (RPS) trip setpoints for the High Flux and Flux-ΔFlux-Flow Functions are properly set prior to reactor criticality if the switch to three RCPs was made while not within the Applicability of CTS 3.4.1.1. This specific Surveillance is not maintained in the ITS. This changes the CTS by deleting the prior to criticality Surveillance.

The purpose of CTS 4.4.1.1.2.b is to ensure the three RCPs operating RPS trip setpoints are properly set prior to reactor criticality if the switch to three RCPs was made while not within the Applicability of CTS 3.4.1.1. This requirement however, is already enforced by other ITS requirements. ITS 3.4.4 requires the setpoints to be adjusted properly for operation with three RCPs. Thus, prior to entering the Applicability of ITS LCO 3.4.4 (MODES 1 and 2), the LCO must be met as required by ITS LCO 3.0.4. Furthermore, ITS LCO 3.3.1 provides the RPS setpoints for operation with three RCPs, and ITS LCO 3.0.4 would also require the setpoint requirement to be met prior to entering the two RPS Functions' (ITS Table 3.3.1-1 Functions 1.a and 8) Applicability (which includes MODES 1 and 2). Therefore, this current requirement is unnecessary and has been deleted. This change is designated as administrative and is acceptable since it does not result in any technical change to the CTS.

A03 The CTS 3.4.1.1 includes a footnote stating "See Special Test Exception 3.10.3." ITS 3.4.4 Applicability does not contain the footnote or a reference to the Special Test Exception.

The purpose of the footnote is to alert the user that a Special Test Exception exists that may modify the Applicability of the Specification. However, CTS 3.10.3 has not been adopted into the ITS (see CTS 3/4.10.3 DOC M01 in Section 3.1), therefore the cross-reference is not needed. Furthermore, it is an ITS convention to not include these types of footnotes or cross-references even if the CTS LCO were maintained in the ITS. This change is designated as administrative as it incorporates an ITS convention with no technical change.

MORE RESTRICTIVE CHANGES

M01 CTS 3.4.1.1 does not specify a default Action if more than one reactor coolant pump is not in operation or if the trips are not reduced in the 4 hour time period required by the CTS 3.4.1.1 Action. Thus, CTS 3.0.3 would be entered requiring entry into HOT STANDBY (MODE 3) within 7 hours. ITS 3.4.4 ACTION A

DISCUSSION OF CHANGES
ITS 3.4.4, RCS LOOPS - MODES 1 AND 2

requires the plant to be in MODE 3 within 6 hours under the same conditions. This changes the CTS by providing one less hour for entry into MODE 3.

The purpose of requiring a shutdown when under the above conditions is to bring the unit to a subcritical condition since the unit is not within the accident analysis assumptions. This change is acceptable because it provides an adequate period of time to be in a MODE in which the LCO does not apply. The Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power in an orderly manner and without challenging unit systems.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 - Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.4.1.1.1 states that the required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours. ITS SR 3.4.4.1 states that each RCS loop shall be verified to be in operation every 12 hours. This changes the CTS by moving the Surveillance Requirement detail to verify that the reactor coolant loops are circulating reactor coolant to the Bases.

The removal of this detail for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be in the Technical Specifications in order to provide adequate protection of the public health and safety. The ITS retains the requirement that an RCS loop be in operation. This will require recirculation of reactor coolant since the ITS Bases specify that verification that a reactor coolant loop is in operation includes flow rate, temperature, or pump status monitoring, which helps ensure that forced flow is providing heat removal. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 3 – Relaxation of Completion Time)* CTS 3.4.1.1 Action a, which applies when shifting from four RCPs operating to three RCPs operating, requires a reduction of the High Flux trip setpoint from the four RCPs operating to three RCPs operating trip setpoint within 4 hours. Under the same conditions, ITS 3.4.4 ACTION A requires the reduction in the trip setpoints within 10 hours.

DISCUSSION OF CHANGES
ITS 3.4.4, RCS LOOPS - MODES 1 AND 2

This changes the CTS by extending the Completion Time to reduce the trip setpoints from "4 hours" to "10 hours."

The purpose of CTS 3.4.1.1 Action a is to ensure the proper trips setpoints for the new RCP configuration are set into the RPS High Flux Function. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the allowed Completion Time. The required Completion Time of 10 hours is reasonable based on the low probability of an accident occurring while operating outside the three RCPs operating trip setpoints, the automatic protection provided by the RPS Flux-ΔFlux-Flow Function (which is automatically reset), and the number of steps required to complete the Required Action, and the THERMAL POWER restriction provided in the LCO (i.e., 80.6% RTP). This proposed time is also consistent with the time allowed to reset the High Flux trip setpoints in ITS 3.2.4 and ITS 3.2.5, when QPT or a power peaking factor parameter is not within the required limits. Under these conditions, similar actions are required by plant personnel to reset the High Flux trip setpoints. This change is designated as less restrictive because additional time is allowed to reduce the trip setpoints.

L02 *(Category 5 - Deletion of Surveillance Requirement)* CTS 4.4.1.1.2 requires verification that the RPS trip setpoints for the High Flux and Flux-ΔFlux-Flow Functions are properly set after shifting from four RCPs operating to three RCPs operating. The ITS does not include this additional Surveillance as part of ITS 3.4.4 ACTION A for the Flux-ΔFlux-Flow Function. This changes the CTS by not including this conditional Surveillance for the Flux-ΔFlux-Flow Function.

The purpose of CTS 4.4.1.1.2 is to ensure the three RCPs operating RPS trip setpoints are properly set following a shift from four RCPs operating to three RCPs operating. However, the Flux-ΔFlux-Flow Function automatically changes its trip setpoint based on the number of operating RCPs. Thus, when one RCP trips, the three RCPs operating Flux-ΔFlux-Flow trip setpoint is automatically enabled - no manual setpoint adjustment is necessary. Thus the only function of this Surveillance is to ensure the automatic adjustment feature of the instrumentation functioned properly. This change is acceptable since the CHANNEL CALIBRATION testing required by ITS 3.3.1, "Reactor Protection System (RPS) Instrumentation," (ITS SR 3.3.1.3) already ensures that the instrumentation can automatically adjust the trip setpoints based on the number of operating RCPs. Therefore, this specific Surveillance is redundant to the normal, routine CHANNEL CALIBRATION Surveillances in the RPS Specification and is not needed. This change is designated as less restrictive because a Surveillance which is required in the CTS will not be required in the ITS.

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

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops - MODES 1 and 2

LCO 3.4.1.1 LCO 3.4.4 Two RCS Loops shall be in operation, with:

a. Four reactor coolant pumps (RCPs) operating ~~, or~~ :

Action a b. Three RCPs operating and ~~THERMAL POWER restricted to [79.9]% RTP.~~ :

INSERT 1

1 2

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
INSERT 2 ~~A~~ B. Requirements of LCO not met for reasons other than Condition A.	~~A~~ B.1 Be in MODE 3.	6 hours

DOC M01 2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify required RCS loops are in operation.	12 hours

4.4.1.1.1

2 INSERT 1

Action a 1. THERMAL POWER is < 80.6% RTP;

Action a.1 2. LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," Function 1.a (High Flux - High Setpoint), Allowable Value of Table 3.3.1-1 is reset for three RCPs operating; and

Action a.2 3. LCO 3.3.1, Function 8 (Flux-ΔFlux-Flow), Allowable Value of Table 3.3.1-1 is reset for three RCPs operating.

2 INSERT 2

Action a	A. Requirements of LCO 3.4.4.b.2 not met.	A.1 Satisfy the requirements of LCO 3.4.4.b.2.	10 hours

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.4, RCS LOOPS - MODES 1 AND 2

1. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01, Section 5.1.3.

2. ISTS LCO 3.4.4 is written for a plant whose design includes an automatic setdown feature for the nuclear overpower trip setpoint. That is, when shifting from four reactor coolant pump (RCP) operation to three RCP operation, the trip setpoints for the Reactor Protection System (RPS) instrumentation automatically adjust based on RCP configuration. This is described in the ISTS Bases, Background section, last paragraph. The Davis-Besse design does not include this automatic setdown feature for the High Flux trip setpoints - the setpoints must be manually adjusted. The current licensing basis provides for time to make a manual adjustment after shifting from four RCPs operating to three RCPs operating (CTS 3.4.1.1 Action a). ITS 3.4.4 has been written to allow the same two options as ISTS LCO 3.4.4: four RCPs must be operating (ITS LCO 3.4.4.a or three RCPs must be operating with a maximum power level restriction (ITS LCO 3.4.4.b and LCO 3.4.4.b.1). ITS LCO 3.4.4 also requires the trip setpoints of the High Flux and Flux-ΔFlux-Flow Functions to be set within the three RCP operating limits when operating with only three RCPs (ITS LCO 3.4.4.b.2 and LCO 3.4.4.b.3). Furthermore, a new ACTION has been added that provide 10 hours to manually reset the High Flux trip setpoints to within the Allowable Value for three RCP operation. While the current licensing basis only provides 4 hours to reset the trip setpoints (CTS 3.4.1.1 Action a), the 10 hours provided in ITS 3.4.4 ACTION A is consistent with the time provided in ISTS 3.2.4 and ISTS 3.2.5 to reset the High Flux trip setpoints when a QPT or power peaking factor limit is not met. Due to this change, ISTS 3.4.4 ACTION A has been renumbered as ACTION B and its associated Condition modified to only apply for reasons other than that provided in ITS 3.4.4 Condition A. In addition, the format of ITS 3.4.4 is also consistent with the format of NUREG-1433, ISTS 3.4.1, which has a similar requirement to manually change a trip setpoint when a recirculation pump (the BWR equivalent to an RCP) is not in operation.

Attachment 1, Volume 9, Rev. 0, Page 76 of 415

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

All changes are 1 unless otherwise noted

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops - MODES 1 and 2

BASES

BACKGROUND

The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the RCS include:

a. Moderating the neutron energy level to the thermal state, to increase the probability of fission ; 3
b. Improving the neutron economy by acting as a reflector ; 3
c. Carrying the soluble neutron poison, boric acid ; 3
d. Providing a second barrier against fission product release to the environment ; and 3
e. Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

The RCS configuration for heat transport uses two RCS loops. Each RCS loop contains an SG and two reactor coolant pumps (RCPs). An RCP is located in each of the two SG cold legs. The pump flow rate has been sized to provide core heat removal with appropriate margin to departure from nucleate boiling (DNB) during power operation and for anticipated transients originating from power operation. This Specification requires two RCS loops with either three or four pumps to be in operation. With three pumps in operation the reactor power level is restricted to < 80.6 ~~[79.9]~~% RTP to preserve the core power to flow relationship, thus maintaining the margin to DNB. The intent of the Specification is to require core heat removal with forced flow during power operation. Specifying the minimum number of pumps is an effective technique for designating the proper forced flow rate for heat transport, and specifying two loops provides for the needed amount of heat removal capability for the allowed power levels. Specifying two RCS loops also provides the minimum necessary paths (two SGs) for heat removal. 2

The Reactor Protection System (RPS) Flux - ΔFlux - Flow (Table 3.3.1-1 Function 8) ~~nuclear overpower~~ trip setpoint is automatically reduced when one pump is taken out of service; manual resetting is not necessary. However, the RPS High Flux - High Setpoint (Table 3.3.1-1 Function 1.a) trip setpoint must be manually reset.

BASES

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the Design Bases Accident (DBA) initial conditions including: RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of pumps in service.

Both transient and steady state analyses have been performed to establish the effect of flow on DNB. The transient or accident analysis for the plant has been performed assuming either three or four pumps are in operation. The majority of the plant safety analysis is based on initial conditions at high core power or zero power. The accident analyses that are of most importance to RCP operation are the four pump coastdown, single pump locked rotor, and single pump (broken shaft or coastdown) (Ref. 1).

Steady state DNB analysis has been performed for four, three, and two pump combinations. For four pump operation, the steady state DNB analysis, which generates the pressure and temperature SL (i.e., the departure from nucleate boiling ratio (DNBR) limit), assumes a maximum power level of ~~[112]% RTP~~ 110.2% of 2817 MWt. This is the design overpower condition for four pump operation. The ~~[112]%~~ value is the accident analysis setpoint of the nuclear overpower (high flux) trip and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit. 2

The three pump pressure temperature limit is tied to the steady state DNB analysis, which is evaluated each cycle. The flow used is the minimum allowed for three pump operation. The actual RCS flow rate will exceed the assumed flow rate. With three pumps operating, overpower protection is automatically provided by the ~~power to flow ratio of the RPS nuclear overpower based on RCS flow and AXIAL POWER IMBALANCE setpoint.~~ INSERT 1 1 The maximum power level for three pump operation is ~~[79.9]~~ < 80.6% RTP and is based on the three pump flow as a fraction of the four pump flow at full power. 2

Although the Specification limits operation to a minimum of three pumps total, existing design analyses show that operation with one pump in each loop (two pumps total) is acceptable when core THERMAL POWER is restricted to be proportionate to the flow. However, continued power operation with two RCPs removed from service is not allowed by this Specification.

RCS Loops - MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

Attachment 1, Volume 9, Rev. 0, Page 79 of 415

1 INSERT 1

RPS Flux - ΔFlux - Flow Function. Overpower protection is also provided by the High Flux - High Setpoint Function, which must be manually reset for three pump operation.

BASES

LCO

The purpose of this LCO is to require adequate forced flow for core heat removal. Flow is represented by the number of RCPs in operation in both RCS loops for removal of heat by the two SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power; if only three pumps are available, power must be reduced. and certain RPS setpoints must be reset 4

APPLICABILITY

In MODES 1 and 2, the reactor is critical and has the potential to produce maximum THERMAL POWER. To ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops - MODE 3,"; 3
LCO 3.4.6, "RCS Loops - MODE 4,";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled,";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled,";
LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation - High Water Level" ~~(MODE 6)~~; and 4
LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level". ~~(MODE 6).~~

ACTIONS

INSERT 2

B ~~A~~.1 5

If the requirements of the LCO are not met for reasons other than Condition A, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

The 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 safety systems.

5 **INSERT 2**

A.1

If only three RCPs are in operation and the RPS High Flux - High Setpoint trip setpoints have not been reset to within the Allowable Value provided in Table 3.3.1-1 Function 1.a for three RCP operation, the trip setpoints must be reset within 10 hours. This ensures the proper automatic overpower protection is provided by the RPS. The 10 hour Completion Time is reasonable based on the low probability of an accident occurring while operating outside the three RCP limit, the automatic protection provided by the RPS Flux - ΔFlux - Flow Function (which is automatically reset), and the number of steps required to complete the Required Action.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification every 12 hours of the required number of loops in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to DNB. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.

REFERENCES

U 1. FSAR, Chapter [ ]. 15.2.5 Section 1 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.4 BASES, RCS LOOPS - MODES 1 AND 2

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

2. The brackets have been removed and the proper plant specific information/value has been provided.

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

4. Changes made to be consistent with the Specification.

5. Changes made to be consistent with changes made to the Specification.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.4, RCS LOOPS - MODES 1 AND 2

There are no specific NSHC discussions for this Specification.

Attachment 1, Volume 9, Rev. 0, Page 86 of 415

ATTACHMENT 5

ITS 3.4.5, RCS LOOPS - MODE 3

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

A01

ITS 3.4.5

ITS

3/4.4 REACTOR COOLANT SYSTEM

SHUTDOWN AND HOT STANDBY

LIMITING CONDITION FOR OPERATION

LCO 3.4.5

3.4.1.2 a. At least two ~~of the~~ coolant loops ~~listed below~~ shall be OPERABLE:

1. ~~Reactor Coolant Loop 1 and its associated steam generator,~~

2. ~~Reactor Coolant Loop 2 and its associated steam generator,~~

LA01

3. Decay Heat Removal Loop 1,*

4. Decay Heat Removal Loop 2,*

See ITS 3.4.6, ITS 3.4.7, and ITS 3.4.8

b. At least one ~~of the above~~ coolant loops shall be in operation.**

c. Not more than one decay heat removal pump may be operated with the sole suction path through DH-11 and DH-12 unless the control power has been removed from the DH-11 and DH-12 valve operator, or manual valves DH-21 and DH-23 are opened.

See ITS 3.4.7 and ITS 3.4.8

d. ~~The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.~~

A02

See ITS 3.4.7 and ITS 3.4.8

APPLICABILITY: MODES 3, 4 and 5

See ITS 3.4.6

ACTION:

ACTION A, Required Action C.2

ACTION B

ACTION C

a. With ~~less than the above required~~ one coolant loops OPERABLE, ~~immediately initiate corrective action~~ to return the required coolant loops to OPERABLE status ~~as soon as possible~~ 72 hours, or be in ~~COLD SHUTDOWN~~ MODE 4 within ~~20~~ 12 hours.

A03

L01

b. With none of the above required coolant loops in operation, or two required RCS loops inoperable, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

See ITS 3.4.7 and ITS 3.4.8

*The normal or emergency power source may be inoperable in MODE 5. ~~This loop may not be selected in MODE 3 unless the primary side temperature and pressure are within the decay heat removal system's design conditions.~~

A04

**The decay heat removal pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

See ITS 3.4.7 and ITS 3.4.8

A01

ITS

3/4.4 REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 The required decay heat removal loop(s) shall be determined OPERABLE per Specification 4.0.5. See ITS 3.4.6, ITS 3.4.7, and ITS 3.4.8

SR 3.4.5.2 4.4.1.2.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side level to be greater than or equal to (a) 18 inches above the lower tube sheet once per 12 hours if an associated reactor coolant pump is operating, or, (b) 35 inches above the lower tube sheet once per 12 hours if no reactor coolant pumps are operating.

SR 3.4.5.1 4.4.1.2.3 At least one coolant loop shall be verified to be in operation ~~and circulating reactor coolant~~ at least once per 12 hours. LA02

Add proposed SR 3.4.5.3 M01

DISCUSSION OF CHANGES
ITS 3.4.5, RCS LOOPS - MODE 3

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.4.1.2.d states that the provisions of Specifications 3.0.3 and 3.0.4 are not applicable. ITS 3.4.5 does not include this exception. This changes the CTS by deleting the specific exception to Specifications 3.0.3 and 3.0.4.

This change is acceptable because it results in no technical change to the Technical Specifications. CTS 3.0.3 (and ITS 3.0.3) provides actions for when an Action is not provided in the CTS for the given unit conditions. Furthermore, it only requires a shutdown to COLD SHUTDOWN (MODE 5). Since the Applicability of CTS 3.4.1.2 includes MODE 5, this exception is needed to ensure the unit does not enter CTS 3.0.3 if an Action of CTS 3.4.1.2 was not completed. It essentially requires the Actions of CTS 3.4.1.2 to be met and not to default to the Actions of CTS 3.0.3. In the ITS, the CTS requirements have been divided up into MODE specific Specifications. Since ITS 3.4.5 covers only MODE 3, the specific exception to ITS 3.0.3 is not needed. CTS 3.0.4 provides requirements to preclude changing MODES with inoperable equipment. However, ITS LCO 3.0.4 has been modified to allow MODE changes under certain circumstances. This is justified in the Discussion of Changes for ITS Section 3.0. Therefore, this specific exception to CTS 3.0.4 is not needed in the ITS. This change is designated as administrative because it does not result in a technical change to the CTS.

A03 CTS 3.4.1.2 Action a states that when less than the required reactor coolant loops are OPERABLE, action must be immediately initiated to restore the required loops. CTS 3.4.1.2 Action b states that when no coolant loops are in operation, all operations involving a reduction in boron concentration of the RCS must be suspended and action must be immediately initiated to return the required loop to operation. ITS 3.4.5 ACTION A specifies the Required Action for one required RCS loop inoperable. The Required Action is to restore the RCS loop to OPERABLE status within 72 hours. ITS 3.4.5 ACTION C specifies the Required Actions for two required RCS loops inoperable and for no required RCS loop in operation. The Required Actions are to immediately suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1, and to immediately initiate action to restore one RCS loop to OPERABLE status and operation. This changes the CTS by revising the Actions to immediately suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1 when two RCS loops are inoperable, and breaking up the Actions for one and two inoperable RCS loops into two separate Actions. The change to when one RCS loop is inoperable

DISCUSSION OF CHANGES
ITS 3.4.5, RCS LOOPS - MODE 3

(change in time from immediately to 72 hours) is justified in Discussion of Change L01.

This change is acceptable because it results in no technical changes to the CTS. When both required RCS loops are inoperable, in all likelihood no RCS loops will be in operation. With no RCS loops in operation at the same time as both required RCS loops are inoperable, the same ITS ACTION (ACTION C) would be required. Therefore, since ITS 3.4.5 ACTION C would also require entry when no RCS loops are in operation, the identical actions would be required (i.e., immediately suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1). This change is designated as administrative because it does not result in any technical changes to the CTS.

A04 CTS LCO 3.4.1.2 Applicability Note * states that decay heat removal loops may not be used in MODE 3 to meet the LCO requirements, unless the primary side temperature and pressure are within the Decay Heat Removal System's design conditions. This Note is not included in the ITS. This changes the CTS by deleting the Applicability Note describing when decay heat removal loops can be used to meet the LCO requirements.

The purpose of the Note in CTS was to ensure DHR cooling is placed in service only if the required design parameters for DHR are met. As described in the ITS 3.4.5 Bases, LCO section, only the RCS loops are allowed to be used to meet the LCO requirements. The decay heat removal pumps are not described as an acceptable means for meeting the LCO. Therefore, the Applicability Note * is not needed for this MODE 3 Specification. This change is designated as administrative because no technical changes are being made to the CTS.

MORE RESTRICTIVE CHANGES

M01 ITS SR 3.4.5.3 requires verification that correct breaker alignment and indicated power are available to each required pump. A Note further explains that the Surveillance is not required to be performed until 24 hours after a required pump is not in operation. This Surveillance is not required by the CTS. This changes the CTS by requiring verification of correct breaker alignment and indicated power availability on required reactor coolant pumps that are not in operation.

The purpose of the ITS SR 3.4.5.3 is to ensure a standby pump is available to provide RCS cooling should the operating pump fail. This change is acceptable because the verification of proper breaker alignment and power availability ensures that an additional reactor coolant pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. This change is designated as more restrictive because it requires performance of the Surveillance on the non-operating reactor coolant pump.

RELOCATED SPECIFICATIONS

None

DISCUSSION OF CHANGES
ITS 3.4.5, RCS LOOPS - MODE 3

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.4.1.2.a.1 and 3.4.1.2.a.2 contain a description of what constitutes an OPERABLE coolant loop. ITS 3.4.5 does not include this description of what constitutes an OPERABLE coolant loop. This changes the CTS by moving the details of what constitutes an OPERABLE coolant loop to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains a requirement for the RCS loops to be OPERABLE. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 3 - Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.4.1.2.3 states that the required coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours. ITS SR 3.4.5.1 states that one RCS loop shall be verified to be in operation every 12 hours. This changes the CTS by moving the Surveillance Requirement details, to verify that the coolant loops are circulating reactor coolant to the Bases.

The removal of this detail for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be in the Technical Specifications in order to provide adequate protection of the public health and safety. The ITS retains the requirement that an RCS loop be in operation. This will require recirculation of reactor coolant since the ITS Bases specify that verification that a reactor coolant loop is in operation includes flow rate, temperature, or pump status monitoring, which helps ensure that forced or natural circulation flow is providing heat removal. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 1 – Relaxation of LCO Requirements)* CTS 3.4.1.2 Action a, which applies when one or both required coolant loops are inoperable, states immediately initiate corrective action to return the required coolant loops to

DISCUSSION OF CHANGES
ITS 3.4.5, RCS LOOPS - MODE 3

OPERABLE status as soon as possible, or be in COLD SHUTDOWN within 20 hours. ITS 3.4.5 ACTION A, which applies when one RCS loop is inoperable, requires restoration of the RCS loop to OPERABLE status within 72 hours. If not restored, ITS 3.4.5 ACTION B requires the unit to be in MODE 4 within 12 hours. This changes the CTS by allowing 72 hours to restore one inoperable RCS loop in lieu of requiring immediate action to be taken to restore the RCS loop, and allowing 12 hours to reach MODE 4 in lieu of 20 hours to reach MODE 5. Once in MODE 4, ITS 3.4.6 would become applicable.

The purpose of CTS 3.4.1.2 Action a is to provide appropriate compensatory measures when an RCS loop is inoperable. This change is acceptable since another RCS loop remains OPERABLE and capable of removing the decay heat. In addition, this remaining RCS loop is still required to be in operation with a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core. The proposed 72 hour Completion Time is reasonable, considering the low probability of an event resulting in loss of the remaining RCS loop. Furthermore, the Applicability of ITS 3.4.5 is MODE 3. Therefore, the requirement to only require placing the unit in MODE 4 in lieu of MODE 5 (COLD SHUTDOWN) is acceptable because being in MODE 4 exits the Applicability. The proposed Completion Time of 12 hours is reasonable, based on operating experience, to achieve cooldown and depressurization from MODE 3 without challenging plant systems. This change is designated as less restrictive since a longer Completion Time is being provided in the ITS than in the CTS.

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

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops - MODE 3

3.4.1.2.a, 3.4.1.2.b

LCO 3.4.5 Two RCS loops shall be OPERABLE and one RCS loop shall be in operation.

-----NOTE-----

~~All reactor coolant pumps (RCPs) may be removed from operation 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:~~

~~a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1 and~~

~~b. Core outlet temperature is maintained at least [10]°F below saturation temperature.~~

1

APPLICABILITY: MODE 3.

ACTIONS

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a	A. One RCS loop inoperable.	A.1 Restore RCS loop to OPERABLE status.	72 hours
Action a	B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	12 hours

CTS

ACTIONS (continued)

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a, Action b	C. Two RCS loops inoperable. OR Required RCS loop not in operation.	C.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1. AND C.2 Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY	
4.4.1.2.3	SR 3.4.5.1	Verify one RCS loop is in operation.	12 hours	
DOC M01	SR 3.4.5.2 3	-----------------------------NOTE----------------------------- Not required to be performed until 24 hours after a required pump is not in operation. --- Verify correct breaker alignment and indicated power available to each required pump.	7 days	1
4.4.1.2.2	SR 3.4.5.2	Verify, for each required RCS loop, SG secondary side water level is: a) ≥ 18 inches above the lower tube sheet if associated reactor coolant pump is operating; or b) ≥ 35 inches above the lower tube sheet if reactor coolant pumps are not operating.	12 hours	1

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.5, RCS LOOPS - MODE 3

1. This LCO Note allowance has been deleted since it is not required. Davis-Besse is allowed to credit natural circulation flow to meet the LCO requirements. This was approved by the NRC as documented in the NRC Safety Evaluation for Amendment 38. Furthermore, ITS SR 3.4.5.2 has been added to ensure adequate SG water level, consistent with current licensing basis.

Attachment 1, Volume 9, Rev. 0, Page 98 of 415

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops - MODE 3

BASES

BACKGROUND

The primary function of the reactor coolant in MODE 3 is removal of decay heat and transfer of this heat, via the steam generators (SGs), to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 3, reactor coolant pumps (RCPs) are used to provide forced circulation for heat removal during heatup and cooldown. The number of RCPs in operation will vary depending on operational needs, and ~~the~~ ~~intent of this~~ LCO ~~is to provide~~ forced flow from at least one RCP for core heat removal and transport. The flow provided by one RCP is adequate for heat removal and for boron mixing. However, two RCS loops are required to be OPERABLE to provide redundant paths for heat removal.

if forced flow is used to meet the

is provided 5

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Reactor coolant natural circulation is not normally used; however, the natural circulation flow rate is sufficient for core cooling. If entry into natural circulation is required, the reactor coolant at the highest elevation of the hot leg must be maintained subcooled for single phase circulation. When in natural circulation, it is preferable to remove heat using both SGs to avoid idle loop stagnation that might occur if only one SG were in service. One generator will provide adequate heat removal. ~~Boron reduction in natural circulation is prohibited because mixing to obtain a homogeneous concentration in all portions of the RCS cannot be ensured.~~

and boron mixing 5

5

APPLICABLE SAFETY ANALYSES

No safety analyses are performed with initial conditions in MODE 3.

related to loss of RCS loops

Failure to provide heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

RCS Loops - MODE 3 satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The purpose of this LCO is to require two loops to be available for heat removal thus providing redundancy. The LCO requires the two loops to be OPERABLE with the intent of requiring both SGs to be capable of transferring heat from the reactor coolant at a controlled rate. Forced reactor coolant flow is the ~~required~~ normal way to transport heat, although natural circulation flow provides adequate removal and can be used to meet the LCO requirements. A minimum of one running RCP meets the LCO requirement for one loop in operation, when forced flow is being used to meet the LCO requirements. Furthermore, the requirements for a loop in operation are also met when natural circulation is established. 5 5

~~The Note permits a limited period of operation without RCPs. All RCPs may be removed from operation for ≤ 8 hours per 24 hour period for the transition to or from the Decay Heat Removal (DHR) System, and otherwise may be de-energized for ≤ 1 hour per 8 hour period. This means that natural circulation has been established. When in natural circulation, boron reduction with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, is prohibited because an even concentration distribution throughout the RCS cannot be ensured. Core outlet temperature is to be maintained at least [10]°F below the saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.~~ 5

~~In MODES 3, 4, and 5, it is sometimes necessary to stop all RCP or DHR pump forced circulation (e.g., change operation from one DHR train to the other, to perform surveillance or startup testing, to perform the transition to and from DHR System cooling, or to avoid operation below the RCP minimum net positive suction head limit). The time period is acceptable because natural circulation is adequate for heat removal, or the reactor coolant temperature can be maintained subcooled and boron stratification affecting reactivity control is not expected.~~

If forced flow is used, ~~A~~an OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE INSERT 1. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required. INSERT 2 5 1 5

APPLICABILITY

In MODE 3, the heat load is lower than at power; therefore, one RCS loop in operation is adequate for transport and heat removal. A second RCS loop is required to be OPERABLE but not in operation for redundant heat removal capability.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.6, "RCS Loops - MODE 4"; and
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled." 3

Attachment 1, Volume 9, Rev. 0, Page 101 of 415

5

INSERT 1

Alternately, if natural circulation is used, an OPERABLE RCS loop consists of an SG that is OPERABLE.

5

INSERT 2

For forced flow, an OPERABLE steam generator requires ≥ 18 inches of secondary side water level above the lower tube sheet. For natural circulation flow, an OPERABLE steam generator requires ≥ 35 inches of secondary side water level above the lower tube sheet. In both cases, the steam generator maximum level must be maintained low enough such that the steam generator remains capable of decay heat removal by maintaining a steam flow path (i.e., ≤ 625 inches full range level).

BASES

APPLICABILITY (continued)

LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled," : 3
LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation - High Water Level" [MODE 6], and : 4 3
LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level" [MODE 6]. . 4

ACTIONS

A.1

or natural circulation 5

If one RCS loop is inoperable, redundancy for forced flow heat removal is lost. The Required Action is restoration of the RCS loop to OPERABLE status within a Completion Time of 72 hours. This time allowance is a justified period to be without the redundant nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core.

B.1

Required Action 4

If restoration of an RCS loop as required in A.1 is not possible within 72 hours, the unit must be brought to MODE 4. In MODE 4, the plant may be placed on the DHR System. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to achieve cooldown and depressurization from the existing plant conditions and without challenging plant systems.

C.1 and C.2

If two RCS loops are inoperable or a required RCS loop is not in operation, ~~except as provided in the Note in the LCO section,~~ all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be immediately suspended. Action to restore one RCS loop to operation shall be immediately initiated and continued until one RCS loop is restored to OPERABLE status and to operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without ~~forced~~ circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal. 5 5

Attachment 1, Volume 9, Rev. 0, Page 103 of 415

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification every 12 hours that the required ~~number of~~ loop~~s and pumps~~ is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced or natural circulation flow is providing heat removal. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess RCS loop status. In addition, control room indication and alarms will normally indicate loop status. 4 1

SR 3.4.5.2 3 6

Verification that each required RCP is OPERABLE ensures that the single failure criterion is met and that an additional RCS loop can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES None.

SR 3.4.5.2

SR 3.4.5.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side water level is either $\geq$ 18 inches above the lower tube sheet when the associated reactor coolant pump is operating (forced flow) or $\geq$ 35 inches above the lower tube sheet if reactor coolant pumps are not operating (natural circulation flow). If the SG water level is not within the associated limit, it may not be capable of providing the heat sink necessary for removal of decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level. 6

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.5 BASES, RCS LOOPS - MODE 3

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

2. The brackets have been removed and the proper plant specific information/value has been provided.

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

4. Changes made to be consistent with the Specification.

5. Changes have been made to allow natural circulation flow to meet the LCO requirements. In addition, due to this change the LCO Note was deleted, thus the Note description in the Bases has been deleted.

6. Changes made to be consistent with changes made to the Specification.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.5, RCS LOOPS - MODE 3

There are no specific NSHC discussions for this Specification.

Attachment 1, Volume 9, Rev. 0, Page 107 of 415

ATTACHMENT 6

ITS 3.4.6, RCS LOOPS - MODE 4

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

ITS 3.4.6

ITS

A01

3/4.4 REACTOR COOLANT SYSTEM

SHUTDOWN AND HOT STANDBY

LIMITING CONDITION FOR OPERATION

LCO 3.4.6

3.4.1.2 a. At least two ~~of the~~ coolant loops ~~listed below~~ shall be OPERABLE:

~~1. Reactor Coolant Loop 1 and its associated steam generator,~~

~~2. Reactor Coolant Loop 2 and its associated steam generator,~~

~~3. Decay Heat Removal Loop 1,*~~

~~4. Decay Heat Removal Loop 2,*~~

LA01

b. At least one ~~of the above~~ coolant loops shall be in operation.**

~~c. Not more than one decay heat removal pump may be operated with the sole suction path through DH-11 and DH-12 unless the control power has been removed from the DH-11 and DH-12 valve operator, or manual valves DH-21 and DH-23 are opened.~~

LA01

~~d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.~~

A02

APPLICABILITY: MODES 3, 4 and 5

See ITS 3.4.7 and ITS 3.4.8

See ITS 3.4.5

ACTION:

ACTION A, Required Action B.2

a. With one less than the above required coolant loops OPERABLE, immediately initiate corrective action to return the required coolant loops to OPERABLE status as soon as possible, or be in COLD SHUTDOWN within 20 24 hours.

A03

L01

M01

ACTION B

b. With none of the above required coolant loops in operation, or, two required RCS loops inoperable, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

L02

*The normal or emergency power source may be inoperable in MODE 5. This loop may not be selected in MODE 3 unless the primary side temperature and pressure are within the decay heat removal system's design conditions.

See ITS 3.4.7 and ITS 3.4.8

See ITS 3.4.5

**The decay heat removal pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

See ITS 3.4.7 and ITS 3.4.8

ITS 3.4.6

ITS

A01

3/4.4 REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 The required decay heat removal loop(s) shall be determined OPERABLE per Specification 4.0.5. A04

SR 3.4.6.2 4.4.1.2.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side level to be greater than or equal to (a) 18 inches above the lower tube sheet once per 12 hours if an associated reactor coolant pump is operating, or, (b) 35 inches above the lower tube sheet once per 12 hours if no reactor coolant pumps are operating.

SR 3.4.6.1 4.4.1.2.3 At least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours. LA02

Add proposed SR 3.4.6.3 M02

DISCUSSION OF CHANGES
ITS 3.4.6, RCS LOOPS - MODE 4

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.4.1.2.d states that the provisions of Specifications 3.0.3 and 3.0.4 are not applicable. ITS 3.4.6 does not include this exception. This changes the CTS by deleting the specific exception to Specifications 3.0.3 and 3.0.4.

This change is acceptable because it results in no technical change to the Technical Specifications. CTS 3.0.3 (and ITS 3.0.3) provides actions for when an Action is not provided in the CTS for the given unit conditions. Furthermore, it only requires a shutdown to COLD SHUTDOWN (MODE 5). Since the Applicability of CTS 3.4.1.2 includes MODE 5, this exception is needed to ensure the unit does not enter CTS 3.0.3 if an Action of CTS 3.4.1.2 was not completed. It essentially requires the Actions of CTS 3.4.1.2 to be met and not to default to the Actions of CTS 3.0.3. In the ITS, the CTS requirements have been divided up into MODE specific Specifications. Since ITS 3.4.6 covers only MODE 4, the specific exception to ITS 3.0.3 is not needed. CTS 3.0.4 provides requirements to preclude changing MODES with inoperable equipment. However, ITS LCO 3.0.4 has been modified to allow MODE changes under certain circumstances. This is justified in the Discussion of Changes for ITS Section 3.0. Therefore, this specific exception to CTS 3.0.4 is not needed in the ITS. This change is designated as administrative because it does not result in a technical change to the CTS.

A03 CTS 3.4.1.2 Action a states that when less than the required reactor coolant loops are OPERABLE, action must be immediately initiated to restore the required loops. CTS 3.4.1.2 Action b states that when no coolant loops are in operation, all operations involving a reduction in boron concentration of the RCS must be suspended and action must be immediately initiated to return the required loop to operation. ITS 3.4.6 ACTION A specifies the Required Action for one required RCS loop inoperable. The Required Action is to immediately initiate action to restore the second RCS loop to OPERABLE status. ITS 3.4.6 ACTION B specifies the Required Actions for two required RCS loops inoperable and for no required RCS loop in operation. The Required Actions are to immediately suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1, and to immediately initiate action to restore one RCS loop to OPERABLE status and operation. This changes the CTS by revising the Actions to immediately suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1 when two RCS loops are inoperable and to break up the Actions for one and two inoperable RCS loops into two separate Actions.

DISCUSSION OF CHANGES
ITS 3.4.6, RCS LOOPS - MODE 4

This change is acceptable because it results in no technical changes to the CTS. When both required RCS loops are inoperable, in all likelihood no RCS loops will be in operation. With no RCS loops in operation at the same time as both required RCS loops are inoperable, the same ITS ACTION (ACTION B) would be required. Therefore, since ITS 3.4.6 ACTION B would also require entry when no RCS loops are in operation, the identical actions would be required (i.e., immediately suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1). This change is designated as administrative because it does not result in any technical changes to the CTS.

A04 CTS 4.4.1.2.1 states that the required decay heat removal loop(s) shall be determined OPERABLE per Specification 4.0.5, the inservice testing Surveillance Requirements for ASME Code Class 1, 2, and 3 components. ITS 3.4.6 does not contain this explicit Surveillance Requirement. This changes the CTS by deleting the explicit requirement to perform the inservice testing Surveillance Requirements for ASME Code Class 1, 2, and 3 components.

The purpose of CTS 4.4.1.2.1 is to ensure the appropriate inservice testing Surveillance Requirements for ASME Code Class 1, 2, and 3 components are performed for the required decay heat removal loops. The inservice testing requirements of CTS 4.0.5 are retained in ITS 5.5.7, "Inservice Testing Program." See the Discussion of Changes for ITS 5.5 for any changes to the requirements of CTS 4.0.5. The explicit cross reference is not necessary because when the system is determined to be inoperable when tested in accordance with the inservice testing program, the plant procedures will require the Decay Heat Removal System to be declared inoperable and the appropriate ITS 3.4.6 ACTIONS will be entered when applicable. This change is designated as administrative because it does not result in technical changes to the CTS.

<u>MORE RESTRICTIVE CHANGES</u>

M01 When one RCS loop is inoperable, CTS 3.4.1.2 Action a requires a unit cooldown to COLD SHUTDOWN (MODE 5) only if immediate action is not initiated to restore the inoperable RCS loop as soon as possible. As long as action is being taken to restore the loop, entry into MODE 5 is not required. Under the same conditions, ITS 3.4.6 ACTION A will require both of the CTS Actions to be taken - immediately initiating action to restore the inoperable RCS loop and a cooldown to MODE 5. This changes the CTS by requiring a unit cooldown to MODE 5 anytime one RCS loop is inoperable.

The purpose of CTS 3.4.1.2 Action a is to provide compensatory measures when an RCS loop is inoperable. The change is acceptable because placing the unit in MODE 5 is a conservative action with regard to decay heat removal. When a single RCS loop is inoperable, the other RCS loop is still capable of removing decay heat. This change is designated more restrictive because a cooldown to MODE 5 that is not required in the CTS will be required in the ITS.

M02 ITS SR 3.4.6.3 requires verification that correct breaker alignment and indicated power are available to each required pump. A Note further explains that the

DISCUSSION OF CHANGES
ITS 3.4.6, RCS LOOPS - MODE 4

Surveillance is not required to be performed until 24 hours after a required pump is not in operation. This Surveillance is not required by the CTS. This changes the CTS by requiring verification of correct breaker alignment and indicated power availability on required pumps that are not in operation.

The purpose of ITS SR 3.4.6.3 is to ensure a standby pump is available to provide RCS cooling should the operating pump fail. This change is acceptable because the verification of proper breaker alignment and power availability ensures that an additional reactor coolant pump or DHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. This change is designated as more restrictive because it requires performance of the Surveillance on the non-operating pump.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.4.1.2.a and 3.4.1.2.c contain a description of what constitutes an OPERABLE coolant loop. ITS 3.4.6 does not include this description of what constitutes an OPERABLE coolant loop. This changes the CTS by moving the details of what constitutes an OPERABLE coolant loop to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains a requirement for the RCS loops to be OPERABLE. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 3 - Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.4.1.2.3 states that the required coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours. ITS SR 3.4.6.1 states that the required DHR or RCS loop shall be verified to be in operation every 12 hours. This changes the CTS by moving the Surveillance Requirement detail to verify that the coolant loops are circulating reactor coolant to the Bases.

The removal of this detail for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be in the Technical Specifications in order to provide adequate

Attachment 1, Volume 9, Rev. 0, Page 114 of 415

DISCUSSION OF CHANGES
ITS 3.4.6, RCS LOOPS - MODE 4

protection of the public health and safety. The ITS retains the requirement that a DHR or RCS loop be in operation. This will require recirculation of reactor coolant since the ITS Bases specify that verification that a reactor coolant loop is in operation includes flow rate, temperature, or pump status monitoring, which helps ensure that forced or natural circulation flow is providing heat removal. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 3 - Relaxation of Completion Time)* CTS 3.4.1.2 Action a requires a cooldown to COLD SHUTDOWN (MODE 5) within 20 hours under certain conditions. When a cooldown to MODE 5 is required in ITS 3.4.6 ACTION A, 24 hours are provided to be in MODE 5. This changes the CTS by extending the time allowed to reach MODE 5 from 20 hours to 24 hours.

The purpose of the CTS 3.4.1.2 Action a time limit to reach MODE 5 is to provide an appropriate amount of time for the unit to be cooled down to MODE 5 conditions in a controlled manner. This change is acceptable because the proposed time is still limited, and provides additional time to reach MODE 5 in an orderly manner and without challenging plant systems. During this additional 4 hours, another RCS loop is still OPERABLE, thus capable of removing the decay heat. Furthermore, the proposed time is consistent with the time normally provided to reach MODE 5 from MODE 4 in other CTS Specifications, such as CTS 3.0.3. This change is designated as less restrictive since more time is provided in the ITS to reach MODE 5 than is provided in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* CTS 3.4.1.2 Action b states that when no coolant loops are in operation, all operations involving a reduction in boron concentration of the RCS must be suspended. ITS 3.4.6 Required Action B.1 states that operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," must be suspended. This relaxes the CTS Action by revising the action from suspending reductions in boron concentration to suspending introduction of coolant into the RCS with a boron concentration less than required to meet LCO 3.1.1.

The purpose of CTS 3.4.1.2 Action b is to ensure that "pockets" of coolant with boron concentration less than that required to maintain the SDM are not created when there is no forced or natural circulation flow through the reactor. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition and the low probability of a DBA

DISCUSSION OF CHANGES
ITS 3.4.6, RCS LOOPS - MODE 4

occurring during the repair period. As long as coolant with boron concentration less than that required to meet the SDM requirement in LCO 3.1.1 is not introduced into the RCS, there is no possibility of creating "pockets" of coolant with less than the required boron concentration. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

Attachment 1, Volume 9, Rev. 0, Page 116 of 415

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

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops - MODE 4

3.4.1.2 LCO 3.4.6 Two loops consisting of any combination of RCS loops and decay heat removal (DHR) loops shall be OPERABLE and one loop shall be in operation.

~~-----NOTE-----~~
~~All reactor coolant pumps (RCPs) may be removed from operation 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:~~

~~a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1 and~~

~~b. Core outlet temperature is maintained at least 10°F below saturation temperature.~~
~~----------~~ 2

APPLICABILITY: MODE 4.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a	A. One required loop inoperable.	A.1 Initiate action to restore a second loop to OPERABLE status. AND A.2 -----NOTE----- Only required if one DHR loop is OPERABLE. ---------- Be in MODE 5.	Immediately 24 hours

CTS

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
Actions a and b	B. Two required loops inoperable. OR Required loop not in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)." AND B.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately Immediately	1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.4.1.2.3	SR 3.4.6.1	Verify required DHR or RCS loop is in operation.	12 hours	
DOC M02	SR 3.4.6.3	-----NOTE----- Not required to be performed until 24 hours after a required pump is not in operation. ---------- Verify correct breaker alignment and indicated power available to each required pump.	7 days	2
4.4.1.2.2	SR 3.4.6.2	Verify, for each required RCS loop, SG secondary side water level is: a) ≥ 18 inches above the lower tube sheet if associated reactor coolant pump is operating; or b) ≥ 35 inches above the lower tube sheet if reactor coolant pumps are not operating.	12 hours	2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.6, RCS LOOPS - MODE 4

1. The title of the LCO has been provided since this is the first reference to the LCO.

2. This LCO Note allowance has been deleted since it is not required. Davis-Besse is allowed to credit natural circulation flow to meet the LCO requirements. This was approved by the NRC as documented in the NRC Safety Evaluation for Amendment 38. Furthermore, ITS SR 3.4.6.2 has been added to ensure adequate SG water level, consistent with current licensing basis.

Attachment 1, Volume 9, Rev. 0, Page 120 of 415

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the steam generators (SGs) or decay heat removal (DHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

forced

If forced flow is used to meet the

is provided

INSERT 1

In MODE 4, either reactor coolant pumps (RCPs) or DHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RCP or one DHR pump for decay heat removal and transport. The flow provided by one RCP or one DHR pump is adequate for heat removal. The other intent of this LCO is to require that two paths (loops) be available to provide redundancy for heat removal. 5

related to loss of RCS loops

APPLICABLE SAFETY ANALYSES

No safety analyses are performed with initial condition in MODE 4. 2

RCS Loops - MODE 4 satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require that two loops, RCS or DHR, be OPERABLE in MODE 4 and one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS or DHR System loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. The second loop that is required to be OPERABLE provides redundant paths for heat removal. 5 3

or natural

a

~~The Note permits a limited period of operation without RCPs. All RCPs may be removed from operation for ≤ 8 hours per 24 hour period for the transition to or from the DHR System and otherwise may be de-energized for ≤ 1 hour per 8 hour period. This means that natural circulation has been established using the SGs. The Note prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained when forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained at least 10°F below saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.~~ 5

5

INSERT 1

Reactor coolant natural circulation is not normally used; however, the natural circulation flow rate is sufficient for core cooling and boron mixing. If entry into natural circulation is required, the reactor coolant at the highest elevation of the hot leg must be maintained subcooled for single phase circulation. When in natural circulation, it is preferable to remove heat using both SGs to avoid idle loop stagnation that might occur if only one SG were in service. One generator will provide adequate heat removal.

BASES

LCO (continued)

~~The Note also permits the DHR pumps to be stopped for ≤ 1 hour per 8 hour period. When the DHR pumps are stopped, no alternate heat removal path exists, unless the RCS and SGs have been placed in service in forced or natural circulation. The response of the RCS without the DHR System depends on the core decay heat load and the length of time that the DHR pumps are stopped. As decay heat diminishes, the effects on RCS temperature and pressure diminish. Without cooling by DHR, higher heat loads will cause the reactor coolant temperature and pressure to increase at a rate proportional to the decay heat load. Because pressure can increase, the applicable system pressure limits (pressure and temperature (P/T) or low temperature overpressure protection (LTOP) limits) must be observed and forced DHR flow or heat removal via the SGs must be re-established prior to reaching the pressure limit. The circumstances for stopping both DHR trains are to be limited to situations where:~~

~~a. Pressure and pressure and temperature increases can be maintained well within the allowable pressure (P/T and LTOP) and 10°F subcooling limits or~~

~~b. An alternate heat removal path through the SG is in operation.~~ 5

If forced flow is used, ~~An~~ an OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE. 5 2

INSERT 2

Similarly for the DHR System, an OPERABLE DHR loop is comprised of the OPERABLE DHR pump(s) capable of providing forced flow to the DHR cooler ~~heat exchanger~~(s). DHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. INSERT 3 2

APPLICABILITY

In MODE 4, this LCO applies because it is possible to remove core decay heat and to provide proper boron mixing with either the RCS loops and SGs or the DHR System.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2;"
LCO 3.4.5, "RCS Loops - MODE 3;"
LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled;" 1
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled;"
LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation – High Water Level" ~~(MODE 6)~~; and 3
LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation – Low Water Level". ~~(MODE 6).~~ 3

Attachment 1, Volume 9, Rev. 0, Page 124 of 415

2 **INSERT 2**

Alternately, if natural circulation is used, an OPERABLE RCS loop consists of an SG that is OPERABLE. For forced flow, an OPERABLE steam generator requires ≥ 18 inches of secondary water level above the lower tube sheet. For natural circulation flow, an OPERABLE steam generator requires ≥ 35 inches of secondary water level above lower tube sheet. In both cases, the steam generator maximum level must be maintained low enough such that the steam generator remains capable of decay heat removal by maintaining a steam flow path (i.e., ≤ 625 inches full range level).

2 **INSERT 3**

Furthermore, the two DHR loops share the same suction path through DH-11 and DH-12. Therefore, when both DHR loops are being used to meet the LCO requirements, control power is required to be removed from DH-11 and DH-12 valve operators, or manual valves DH-21 and DH-23 are required to be open. Additionally, since the DHR System is a manually operated system (i.e., it is not automatically actuated), each DHR loop is OPERABLE if it can be manually aligned (remote or local) to the decay heat removal mode.

ACTIONS

A.1

If only one required RCS loop or DHR loop is OPERABLE and in operation, redundancy for heat removal is lost. Action must be initiated to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

A.2

If restoration is not accomplished and a DHR loop is OPERABLE, the unit must be brought to MODE 5 within the following 24 hours. Bringing the unit to MODE 5 is a conservative action with regard to decay heat removal. With only one DHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining DHR loop, it would be safer to initiate that loss from MODE 5 rather than MODE 4. The Completion Time of 24 hours is reasonable, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems.

This Required Action is modified by a Note which indicates that the unit must be placed in MODE 5 only if a DHR loop is OPERABLE. With no DHR loop OPERABLE, the unit is in a condition with only limited cooldown capabilities. Therefore, the actions are to be concentrated on the restoration of a DHR loop, rather than a cooldown of extended duration.

B.1 and B.2

If two required RCS or DHR loops are inoperable or a required loop is not in operation, ~~except during conditions permitted by the Note in the LCO section,~~ all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RCS or DHR loop to OPERABLE status and operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant, into the RCS, with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to ensure continued safe operation. With coolant added without ~~forced~~ circulation, unmixed coolant could be introduced to the core, however, coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must continue until one loop is restored to operation.

4

5

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This Surveillance requires verification every 12 hours of the required DHR or RCS loop in operation to ensure forced flow is providing decay heat removal. Verification includes flow rate, temperature, or pump status monitoring. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess RCS loop status. In addition, control room indication and alarms will normally indicate loop status. or natural circulation 5 4

INSERT 4 4

SR 3.4.6.2 3

Verification that each required pump is OPERABLE ensures that an additional RCS or DHR loop can be placed in operation if needed to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other administrative controls and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES None.

4 **INSERT 4**

SR 3.4.6.2

SR 3.4.6.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side water level is either ≥ 18 inches above the lower tube sheet when the associated reactor coolant pump is operating (forced flow) or ≥ 35 inches above the lower tube sheet if reactor coolant pumps are not operating (natural circulation flow). If the SG water level is not within the associated limit, it may not be capable of providing the heat sink necessary for removal of decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.6 BASES, RCS LOOPS - MODE 4

1. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01, Section 5.1.3.

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

3. Changes made to be consistent with the Specification.

4. Changes made to be consistent with changes made to the Specification.

5. Changes have been made to allow natural circulation flow to meet the LCO requirements. In addition, due to these changes, the LCO Note was deleted; thus the Note description in the Bases has also been deleted.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.6, RCS LOOPS - MODE 4

There are no specific NSHC discussions for this Specification.

ATTACHMENT 7

ITS 3.4.7, RCS LOOPS - MODE 5, LOOPS FILLED

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

ITS 3.4.7

A01

ITS

3/4.4 REACTOR COOLANT SYSTEM

SHUTDOWN AND HOT STANDBY

LIMITING CONDITION FOR OPERATION

LCO 3.4.7

3.4.1.2 a. At least two ~~of the~~ coolant loops ~~listed below~~ shall be OPERABLE: LA01

~~1. Reactor Coolant Loop 1 and its associated steam generator,~~

~~2. Reactor Coolant Loop 2 and its associated steam generator,~~ LA01

~~3. Decay Heat Removal Loop 1,*~~

~~4. Decay Heat Removal Loop 2,*~~

b. At least one ~~of the above~~ coolant loops shall be in operation.**

~~c. Not more than one decay heat removal pump may be operated with the sole suction path through DH-11 and DH-12 unless the control power has been removed from the DH-11 and DH-12 valve operator, or manual valves DH-21 and DH-23 are opened.~~ LA01

~~d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.~~ A02

APPLICABILITY: MODES 3, 4 and 5 A07 See ITS 3.4.6 See ITS 3.4.5

ACTION:

ACTION A, Required Action B.2

a. With one less than the above required coolant loops OPERABLE, immediately initiate corrective action to return the required coolant loops to OPERABLE status as soon as possible, or be in COLD SHUTDOWN within 20 hours. A03 See ITS 3.4.5 and ITS 3.4.6

ACTION B

b. With none of the above required coolant loops in operation or two required DHR loops inoperable, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation. L01

A04

*The normal or emergency power source may be inoperable in MODE 5. This loop may not be selected in MODE 3 unless the primary side temperature and pressure are within the decay heat removal system's design conditions. See ITS 3.4.5

~~**The decay heat removal pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.~~ M01

ITS 3.4.7

A01

ITS

3/4.4 REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 The required decay heat removal loop(s) shall be determined OPERABLE per Specification 4.0.5. A05

SR 3.4.7.2 4.4.1.2.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side level to be greater than or equal to (a) 18 inches above the lower tube sheet once per 12 hours if an associated reactor coolant pump is operating, or (b) 35 inches above the lower tube sheet once per 12 hours if no reactor coolant pumps are operating.

See ITS 3.4.5 and ITS 3.4.6

A06

SR 3.4.7.1 4.4.1.2.3 At least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours. LA02

Add proposed SR 3.4.7.3 M02

Attachment 1, Volume 9, Rev. 0, Page 135 of 415

DISCUSSION OF CHANGES
ITS 3.4.7, RCS LOOPS - MODE 5, LOOPS FILLED

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.4.1.2.d states that the provisions of Specifications 3.0.3 and 3.0.4 are not applicable. ITS 3.4.7 does not include this exception. This changes the CTS by deleting the specific exception to Specifications 3.0.3 and 3.0.4.

This change is acceptable because it results in no technical change to the Technical Specifications. ITS LCO 3.0.3 (which is equivalent to CTS 3.0.3) specifically states that it is not Applicable in MODE 5, which is the Applicability of ITS 3.4.7. Therefore, this exception to CTS 3.0.3 is redundant and unnecessary. CTS 3.0.4 provides requirements to preclude changing MODES with inoperable equipment. However, ITS LCO 3.0.4 has been modified to allow MODE changes under certain circumstances. This is justified in the Discussion of Changes for ITS Section 3.0. Therefore, this specific exception to CTS 3.0.4 is not needed in the ITS. This change is designated as administrative because it does not result in a technical change to the CTS.

A03 CTS 3.4.1.2 Action a states that when less than the required reactor coolant loops are OPERABLE, action must be immediately initiated to restore the required loops. CTS 3.4.1.2 Action b states that when no coolant loops are in operation, all operations involving a reduction in boron concentration of the RCS must be suspended and action must be immediately initiated to return the required loop to operation. ITS 3.4.7 ACTION A specifies the Required Actions when one of the two required loops is inoperable. Required Action A.1 is to immediately initiate action to restore the second loop to OPERABLE status. ITS 3.4.7 ACTION B specifies the Required Actions when two required loops are inoperable and when no required loop is in operation. The Required Actions are to immediately suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1, and to immediately initiate action to restore one loop to OPERABLE status and operation. This changes the CTS by revising the Actions to immediately suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1 when two required loops are inoperable and to break up the Actions for one and two inoperable required loops into two separate Actions.

This change is acceptable because it results in no technical changes to the CTS. When both required loops are inoperable, in all likelihood no loops will be in operation. With no loops in operation at the same time as both required loops are inoperable, the same ITS ACTION (ACTION B) would be required. Therefore, since ITS 3.4.7 ACTION B would also require entry when no loops are in operation, the identical actions would be required (i.e., immediately suspend

DISCUSSION OF CHANGES
ITS 3.4.7, RCS LOOPS - MODE 5, LOOPS FILLED

operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1). This change is designated as administrative because it does not result in any technical changes to the CTS.

A04 CTS 3.4.1.2 footnote * states the decay heat removal (DHR) loops normal or emergency power may be inoperable in MODE 5. ITS 3.4.7 has not retained this specific footnote allowance. This changes the CTS by deleting a specific footnote allowance concerning power supplies.

This change is acceptable because the ITS definition of OPERABLE - OPERABILITY requires an OPERABLE component to have only a normal or an emergency power source. This change to the CTS definition of OPERABLE - OPERABILITY is discussed in the ITS Section 1.0 Discussion of Changes. Given this change to the definition of OPERABLE - OPERABILITY, a specific allowance for the DHR loops is not required. This change is designated as an administrative change since it does not result in a technical change to the CTS.

A05 CTS 4.4.1.2.1 states that the required decay heat removal loop(s) shall be determined OPERABLE per Specification 4.0.5, the inservice testing Surveillance Requirements for ASME Code Class 1, 2, and 3 components. ITS 3.4.7 does not contain this explicit Surveillance Requirement. This changes the CTS by deleting the explicit requirement to perform the inservice testing Surveillance Requirements for ASME Code Class 1, 2, and 3 components.

The purpose of CTS 4.4.1.2.1 is to ensure the appropriate inservice testing Surveillance Requirements for ASME Code Class 1, 2, and 3 components are performed for the required decay heat removal loops. The inservice testing requirements of CTS 4.0.5 are retained in ITS 5.5.7, "Inservice Testing Program." See the Discussion of Changes for ITS 5.5 for any changes to the requirements of CTS 4.0.5. The explicit cross reference is not necessary because when the system is determined to be inoperable when tested in accordance with the inservice testing program, the plant procedures will require the Decay Heat Removal System to be declared inoperable and the appropriate ITS 3.4.7 ACTIONS will be entered when applicable. This change is designated as administrative because it does not result in technical changes to the CTS.

A06 CTS 4.4.1.2.2, in part, specifies the steam generator water level requirements for when the reactor coolant pumps (RCPs) are not operating. ITS LCO 3.4.7 and SR 3.4.7.2 provide the same steam generator water level requirements, but do not state that this level is for when the RCPs are not operating. This changes the CTS by deleting the amplifying information that the RCPs are not operating.

The change is acceptable since the unit is in MODE 5 and the RCPs are not routinely operated in MODE 5, and the ITS 3.4.7 Bases, LCO section, clearly defines the required loop does not include an RCP, only the steam generators. This change is designated as administrative because it does not result in any technical changes to the CTS.

A07 CTS 3.4.1.2 includes all MODE 5 coolant loop requirements in one Specification. ITS 3.4.7 includes only the MODE 5, Loops Filled requirements. The MODE 5,

DISCUSSION OF CHANGES
ITS 3.4.7, RCS LOOPS - MODE 5, LOOPS FILLED

Loops Not Filled requirements are included in ITS 3.4.8. This changes the CTS by splitting the MODE 5 requirements into two Specifications.

This change is acceptable since all facets of MODE 5 operation are covered in the two ITS Specifications. This change is designated as administrative because it does not result in any technical changes.

MORE RESTRICTIVE CHANGES

M01 CTS 3.4.1.2 states the number of coolant loops that shall be OPERABLE, and states that at least one loop must be in operation. This requirement is modified by footnote ** that states that the DHR pumps may be de-energized for up to 1 hour, provided certain requirements are met. ITS 3.4.7 does not include this allowance.

The purpose of the 1 hour allowance is to allow the DHR pump to be removed from operation to perform various activities, such as to place the other DHR pump in service. However, this allowance is not necessary since the CTS already allows natural circulation flow to be used as a means to meet the LCO 3.4.1.2.b requirement that a loop be in operation. This CTS footnote essentially allows all DHR pumps to be de-energized and natural circulation to not be occurring for up to 1 hour. This change is acceptable since it will ensure either a DHR pump is in operation or one RCS loop will be in natural circulation at all times; otherwise ACTIONS to restore a loop to operation will be required. This change is designated as more restrictive because it will ensure at least one loop is in operation (either a DHR loop with forced flow or an RCS loop with natural circulation flow).

M02 ITS SR 3.4.7.3 requires verification that correct breaker alignment and indicated power are available to each required pump. A Note further explains that the Surveillance is not required to be performed until 24 hours after a required pump is not in operation. This Surveillance is not required by the CTS. This changes the CTS by requiring verification of correct breaker alignment and indicated power availability on required DHR pumps that are not in operation.

The purpose of ITS SR 3.4.7.3 is to ensure a standby pump is available to provide RCS cooling should the operating pump fail. This change is acceptable because the verification of proper breaker alignment and power availability ensures that an additional DHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. This change is designated as more restrictive because it requires performance of the Surveillance on the non-operating pump.

RELOCATED SPECIFICATIONS

None

DISCUSSION OF CHANGES
ITS 3.4.7, RCS LOOPS - MODE 5, LOOPS FILLED

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.4.1.2.a and 3.4.1.2.c contain a description of what constitutes an OPERABLE coolant loop. ITS 3.4.7 does not include this description of what constitutes an OPERABLE coolant loop. This changes the CTS by moving the details of what constitutes an OPERABLE coolant loop to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains a requirement for the RCS loops or decay heat removal loops to be OPERABLE. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 3 - Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.4.1.2.3 states that the required coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours. ITS SR 3.4.7.1 states that the required DHR loop shall be verified to be in operation every 12 hours. This changes the CTS by moving the Surveillance Requirement to verify that the coolant loops are circulating reactor coolant to the Bases.

The removal of this detail for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be in the Technical Specifications in order to provide adequate protection of the public health and safety. The ITS retains the requirement that a DHR loop be in operation. This will require recirculation of reactor coolant since the ITS Bases specify that verification that a DHR loop is in operation includes flow rate, temperature, or pump status monitoring, which helps ensure that forced flow is providing heat removal. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.4.1.2 Action b states that when no coolant loops are in operation, all operations involving a reduction in boron concentration of the RCS must be suspended. ITS 3.4.7 Required

DISCUSSION OF CHANGES
ITS 3.4.7, RCS LOOPS - MODE 5, LOOPS FILLED

Action B.1 states that operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," must be suspended. This relaxes the CTS Action by revising the action from suspending reductions in boron concentration to suspending introduction of coolant into the RCS with a boron concentration less than required to meet LCO 3.1.1.

The purpose of CTS 3.4.1.2 Action b is to ensure that "pockets" of coolant with boron concentration less than that required to maintain the SDM are not created when there is no forced or natural circulation flow through the reactor. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition and the low probability of a DBA occurring during the repair period. As long as coolant with boron concentration less than that required to meet the SDM requirement in LCO 3.1.1 is not introduced into the RCS, there is no possibility of creating "pockets" of coolant with less than the required boron concentration. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

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

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops - MODE 5, Loops Filled

Two loops consisting of any combination of RCS loops and decay heat removal (DHR) loops shall be OPERABLE and one loop shall be in operation.

3.4.1.2 LCO 3.4.7

~~One decay heat removal (DHR) loop shall be OPERABLE and in operation, and either:~~

~~a. One additional DHR loop shall be OPERABLE or~~

~~b. The secondary side water level of each steam generator (SG) shall be ≥ [50]%.~~ 1

-----NOTES-----

~~1. The DHR pump of the loop in operation may be removed from operation for ≤ 1 hour per 8 hour period provided:~~

~~a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1 and~~

~~b. Core outlet temperature is maintained at least 10°F below saturation temperature.~~ 1

~~2. One required DHR loop may be inoperable for up to 2 hours for surveillance testing provided that the other DHR loop is OPERABLE and in operation.~~

~~3. All DHR loops may be not in operation during planned heatup to MODE 4 when at least one RCS loop is in operation.~~

APPLICABILITY: MODE 5 with RCS loops filled.

CTS

All changes are 1 unless otherwise noted

ACTIONS

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a	A. One required DHR loop inoperable. ~~AND~~ ~~One DHR loop OPERABLE.~~	A.1 Initiate action to restore a second DHR loop to OPERABLE status.	Immediately
		~~OR~~ ~~A.2 Initiate action to restore required SGs secondary side water levels to within limits.~~	~~Immediately~~
	~~B. One or more required SGs with secondary side water level not within limit.~~ ~~AND~~ ~~One DHR loop OPERABLE.~~	~~B.1 Initiate action to restore a second DHR loop to OPERABLE status.~~ ~~OR~~ ~~B.2 Initiate action to restore required SGs secondary side water level to within limit.~~	~~Immediately~~ ~~Immediately~~
Action a, Action b	~~C.~~ B. ~~No~~ Two required DHR loops ~~OPERABLE~~ inoperable. OR Required DHR loop not in operation.	~~C.1~~ B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1~~.~~ . "SHUTDOWN MARGIN (SDM)." 2 AND ~~C.2~~ B.2 Initiate action to restore one DHR loop to OPERABLE status and operation.	Immediately Immediately

CTS

SURVEILLANCE REQUIREMENTS

CTS		SURVEILLANCE	FREQUENCY	
4.4.1.2.3	SR 3.4.7.1	or RCS Verify required DHR loop is in operation.	12 hours	1
4.4.1.2.2	SR 3.4.7.2	, for each required RCS loop, Verify ~~required~~ SG secondary side water level~~s are~~ is ≥ ~~[50]%~~ 35 inches above the lower tube sheet.	12 hours	1 3
DOC M02	SR 3.4.7.3	------------------------------NOTE------------------------------ Not required to be performed until 24 hours after a required pump is not in operation. -- Verify correct breaker alignment and indicated power available to each required DHR pump.	 7 days	

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.7, RCS LOOPS - MODE 5, LOOPS FILLED

1. The Specification has been modified to allow credit for natural circulation flow to meet the LCO requirements. Thus, any combination of DHR and RCS loops can be used to meet both the OPERABLE and in operation requirements, similar to the ITS 3.4.6 requirements. This was approved by the NRC as documented in the Safety Evaluation for Amendment 38. Furthermore, due to this change, the NOTES have been deleted and the ACTIONS have been modified to reflect the natural circulation option. The proposed ACTIONS are consistent with the ACTIONS of ITS 3.4.6, which has similar LCO requirements. In addition, ITS SR 3.4.7.1 and SR 3.4.7.2 have been modified to reflect the natural circulation allowances.

2. The title of the LCO has been provided since this is the first reference to the LCO.

3. Removed brackets and provided plant specific information.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat either to the steam generator (SG) secondary side coolant or the component cooling water via the decay heat removal (DHR) ~~heat exchangers~~ coolers. While the principal means for decay heat removal is via the DHR System, the SGs are specified as a backup means for redundancy. Although the SGs cannot remove heat unless steaming occurs ~~(which is not possible in MODE 5)~~, they are available as a temporary heat sink and can be used by allowing the RCS to heat up into the temperature region of MODE 4 where steaming can be effective for heat removal. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid. 2 2

In MODE 5 with RCS loops filled, DHR loops are the principal means for heat removal. The number of loops in operation can vary to suit the operational needs. If forced flow is used to meet the ~~The intent of this~~ LCO ~~is to provide~~ is provided forced flow from at least one DHR loop for decay heat removal and transport. The flow provided by one DHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for heat removal. 5

INSERT 1 5

The LCO provides for either SG heat removal or DHR System heat removal. In this MODE, reactor coolant pump (RCP) operation may be restricted because of net positive suction head (NPSH) limitations, and the SG will not be able to provide steam for the turbine driven feed pumps. However, to ensure that the SGs can be used as a heat sink, an electrically ~~motor~~ driven feed~~water~~ pump is needed, because it is independent of steam. The Startup Feed Pump ~~Condensate pumps, startup pumps,~~ or the Motor Driven Feedwater Pump ~~motor driven auxiliary feedwater pump~~ can be used. If RCPs are available, the steam generator level need not be adjusted. If RCPs are not available, the water level must be adjusted for natural circulation. The high entry point in the generator is ~~should be~~ accessible from the ~~feedwater pumps~~ so that natural circulation can be further stimulated, if needed. The SGs are primarily a backup to the DHR pumps, which are used for forced flow. By requiring the SGs to be a backup heat removal path, the option to increase RCS pressure and temperature for heat removal in MODE 4 is provided. 2

5 **INSERT 1**

Reactor coolant natural circulation is not normally used; however, the natural circulation flow rate is sufficient for core cooling and boron mixing. If entry into natural circulation is required, the reactor coolant at the highest elevation of the hot leg must be maintained subcooled for single phase circulation. When in natural circulation, it is preferable to remove heat using both SGs to avoid idle loop stagnation that might occur if only one SG were in service. One generator will provide adequate heat removal.

BASES

APPLICABLE SAFETY ANALYSES

No safety analyses related to loss of RCS loops are performed with initial conditions in MODE 5. 2

RCS Loops - MODE 5 (Loops Filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require that ~~at least one of the DHR~~ two loops, RCS or DHR, be OPERABLE and one of these loops be in operation ~~with an additional DHR loop OPERABLE or both SGs with secondary side water level ≥ [50]%~~. ~~One DHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. The second DHR loop is normally maintained as a backup to the operating DHR loop to provide redundancy for decay heat removal. However, if the standby DHR loop is not OPERABLE, a sufficient alternate method of providing redundant heat removal paths is to provide both SGs with their secondary side water levels ≥ [50]%. Should the operating DHR loop fail, the SGs could be used to remove the decay heat.~~ INSERT 2 5

~~Note 1 permits the DHR pumps to be removed from operation for up to 1 hour per 8 hour period. The circumstances for stopping both DHR trains are to be limited to situations where: (a) Pressure and temperature increases can be maintained well within the allowable pressure (P/T and low temperature overpressure protection) and 10°F subcooling limits or (b) Alternate heat paths through the SGs are in operation.~~

~~The Note prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained when DHR forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained at least 10°F below saturation temperature so that no vapor bubble would form and possibly cause a natural circulation flow obstruction. In this MODE, the generators are used as a backup for decay heat removal and, to ensure their availability, the RCS loop flow path is to be maintained with subcooled liquid.~~ 5

~~In MODE 5, it is sometimes necessary to stop all RCP or DHR pump forced circulation. This is permitted to change operation from one DHR train to the other, perform surveillance or startup testing, perform the transition to and from the DHR System, or to avoid operation below the RCP minimum NPSH limit. The time period is acceptable because natural circulation is acceptable for heat removal, the reactor coolant temperature can be maintained subcooled, and boron stratification affecting reactivity control is not expected.~~

5 **INSERT 2**

The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS or DHR System loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced or natural circulation. The second loop that is required to be OPERABLE provides a redundant path for heat removal.

Attachment 1, Volume 9, Rev. 0, Page 150 of 415

BASES

LCO (continued)

Note 2 allows one DHR loop to be inoperable for a period of up to 2 hours provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 3 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting DHR loops to not be in operation when at least one RCP is in operation. This Note provides for the transition to MODE 4 where an RCP is permitted to be in operation and replaces the RCS circulation function provided by the DHR loops. 5

INSERT 3

An OPERABLE DHR loop is composed of an OPERABLE DHR pump and an OPERABLE DHR heat exchanger. INSERT 4 5 2

DHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. A SG can perform as a heat sink when it has an adequate water level and is OPERABLE. 2 5

INSERT 5

APPLICABILITY

In MODE 5 with loops filled, forced circulation is provided by this LCO to remove decay heat from the core and to provide proper boron mixing. One loop of DHR provides sufficient circulation for these purposes.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2,"
LCO 3.4.5, "RCS Loops - MODE 3," 3
LCO 3.4.6, "RCS Loops - MODE 4,"
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled,"
LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation - High Water Level" (MODE 6), and 4
LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level" (MODE 6). 4

;

.

ACTIONS

A.1, A.2, B.1, and B.2

only

required RCS loop or

and in operation,

If one DHR loop is OPERABLE and any required SG has secondary side water level < [50]% or one required DHR loop inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second DHR loop to OPERABLE status or initiate action to restore the secondary side water level in the SGs, and action must be taken immediately. Either Required Action will restore redundant decay heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal. 5

(5) **INSERT 3**

An OPERABLE RCS loop consists of an SG that is OPERABLE. An OPERABLE SG requires ≥ 35 inches of secondary side water level above the lower tube sheet. In addition, the steam generator maximum level must be maintained low enough such that the steam generator remains capable of heat removal by maintaining a steam flow path (i.e., ≤ 625 inches full range level). Furthermore, the SG must be capable of transferring heat from the reactor coolant at a controlled rate.

(2) **INSERT 4**

cooler. Furthermore, the two DHR loops share the same suction path through DH-11 and DH-12. Therefore, when both DHR loops are being used to meet the LCO requirements, control power is required to be removed from DH-11 and DH-12 valve operators, or manual valves DH-21 and DH-23 are required to be open.

(2) **INSERT 5**

Additionally, since the DHR System is a manually operated system (i.e., it is not automatically actuated), each DHR loop is OPERABLE if it can be manually aligned (remote or local) to the decay heat removal mode.

BASES

ACTIONS (continued)

B

C.1 and C.2

If no required DHR loop is in operation, except as provided in Note 1, or no required DHR loop is OPERABLE, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore a DHR loop to OPERABLE status and operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal. 5

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

RCS or

This SR requires verification every 12 hours that the required DHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation. In addition, control room indication and alarms will normally indicate loop status. 5

or natural circulation

SR 3.4.7.2

required

35 inches above the lower tube sheet

Verifying the SGs are OPERABLE by ensuring their secondary side water levels are ≥ [50]% ensures that redundant heat removal paths are available if the second DHR loop is not OPERABLE. If both DHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. 5 1 5 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.7.3

35 inches above the lower tube sheet

Verification that each required DHR pump is OPERABLE ensures that redundant paths for heat removal are available. The requirement also ensures that the additional loop can be placed in operation if needed to maintain decay heat removal and reactor coolant circulation. If the secondary side water level is ≥ [50]% in both SGs, this Surveillance is not needed. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience. 1

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES None.

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.7 BASES, RCS LOOPS - MODE 5, LOOPS FILLED

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

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

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

4. Changes made to be consistent with the Specification.

5. Changes have been made to allow natural circulation flow to meet the LCO requirements. In addition, due to these changes, other associated changes to the NOTES, ACTIONS, and Surveillances have been made to be consistent with changes made to the Specification.

Attachment 1, Volume 9, Rev. 0, Page 155 of 415

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.7, RCS LOOPS - MODE 5, LOOPS FILLED

There are no specific NSHC discussions for this Specification.

ATTACHMENT 8

ITS 3.4.8, RCS LOOPS - MODE 5, LOOPS NOT FILLED

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

ITS 3.4.8

ITS

A01

3/4.4 REACTOR COOLANT SYSTEM

SHUTDOWN AND HOT STANDBY

LIMITING CONDITION FOR OPERATION

LCO 3.4.8

3.4.1.2 a. At least two ~~of the~~ coolant loops ~~listed below~~ shall be OPERABLE: LA01

1. Reactor Coolant Loop 1 and its associated steam generator,

2. Reactor Coolant Loop 2 and its associated steam generator,

See ITS 3.4.5, ITS 3.4.6, and ITS 3.4.7

3. ~~Decay Heat Removal Loop 1,*~~

4. ~~Decay Heat Removal Loop 2.*~~ LA01

b. At least one ~~of the above~~ coolant loops shall be in operation.** L01

c. ~~Not more than one decay heat removal pump may be operated with the sole suction path through DH-11 and DH-12 unless the control power has been removed from the DH-11 and DH-12 valve operator, or manual valves DH-21 and DH-23 are opened.~~ LA01

Add proposed LCO 3.4.8

d. ~~The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.~~ A02

APPLICABILITY: MODES 3, 4 and 5 A03

See ITS 3.4.6

See ITS 3.4.5

ACTION:

ACTION A

a. With one less than the above required coolant loops OPERABLE, immediately initiate corrective action to return the required coolant loops to OPERABLE status as soon as possible, ~~or be in COLD SHUTDOWN within 20 hours.~~ A04

See ITS 3.4.5 and ITS 3.4.6

ACTION B

b. With none of the above required coolant loops in operation, or two required DHR loops inoperable, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

A05

~~*The normal or emergency power source may be inoperable in MODE 5.~~ This loop may not be selected in MODE 3 unless the primary side temperature and pressure are within the decay heat removal system's design conditions.

See ITS 3.4.5

LCO 3.4.8 NOTE 1

**The decay heat removal pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration required to meet SDM of LCO 3.1.1; and (2) core outlet temperature is maintained at least 10°F below saturation temperature. M01 L02

Add proposed LCO 3.4.8 NOTE 1 part c M01

M02

Attachment 1, Volume 9, Rev. 0, Page 160 of 415

ITS

A01

3/4.4 REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

A06

4.4.1.2.1 The required decay heat removal loop(s) shall be determined OPERABLE per Specification 4.0.5.

See ITS 3.4.5, ITS 3.4.6, and ITS 3.4.7

4.4.1.2.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side level to be greater than or equal to (a) 18 inches above the lower tube sheet once per 12 hours if an associated reactor coolant pump is operating, or (b) 35 inches above the lower tube sheet once per 12 hours if no reactor coolant pumps are operating.

See ITS 3.4.5 and ITS 3.4.6

See ITS 3.4.7

SR 3.4.8.1 4.4.1.2.3 At least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

LA02

Add proposed SR 3.4.8.2

M03

DISCUSSION OF CHANGES
ITS 3.4.8, RCS LOOPS - MODE 5, LOOPS NOT FILLED

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.4.1.2.d states that the provisions of Specifications 3.0.3 and 3.0.4 are not applicable. ITS 3.4.8 does not include this exception. This changes the CTS by deleting the specific exception to Specifications 3.0.3 and 3.0.4.

This change is acceptable because it results in no technical change to the Technical Specifications. ITS LCO 3.0.3 (which is equivalent to CTS 3.0.3) specifically states that it is not Applicable in MODE 5, which is the Applicability of ITS 3.4.8. Therefore, this exception to CTS 3.0.3 is redundant and unnecessary. CTS 3.0.4 provides requirements to preclude changing MODES with inoperable equipment. However, ITS LCO 3.0.4 has been modified to allow MODE changes under certain circumstances. This is justified in the Discussion of Changes for ITS Section 3.0. Therefore, this specific exception to CTS 3.0.4 is not needed in the ITS. This change is designated as administrative because it does not result in a technical change to the CTS.

A03 CTS 3.4.1.2 includes all MODE 5 coolant loop requirements in one Specification. ITS 3.4.8 includes only the MODE 5, Loops Not Filled requirements. The MODE 5, Loops Filled requirements are included in ITS 3.4.7. This changes the CTS by splitting the MODE 5 requirements into two Specifications.

This change is acceptable since all facets of MODE 5 operation are covered in the two ITS Specifications. This change is designated as administrative because it does not result in any technical changes.

A04 CTS 3.4.1.2 Action a states that when less than the required reactor coolant loops are OPERABLE, action must be immediately initiated to restore the required loops. CTS 3.4.1.2 Action b states that when no coolant loops are in operation, all operations involving a reduction in boron concentration of the RCS must be suspended and action must be immediately initiated to return the required loop to operation. ITS 3.4.8 ACTION A specifies the Required Actions when one of the two required DHR loops is inoperable. Required Action A.1 is to immediately initiate action to restore the DHR loop to OPERABLE status. ITS 3.4.8 ACTION B specifies the Required Actions when two required DHR loops are inoperable and when no required DHR loop is in operation. The Required Actions are to immediately suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1, and to immediately initiate action to restore one DHR loop to OPERABLE status and operation. This changes the CTS by revising the Actions to immediately suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required

DISCUSSION OF CHANGES
ITS 3.4.8, RCS LOOPS - MODE 5, LOOPS NOT FILLED

to meet the requirements of LCO 3.1.1 when two required DHR loops are inoperable and to break up the Actions for one and two inoperable required DHR loops into two separate Actions.

This change is acceptable because it results in no technical changes to the CTS. When both required DHR loops are inoperable, in all likelihood no DHR loops will be in operation. With no DHR loops in operation at the same time as both required DHR loops are inoperable, the same ITS ACTION (ACTION B) would be required. Therefore, since ITS 3.4.8 ACTION B would also require entry when no DHR loops are in operation, the identical actions would be required (i.e., immediately suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1). This change is designated as administrative because it does not result in any technical changes to the CTS.

A05 CTS 3.4.1.2 footnote * states the decay heat removal (DHR) loops normal or emergency power may be inoperable in MODE 5. ITS 3.4.8 has not retained this specific footnote allowance. This changes the CTS by deleting a specific footnote allowance concerning power supplies.

This change is acceptable because the ITS definition of OPERABLE - OPERABILITY requires an OPERABLE component to have only a normal or an emergency power source. This change to the CTS definition of OPERABLE - OPERABILITY is discussed in the ITS Section 1.0 Discussion of Changes. Given this change to the definition of OPERABLE - OPERABILITY, a specific allowance for the DHR loops is not required. This change is designated as an administrative change since it does not result in a technical change to the CTS.

A06 CTS 4.4.1.2.1 states that the required decay heat removal loop(s) shall be determined OPERABLE per Specification 4.0.5, the inservice testing Surveillance Requirements for ASME Code Class 1, 2, and 3 components. ITS 3.4.8 does not contain this explicit Surveillance Requirement. This changes the CTS by deleting the explicit requirement to perform the inservice testing Surveillance Requirements for ASME Code Class 1, 2, and 3 component.

The purpose of CTS 4.4.1.2.1 is to ensure the appropriate inservice testing Surveillance Requirements for ASME Code Class 1, 2, and 3 components are performed for the required decay heat removal loops. The inservice testing requirements of CTS 4.0.5 are retained in ITS 5.5.7, "Inservice Testing Program." See the Discussion of Changes for ITS 5.5 for any changes to the requirements of CTS 4.0.5. The explicit cross reference is not necessary because when the system is determined to be inoperable when tested in accordance with the inservice testing program, the plant procedures will require the Decay Heat Removal System to be declared inoperable and the appropriate ITS 3.4.8 ACTIONS will be entered when applicable. This change is designated as administrative because it does not result in technical changes to the CTS.

DISCUSSION OF CHANGES
ITS 3.4.8, RCS LOOPS - MODE 5, LOOPS NOT FILLED

MORE RESTRICTIVE CHANGES

M01 CTS 3.4.1.2 footnote ** contains an allowance for the decay heat removal pumps to be de-energized for up to one hour. ITS LCO 3.4.8 Note 1 allows all DHR pumps to be removed from operation for ≤ 15 minutes only when switching from one loop to the other, and also requires that no draining operations to further reduce the RCS water volume are permitted (part c). This changes the CTS by reducing the time allowed for the DHR pump to be de-energized from 1 hour to 15 minutes, restricts the allowance to only pump switching operations, and adds a restriction that no draining operations are permitted to further reduce the RCS water volume.

The purpose of the CTS 3.4.1.2 footnote ** in MODE 5 with loops not filled is to allow the DHR loops to be switched from one to the other. This change is acceptable because ITS LCO 3.4.8 Note 1 provides sufficient time to perform loop switching operations and provides adequate controls. Stopping all operating DHR loops when the RCS is not filled should be limited to short periods of time because of the reduced inventory of water available to absorb decay heat. Stopping all DHR pumps during loop swapping operations may be necessary. Fifteen minutes is sufficient time to perform the loop swapping operation without excessive increases in RCS average temperature due to lack of decay heat removal. Adding the additional condition that no draining operations be performed when the pumps are stopped is reasonable given the low RCS water level and the unavailability of the DHR pumps to add inventory to the RCS, if needed. This change is more restrictive because it reduces the time a DHR loop may be out of service and adds an additional restriction.

M02 CTS 3.4.1.2 footnote ** part (2) allows the DHR pumps to be de-energized provided the core outlet temperature is maintained at least 10°F below saturation temperature. ITS LCO 3.4.8 Note 1 provides a similar allowance, but requires the maximum RCS temperature to be ≤ 190°F. This changes the CTS by requiring the RCS temperature to be ≤ 190°F instead of 10°F below saturation temperature.

The purpose of CTS 3.4.1.2 footnote ** part 2 is to help ensure the RCS temperature does not reach the boiling point. With the RCS loops not filled, the RCS pressure would be at atmospheric pressure. Thus 10°F below saturation temperature is 202°F. This change is acceptable because the proposed change increases the margin to the boiling point since it requires the maximum RCS temperature be ≤ 190°F. Furthermore, the 190°F limit is 10°F below the MODE 5 to MODE 4 transition temperature of 200°F. This change is more restrictive because it requires the unit to be maintained at a lower RCS temperature when the required DHR pump is not in operation.

M03 ITS SR 3.4.8.2 requires verification that correct breaker alignment and indicated power are available to each required pump. A Note further explains that the Surveillance is not required to be performed until 24 hours after a required pump is not in operation. This Surveillance is not required by the CTS. This changes the CTS by requiring verification of correct breaker alignment and indicated power availability on required DHR pumps that are not in operation.

DISCUSSION OF CHANGES
ITS 3.4.8, RCS LOOPS - MODE 5, LOOPS NOT FILLED

The purpose of ITS SR 3.4.8.2 is to ensure a standby pump is available to provide RCS cooling should the operating pump fail. This change is acceptable because the verification of proper breaker alignment and power availability ensures that an additional DHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. This change is designated as more restrictive because it requires performance of the Surveillance on the non-operating pump.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.4.1.2.a and 3.4.1.2.c contain a description of what constitutes an OPERABLE coolant loop. ITS 3.4.8 does not include this description of what constitutes an OPERABLE coolant loop. This changes the CTS by moving the details of what constitutes an OPERABLE coolant loop to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains a requirement for the RCS loops to be OPERABLE. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 3 - Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.4.1.2.3 states that the required coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours. ITS SR 3.4.8.1 states that the required DHR loop shall be verified to be in operation every 12 hours. This changes the CTS by moving the Surveillance Requirement to verify that the coolant loops are circulating reactor coolant to the Bases.

The removal of this detail for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be in the Technical Specifications in order to provide adequate protection of the public health and safety. The ITS retains the requirement that a DHR loop be in operation. This will require recirculation of reactor coolant since the ITS Bases specify that verification that a DHR loop is in operation includes flow rate, temperature, or pump status monitoring, which helps ensure that forced flow is providing heat removal. Also, this change is acceptable because these

DISCUSSION OF CHANGES
ITS 3.4.8, RCS LOOPS - MODE 5, LOOPS NOT FILLED

types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 1 – Relaxation of LCO Requirements)* CTS 3.4.1.2 places OPERABILITY requirements for the DHR loops to be OPERABLE and operating. ITS 3.4.8 specifies the same requirements; however, a new allowance is provided. ITS LCO 3.4.8 Note 2 allows one of the required DHR loops to be inoperable for up to 2 hours for Surveillance testing provided the other DHR loop is OPERABLE and in operation. This changes the CTS by adding this new allowance.

The purpose of CTS LCO 3.4.1.2 is to ensure there is sufficient forced circulation to provide forced flow for decay heat removal and transport. This change is acceptable because the LCO requirements continue to ensure that the structures, systems, and components are maintained consistent with the UFSAR analyses and licensing basis. This allowance provided by ITS 3.4.8 Note 2 still ensures a DHR loop is OPERABLE and in operation. Thus, decay heat removal and transport is still provided during this 2 hour time period. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

L02 *(Category 1 – Relaxation of Required Action)* CTS LCO 3.4.1.2 footnote **, in part, states that all decay heat removal (DHR) pumps may be de-energized for up to 1 hour provided no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration. CTS 3.4.1.2 Action b states that when no coolant loops are in operation, all operations involving a reduction in boron concentration of the RCS must be suspended. The ITS LCO 3.4.8 Note 1 allows all DHR pumps to be removed from operation for a certain period of time provided no operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)." ITS 3.4.8 Required Action B.1 states that operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1 must be suspended. This relaxes the CTS Action and LCO footnote by revising the action and footnote from suspending reductions in boron concentration to suspending introduction of coolant into the RCS with a boron concentration less than required to meet LCO 3.1.1.

The purpose of the CTS LCO 3.4.1.2 footnote ** and CTS 3.4.1.2 Action b is to ensure that "pockets" of coolant with boron concentration less than that required to maintain the SDM are not created when there is no forced flow through the reactor. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded

DISCUSSION OF CHANGES
ITS 3.4.8, RCS LOOPS - MODE 5, LOOPS NOT FILLED

conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition and the low probability of a DBA occurring during the repair period. As long as coolant with boron concentration less than that required to meet the SDM requirement in LCO 3.1.1 is not introduced into the RCS, there is no possibility of creating "pockets" of coolant with less than the required boron concentration. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

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

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops - MODE 5, Loops Not Filled

3.4.1.2 LCO 3.4.8 Two decay heat removal (DHR) loops shall be OPERABLE and one DHR loop shall be in operation.

-----NOTES-----

3.4.1.2 footnote ** 1. All DHR pumps may be removed from operation for ≤ 15 minutes when switching from one loop to another provided:

[a. The maximum RCS temperature is ≤ [160] 190 °F.] : 1 2

b. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM);" and 4 2

c. No draining operations to further reduce the RCS water volume are permitted.

DOC L01 2. One DHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other DHR loop is OPERABLE and in operation. 3

APPLICABILITY: MODE 5 with RCS loops not filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a — A. One required DHR loop inoperable.	A.1 Initiate action to restore DHR loop to OPERABLE status.	Immediately

CTS

ACTIONS (continued)

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a, Action b	B. No required DHR loop OPERABLE. OR Required DHR loop not in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
		AND	
		B.2 Initiate action to restore one DHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY
4.4.1.2.3	SR 3.4.8.1	Verify required DHR loop is in operation.	12 hours
DOC M03	SR 3.4.8.2	-----NOTE----- Not required to be performed until 24 hours after a required pump is not in operation. ----- Verify correct breaker alignment and indicated power available to each required DHR pump.	7 days

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.8, RCS LOOPS - MODE 5, LOOPS NOT FILLED

1. Removed brackets and provided plant specific limit.
2. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01, Section 5.1.3.
3. Typographical error corrected.
4. The title of the LCO has been provided since this is the first reference to the LCO.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops - MODE 5, Loops Not Filled

BASES

BACKGROUND

coolers

In MODE 5 with loops not filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the decay heat removal (DHR) ~~heat exchangers~~. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid. 1

RCS draining is initiated (hot legs not completely filled). Additionally, the RCS inventory is further reduced to a

Loops are not filled when the ~~reactor coolant~~ water level ~~is~~ within the horizontal portion of the hot leg , as might be the case for refueling or maintenance on the reactor coolant pumps or SGs. GL 88-17 (Ref. 1) expresses concerns for loss of decay heat removal for this operating condition. With water at this low level, the margin above the decay heat suction piping connection to the hot leg is small. The possibility of loss of level or inlet vortexing exists and if it were to occur, the operating DHR pump could become air bound and fail resulting in a loss of forced flow for heat removal. As a consequence the water in the core will heat up and could boil with the possibility of core uncovering due to boil off. Because the containment hatch may be open at this time, a pathway to the outside for fission product release exists if core damage were to occur. 1

In MODE 5 with loops not filled, only DHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one DHR pump for decay heat removal and transport, to require that two paths be available to provide redundancy for heat removal.

APPLICABLE SAFETY ANALYSES

No safety analyses are performed with initial conditions in MODE 5 with loops not filled. The flow provided by one DHR pump is adequate for heat removal and for boron mixing.

RCS Loops - MODE 5 (Loops Not Filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require that a minimum of two DHR loops be OPERABLE and that one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the DHR system unless forced flow is used. A minimum of one running decay heat removal pump meets the LCO requirement for one loop in operation. An additional DHR loop is required to be OPERABLE to provide redundancy for heat removal. 3

BASES

LCO (continued)

Note 1 permits the DHR pumps to be removed from operation for ≥ 15 minutes when switching from one train to the other. The circumstances for stopping both DHR pumps are to be limited to situations where the outage time is short [and temperature is maintained ≥ [160]°F]. The Note prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained or draining operations when DHR forced flow is stopped.

≤ loop ≤ 190 meet and that could reduce the RCS water volume 4 3 5 4

Note 2 allows one DHR loop to be inoperable for a period of 2 hours provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when these tests are safe and possible.

An OPERABLE DHR loop is composed of an OPERABLE DHR pump capable of providing forced flow to an OPERABLE DHR heat exchanger. DHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

INSERT 1 cooler INSERT 2 1 1

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the DHR System.

Operation in other MODES is covered by:

LCO 3.4.4,	"RCS Loops - MODES 1 and 2,"
LCO 3.4.5,	"RCS Loops - MODE 3,"
LCO 3.4.6,	"RCS Loops - MODE 4,"
LCO 3.4.7,	"RCS Loops - MODE 5, Loops Filled,"
LCO 3.9.4,	"Decay Heat Removal (DHR) and Coolant Circulation - High Water Level" (MODE 6), and
LCO 3.9.5,	"Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level" (MODE 6).

; . 2 4 4

ACTIONS

A.1

If one required DHR loop is inoperable, redundancy for heat removal is lost. Required Action A.1 is to immediately initiate activities to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

1 **INSERT 1**

Furthermore, the two DHR loops share the same suction path through DH-11 and DH-12. Therefore, when both DHR loops are being used to meet the LCO requirements, control power is required to be removed from DH-11 and DH-12 valve operators, or manual valves DH-21 and DH-23 are required to be open.

1 **INSERT 2**

Additionally, since the DHR System is a manually operated system (i.e., it is not automatically actuated), each DHR loop is OPERABLE if it can be manually aligned (remote or local) to the decay heat removal mode.

BASES

ACTIONS (continued)

B.1 and B.2

If no required loop is OPERABLE or the required loop is not in operation, except as provided by Note 1 in the LCO, the Required Action s require~~s~~ immediate suspension of all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 and ~~requires~~ immediate initiation of action to ~~immediately~~ restore one DHR loop to OPERABLE status and operation. ~~The Required Action for restoration does not apply to the condition of both loops not in operation when the exception Note in the LCO is in force.~~ Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operations for decay heat removal. The action to restore must continue until one loop is restored. 4 4 6

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This Surveillance requires verification every 12 hours that the required loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.

SR 3.4.8.2

Verification that each required pump is OPERABLE ensures that redundancy for heat removal is provided. The requirement also ensures that an additional loop can be placed in operation if needed to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

BASES

REFERENCES	1. Generic Letter 88-17, October 17, 1988.

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.8 BASES, RCS LOOPS - MODE 5, LOOPS NOT FILLED

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

2. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01, Section 5.1.3.

3. Typographical error corrected.

4. Changes made to be consistent with the Specification.

5. Changes made to be consistent with changes made to the Specification.

6. This description is not necessary. When using the Note allowance, ACTION B is not required to be entered (as described in the first sentence of ACTIONS B.1 and B.2 Bases). In addition, the deleted wording implies that only Required Action B.2 does not apply, when in actuality, neither of the Required Actions apply.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.8, RCS LOOPS - MODE 5, LOOPS NOT FILLED

There are no specific NSHC discussions for this Specification.

Attachment 1, Volume 9, Rev. 0, Page 180 of 415

ATTACHMENT 9

ITS 3.4.9, PRESSURIZER

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

A01

ITS 3.4.9

ITS

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

LCO 3.4.9 — 3.4.4 The pressurizer shall be OPERABLE with: A02

~~a. A steam bubble,~~ L01

LCO 3.4.9.a — b. A water level ~~between 45 and~~ 228 inches.

Add proposed LCO 3.4.9.b M01

APPLICABILITY: MODES 1 and 2.

Add proposed MODE 3 Applicability M02

ACTION:

ACTION A — With the pressurizer inoperable, restore the pressurizer to OPERABLE status within 1 hour or

ACTION B — be in at least HOT STANDBY ~~with the control rod drive trip breakers open~~ within the next 6 hours. A03

Be in MODE 4 in 12 hours M02

Add proposed ACTIONS C and D M01

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1 — 4.4.4 The pressurizer shall be demonstrated OPERABLE by verifying pressurizer level to be within limits at least once per 12 hours.

Add proposed SR 3.4.9.2 M01

DISCUSSION OF CHANGES
ITS 3.4.9, PRESSURIZER

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.4.4.a states that the pressurizer shall be OPERABLE with a steam bubble. ITS 3.4.9 does not retain this requirement. This changes the CTS by not specifically requiring the pressurizer to be OPERABLE with a steam bubble.

This change is acceptable because when the unit is in MODE 1, 2, or 3 and the pressurizer water level is maintained at less than 228 inches, a steam bubble will exist. Since the ITS still requires the pressurizer water level to be less than 228 inches, a steam bubble will be present and there is no need to specifically require the steam bubble. The change is designated as administrative because it does not result in a technical change to the CTS.

A03 CTS 3.4.4 Action states that if the inoperable pressurizer is not restored to OPERABLE status within the allowed time, to be in HOT STANDBY (MODE 3) with the control rod drive trip breakers open within the next 6 hours. Under similar conditions, ITS 3.4.9 ACTION B states to be in MODE 3 within 6 hours and in MODE 4 within 12 hours. This changes the CTS by eliminating the requirement to open the control rod drive trip breakers. The change associated with entering MODE 4 is discussed in DOC M02.

This change is acceptable because it results in no technical change to the Technical Specifications. Although CTS 3.4.4 Action appears to require the control rod drive trip breakers to be opened within 6 hours (if the pressurizer is not restored to OPERABLE status within the allowed restoration time), they are not actually required to be opened. The Applicability of CTS 3.4.4 is MODES 1 and 2. CTS 3.0.1 states that "Limiting Conditions for Operation and ACTION requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for each specification." Therefore, the CTS 3.4.4 Action to open the control rod drive trip breakers ceases to be applicable once the unit enters MODE 3, and the Action is exited. As a result, deleting this action results in no operational difference from the CTS Action. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.4.4 does not contain requirements for the pressurizer heaters. ITS LCO 3.4.9.b has been added requiring the pressurizer to be OPERABLE with a minimum of 112 kW of essential pressurizer heaters OPERABLE. ITS 3.4.9 ACTIONS C and D have been added to provide compensatory measures when the new requirement is not met. ITS 3.4.9 ACTION C, which applies when the

DISCUSSION OF CHANGES
ITS 3.4.9, PRESSURIZER

capacity of pressurizer heaters is less than 150 kW, requires restoration of the essential pressurizer heater capability within 72 hours. If the heater capability is not restored within 72 hours, ITS 3.4.9 ACTION D requires the unit to be in MODE 4 within 12 hours. In addition, SR 3.4.9.2 has been added, and requires verification that the essential pressurizer heater capacity is greater than or equal to 150 kW every 24 months.

This change is acceptable because the pressurizer heaters are used to maintain the steam and water at the saturation temperature corresponding to the desired RCS pressure. The addition of the LCO, ACTIONS and Surveillance Requirement will assure that this capability is available. This change is designated as more restrictive because additional LCO requirement and associated ACTIONS and a Surveillance Requirement have been added.

M02 CTS 3.4.4 only requires the pressurizer to be OPERABLE in MODES 1 and 2. If the pressurizer is inoperable, the CTS Actions allows 1 hour to restore the pressurizer to OPERABLE status or the unit must be in HOT STANDBY (MODE 3) with the control rod drive trip breakers open within the next 6 hours. ITS 3.4.9 requires the pressurizer to be OPERABLE in MODES 1, 2, and 3. If the pressurizer is not restored to OPERABLE status under the same conditions as the CTS (water level not within limit) within 1 hour, the unit must be in MODE 3 in 6 hours and in MODE 4 in 12 hours. This changes the CTS by expanding the Applicability of the Pressurizer to include MODE 3 and requiring the unit to exit this new Applicability within 12 hours. The deletion of the Action to open the control rod drive trip breakers is discussed in DOC A03.

The purpose of the ITS MODE 3 Applicability is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation. This change is acceptable because it provides appropriate requirements in MODE 3 to achieve this purpose. This change is designated as more restrictive because it requires the pressurizer to be OPERABLE under more conditions (MODE 3) than is currently required.

<u>RELOCATED SPECIFICATIONS</u>

None

<u>REMOVED DETAIL CHANGES</u>

None

<u>LESS RESTRICTIVE CHANGES</u>

L01 *(Category 1 – Relaxation of LCO Requirements)* CTS 3.4.4.b states that the pressurizer shall be OPERABLE with a water level between 45 and 228 inches. ITS LCO 3.4.9.a states that the pressurizer shall be OPERABLE with a pressurizer water level $\leq$ 228 inches. This changes the CTS by eliminating the lower water level limit of 45 inches.

DISCUSSION OF CHANGES
ITS 3.4.9, PRESSURIZER

The purpose of the CTS 3.4.4.b lower limit is to preserve the steam space during normal operation, allowing both sprays and heaters to maintain the design operating pressure. The lower level limit prevents the low level interlock from de-energizing the pressurizer heaters during steady state operations. This change is acceptable because the low water level limit is not necessary for accident mitigation. The pressurizer water level is routinely monitored by operations personnel to ensure a low level in the pressurizer does not occur, similar to other plant parameters not specified in the Technical Specifications. Therefore, the low level limit is not necessary to be included in the Technical Specifications. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than are being applied in the CTS.

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

CTS

Pressurizer
3.4.9

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.4 — LCO 3.4.9 The pressurizer shall be OPERABLE with:

LCO 3.4.4.b — a. Pressurizer water level ≤ ~~[290]~~ 228 inches ~~and~~ ; ① ④

DOC M01 — b. A minimum of ~~[126]~~ 150 kW of essential pressurizer heaters OPERABLE ~~[and capable of being powered from an emergency power supply]~~. ②

~~-----NOTE-----~~
~~OPERABILITY requirements on pressurizer heaters do not apply in MODE 4.~~
~~----------~~ ③

APPLICABILITY: MODES 1, 2, and 3~~[,~~
~~MODE 4 with RCS temperature ≥ [275]°F]~~. ③

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
Action	A. Pressurizer water level not within limit.	A.1 Restore level to within limit.	1 hour	
Action	B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 ~~[with RCS temperature ≤ [275]°F]~~.	6 hours ~~[24]~~ 12 hours	③
DOC M01	C. Capacity of essential pressurizer heaters ~~[capable of being powered by emergency power supply]~~ less than limit.	C.1 Restore essential pressurizer heater ~~capability~~ capacity.	72 hours	②

Attachment 1, Volume 9, Rev. 0, Page 188 of 415

CTS

ACTIONS (continued)

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC M01	D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY	
4.4.4	SR 3.4.9.1	Verify pressurizer water level ≤ ~~[290]~~ 228 inches.	12 hours	1
DOC M01	SR 3.4.9.2	[Verify ~~≥ [126] kW of pressurizer heaters are capable of being powered from an emergency power supply.~~ capacity of essential pressurizer heaters is ≥ 150 kW.	~~[18]~~ 24 months]	2 1
	~~SR 3.4.9.3~~	~~[Verify emergency power supply for pressurizer heaters is OPERABLE.~~	~~[18] months]~~	3

Attachment 1, Volume 9, Rev. 0, Page 189 of 415

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.9, PRESSURIZER

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

2. The bracketed requirement that the pressurizer heaters be capable of being powered from an emergency power supply has been deleted. The essential heaters, which are the heaters used to meet the LCO requirement, are always powered from the emergency power supply (i.e., they are powered from the essential buses). This is consistent with the ISTS SR 3.4.9.3 Bases, which states that the SR is not applicable if the heaters are permanently powered by 1E power supplies.

3. ISTS 3.4.9 includes the Applicability of MODE 4 with RCS temperature $\geq$ 275°F. The ISTS Bases states that the reason for the MODES 3 and 4 Applicability is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbations. However, the temperature cross-over point between MODES 3 and 4 for Davis-Besse is 280°F. In the ISTS, the temperature cross-point is 330°F. Thus, the Davis-Besse MODE 3 Applicability requirement is essentially equivalent to the ISTS 3.4.9 Applicability of MODE 4 with RCS temperature $\geq$ 275°F (only a 5°F difference exists). Therefore, ITS 3.4.9 does not include the MODE 4 Applicability; only the MODES 1, 2, and 3 Applicability is maintained. Due to this change, the NOTE to the LCO has been deleted and the associated Required Action (Required Action B.2) and Completion Time has been modified to be consistent with the normal time provided in the ISTS to be in MODE 4.

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

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

and
The pressure control components addressed by this LCO include the pressurizer water level, the required heaters, and their controls and emergency power supplies. Pressurizer safety valves and pressurizer power operated relief valves (PORVs) are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Pressurizer Power Operated Relief Valve (PORV)," respectively. 3 2 3
pilot
the
Pilot

The maximum water level limit has been established to ensure that a liquid to vapor interface exists to permit RCS pressure control during normal operation and proper pressure response for anticipated design basis transients. The water level limit thus serves two purposes:

a. Pressure control during normal operation maintains subcooled reactor coolant in the loops and thus is in the preferred state for heat transport, and 5

;

b. By restricting the level to a maximum, expected transient reactor coolant volume increases (pressurizer insurge) will not cause excessive level changes that could result in degraded ability for pressure control.

The maximum water level limit permits pressure control equipment to function as designed. The limit preserves the steam space during normal operation, thus both sprays and heaters can operate to maintain the design operating pressure. The level limit also prevents filling the pressurizer (water solid) for anticipated design basis transients, thus ensuring that pressure relief devices (PORVs or code safety valves) can control pressure by steam relief rather than water relief. If the level limits were exceeded prior to a transient that creates a large pressurizer insurge volume leading to water relief, the maximum RCS pressure might exceed the design Safety Limit (SL) of 2750 psig or damage may occur to the PORVs or pressurizer code safety valves.

There are two essential heater banks, with each bank powered from a separate essential bus and each bank having a capacity of 126 kW.

2

BASES

BACKGROUND (continued)

essential

2

The pressurizer heaters are used to maintain a pressure in the RCS so reactor coolant in the loops is subcooled and thus in the preferred state for heat transport to the steam generators (SGs). This function must be maintained with a loss of offsite power. Consequently, the emphasis of this LCO is to ensure that the essential power supplies and the associated heaters are adequate to maintain pressure for RCS loop subcooling with an extended loss of offsite power.

150

A minimum required available capacity of ~~[126]~~ kW ensures that the RCS pressure can be maintained. Unless adequate heater capacity is available, reactor coolant subcooling cannot be maintained indefinitely. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to loss of single phase natural circulation and decreased capability to remove core decay heat. 1

APPLICABLE SAFETY ANALYSES

In MODES 1 and 2, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. No safety analyses are performed in lower MODES. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present. 2

U

Safety analyses presented in the FSAR do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure. 2

The maximum level limit is of prime interest for the loss of main feedwater (LOMFW) event. Conservative safety analyses assumptions for this event indicate that it produces the largest increase of pressurizer level caused by a moderate frequency event. Thus this event has been selected to establish the pressurizer water level limit. Assuming proper response action by emergency systems, the level limit prevents water relief through the pressurizer safety valves. Since prevention of water relief is a goal for abnormal transient operation, rather than an SL, the value for pressurizer level is nominal and is not adjusted for instrument error.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Evaluations performed for the design basis large break loss of coolant accident (LOCA), which assumed a [higher] maximum level than assumed for the LOMFW event, have been made. The higher pressurizer level assumed for the [LOCA] is the basis for the volume of reactor coolant released to the containment. The containment analysis performed using the mass and energy release demonstrated that the maximum resulting containment pressure was within design limits. lower 2 LOMFW 2

The requirement for emergency power supplies is based on NUREG-0737 (Ref. 1). The intent is to allow maintaining the reactor coolant in a subcooled condition with natural circulation at hot, high pressure conditions for an undefined, but extended, time period after a loss of offsite power. While loss of offsite power is an initial condition or coincident event assumed in many accident analyses, maintaining hot, high pressure conditions over an extended time period is not evaluated as part of FSAR accident analyses. 2

U

The maximum pressurizer water level limit satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 1), is the reason for providing an LCO.

LCO

228

The LCO requirement for the pressurizer to be OPERABLE with a water level ≤ [290] inches ensures that a steam bubble exists. Limiting the maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions. 1

150 essential

The LCO requires a minimum of [126] kW of pressurizer heaters OPERABLE [and capable of being powered from an emergency power supply]. As such, the LCO addresses both the heaters and the power supplies. The minimum heater capacity required is sufficient to maintain the system near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. The exact design value of [126] kW is derived from the use of nine heaters rated at 14 kW each. The amount needed to maintain pressure is dependent on the insulation losses, which can vary due to tightness of fit and condition. 3 2

. Since each essential bank has a capability of 126 kW, both essential banks are used to meet the LCO requirement.

per bank 1 2

Attachment 1, Volume 9, Rev. 0, Page 194 of 415

BASES

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus Applicability has been designated for MODES 1 and 2. The Applicability is also provided for MODE 3 ~~and, for pressurizer water level, for MODE 4 with RCS temperature ≥ [275]°F~~. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbations, such as reactor coolant pump startup. ~~The temperature of [275]°F has been designated as the cutoff for applicability because LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," provides a requirement for pressurizer level below [275]°F. The LCO does not apply to MODE 5 with loops filled because LCO 3.4.12 applies. The LCO does not apply to MODES 5 and 6 with partial loop operation.~~ 3

In MODES 1, 2, and 3, there is the need to maintain the availability of pressurizer heaters capable of being powered from an emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. ~~The Applicability is modified by a Note stating that the OPERABILITY requirements on pressurizer heaters do not apply in MODE 4.~~ For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Decay Heat Removal System is in service, and therefore the LCO is not applicable. 3

ACTIONS

A.1

With pressurizer water level in excess of the maximum limit, action must be taken to restore pressurizer operation to within the bounds assumed in the analysis. This is done by restoring the pressurizer water level to within the limit. The 1 hour Completion Time is considered to be a reasonable time for draining excess liquid.

B.1 and B.2

If the water level cannot be restored, reducing core power constrains heat input effects that drive pressurizer insurge that could result from an anticipated transient. ~~By shutting down the reactor and reducing reactor coolant temperature to at least MODE 3, the potential thermal energy of the reactor coolant mass for LOCA mass and energy releases is reduced.~~

INSERT 1 3

3 **INSERT 1**

Therefore, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within 12 hours. The 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 plant systems. Similarly, the Completion Time of 12 hours to reach MODE 4 is reasonable based on operating experience to achieve power reduction from full power conditions in an orderly manner and without challenging plant systems.

Attachment 1, Volume 9, Rev. 0, Page 196 of 415

BASES

ACTIONS (continued)

~~Six hours is a reasonable time based upon operating experience to reach MODE 3 from full power without challenging plant systems and operators. Further pressure and temperature reduction to MODE 4 with RCS temperature ≤ [275]°F places the plant into a MODE where the LCO is not applicable. The [24] hour Completion Time to reach the nonapplicable MODE is reasonable based upon operating experience.~~ 3

C.1

If ~~the [emergency] power supplies to the heaters are not capable of providing [126] kW, or~~ the essential pressurizer heaters are inoperable, restoration is required in 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand caused by loss of offsite power will not occur in this period. Pressure control may be maintained during this time using normal station powered heaters. 3

D.1 and D.2

If essential pressurizer heater capability cannot be restored within the allowed Completion Time of Required Action C.1, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within ~~the following 6~~ 12 hours. The 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 plant systems. Similarly, the Completion Time of 12 hours to reach MODE 4 is reasonable based on operating experience to achieve power reduction from full power conditions in an orderly manner and without challenging plant systems. 3 4

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer water level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess the level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level indications.

BASES

SURVEILLANCE REQUIREMENTS (continued)

[SR 3.4.9.2 1

essential

The SR requires the power supplies are capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. (This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance.) The Frequency of [[18] months] is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.]

3

24 3

1

~~[SR 3.4.9.3~~

~~This SR is not applicable if the heaters are permanently powered by 1E power supplies.~~ 3

~~This Surveillance demonstrates that the heaters can be manually transferred to, and energized by, emergency power supplies. The Frequency of [18] months is based on a typical fuel cycle and is consistent with similar verifications of emergency power.]~~

REFERENCES 1. NUREG-0737, November 1980.

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.9 BASES, PRESSURIZER

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

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

3. Changes are made to reflect changes made to the Specification.

4. Changes made to be consistent with the Specification.

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

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.9, PRESSURIZER

There are no specific NSHC discussions for this Specification.

ATTACHMENT 10

ITS 3.4.10, PRESSURIZER SAFETY VALVES

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

A01

ITS 3.4.10

ITS

REACTOR COOLANT SYSTEM

SAFETY VALVES AND PILOT OPERATED RELIEF VALVE - OPERATING

LIMITING CONDITION FOR OPERATION

LCO 3.4.10 3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of ≤ 2525 psig.* When not isolated, the pressurizer pilot operated relief valve shall have a trip setpoint of ≥ 2435 psig and an allowable value of ≥2435 psig.** — See ITS 3.4.11

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

ACTION A — With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN

ACTION B — within 12 hours.

Add proposed Required Action B.1 — M01

Add proposed ACTION B for two pressurizer safety valves inoperable. — M02

SURVEILLANCE REQUIREMENTS

Add proposed reset limit — M03

SR 3.4.10.1 4.4.3 For the pressurizer code safety valves, there are no additional Surveillance Requirements other than those required by Specification 4.0.5. For the pressurizer pilot operated relief valve a CHANNEL CALIBRATION check shall be performed each REFUELING INTERVAL. — See ITS 3.4.11

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. — LA01

** Allowable value for CHANNEL CALIBRATION check. — See ITS 3.4.11

DISCUSSION OF CHANGES
ITS 3.4.10, PRESSURIZER SAFETY VALVES

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.4.3 Action requires, in part, that with one pressurizer code safety valve inoperable, to either restore it within 15 minutes or be in HOT SHUTDOWN (MODE 4) within 12 hours. ITS 3.4.10 ACTION A requires that with one pressurizer safety valve inoperable, to restore the valve to OPERABLE status within 15 minutes. If not restored, ITS 3.4.10 ACTION B requires the unit to be in MODE 3 within 6 hours and MODE 4 within 12 hours. This changes the CTS by requiring entry into MODE 3 within 6 hours when a shutdown is required.

This change is acceptable because the requirement to place the unit in MODE 3 ensures an intermediate shutdown condition is reached in a shorter period of time. The 6 hour Completion Time is based on operating experience and the need to reach the required condition from full power in an orderly manner and without challenging unit systems. This change is designated as more restrictive because it imposes a time requirement on when the unit must be in MODE 3.

M02 CTS 3.4.3 Action does not provide any actions for when two pressurizer safety valves are inoperable. Therefore, CTS 3.0.3 would be entered requiring entry into HOT STANDBY (MODE 3) within 7 hours and HOT SHUTDOWN (MODE 4) within 13 hours. ITS 3.4.10 ACTION B, which applies when two pressurizer safety valves are inoperable, requires a shutdown to MODE 3 within 6 hours and to MODE 4 within 12 hours. This changes the CTS by providing one less hour to shut down the unit to both MODE 3 and MODE 4 following discovery of two inoperable pressurizer safety valves.

The purpose of requiring a shutdown when both pressurizer safety valves are inoperable is due to the plant is not meeting the overpressure protection analysis assumptions. This change is acceptable because it provides an adequate period of time to be in a MODE in which the requirement does not apply, commensurate with the severity of the inoperability. The Completion Times of 6 hours and 12 hours are reasonable, based on operating experience, for reaching MODES 3 and 4, respectively, from full power in an orderly manner and without challenging unit systems. This change has been designated as more restrictive because it reduces the Completion Times to be in MODES 3 and 4.

M03 CTS 4.4.3 requires a verification that the pressurizer safety valve lift setting is within the limit of CTS 3.4.3 (i.e., < 2525 psig). ITS SR 3.4.10.1 includes a similar requirement, but also requires that following testing, the lift setting must

DISCUSSION OF CHANGES
ITS 3.4.10, PRESSURIZER SAFETY VALVES

be within ± 1% of the nominal setting (2500 psig). This changes the CTS by requiring a minimum pressurizer safety valve setpoint after testing of ≥ 2475 psig.

The purpose of CTS 4.4.3 is to ensure the pressurizer safety valves are set within the accident analysis setpoint. This change is acceptable because the valves must be set in accordance with the Inservice Test Program requirements. The pressurizer safety valves are ASME Code Section III relief valves, thus they must be set to ± 1% of the nominal setpoint following testing. This change is designated as more restrictive since a new requirement is specified in the ITS that is not included in the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS LCO 3.4.3 is modified by a note (footnote *) that states that the pressurizer safety valves lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. This information is not provided in ITS 3.4.10. This changes the CTS by moving this information to the Bases.

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 3.4.10 still retains a requirement for the valves to be OPERABLE. Under the definition of OPERABILITY, the pressurizer safety valves must be capable of lifting at the assumed conditions, which includes the ambient operating conditions of the pressurizer safety valves themselves. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being moved from the Technical Specifications to the ITS Bases.

LESS RESTRICTIVE CHANGES

None

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

3.4.3 LCO 3.4.10 Two pressurizer safety valves shall be OPERABLE with lift settings ≥ [2475] psig and ≤ [2525] psig. 1

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > [283]°F. 2

--NOTE--
The lift settings are not required to be within the LCO limits for entry into MODES 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. 3

ACTIONS

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
Action	A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
Action DOC M02	B. Required Action and associated Completion Time of Condition A not met. OR Two pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 with any RCS cold leg temperature ≤ [283]°F.	6 hours [24] hours 12

4 2

SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY
4.4.3	SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program

Attachment 1, Volume 9, Rev. 0, Page 209 of 415

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.10, PRESSURIZER SAFETY VALVES

1. ISTS LCO 3.4.10 requires both a minimum and maximum lift setting value for the pressurizer safety valves. Davis-Besse is only including the maximum lift setting in ITS LCO 3.4.10, consistent with current licensing basis. The overpressure protection analysis assumes a maximum lift setting for the pressurizer safety valves; a minimum lift setting is not assumed. However, the minimum lift setting is being included in the Davis-Besse ITS as part of ITS SR 3.4.10.1, the pressurizer safety valve lift setting Surveillance. ITS SR 3.4.10.1 requires the as-left lift setting to be ± 1%, which is consistent with the ASME Code requirements. Thus, the pressurizer safety valves will be considered OPERABLE provided their lift settings are ≤ 2525 psig, but when tested the as-left lift settings will be ≥ 2475 psig and ≤ 2525 psig.

2. ISTS 3.4.10 Applicability of MODE 4 with all RCS cold leg temperatures > 283°F is not included in the Davis-Besse ITS. This is consistent with the current licensing basis. The temperature cross-over point between MODES 3 and 4 for Davis-Besse is 280°F. In the ISTS, the temperature cross-point is 330°F. Thus, the Davis-Besse MODE 3 Applicability requirement is actually more restrictive than the ISTS 3.4.10 Applicability of MODE 4 with RCS temperature ≥ 283°F. Therefore, ITS 3.4.10 does not include the MODE 4 Applicability; only the MODES 1, 2, and 3 Applicability is maintained. In addition, due to this change, ISTS 3.4.10 Required Action and associated Completion Time have been changed to only require being in MODE 4 within 12 hours. The 12 hour Completion Time is consistent with the time to be in MODE 4 in other actions (e.g., ITS LCO 3.0.3).

3. As described in the Applicability Section of the ISTS Bases, this Note is included to allow testing of the pressurizer safety valves at high pressure and temperature near their normal operating range. The Davis-Besse pressurizer safety valves discharge directly to the containment atmosphere. In-situ testing is not performed at Davis-Besse; the pressurizer safety valves lift settings are verified at a vendor test facility. Thus, the Note allowance is not needed and has been deleted.

4. This change has been made consistent with the Writer's Guide for the Improved Technical Specifications TSTF-GG-05-01, Section 4.1.6.i.5.ii.

Attachment 1, Volume 9, Rev. 0, Page 210 of 415

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The purpose of the two spring loaded pressurizer safety valves is to provide RCS overpressure protection. Operating in conjunction with the Reactor Protection System (RPS), two valves are used to ensure that the Safety Limit (SL) of 2750 psig is not exceeded for analyzed transients during operation in MODES 1 and 2. Two safety valves are used for portions of MODE 3 ~~and portions of MODE 4~~. For the remainder of MODE ~~4~~ 3, MODE S 4 and MODE 5, and MODE 6 with the reactor head on, overpressure protection is provided by operating procedures and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) ~~System~~." (1) (1)

The self actuated pressurizer safety valves are designed in accordance with the requirements set forth in the ASME Boiler and Pressure Vessel Code, Section III (Ref. 1). The required lift pressure is 2500 psig ~~±~~ + 1%. The safety valves discharge steam from the pressurizer ~~to a quench tank located in the containment. The discharge flow is indicated by an increase in temperature downstream of the safety valves and by an increase in the quench tank temperature and level.~~ into a separate tee opening directly into containment. Flow through the pressurizer safety valves generates acoustic levels or vibration that is detected by piezoelectric sensors on the discharge pipe. These sensors provide valve position indication (open/closed) in the control room. (1) (1)

The ~~upper and lower~~ pressure limit~~s~~ ~~are~~ is based on the ~~±~~ + 1% tolerance requirement for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3 (nominal operating temperature and pressure). This requires either that the valves be set hot or that a correlation between hot and cold settings be established. (1) (3)

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the ASME pressure limit could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

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 (3)

BASES

APPLICABLE SAFETY ANALYSES (continued)

insurges that could occur during a startup, rod withdrawal, ejected rod, 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~~ from a subcritical condition. 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. 3

Pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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~~ limit~~s~~ ~~are~~ is 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. 3 1 3

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.

APPLICABILITY

In MODES 1, 2, and 3, ~~and portions of MODE 4 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. MODE 3 ~~and portions of MODE 4 are~~ is conservatively included, although the listed accidents may not require both safety valves for protection. 1 1

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. 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~~ 1

BASES

LCO (continued)

~~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.~~ 1

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 ~~[24]~~ 12 hours allowed is reasonable, based on operating experience, to reach MODE 4 without challenging plant systems. ~~With any RCS cold leg temperature at or below [283]°F, overpressure protection is provided by LTOP. The change from MODE 1, 2, or 3 to MODE 4 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.~~ 1 1 1

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 the ASME OM Code (Ref. ~~1~~ 2), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified. 3

The pressurizer safety valve setpoint is ~~± [3]~~ +1% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance ~~to allow for drift~~ in accordance with Reference 1. 1 2 3

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III.

~~1~~ 2. ASME Code for Operation and Maintenance of Nuclear Power Plants, 1995 Edition with 1996 Addenda. 3 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.10 BASES, PRESSURIZER SAFETY VALVES

1. Changes are made to be consistent with changes to the Specification.

2. The brackets have been removed and the proper plant specific information/value has been provided.

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

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.10, PRESSURIZER SAFETY VALVES

There are no specific NSHC discussions for this Specification.

ATTACHMENT 11

ITS 3.4.11, PRESSURIZER PILOT OPERATED RELIEF VALVE (PORV)

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

A01

ITS 3.4.11

ITS

REACTOR COOLANT SYSTEM

SAFETY VALVES AND PILOT OPERATED RELIEF VALVE - OPERATING

LIMITING CONDITION FOR OPERATION

LCO 3.4.11

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of < 2525 psig.* When not isolated, the pressurizer pilot operated relief valve shall have a trip setpoint of ≥ 2435 psig and an allowable value of ≥ 2435 psig.**

See ITS 3.4.10

LA01

LA02

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

See ITS 3.4.10

Add proposed ACTIONS A, B, and C

M01

SURVEILLANCE REQUIREMENTS

4.4.3 For the pressurizer code safety valves, there are no additional Surveillance Requirements other than those required by Specification 4.0.5. For the pressurizer pilot operated relief valve a CHANNEL CALIBRATION check shall be performed each REFUELING INTERVAL.

See ITS 3.4.10

LA02

Add proposed SR 3.4.11.1 and SR 3.4.11.2

M02

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

See ITS 3.4.10

** Allowable value for CHANNEL CALIBRATION check.

LA02

DISCUSSION OF CHANGES
ITS 3.4.11, PRESSURIZER PILOT OPERATED RELIEF VALVE (PORV)

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.4.3 does not provide any actions for when the pressurizer pilot operated relief valve (PORV) or block valve are inoperable and not isolated. Therefore, CTS 3.0.3 would be entered, requiring entry into HOT STANDBY (MODE 3) within 7 hours and HOT SHUTDOWN (MODE 4) within 13 hours. With the PORV inoperable, ITS 3.4.11 ACTION A requires the block valve to be closed within 1 hour and power removed from the block valve within 1 hour. With the block valve inoperable, ITS 3.4.11 ACTION B requires the block valve to be closed within 1 hour and power removed from the block valve within 1 hour. If either of these actions are not met, ITS 3.4.11 ACTION C requires a shutdown to MODE 3 within 6 hours and to MODE 4 within 12 hours. This changes the CTS by stating the ACTIONS rather than deferring to CTS 3.0.3 and by adding the requirement to remove power from the block valve.

The purpose of CTS 3.0.3 is to place the unit outside the MODE of Applicability within a reasonable amount of time in a controlled manner. CTS 3.4.3 is silent on these actions, deferring to CTS 3.0.3 for the actions to accomplish this. This portion of the change is acceptable because the ACTIONS specified in ITS 3.4.11 adopt ISTS structure for placing the unit outside the MODE of Applicability without changing the time specified to enter MODE 3 and MODE 4. Furthermore, power must be removed from the block valve to reduce the potential of inadvertent depressurization that would occur if the PORV failed open. This is acceptable because it ensures an inadvertent depressurization cannot occur due to a failed open PORV. This change is designated as more restrictive because an additional requirement is included in the ITS that is not in the CTS.

M02 CTS 4.4.3 does not specify Surveillance Requirements to cycle the pressurizer pilot operated relief valve (PORV) and the block valve. ITS SR 3.4.11.1 requires performance of one complete cycle of the block valve every 92 days. This Surveillance Requirement is modified by a Note stating that the Surveillance is not required to be performed with the block valve closed in accordance with the Required Action of the LCO. ITS SR 3.4.11.2 requires cycling of the PORV every 24 months. This changes the CTS by adding specific requirements to cycle the block valve and the PORV.

The purpose of ITS SR 3.4.11.1 and SR 3.4.11.2 is to ensure the PORV and associated block valve are operating correctly so the potential for a small break

DISCUSSION OF CHANGES
ITS 3.4.11, PRESSURIZER PILOT OPERATED RELIEF VALVE (PORV)

LOCA through the PORV pathway is minimized, or if a small break LOCA were to occur through a failed open PORV, the block valve could be manually operated to isolate the path. In addition, ITS SR 3.4.11.2 ensures the PORV can be opened as necessary if needed during a steam generator tube rupture (SGTR) event. This change is acceptable because it provides specific requirements for testing of the block valve and the PORV. This change is designated as more restrictive because it adds Surveillance Requirements for the block valve and the PORV to the ITS that are not in the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.4.3 provides the trip setpoint for pilot operated relief valve (PORV). ITS 3.4.11 does not retain this detail. This changes the CTS by moving the details of the trip setpoint to the Bases.

The removal of this detail, which is related to system design, from the Technical Specification is acceptable because this type of information is not necessary to be in the Technical Specifications to provide adequate protection of public health and safety. The PORV is not assumed to open automatically in any safety analysis. It is utilized to depressurize the RCS for mitigation of a SGTR event when offsite power is unavailable. However, UFSAR analysis for the SGTR assumes that offsite power is available. The ITS still retains a requirement for the PORV to be OPERABLE. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being moved from the Technical Specifications to the ITS Bases.

LA02 *(Type 4 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAPM, IST Program, or IIP)* CTS 3.4.3 provides the Allowable Value for PORV opening and footnote ** states that this Allowable Value is for the CHANNEL CALIBRATION. CTS 4.4.3 requires a CHANNEL CALIBRATION of the pressurizer pilot operated relief valve (PORV) each REFUELING INTERVAL. ITS 3.4.11 does not retain these requirements. This changes the CTS by moving the CHANNEL CALIBRATION and associated Allowable Value to the Technical Requirements Manual (TRM).

The removal of these details from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The PORV is not assumed to open automatically in any safety analysis. It is utilized

DISCUSSION OF CHANGES
ITS 3.4.11, PRESSURIZER PILOT OPERATED RELIEF VALVE (PORV)

to depressurize the RCS for mitigation of a SGTR event when offsite power is unavailable. However, UFSAR analysis for the SGTR assumes that offsite power is available. ITS 3.4.11 now requires that the PORV and the block valve be cycled through at least one complete cycle instead of the CHANNEL CALIBRATION. (See DOC M02 for the addition of the ITS SR 3.4.11.1 and ITS SR 3.4.11.2). Also, this change is acceptable because the removed information will be adequately controlled in the TRM. The TRM is currently incorporated by reference into the UFSAR, thus any changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because a Surveillance Requirement, including its acceptance criteria, is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

Attachment 1, Volume 9, Rev. 0, Page 223 of 415

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

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

Pilot

3.4.11 Pressurizer ~~Power~~ Operated Relief Valve (PORV) 1

3.4.3 LCO 3.4.11 The PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC M01	A. PORV inoperable.	A.1 Close block valve. AND A.2 Remove power from block valve.	1 hour 1 hour
DOC M01	B. Block valve inoperable.	B.1 Close block valve. AND B.2 Remove power from block valve.	1 hour 1 hour
DOC M01	C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

CTS

SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY	
DOC M02	SR 3.4.11.1	---NOTE--- Not required to be performed with block valve closed in accordance with the Required Actions of this LCO. --- Perform one complete cycle of the block valve.	92 days	
DOC M02	SR 3.4.11.2	Perform one complete cycle of the PORV.	~~18~~ 24 months	2
	SR 3.4.11.3	~~[Verify PORV and block valve are capable of being powered from an emergency power source.~~	~~18 months]~~	3

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.11, PRESSURIZER PILOT OPERATED RELIEF VALVE (PORV)

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

2. ISTS SR 3.4.11.2 requires performance of one complete cycle of the PORV every 18 months. The ISTS SR 3.4.11.2 Bases states the 18 month frequency is based on a typical refueling cycle. Therefore, the Frequency has been changed to align it with the Davis-Besse refueling cycle, which is 24 months.

3. The bracketed Surveillance Requirement that the PORV and associated block valve are verified to be capable of being powered from an emergency power source has been deleted. The PORV and associated block valve are always powered from the emergency power supply (i.e., they are powered from the essential buses). This is consistent with the ISTS SR 3.4.11.3 Bases, which states that the SR is not applicable if the valves are permanently powered by 1E power supplies.

Attachment 1, Volume 9, Rev. 0, Page 227 of 415

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

Pilot

B 3.4.11 Pressurizer ~~Power~~ Operated Relief Valve (PORV) 1

BASES

BACKGROUND

The pressurizer is equipped with three devices for pressure relief functions: two American Society of Mechanical Engineers (ASME) pressurizer safety valves that are safety grade components and one PORV that is not a safety grade device. The PORV is an electromatic pilot operated valve that is automatically opened at a specific set pressure when the pressurizer pressure increases and is automatically closed on decreasing pressure. The PORV may also be manually operated using controls installed in the control room.

An electric motor operated, normally open, block valve is installed between the pressurizer and the PORV. The function of the block valve is to isolate the PORV. Block valve closure is accomplished manually using controls in the control room and may be used to isolate a leaking PORV to permit continued power operation. Most importantly, the block valve is to be used to isolate a stuck open PORV to isolate the resulting small break loss of coolant accident (LOCA). Closure terminates the RCS depressurization and coolant inventory loss.

the essential buses, which are powered from either the offsite

power

sources

or the

The PORV, its block valve, and their controls are powered from ~~normal~~ power ~~supplies~~ ~~but are also capable of being powered from~~ emergency ~~supplies~~. Power supplies for the PORV are separate from those for the block valve. Power supply requirements are defined in NUREG-0737, Paragraph ~~III~~, G.1 (Ref. 1). 5 6 2

II

(≥ 2435 psig)

The PORV setpoint is above the high pressure reactor trip setpoint and below the opening setpoint for the pressurizer safety valve as required by IE Bulletin 79-05B (Ref. 2). The purpose of the relationship of these setpoints is to limit the number of transient pressure increase challenges that might open the PORV, which, if opened, could fail in the open position. A pressure increase transient would cause a reactor trip, reducing core energy, and for many expected transients, prevent the pressure increase from reaching the PORV setpoint. The PORV setpoint thus limits the frequency of challenges from transients and limits the possibility of a small break LOCA from a failed open PORV. 2

INSERT 1

Placing the setpoint below the pressurizer safety valve opening setpoint reduces the frequency of challenges to the safety valves, which, unlike the PORV, cannot be isolated if they were to fail open. The PORV setpoint is therefore important for limiting the possibility of a small break LOCA.

2 **INSERT 1**

The PORV is also set such that it will open before the pressurizer safety valves are opened. However, it should not open on any anticipated transients. Reference 3 identified that the turbine trip from full power would cause the largest overpressure transient. The Reference 3 analysis demonstrated that with an RPS RC High Pressure trip setpoint of 2355 psig, the resulting overshoot in RCS pressure would be limited to 50 psi. Consequently, the minimum PORV setpoint needs to accommodate both the RCS pressure overshoot and the RPS instrument string error of 30 psi.

Attachment 1, Volume 9, Rev. 0, Page 230 of 415

BASES

BACKGROUND (continued)

The primary purpose of this LCO is to ensure that the PORV and the block valve are operating correctly so the potential for a small break LOCA through the PORV pathway is minimized, or if a small break LOCA were to occur through a failed open PORV, the block valve could be manually operated to isolate the path.

The PORV may be manually operated to depressurize the RCS as deemed necessary by the operator in response to normal or abnormal transients. The PORV may be used for depressurization when the pressurizer spray is not available; a condition that would be encountered during loss of offsite power. Steam generator tube rupture (SGTR) is one event that may require use of the PORV if the sprays are unavailable.

The PORV may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

The PORV functions as an automatic overpressure device and limits challenges to the safety valves. Although the PORV acts as an overpressure device for operational purposes, safety analyses [] do not take credit for PORV actuation, but [] do take credit for the safety valves. 3

The PORV also provides low temperature overpressure protection (LTOP) during heatup and cooldown. LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," addresses this function. 2

APPLICABLE SAFETY ANALYSES

The PORV small break LOCA break size is bounded by the spectrum of piping breaks analyzed for plant licensing. Because the PORV small break LOCA is located at the top of the pressurizer, the RCS response characteristics are different from RCS loop piping breaks; analyses have been performed to investigate these characteristics.

The possibility of a small break LOCA through the PORV is reduced when the PORV flow path is OPERABLE and the PORV opening setpoint is established to be reasonably remote from expected transient challenges. The possibility is minimized if the flow path is isolated.

The PORV opening setpoint has been established in accordance with Reference 2. It has been set so expected RCS pressure increases from anticipated transients will not challenge the PORV, minimizing the possibility of a small break LOCA through the PORV.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Overpressure protection is provided by safety valves, and analyses do not take credit for the PORV opening for accident mitigation.

Operational analyses that support the emergency operating procedures utilize the PORV to depressurize the RCS for mitigation of SGTR when the pressurizer spray system is unavailable (loss of offsite power). U FSAR safety analyses for SGTR have been performed assuming that offsite power is available and thus pressurizer sprays (or the PORV) are available. 2

The PORV and its block valve satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires the PORV and its associated block valve to be OPERABLE. The block valve is required to be OPERABLE so it may be used to isolate the flow path if the PORV is not OPERABLE. If the block valve is not OPERABLE, the PORV may be used for temporary isolation.

APPLICABILITY

In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. A likely cause for PORV LOCA is a result of pressure increase transients that cause the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. Pressure increase transients can occur any time the steam generators are used for heat removal. The most rapid increases will occur at higher operating power and pressure conditions of MODES 1 and 2.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the applicability is pertinent to MODES 1, 2, and 3. The LCO is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. ~~The PORV setpoint is reduced for LTOP in MODES 4, 5, and 6 with the reactor vessel head in place. LCO 3.4.12 addresses the PORV requirements in these MODES.~~ 1

ACTIONS

A.1 and A.2

With the PORV inoperable, the PORV must be restored or the flow path isolated within 1 hour. The block valve ~~should~~ must be closed and power must be removed from the block valve to reduce the potential ~~for~~ of inadvertent ~~PORV opening and depressurization~~ depressurization that would occur if the PORV failed open. 4 5

BASES

ACTIONS (continued)

B.1 and B.2

If the block valve is inoperable, it must be restored to OPERABLE status within 1 hour. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to close the block valve and remove power within 1 hour rendering the PORV isolated. The 1 hour Completion Times are consistent with an allowance of some time for correcting minor problems, restoring the valve to operation, and establishing correct valve positions and restricting the time without adequate protection against RCS depressurization.

C.1 and C.2

If the Required Action and associated Completion Time cannot be met, 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 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 reasonable, based on operating experience, to reach MODE 4 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that it can be closed if needed. The basis for the Frequency of 92 days is the ASME Code (Ref. ~~3~~ 4). Block valve cycling, as stated in the Note, is not required to be performed when it is closed for isolation; cycling could increase the hazard of an existing degraded flow path. 2

SR 3.4.11.2

PORV cycling demonstrates its function. Any combination of indications (e.g., acoustic, system response) may be used to confirm a complete cycle of the PORV. The Frequency of ~~18~~ 24 months is based on a typical refueling cycle and industry accepted practice. 2 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.11.3

This Surveillance is not required for plants with permanent 1E power supplies to the valves.

This SR demonstrates that emergency power can be provided and is performed by transferring power from the normal supply to the emergency supply and cycling the valves. The Frequency of 18 months is based on a typical refueling cycle and industry accepted practice. 1

REFERENCES

1. NUREG-0737, Paragraph ~~III~~ II, G.1, November 1980. 6

2. NRC IE Bulletin 79-05B, April 21, 1979.

3. BAW-1890, September 1985. 2

4 ~~3~~. ASME Code for Operation and Maintenance of Nuclear Power Plants. 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.11 BASES, PRESSURIZER PILOT OPERATED RELIEF VALVE (PORV)

1. Changes are made to be consistent with changes to the Specification.
2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.
3. The brackets have been removed and the proper plant specific information/value has been provided.
4. Changes made to be consistent with the Specification.
5. Editorial change for clarity.
6. Typographical error corrected.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.11, PRESSURIZER PILOT OPERATED RELIEF VALVE (PORV)

There are no specific NSHC discussions for this Specification.

ATTACHMENT 12

ITS 3.4.12, LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP)

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

A01

ITS

REACTOR COOLANT SYSTEM

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

LCO 3.4.12 — 3.4.2 Decay Heat Removal System relief valve DH-4849 shall be OPERABLE with a lift setting of ≤ 330 psig* and isolation valves DH-11 and DH-12 open and control power to their valve operators removed. LA01 LA01

APPLICABILITY: MODES 4 and 5. — Add proposed MODE 6 Applicability — M01

ACTION:

A. With DH-4849 not OPERABLE: LA01

ACTION C — 1. Make the valve OPERABLE within eight hours; or

ACTION D — 2. a. Within next one hour, disable the capability of both high pressure injection (HPI) pumps to inject water into the reactor coolant system; and

b. Within next eight hours:

1. Disable the automatic transfer of makeup pump suction to the borated water storage tank on low makeup tank level; and

2. Reduce makeup tank level to ≤ 73 inches and reduce reactor coolant system pressure and pressurizer level within the acceptable region on Figures 3.4-2a (in MODE 4) and 3.4-2b (in MODE 5).

ACTION A — B. With DH-11 or DH-12 closed, open DH-21 and DH-23 within one hour. LA01

ACTION B — C. With the control power not removed from DH-11 and DH-12, remove the power to the valve operators at the Motor Control Centers within one hour.

SURVEILLANCE REQUIREMENTS

4.4.2 Decay Heat Removal System relief valve DH-4849 shall be determined OPERABLE: LA01

SR 3.4.12.2 — a. per the surveillance requirements of Specification 4.0.5.

b. at least once per 24 hours by verifying either:

SR 3.4.12.1 — 1. isolation valves DH-11 and DH-12 open with control power removed from their valve operators; or LA01

Required Action B.2 — 2. valves DH-21 and DH-23 open.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. LA01

A01

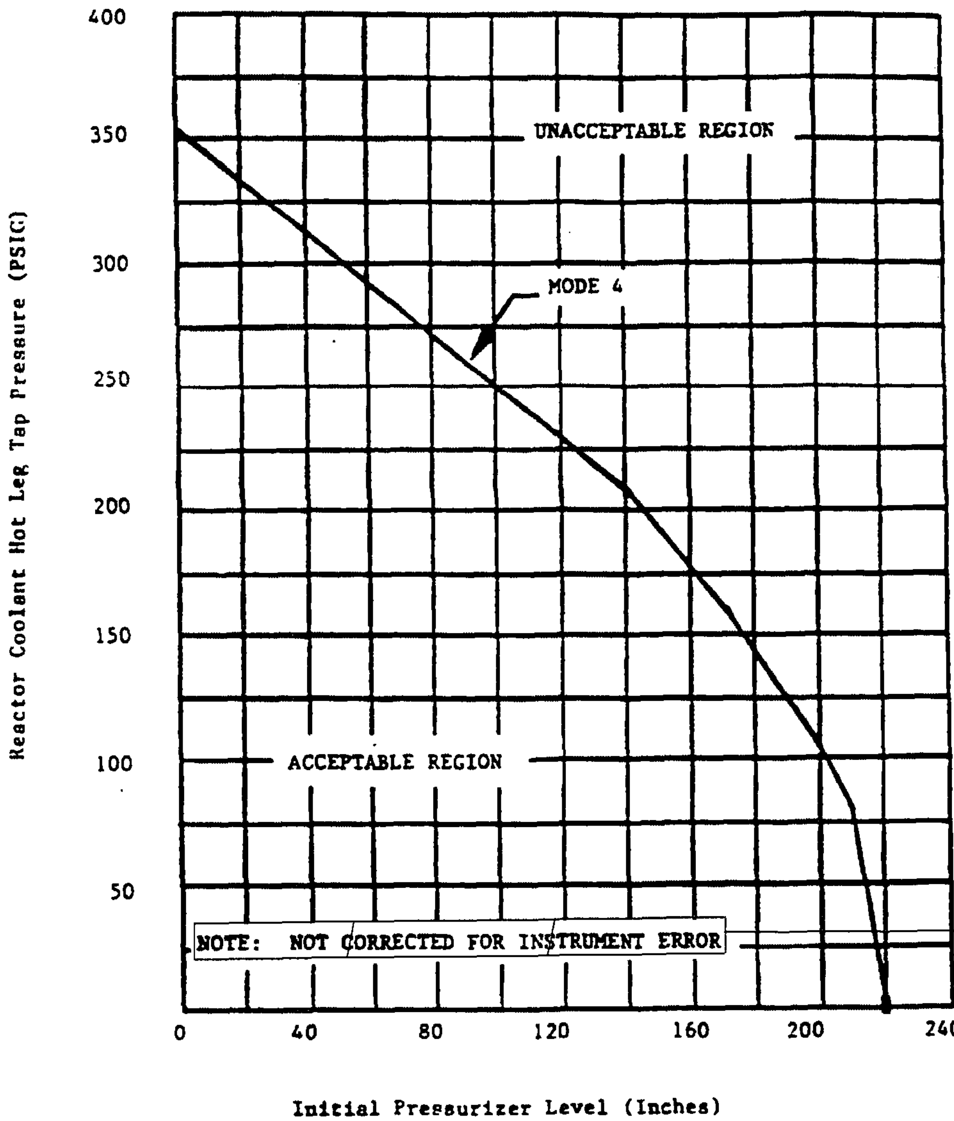
ITS 3.4.12

ITS

Figure 3.4.12-1

Figure 3.4-2a

Reactor Coolant System Pressure - Pressurizer Level Limits for inoperable Decay Heat Removal System Relief Valve in MODE 4



LA01

A01

ITS 3.4.12

ITS

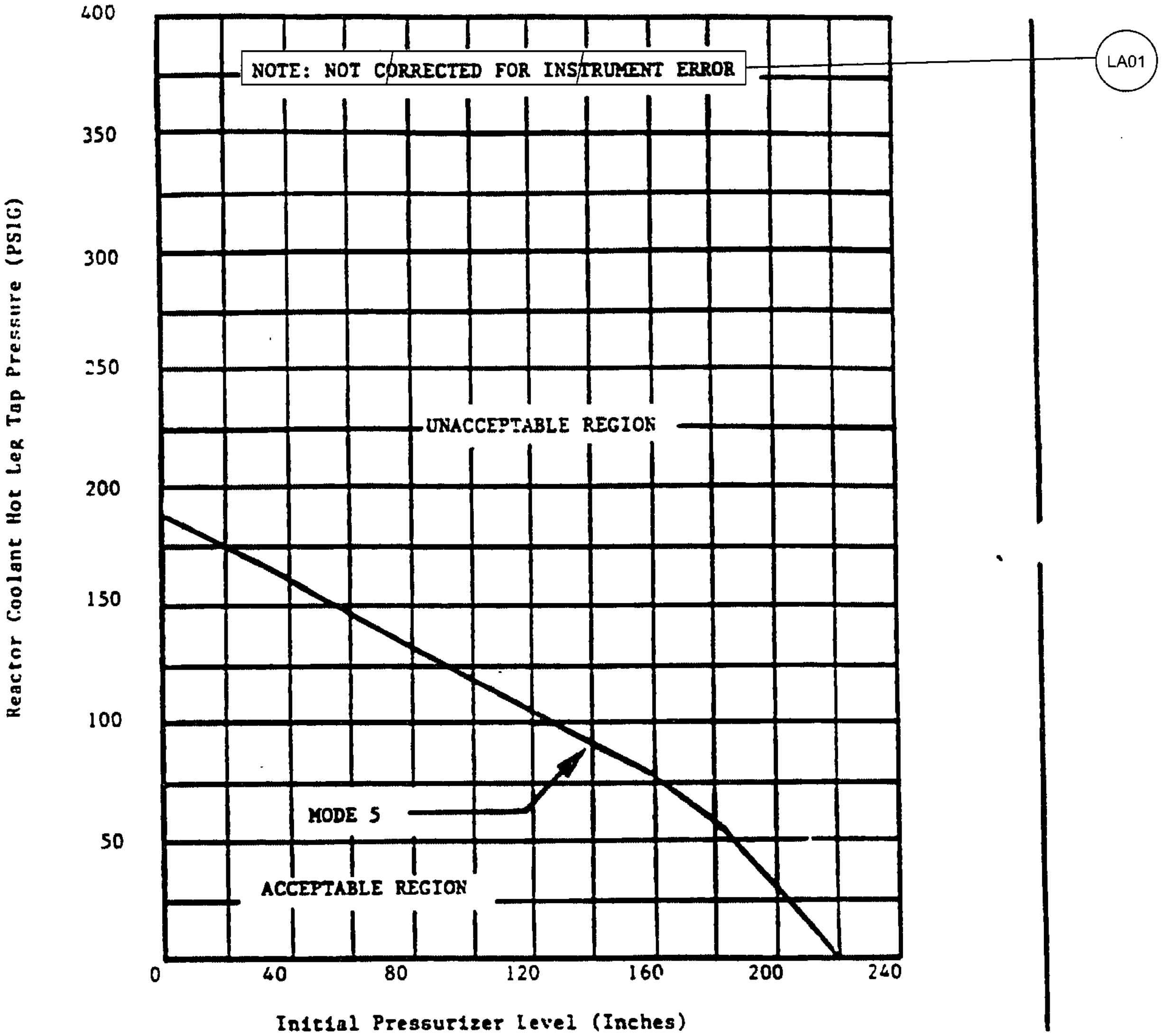
Figure 3.4.12-2

Figure 3.4-2b

Reactor Coolant System Pressure - Pressurizer Level Limits for inoperable Decay Heat Removal System Relief Value in MODE 5

Add proposed MODE 6

M01



DISCUSSION OF CHANGES
ITS 3.4.12, LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP)

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.4.2 is applicable in MODES 4 and 5. CTS Figure 3.4-2b is applicable in MODE 5. ITS LCO 3.4.12 is applicable in MODES 4 and 5, and MODE 6 when the reactor vessel head is on. In addition, Figure 3.4.12-2 is applicable in MODE 5 and MODE 6 when the reactor vessel head is on. This change expands the Applicability of the low temperature overpressure protection components to be OPERABLE in MODE 6 when the reactor vessel head is on.

The purpose of CTS 3.4.2 is to ensure that there is a sufficient low temperature protection during shutdown conditions. The definition of MODE 6 in ITS Table 1.1-1 clearly states that MODE 6 is when one or more reactor vessel head closure bolts are less than fully tensioned. Therefore, this change will require the MODE 6 Applicability when one or more reactor vessel head closure bolts are less than fully tensioned, until the vessel head is removed. This change is necessary because an overpressure event could occur in this situation and a relief path is still necessary until the head is physically removed. This change is designated as more restrictive because it adds additional requirements to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.4.2 is modified by a note (footnote *) that states that the decay heat removal relief valve lift setting pressure shall correspond to normal operating temperature and pressure. CTS LCO 3.4.2, Actions A, B, and C, and Surveillance Requirement 4.4.2 provides specific valve numbers for certain Decay Heat Removal System valves. CTS 3.4.2 Action c requires power to the valve operators be removed at the motor control centers. CTS Figures 3.4-2a and 3.4-2b (used when a Decay Heat Removal System relief valve is inoperable) include a Note that states the Figures are not corrected for instrument error. ITS 3.4.12 does not include these details. Furthermore, ITS 3.4.12 uses the plant specific names for the associated valves, and requires control power to be

DISCUSSION OF CHANGES
ITS 3.4.12, LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP)

removed from the RCS to DHR system isolation valves. This changes the CTS by moving the valve numbers, the information concerning the lift settings, the details concerning how to remove power from the valves, and that the Figures are not corrected for instrument error to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specification to provide adequate protection of public health and safety. ITS 3.4.12 still retains a requirement for the valves to be OPERABLE, uses the plant specific names for the valves, requires control power to be removed from the valves, and the Figures to be used when a Decay Heat Removal System relief valve is inoperable. Under the definition of OPERABILITY, the Decay Heat Removal System relief valve must be capable of lifting at the assumed conditions, which includes ambient operating conditions of the Decay Heat Removal System relief valve itself. Also this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being moved from the Technical Specifications to the ITS Bases.

LESS RESTRICTIVE CHANGES

None

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

1

CTS

LTOP System
3.4.12

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

3.4.2 LCO 3.4.12

INSERT 1

~~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) isolated and:~~

-----NOTES-----

1. ~~[Two makeup pumps] may be capable of injecting for ≤ 1 hour for pump swap operations.~~

2. ~~CFT may be unisolated 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 the PTLR.~~

---

a. ~~Pressurizer level ≤ [220] inches and an OPERABLE power operated relief valve (PORV) with a lift setpoint of ≤ [555] psig or~~

b. ~~The RCS depressurized and an RCS vent of ≥ [0.75] square inch.~~

S

and 5,

APPLICABILITY: MODE 4 ~~when any RCS cold leg temperature is ≤ [283]°F,~~
~~MODE 5,~~
MODE 6 when the reactor vessel head is on.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ~~More than [one] makeup pump capable of injecting into the RCS.~~	A.1	~~Initiate action to verify only [one] makeup pump is capable of injecting into the RCS.~~	Immediately
B. ~~HPI activated.~~	B.1	~~Initiate action to verify HPI deactivated.~~	Immediately

INSERT 2

Attachment 1, Volume 9, Rev. 0, Page 246 of 415

CTS

(1) INSERT 1

3.4.2 The Decay Heat Removal (DHR) System relief valve shall be OPERABLE with:

a. A lift setting of ≤ 330 psig; and

b. The Reactor Coolant System (RCS) to DHR System isolation valves open with control power removed.

(1) INSERT 2

CTS	Condition	Required Action	Completion Time
Action B	A. DHR System relief valve inoperable due to one or more RCS to DHR System isolation valves closed.	A.1 Open RCS to DHR System isolation bypass valves. AND A.2 Verify RCS to DHR System isolation bypass valves open.	1 hour Once per 24 hours
Action C	B. DHR System relief valve inoperable due to one or more RCS to DHR System isolation valves with control power not removed.	B.1 Remove control power from RCS to DHR System isolation valves.	1 hour
Action A.1	C. DHR System relief valve inoperable for reasons other than Condition A or B.	C.1 Restore DHR System relief valve to OPERABLE status.	8 hours

CTS

1 **INSERT 2** (continued)

Action A.2	D. Required Action and Associated Completion Time not met.	D.1 Disable capability of both high pressure injection pumps to inject water into the RCS.	1 hour
		AND	
		D.2 Disable makeup pump suction automatic transfer to the borated water storage tank on low makeup tank level.	8 hours
		AND	
		D.3 Verify makeup tank level $\leq$ 73 inches.	8 hours
		AND	
		D.4 Verify RCS pressure and pressurizer level in Acceptable Region of Figure 3.4.12-1 or 3.4.12-2, as applicable.	8 hours

CTS

1

LTOP System
3.4.12

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. A CFT not isolated when CFT pressure is greater than or equal to the maximum RCS pressure for existing temperature allowed in the PTLR.	C.1	Isolate affected CFT.	1 hour
D. Required Action C.1 not met within the required Completion Time.	D.1	Increase RCS temperature to > 175°F.	12 hours
	OR		
	D.2	Depressurize affected CFT to < [555] psig.	12 hours
E. Pressurizer level > [220] inches.	E.1	Restore pressurizer level to ≤ [220] inches.	1 hour
F. Required Action E.1 not met within the required Completion Time.	F.1	Close and maintain closed the makeup control valve and its associated isolation valve.	12 hours
	AND		
	F.2	Stop RCS heatup.	12 hours
G. PORV inoperable.	G.1	Restore PORV to OPERABLE status.	1 hour
H. Required Action G.1 not met within the required Completion Time.	H.1	Reduce makeup tank level to ≤ [70] inches.	12 hours
	AND		
	H.2	Deactivate low low makeup tank level interlock to the borated water storage tank suction valves.	12 hours

CTS

1

LTOP System
3.4.12

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. Pressurizer level > [220] inches. AND PORV inoperable. OR LTOP System inoperable for any reason other than Condition A through Condition H.	I.1 Depressurize RCS and establish RCS vent of ≥ [0.75] square inch.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
	SR 3.4.12.1	Verify a maximum of [one] makeup pump is capable of injecting into the RCS.	12 hours
	SR 3.4.12.2	Verify HPI is deactivated.	12 hours
	SR 3.4.12.3	Verify each CFT is isolated.	12 hours
	SR 3.4.12.4	Verify pressurizer level is ≤ [220] inches.	30 minutes during RCS heatup and cooldown AND 12 hours
4.4.2.b	SR 3.4.12.5 1	Verify PORV block valve is open. RCS to DHR isolation valves open with control power removed.	12 hours 24

CTS

1

LTOP System
3.4.12

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
	SR 3.4.12.6 ~~Verify required RCS vent ≥ [0.75] square inch is open.~~	12 hours for unlocked open vent valve(s) AND 31 days for other vent path(s)
4.4.2.a	SR 3.4.12.7 (2) ~~Perform CHANNEL FUNCTIONAL TEST for PORV.~~ Verify DHR System relief valve lift setpoint ≤ 330 psig in accordance with the Inservice Testing (IST) Program.	~~Within [12] hours after decreasing RCS temperature to ≤ [283]°F AND 31 days thereafter~~ In accordance with the IST Program
	~~SR 3.4.12.8 Perform CHANNEL CALIBRATION for PORV.~~	~~[18] months~~

INSERT 3

INSERT 4

3.4.12

CTS

1 **INSERT 3**

Figure 3.4-2a

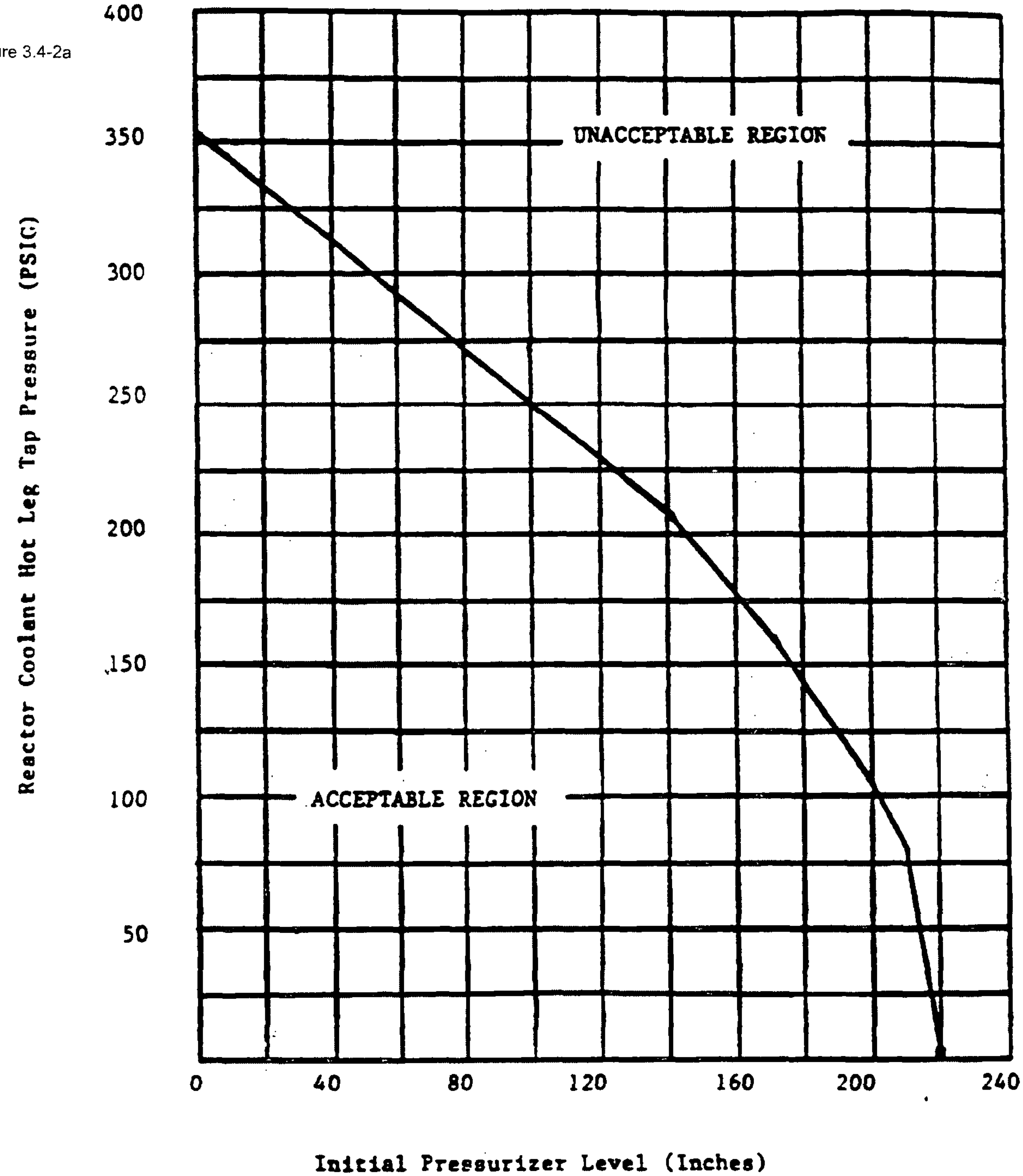


Figure 3.4.12-1
RCS Pressure Versus Pressurizer Level Limit
for Inoperable DHR System Relief Valve in MODE 4

Attachment 1, Volume 9, Rev. 0, Page 252 of 415

CTS

1

INSERT 4

Figure 3.4-2b

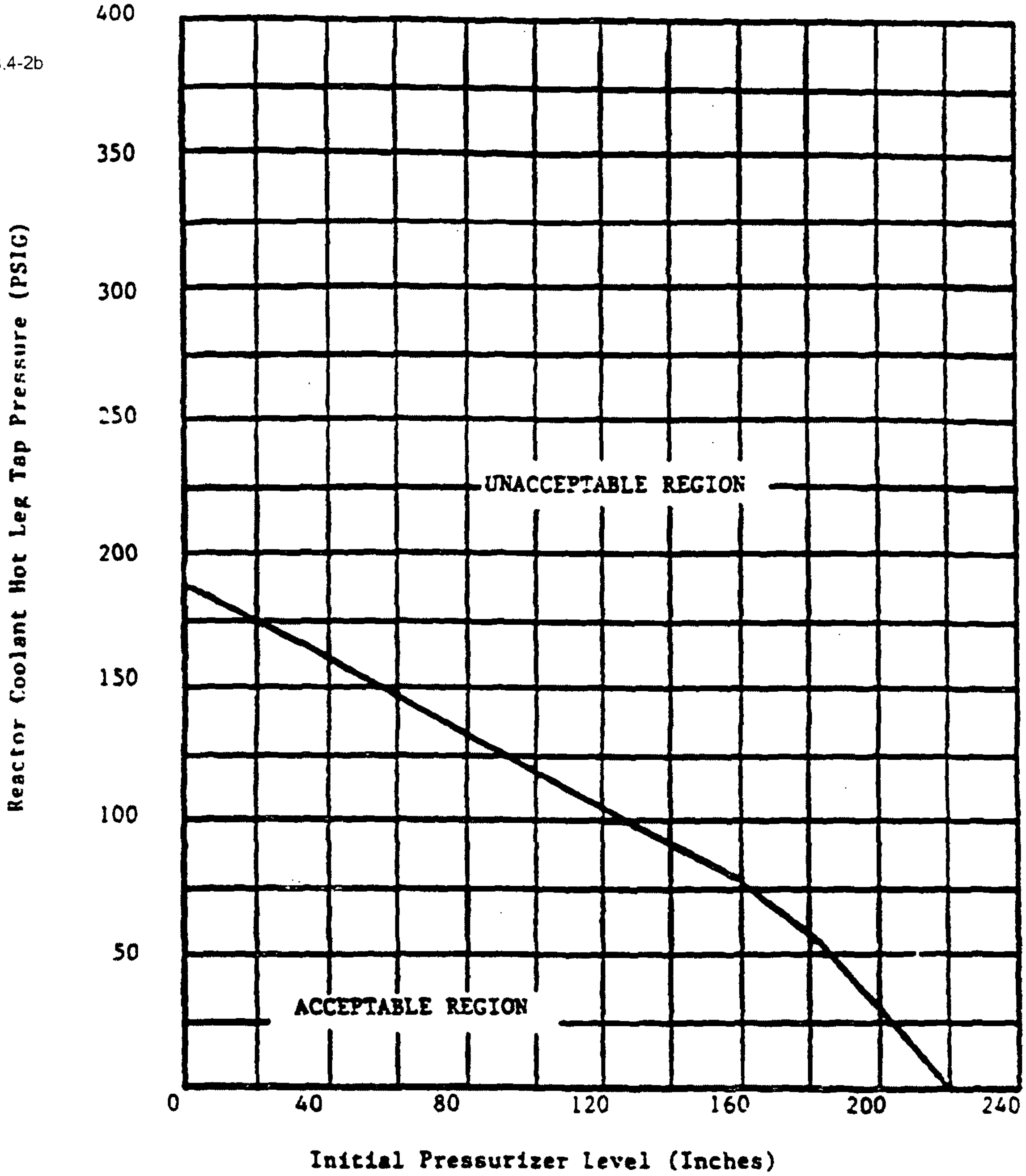


Figure 3.4.12-2
RCS Pressure Versus Pressurizer Level Limit
for Inoperable DHR System Relief Valve in MODE 5 and
MODE 6 when the reactor vessel head is on

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.12, LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP)

1. ISTS 3.4.12 has been changed to be consistent with the Davis-Besse current licensing basis and analysis basis. The Davis-Besse low temperature overpressure protection analysis between 280°F (MODE 4 entry temperature) and 140°F only requires the Decay Heat Removal (DHR) System relief valve to be OPERABLE with a setpoint of ≤ 330 psig to protect the RCS from an overpressure condition. This relief valve performs the same function as the PORV in the ISTS. Between 280°F and 140°F, the analysis does not require the high pressure injection (HPI) pumps to be incapable of injecting, the core flooding tanks to be isolated, or the pressurizer level to be within a certain limit. The CTS Actions only require the HPI pumps to be disabled and the pressurizer level to be within a certain limit if the DHR System relief valve is inoperable. To ensure the relief valve remains connected to the RCS, the CTS requires the Reactor Coolant System (RCS) to DHR isolation valves to be open with control power removed. If these requirements are not met, the CTS provides specific Actions to take. These requirements have been maintained in the ITS. In addition, the current Surveillances to ensure the LCO is met have also been provided. Finally, Davis-Besse has added the ISTS MODE 6 Applicability to be as consistent with the ISTS as possible, while still maintaining the specific analysis assumption requirements.

Attachment 1, Volume 9, Rev. 0, Page 254 of 415

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

All changes are 1
unless otherwise noted

Low Temperature Overpressure Protection (LTOP) System 2
B 3.4.12

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System 2

BASES

BACKGROUND

-----------------------------------REVIEWER'S NOTE----------------------------------- 3
~~For plants for which the NRC has approved LTOP setpoints based on non-10 CFR 50, Appendix G, methodology, as allowed in NRC Generic Letter 88-11, the following Bases must be revised accordingly.~~

~~The~~ LTOP ~~System~~ controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) requirements of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for providing such protection. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," provides the allowable combinations for operational pressure and temperature during cooldown, shutdown, and heatup to keep from violating the Reference 1 limits.

The reactor vessel material is less tough at reduced temperatures than at normal operating temperature. Also, as vessel neutron irradiation accumulates, the material becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure must be maintained low when temperature is low and must be increased only as temperature is increased.

Operational maneuvering during cooldown, heatup, or any anticipated operational occurrence must be controlled to not violate LCO 3.4.3. Exceeding these limits could lead to brittle fracture of the reactor vessel. LCO 3.4.3 presents requirements for administrative control of RCS pressure and temperature to prevent exceeding the P/T limits.

This LCO provides RCS overpressure protection in the applicable MODES by ensuring an adequate pressure relief capacity ~~and a minimum coolant addition capability. The pressure relief capacity requires either the power operated relief valve (PORV) lift setpoint to be reduced and pressurizer coolant level at or below a maximum limit or the RCS depressurized and with an RCS vent of sufficient size to handle the limiting transient during LTOP.~~

through the Decay Heat Removal (DHR) System relief valve.

~~The LTOP approach to protecting the vessel by limiting coolant addition capability allows a maximum of [one] makeup pump, and requires deactivating HPI, and isolating the core flood tanks (CFTs).~~

All changes are 1 unless otherwise noted

Low Temperature Overpressure Protection (LTOP) System
B 3.4.12 2

BASES

BACKGROUND (continued)

Should more than [one] HPI pump inject on an HPI actuation, the pressurizer level and PORV or another RCS vent cannot prevent overpressurizing the RCS. Even with only one HPI pump OPERABLE, the vent cannot prevent RCS overpressurization.

The pressurizer level limit provides a compressible vapor space or cushion (either steam or nitrogen) that can accommodate a coolant insurge and prevent a rapid pressure increase, allowing the operator time to stop the increase. The PORV, with reduced lift setting, or the RCS vent is the overpressure protection device that acts as backup to the operator in terminating an increasing pressure event.

With HPI deactivated, the ability to provide RCS coolant addition is restricted. To balance the possible need for coolant addition, the LCO does not require the Makeup System to be deactivated. Due to the lower pressures associated with the LTOP MODES and the expected decay heat levels, the Makeup System can provide flow with the OPERABLE makeup pump through the makeup control valve.

PORV Requirements

INSERT 1

As designed for the LTOP System, each PORV is signaled to open if the RCS pressure approaches a limit set in the LTOP actuation circuit. The LTOP actuation circuit monitors RCS pressure and determines when an overpressure condition is approached. When the monitored pressure meets or exceeds the setting, the PORV is signaled to open. Maintaining the setpoint within the limits of the LCO ensures the Reference 1 limits will be met in any event analyzed for LTOP.

When a PORV is opened in an increasing pressure transient, the release of coolant causes the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at ambient containment pressure in an RCS overpressure transient, if the relieving requirements of the maximum credible LTOP transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow of the limiting LTOP transient and maintaining pressure below P/T limits. The required vent capacity may be provided by one or more vent paths.

1 **INSERT 1**

The DHR System relief valve provides overpressure protection for the RCS during low temperature operations. RCS and DHR Systems are monitored for temperature and pressure. Maintaining the relief setpoint within the limits of the LCO ensures the Reference 1 limits will be met in any event in the LTOP analysis.

If system pressure exceeds the lift setpoint of the DHR System relief valve, it will open. As the relief valve opens, coolant is released and pressure decreases. When the relief valve reset is reached, below the LTOP pressure limit, the relief valve closes.

All changes are 1 unless otherwise noted

Low Temperature Overpressure Protection (LTOP) System
B 3.4.12 2

BASES

BACKGROUND (continued)

For the remaining portions of MODE 3, overpressure protection is provided by operating procedures.

~~For an RCS vent to meet the flow capacity, it requires removing a pressurizer safety valve, locking the PORV in the open position and disabling its block valve in the open position, or similarly establishing a vent by opening an RCS vent valve. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.~~

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 3) demonstrate that the reactor vessel can be adequately protected against overpressurization transients during shutdown. In MODES 1, and 2, and portions of MODE 3 ~~and in MODE 4 with RCS temperature exceeding [283]°F~~, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At nominally ~~[283]~~ 280°F ~~and below~~ to 140°F, overpressure prevention falls to ~~an~~ the OPERABLE ~~PORV and a restricted coolant level in the pressurizer or to a depressurized RCS and a sufficient size RCS vent~~ DHR System relief valve. Below 140°F, credible overpressurization sources are secured. ~~Each of these means has a limited overpressure relief capability.~~ 4

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as vessel material toughness decreases due to neutron embrittlement. Each time the P/T limit curves are revised, the LTOP ~~System~~ will be re-evaluated to ensure that its functional requirements can still be met with the ~~PORV and pressurizer level method or the depressurized and vented RCS condition~~ DHR System relief valve and operating procedures.

Transients that are capable of overpressurizing the RCS have been identified and evaluated. These transients relate to either mass input or heat input: actuating the ~~HPI~~ High Pressure Injection (HPI) System, discharging the ~~CFTs~~ Core Flooding Tanks (CFTs), energizing the pressurizer heaters, failing the makeup control valve open, losing decay heat removal, and starting a reactor coolant pump (RCP) ~~with a large temperature mismatch between the primary and secondary coolant systems, and adding nitrogen to the pressurizer~~.

~~HPI actuation and CFT discharge are the transients that result in exceeding P/T limits within < 10 minutes, in which time no operator action is assumed to take place. In the rest, operator action after that time precludes overpressurization. The analyses demonstrate that the time allowed for operator action is adequate, or the events are self limiting and do not exceed P/T limits.~~

~~The following are required during the LTOP MODES to ensure that transients do not occur, which either of the LTOP overpressure protection means cannot handle:~~

INSERT 2

Attachment 1, Volume 9, Rev. 0, Page 259 of 415

1 **INSERT 2**

The DHR System relief valve (DH-4849), which is in the suction line to the decay heat pumps, has been sized to pass 1800 gpm at the nominal set pressure of 320 psig. The flow rate is based on the maximum developed runout flow (900 gpm per pump) with both HPI pumps running simultaneously. This flow rate is considered to cause the worst credible pressure transient. The opening of a CFT isolation valve was not considered because power is removed from the valve once it is closed upon plant cooldown and depressurization. Other postulated occurrences, makeup control valve failing to open, loss of DHR System cooling, all pressurizer heaters energizing, do not produce a pressure excursion as severe as that produced by the two HPI pumps. Although the pressurizer, by procedure, cannot be solid, for the purpose of analysis it was considered to go solid during the transient. The DHR System relief valve is a Seismic Class I Nuclear Class 2 bellows type of safety-relief valve. It should be noted that the postulation of both HPI pumps starting during DHR System operation is made only for the purpose of sizing the DHR System relief valve. The possibility of this event occurring due to either a single operator error or a single spurious signal is precluded by the design of the Safety Features Actuation System.

All changes are 1 unless otherwise noted

Low Temperature Overpressure Protection (LTOP) System 2
B 3.4.12

BASES

APPLICABLE SAFETY ANALYSES (continued)

a. Deactivating all but [one] makeup pump,

b. Deactivating HPI, and

c. Immobilizing CFT discharge isolation valves in their closed positions.

DHR System relief valve

The Reference 3 analyses demonstrate the PORV can maintain RCS pressure below limits when only one makeup pump is actuated. Consequently, the LCO allows only [one] makeup pump to be OPERABLE in the LTOP MODES.

Since the PORV cannot do this for one HPI pump and the RCS vent cannot do this for even one pump, the LCO also requires the HPI actuation circuits deactivated and the CFTs isolated.

The isolated CFTs must have their discharge valves closed and the valve power breakers fixed in their open positions. The analyses show the effect of CFT discharge is over a narrower RCS temperature range (175°F and below) than that of the LCO ([283]°F and below).

The DHR System relief valve is placed in service before RCS temperature is reduced below 280

Fracture mechanics analyses established the temperature of LTOP Applicability at [283]°F. Above this temperature, the pressurizer safety valves provide the reactor vessel pressure protection. The vessel materials were assumed to have a neutron irradiation accumulation equal to 21 effective full power years (EFPYs) of operation.

and operating procedures

This LCO will deactivate the HPI actuation when the RCS temperature is ≤ [283]°F. The consequences of a small break LOCA in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 4 and 5), requirements by having a maximum of [one] makeup pump OPERABLE.

Reference 3 contains the acceptance limits that satisfy the LTOP requirements. Any change to the RCS must be evaluated against these analyses to determine the impact of the change on the LTOP acceptance limits.

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORV is set to open at ≤ [555] psig. The setpoint is derived by modeling the performance of the LTOP System, assuming the limiting allowed LTOP transient of uncontrolled HPI actuation of one pump.

All changes are 1 unless otherwise noted

2

BASES

APPLICABLE SAFETY ANALYSES (continued)

These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoint at or below the derived limit ensures the Reference 1 limits will be met.

The PORV setpoint will be re-evaluated for compliance when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to embrittlement induced by neutron irradiation. Revised P/T limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3 discuss these examinations.

As required by License Condition 2.C(3)(d), prior to operation beyond 21 Effective Full Power Years, a reanalysis and proposed modifications, as necessary, to ensure continued means of protection for LTOP events will be provided to the NRC.

The PORV is considered an active component. Therefore, its failure represents the worst case LTOP single active failure.

Pressurizer Level Performance

Analyses of operator response time show that the pressurizer level must be maintained ≤ [220] inches to provide the 10 minute action time for correcting transients.

The pressurizer level limit will also be re-evaluated for compliance each time P/T limit curves are revised based on the results of the vessel material surveillance.

RCS Vent Performance

With the RCS depressurized, analyses show a vent of [0.75] square inches is capable of mitigating the transient resulting from full opening of the makeup control valve while the makeup pump is providing RCS makeup. The capacity of a vent this size is greater than the flow resulting from this credible transient at 100 psig back pressure, which is less than the maximum RCS pressure on the P/T limit curve in LCO 3.4.3.

The RCS vent size will also be re-evaluated for compliance each time P/T limit curves are revised based on the results of the vessel material surveillance.

The vent is passive and is not subject to active failure.

The LTOP System satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

All changes are 2 unless otherwise noted

BASES

LCO

INSERT 3

The LCO requires an LTOP System OPERABLE with a limited coolant input capability and a pressure relief capability. To limit coolant input, the LCO requires a maximum of [one] makeup pump OPERABLE, the HPI deactivated, and the CFT discharge isolation valves closed and immobilized. For pressure relief, it requires either the pressurizer coolant at or below a maximum level and the PORV OPERABLE with a lift setting at the LTOP limit or the RCS depressurized and a vent established.

The LCO is modified by two Notes. Note 1 allows [two makeup pumps] to be made capable of injecting for ≤ 1 hour during pump swap operations. One hour provides sufficient time to safely complete the actual transfer and to complete the administrative controls and surveillance requirements associated with the swap. The intent is to minimize the actual time that more than [one] makeup pump is physically capable of injection. Note 2 states that CFT isolation 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.

The pressurizer is OPERABLE with a coolant level ≤ [220] inches.

The PORV is OPERABLE when its block valve is open, its lift setpoint is set at ≤ [555] psig and testing has proven its ability to open at that setpoint, and motive power is available to the two valves and their control circuits.

For the depressurized RCS, an RCS vent is OPERABLE when open with an area of at least [0.75] square inches.

APPLICABILITY S

This LCO is applicable in MODE 4 ~~when any RCS cold leg temperature is ≤ [283]°F, in MODE 5~~ and 5, and in MODE 6 when the reactor vessel head is on. The Applicability ~~temperature of [283]°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]°F~~ in MODES 1, 2, and 3. With the vessel head off, overpressurization is not possible.

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]°F~~.

2 **INSERT 3**

For low temperature overpressure protection, Davis-Besse relies on the four-inch DHR System relief valve (DH-4849) with a lift setpoint ≤ 330 psig. This relief valve is located on the DHR System suction line from the RCS. The RCS to DHR System isolation valves (DH-11 and DH-12) must be open and control power removed from the valve operators for the DHR System relief valve to be OPERABLE. Control power can be removed either in the control room or at the motor control center (by removing fuses, opening breakers, or racking breakers out).

All changes are 2 unless otherwise noted

BASES

ACTIONS

A.1 and B.1

INSERT 4

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 causes the greatest pressure increase in the shortest time. Also, the vent cannot mitigate overpressurization from the injection of even one HPI pump.

The immediate Completion Times reflect the urgency of quickly proceeding with the Required Actions.

C.1, D.1, and D.2

An unisolated CFT requires isolation 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 isolation is needed and cannot be accomplished in 1 hour, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed in 12 hours. By increasing the RCS temperature to > 175°F, the CFT pressure of 600 psig cannot exceed the LTOP limits if both tanks are fully injected. Depressurizing the CFTs below the LTOP limit of [555] psig also prevents exceeding the LTOP limits in the same event.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that a limiting LTOP event is not likely in the allowed times.

E.1, F.1, and F.2

With the pressurizer level more than [220] inches, the time for operator action in a pressure increasing event is reduced. The postulated event most affected in the LTOP MODES is failure of the makeup control valve, which fills the pressurizer relatively rapidly. Restoration is required within 1 hour.

2 **INSERT 4**

A.1 and A.2

With the DHR System relief valve inoperable due to one or both RCS to DHR System isolation valves closed, the overpressure protection flow path is isolated. The flow path must be restored by opening the RCS to DHR System isolation bypass valves (DH-21 and DH-23), within 1 hour. After opening, the RCS to DHR System isolation bypass valves must be verified open every 24 hours.

The 1 hour Completion Time reflects the importance of the action and provides time for a timely opening of the RCS to DHR System isolation bypass valves. To ensure they remain in the open position, the positions of the RCS to DHR System isolation bypass valves are required to be verified every 24 hours. RCS to DHR System isolation bypass valves are manual valves and do not have remote position indication.

B.1

With control power available to one or both of the RCS to DHR System isolation valves, the overpressure protection flow path could be inadvertently isolated. The control power must be removed from the valves within 1 hour to ensure the valves will remain open during system operation.

The 1 hour Completion Time reflects the importance of the action and provides time for a timely removal of control power.

C.1

If the DHR System relief valve is inoperable for reasons other than the relief flow path (Condition A or B), the DHR System relief valve must be restored to OPERABLE status within 8 hours. The 8 hour Completion Time is acceptable due to the low probability of an overpressure event.

D.1, D.2, D.3, and D.4

If any Required Action and Completion Time of Condition A, B, or C is not met, other compensatory actions must be taken to minimize the probability and consequences of an LTOP event. Without an OPERABLE relief path for overpressure protection, the RCS water addition capabilities must be limited. Within 1 hour both HPI pumps must be disabled (e.g., by opening motor supply breakers), and within 8 hours the makeup pump suction automatic transfer to the borated water storage tank on low makeup tank level must be disabled. Makeup tank level must be verified to be ≤ 73 inches within 8 hours to minimize volume. Furthermore, without an overpressure relief path, RCS pressure and pressurizer level must be verified to be in the Acceptable Region of Figure 3.4.12-1 or 3.4.12-2 (depending on the MODE) within 8 hours to ensure an overpressure condition cannot occur. These Figures do not include instrument error uncertainties.

All changes are 2 unless otherwise noted

BASES

ACTIONS (continued)

If restoration within 1 hour in either case cannot be accomplished, Required Actions F.1 and F.2 must be performed within 12 hours to close the makeup control valve and its isolation valve. These Required Actions limit the makeup capability, which is not required with a high pressurizer level, and permit cooldown and depressurization to continue. Heatup must be stopped because heat addition decreases the reactor coolant density and increases the pressurizer level.

The Completion Times again are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that a limiting LTOP transient is not likely in the allowed times.

G.1, H.1, and H.2

With the PORV inoperable, overpressure relieving capability is lost, and restoration of the PORV within 1 hour is required. If that cannot be accomplished, the ability of the Makeup System to add water must be limited within the next 12 hours.

If restoration cannot be completed within 1 hour, Required Action H.1 and Required Action H.2 must be performed to limit RCS water addition capability. Makeup is not deactivated to maintain the RCS coolant level. Required Action H.1 and Required Action H.2 require reducing the makeup tank level to 70 inches and deactivating the low low makeup tank level interlock to the borated water storage tank. This makes the available makeup water volume insufficient to exceed the LTOP limit by a makeup control valve full opening.

These Completion Times also consider these activities can be accomplished in these time periods. A limiting LTOP event is not likely in those times.

Some PORV testing or maintenance can only be performed at plant shutdown. Such activity is permitted if Required Action H.1 and Required Action H.2 are taken to compensate for PORV unavailability.

All changes are 2 unless otherwise noted

BASES

ACTIONS (continued)

I.1

With the pressurizer level above [220] inches and the PORV inoperable or the LTOP System inoperable for any reason other than cited in Condition A through H, Required Action I.1 requires the RCS depressurized and vented within 12 hours from the time either Condition started.

One or more vents may be used. A vent size of ≥ [0.75] square inches is specified. This vent size assumes 100 psig backpressure. Because makeup may be required, the vent size accommodates inadvertent full makeup system operation. Such a vent keeps the pressure from full flow of [one] makeup pump with a wide open makeup control valve within the LCO limit.

The PORV has a larger area and may be used for venting by opening and locking it open.

This size RCS vent or the PORVs a vent cannot maintain RCS pressure below LTOP limits if the HPI and CFT systems are inadvertently actuated. Therefore, verification of the deactivation of two HPI pumps, HPI injection, and the CFTs must accompany the depressurizing and venting. Since these systems are required deactivated by the LCO, SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3 require verification of their deactivated status every 12 hours.

The Completion Time is based on operating experience that this activity can be accomplished in this time period and on engineering evaluations indicating that a limiting LTOP transient is not likely in this time.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3

Verifications must be performed that only [one] makeup pump is capable of injecting into the RCS, the HPI is deactivated, and the CFT discharge isolation valves are closed and immobilized. These Surveillances ensure the minimum coolant input capability will not create an RCS overpressure condition to challenge the LTOP System. The Surveillances are required at 12 hour intervals.

The 12 hour intervals are shown by operating practice to be sufficient to regularly assess conditions for potential degradation and verify operation within the safety analysis.

All changes are 2 unless otherwise noted

Low Temperature Overpressure Protection (LTOP) System
B 3.4.12

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.4

Verification of the pressurizer level at ≤ [220] inches by observing control room or other indications ensures a cushion of sufficient size is available to reduce the rate of pressure increase from potential transients.

The 30 minute Surveillance Frequency during heatup and cooldown must be performed for the LCO Applicability period when temperature changes can cause pressurizer level variations. This Frequency may be discontinued when the ends of these conditions are satisfied, as defined in plant procedures. Thereafter, the Surveillance is required at 12 hour intervals.

These Frequencies are shown by operating practice sufficient to regularly assess indications of potential degradation and verify operation within the safety analysis.

1

SR 3.4.12.5

INSERT 5

Verification that the PORV block valve is open ensures a flow path to the PORV. This is required at 12 hour intervals.

The interval has been shown by operating practice sufficient to regularly assess conditions for potential degradation and verify operation is within the safety analysis.

SR 3.4.12.6

The RCS vent of at least [0.75] square inches must be verified open for relief protection only if the vent is being used to satisfy the requirements of this LCO. For a vent valve not locked open, the Frequency is every 12 hours. Valves that are sealed or secured in the open position are considered "locked" in this context. For other vent path(s) (e.g., a vent valve that is locked, sealed, or secured in position, a removed pressurizer safety valve, or open manway), the required Frequency is every 31 days.

Again, the Frequency intervals consider operating practice to determine adequacy to regularly assess conditions for potential degradation and verify operation within the safety analysis.

(2) **INSERT 5**

Verification of the flow path from the RCS to the DHR System relief valve is required every 24 hours. This verification is performed by checking RCS to DHR System isolation valves in the open position with control power removed from the valve operator. This Surveillance ensures the overpressure relief flow path is aligned and remains aligned. Removal of control power ensures the flow path is not inadvertently closed.

The Frequency is adequate based on operating experience. Manual operation is required to close the isolation valves or energize control power. Valve operations are administratively controlled by procedure. In this configuration the isolation valves will not inadvertently close.

Attachment 1, Volume 9, Rev. 0, Page 270 of 415

All changes are 2 unless otherwise noted

BASES

SURVEILLANCE REQUIREMENTS (continued)

The passive vent path arrangement must only be open to be OPERABLE.

2

SR 3.4.12.7

TSTF 443 changes not shown

INSERT 6

A CHANNEL FUNCTIONAL TEST is required within [12] hours after decreasing RCS temperature to ≤ [283]°F and every 31 days thereafter to ensure the setpoint is proper for using the PORV for LTOP. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. PORV actuation is not needed, as it could depressurize the RCS.

The [12] hour Frequency considers the unlikelihood of a low temperature overpressure event during the time. The 31 day Frequency is based on industry accepted practice and is acceptable by experience with equipment reliability.

SR 3.4.12.8

The performance of a CHANNEL CALIBRATION is required every [18] months. The CHANNEL CALIBRATION for the LTOP setpoint ensures that the PORV will be actuated at the appropriate RCS pressure by verifying the accuracy of the instrument string. The calibration can only be performed in shutdown.

The Frequency considers a typical refueling cycle and industry accepted practice.

REFERENCES

1. 10 CFR 50, Appendix G.

2. Generic Letter 88-11.

3. U FSAR, Section 1.5 9.3.5 1

4. 10 CFR 50.46. 1

5. 10 CFR 50, Appendix K.

2 **INSERT 6**

Verification of the DHR System relief valve lift setpoint must be performed to ensure LTOP requirements can be met. Overpressure protection of the RCS is ensured by the DHR System relief valve, which relieves pressure and prevents the RCS from exceeding the Pressure/Temperature Limits.

The DHR System relief valve setpoint is verified in accordance with the Inservice Testing (IST) Program for proper operation and correct lift setting of ≤ 330 psig. This lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. The IST Program specifies the testing and frequency, as directed by ASME Code.

Attachment 1, Volume 9, Rev. 0, Page 272 of 415

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.12 BASES, LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP)

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

2. Changes made to be consistent with changes made to the Specification.

3. This Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed in to what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific submittal.

4. The Davis-Besse methods for LTOP have been included. With RCS temperature between approximately 500°F and 280°F, pressurizer safety valves cannot provide overpressure protection; LTOP is provided by operating procedures. Below 140°F, credible overpressurization sources are secured. These methods for LTOP have been previously reviewed and approved by the NRC, as documented in the NRC Safety Evaluation for Amendment 199, dated July 20, 1995.

Attachment 1, Volume 9, Rev. 0, Page 273 of 415

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.12, LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP)

There are no specific NSHC discussions for this Specification.

ATTACHMENT 13

ITS 3.4.13, RCS OPERATIONAL LEAKAGE

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

ITS

A01

ITS 3.4.13

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.13 3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 150 gallons per day primary to secondary leakage through any one steam generator (SG),

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 10 GPM CONTROLLED LEAKAGE, and L01

f. 5 GPM leakage from any Reactor Coolant System Pressure Isolation Valve as specified in Table 3.4-2. See ITS 3.4.14

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

ACTION B a. With any PRESSURE BOUNDARY LEAKAGE, or with primary to secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION A b. With any Reactor Coolant System operational leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE or primary to secondary leakage, reduce the leakage rate to within limits within 4 hours or be in at

ACTION B least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours ~~except as permitted by paragraph c below.~~

c. In the event that integrity of any pressure isolation valve specified in Table 3.4-2 cannot be demonstrated, POWER OPERATION may continue, provided that at least two valves in each high pressure line having a non-functional valve are in and remain in, the mode corresponding to the isolated condition.[a] See ITS 3.4.14

d. The provisions of Section 3.0.4 are not applicable for entry into MODES 3 and 4 for the purpose of testing the isolation valves in Table 3.4-2.

[a] Motor operated valves shall be placed in the closed position and power supplies deenergized. See ITS 3.4.14

ITS

A01

ITS 3.4.13

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System operational leakages shall be demonstrated to be within each of the above limits by:

a. ~~Monitoring the containment atmosphere gaseous or particulate radioactivity at least once per 12 hours.~~

b. ~~Monitoring the containment sump level and flow indication at least once per 12 hours.~~

L02

c. ~~Measurement of the CONTROLLED LEAKAGE from the reactor coolant pump seals to the makeup system when the Reactor Coolant System pressure is 2185 ± 20 psig at least once per 31 days.~~

L01

SR 3.4.13.1 d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation.[(1)(2)]

SR 3.4.13.2 e. Verifying that primary to secondary leakage is ≤ 150 gallons per day through any one steam generator, at least once per 72 hours. [(2)]

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-2 shall be individually demonstrated OPERABLE by verifying leakage testing (or the equivalent) to be within its limit prior to entering MODE 2:

a. After each refueling outage,

b. Whenever the plant has been in COLD SHUTDOWN for 7 days, or more, and if leakage testing has not been performed in the previous 9 months, and

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve.

d. The provisions of Specification 4.0.4 are not applicable for entry into MODES 3 or 4.

4.4.6.2.3 Whenever the integrity of a pressure isolation valve listed in Table 3.4-2 cannot be demonstrated, determine and record the integrity of the high pressure flowpath on a daily basis. Integrity shall be determined by performing either a leakage test of the remaining pressure isolation valve, or a combined leakage test of the remaining pressure isolation valve in a series with the closed motor-operated containment isolation valve. In addition, record the position of the closed motor-operated containment isolation valve located in the high pressure piping on a daily basis.

See ITS 3.4.14

SR 3.4.13.1 NOTE 2 — (1) Not applicable to primary to secondary leakage.

SR 3.4.13.1 NOTE 1, SR 3.4.13.2 NOTE — (2) Not required to be performed until 12 hours after establishment of steady state operation.

DISCUSSION OF CHANGES
ITS 3.4.13, RCS OPERATIONAL LEAKAGE

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 1 – Relaxation of LCO Requirements)* CTS 3.4.6.2.e requires that Reactor Coolant System leakage shall be limited to 10 gpm of CONTROLLED LEAKAGE. CTS 4.4.6.2.1.c requires a verification that the CONTROLLED LEAKAGE is within the limit every 31 days. ITS LCO 3.4.13 does not retain these requirements. This changes the CTS by deleting this LCO requirement.

The purpose of CTS 3.4.6.2.e and its associated Surveillance is to ensure the CONTROLLED LEAKAGE does not exceed a specified limit. CONTROLLED LEAKAGE is seal water flow from the reactor coolant pumps seals. The CTS 3.4.6.2.e limit of 10 gpm is the design leakage rate through the pump seals and back to the makeup tank via the seal return lines. Thus a higher flow rate would indicate that the pumps seals are deteriorated or failed. However, a maximum seal water leakage (i.e. flow) is not an assumption of any accident or transient analysis, which is the reason it has been maintained in another pressurized water reactor ISTS (NUREG-1431, "Standard Technical Specifications - Westinghouse Plants," ISTS 3.5.5). For Davis-Besse, there is no need to quantify the normal CONTROLLED LEAKAGE since it is normal system operation and there is no loss from the RCS inventory. Furthermore, if the seal water flow increases greater than the current 10 gpm limit due to an upper seal failure, the increased flow would be directed to the containment normal sump. The containment normal sump is the collecting sump that identified LEAKAGE is quantified (and limited to 10 gpm). Thus the increased seal water flow resulting

DISCUSSION OF CHANGES
ITS 3.4.13, RCS OPERATIONAL LEAKAGE

from a failed or leaking upper seal would be detected and proper actions taken as necessary. Therefore this change is acceptable and is designated as less restrictive because an LCO requirement required in the CTS will not be required in the ITS.

L02 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.4.6.2.1.a requires monitoring of the containment atmosphere gaseous or particulate radioactivity at least once per 12 hours. CTS 4.4.6.2.1.b requires monitoring the containment sump level and flow indication at least once per 12 hours. The ITS does not contain these Surveillance Requirements. This changes the CTS by eliminating these Surveillance Requirements.

This change is acceptable because the deleted Surveillance Requirements are not necessary to verify that the LCO is being met. Thus, appropriate Surveillance Requirements continue to be performed in a manner and at a Frequency necessary to give confidence that the LCO is being met. The indications in the deleted Surveillance Requirements are not necessarily indications of failure to meet the LCO on RCS operational LEAKAGE. These items do provide useful information and the containment atmosphere particulate monitor and the containment sump monitors are required to be OPERABLE and tested by ITS 3.4.15, "RCS Leakage Detection Instrumentation." However, under ITS SR 3.0.1, failure to meet the Surveillance results in failure to meet the LCO. As these indications do not necessarily indicate a failure to meet the LCO, it is not appropriate to retain these indications in this Specification. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

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

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

3.4.6.2 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, and

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

;

1

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

ACTIONS

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
Action b	A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
Action a, Action b	B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

CTS

SURVEILLANCE REQUIREMENTS

CTS		SURVEILLANCE	FREQUENCY
4.4.6.2.1.d (including footnotes (1) and (2))	SR 3.4.13.1	-----NOTES----- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. ---------- Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.	72 hours
4.4.6.2.1.e (including footnote (2))	R 3.4.13.2	-----NOTE----- Not required to be performed until 12 hours after establishment of steady state operation. ---------- Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG.	72 hours

Attachment 1, Volume 9, Rev. 0, Page 284 of 415

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.13, RCS OPERATIONAL LEAKAGE

1. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01, Section 5.1.3.

Attachment 1, Volume 9, Rev. 0, Page 285 of 415

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

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 plant 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.

Although not committed to

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant ~~LEAKAGE.~~ Regulatory Guide 1.45 (Ref. 2), it describes acceptable methods for selecting Leakage Detection ~~Systems.~~ 2

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 is necessary. Quickly 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.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

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 plant shutdown before a LOCA occurs. This advantage has been shown by "leak before break" studies.

BASES

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 such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from all steam generators (SGs) is [1 gallon per minute] or increases to [1 gallon per minute] as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis. 1

main 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. M 2

U 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 safety analysis assumption is relatively inconsequential. 2 1

M ~~The SLB is more limiting for site radiation releases.~~ The safety analysis for the SLB accident assumes the entire [1 gpm] primary to secondary LEAKAGE is through the affected generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100. M 2 2 1 2

RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

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 degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

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
return flow 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. 2

d. Primary to Secondary LEAKAGE Through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

BASES

APPLICABILITY (continued)

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leaktight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

If unidentified LEAKAGE or identified LEAKAGE are in excess of the LCO limits, the LEAKAGE must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists or primary to secondary LEAKAGE is not within limit, or if unidentified or identified LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The Completion Times allowed are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE within the LCO limits ensures that the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be performed with the reactor at steady state operating conditions (stable temperature, power level,
and
pressurizer ~~and makeup tank levels, makeup and letdown, [and RCP seal~~ 2
level

BASES

SURVEILLANCE REQUIREMENTS (continued)

injection and return flows]). The Surveillance is modified by two Notes. Note 1 states that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. 2 5

The accuracy of the results will be impacted if any measured parameter used to calculate the RCS LEAKAGE is not in a steady state condition.

~~Steady state operation is required to perform a proper water inventory balance since calculations during maneuvering are not useful.~~ ~~For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP pump seal injection and return flows.~~ 2 4

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable ~~RCS pressure, temperature,~~ power level (± 1%), ~~pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows~~. (2)

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. UFSAR, ~~Chapter~~ Section [15]. (2) (1)
4. NEI 97-06, "Steam Generator Program Guidelines."
5. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.13 BASES, RCS OPERATIONAL LEAKAGE

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

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

3. Changes made to reflect changes made to the Specification.

4. Duplicate discussion deleted. Steady state operation is discussed in the previous paragraph.

5. Editorial change for clarity.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.13, RCS OPERATIONAL LEAKAGE

There are no specific NSHC discussions for this Specification.

ATTACHMENT 14

ITS 3.4.14, RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

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

A01

ITS 3.4.14

ITS

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 150 gallons per day primary to secondary leakage through any one steam generator (SG),

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 10 GPM CONTROLLED LEAKAGE, and

See ITS 3.4.13

LCO 3.4.14 part 1 — f. 5 GPM leakage from any Reactor Coolant System Pressure Isolation Valve as specified in Table 3.4-2. — LA01

SR 3.4.14.2

APPLICABILITY: MODES 1, 2, 3 and 4 — L01

Add proposed ACTION Note 1 — A02

ACTION:

Add proposed ACTION Note 2 — A03

a. With any PRESSURE BOUNDARY LEAKAGE, or with primary to secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours. — See ITS 3.4.13

ACTION A — b. With any Reactor Coolant System operational leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE or primary to secondary leakage, reduce the leakage rate to within limits within 4 hours or be in at

ACTION B — least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours except as permitted by paragraph c below. — L02

ACTION A — c. In the event that integrity of any pressure isolation valve specified in Table 3.4-2 cannot be demonstrated, POWER OPERATION may continue, provided that at least two valves in each high pressure line having a non-functional valve are in and remain in, the mode corresponding to the isolated condition.[a] — Add proposed Required Actions A.1 and A.2 Note

SR 3.4.14.2 Note — d. The provisions of Section 3.0.4 are not applicable for entry into MODES 3 and 4 for the purpose of testing the isolation valves in Table 3.4-2. — M01

ACTION A — [a]Motor operated valves shall be placed in the closed position and power supplies deenergized.

A01

ITS 3.4.14

ITS

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System operational leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous or particulate radioactivity at least once per 12 hours.

b. Monitoring the containment sump level and flow indication at least once per 12 hours.

c. Measurement of the CONTROLLED LEAKAGE from the reactor coolant pump seals to the makeup system when the Reactor Coolant System pressure is 2185 ± 20 psig at least once per 31 days.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation. (1) (2)

e. Verifying that primary to secondary leakage is ≤ 150 gallons per day through any one steam generator, at least once per 72 hours. (2)

See ITS 3.4.13

SR 3.4.14.2 4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve ~~specified in Table 3.4-2~~ shall be individually demonstrated OPERABLE by verifying leakage testing (or the equivalent) to be within its limit prior to entering MODE 2: LA01

a. After each refueling outage,

b. Whenever the plant has been in COLD SHUTDOWN for 7 days, or more, and if leakage testing has not been performed in the previous 9 months, and

c. ~~Prior to returning the valve to service following maintenance, repair or replacement work on the valve.~~ L03

SR 3.4.14.2 Note

d. The provisions of Specification 4.0.4 are not applicable for entry into MODES 3 or 4.

~~4.4.6.2.3 Whenever the integrity of a pressure isolation valve listed in Table 3.4-2 cannot be demonstrated, determine and record the integrity of the high pressure flowpath on a daily basis. Integrity shall be determined by performing either a leakage test of the remaining pressure isolation valve, or a combined leakage test of the remaining pressure isolation valve in a series with the closed motor-operated containment isolation valve. In addition, record the position of the closed motor-operated containment isolation valve located in the high pressure piping on a daily basis.~~ L04

(1) Not applicable to primary to secondary leakage.
(2) Not required to be performed until 12 hours after establishment of steady state operation.

See ITS 3.4.13

ITS A01 ITS 3.4.14

TABLE 3.4-2

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

LA01

SYSTEM	VALVE NUMBERS (b)	MAXIMUM ALLOWABLE LEAKAGE (a)(c)
1. Decay Heat Removal	CF-30	≤ 5.0 gpm
2. Decay Heat Removal	CF-31	≤ 5.0 gpm
3. Decay Heat Removal	DH-76	≤ 5.0 gpm
4. Decay Heat Removal	DH-77	≤ 5.0 gpm

LA02

SR 3.4.14.2

Notes:

SR 3.4.14.2 (a)

1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

(b) Valves CF-30 and CF-31 will be tested with the Reactor Coolant system pressure >1200 psig. Valves DH-76 and DH-77 will be tested with normal Core Flooding Tank pressure which is >575 psig. Minimum differential test pressure across each valve shall not be less than 150 psid.

(c) 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.

LA02

ITS

A01

ITS 3.4.14

TABLE 3.3-3 (Continued)

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF UNITS	UNITS TO TRIP	MINIMUM UNITS OPERABLE	APPLICABLE MODES	ACTION
3. MANUAL ACTUATION					
a. SFAS (except Containment Spray and Emergency Sump Recirculation)	2	2	2	1,2,3,4	12
b. Containment Spray	2	2	2	1,2,3,4	12
4. SEQUENCE LOGIC CHANNELS					
a. Sequencer	4	2/BUS	2/BUS	1,2,3,4	15#
b. Essential Bus Feeder Breaker Trip Degraded Voltage Relay (DVR)	4*****	2/BUS	2/BUS	1,2,3,4	15#
c. Diesel Generator Start, Load Shed on Essential Bus Loss of Voltage Relay (LVR)	4	2/BUS	2/BUS	1,2,3,4	15#
5. INTERLOCK CHANNELS					
a. Decay Heat Isolation Valve	1	1	1	1,2,3	13#
b. Pressurizer Heaters	2	2	2	3******	14

See ITS 3.3.6

See ITS 3.3.8 and ITS 3.8.1

A04

See ITS 3.3.5

LCO 3.4.14 part 2

DAVIS-BESSE, UNIT 1 3/4 3-11 Amendment No. ~~28, 37, 52, 102, 135, 159, 211, 221~~, 275

Attachment 1, Volume 9, Rev. 0, Page 301 of 415

A01

ITS 3.4.14

ITS

TABLE 3.3-3 (Continued)

ACTION STATEMENTS

ACTION 12 - With the number of OPERABLE Units one less than the Total Number of Units, restore the inoperable functional unit to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

See ITS 3.3.6

ACTION C

ACTION 13 - a. With less than the Minimum Units OPERABLE and indicated reactor coolant pressure > 328 psig, both Decay Heat Isolation Valves ~~(DH11 and DH12)~~ shall be verified closed. within 4 hours

LA03

L05

b. With Less than the Minimum Units OPERABLE and indicated reactor coolant pressure < 328 psig operation may continue; however, the functional unit shall be OPERABLE prior to increasing indicated reactor coolant pressure above 328 psig.

ACTION 14 - With less than the Minimum Units OPERABLE and indicated reactor coolant pressure < 328 psig, operation may continue; however, the functional unit shall be OPERABLE prior to increasing indicated reactor coolant pressure above 328 psig, or the inoperable functional unit shall be placed in the tripped state.

See ITS 3.3.5

ACTION 15 - a. With the number of OPERABLE units one less than the Minimum Units Operable per Bus, place the inoperable unit in the tripped condition within one hour. For functional unit 4.a the sequencer shall be placed in the tripped condition by physical removal of the sequencer module. The inoperable functional unit may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

b. With the number of OPERABLE units two less than the Minimum Units Operable per Bus, declare inoperable the Emergency Diesel Generator associated with the functional units not meeting the required minimum units OPERABLE and take the ACTION required of Specification 3.8.1.1.

See ITS 3.3.8 and ITS 3.8.1

ITS

A01

ITS 3.4.14

TABLE 3.3-4

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION ALLOWABLE VALUES

FUNCTIONAL UNIT	ALLOWABLE VALUES##	
INSTRUMENT STRINGS		
a. DELETED	DELETED	
b. Containment Pressure – High	≤ 19.38 psia	See ITS 3.3.5
c. Containment Pressure - High-High	≤ 41.65 psia	
d. RCS Pressure – Low	≥ 1576.2 psig	
e. RCS Pressure - Low-Low	≥ 441.42 psig	
f. BWST Level	≥ 101.6 and ≤ 115.4 in. H_2O	
SEQUENCE LOGIC CHANNELS		
a. Essential Bus Feeder Breaker Trip Degraded Voltage Relay (DVR)	≥ 3712 volts (dropout) and ≤ 3771 volts (pickup) with a time delay of ≥ 6.4 and ≤ 7.9 sec	See ITS 3.3.8
b. Diesel Generator Start, Load Shed on Essential Bus Loss of Voltage Relay (LVR)	≥ 2071 volts (dropout) and ≤ 2492 volts (pickup) with a time delay of ≥ 0.42 and ≤ 0.58 sec	
INTERLOCK CHANNELS		
a. Decay Heat Isolation Valve and Pressurizer Heater	< 328 psig *	

See ITS 3.3.5

SR 3.4.14.3, SR 3.4.14.4

LA03

* Referenced to the RCS Pressure instrumentation tap.

** Allowable Values for CHANNEL FUNCTIONAL TEST

See ITS 3.3.5

DAVIS-BESSE, UNIT 1 3/4 3-13 Amendment No. ~~7,28,36,58,190,218,221,241,~~ ~~243~~. 275

ITS

A01

TABLE 4.3-2 (Continued)

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. SEQUENCE LOGIC CHANNELS				
a. Sequencer	S	NA	M	1, 2, 3, 4
b. Essential Bus Feeder Breaker Trip, Degraded Voltage Relay (DVR)	S	A(3)	M(3)	1, 2, 3, 4
c. Diesel Generator Start, Load Shed on Essential Bus, Loss of Voltage Relay (LVR)	S	A(3)	M(3)	1, 2, 3, 4
5. INTERLOCK CHANNELS	SR 3.4.14.1	SR 3.4.14.5	SR 3.4.14.3, SR 3.4.14.4	
a. Decay Heat Isolation Valve	S	R	**	1, 2, 3
b. Pressurizer Heater	S	R	**	3 ##

See ITS 3.3.8 and ITS 3.8.1

See ITS 3.3.5

TABLE NOTATION

(1) Manual actuation switches shall be tested at least once per REFUELING INTERVAL. All other circuitry associated with manual safeguards actuation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days.

See ITS 3.3.6

(2) The CHANNEL FUNCTIONAL TEST shall include exercising the transmitter by applying either vacuum or pressure to the appropriate side of the transmitter.

See ITS 3.3.5

(3) The as-left instrument setting shall be returned to a setting within the tolerance band of the trip setpoint established to protect the safety limit.

See ITS 3.3.8

** See Specification 4.5.2.d.1

SR 3.4.14.3, SR 3.4.14.4

When either Decay Heat Isolation Valve is open.

See ITS 3.3.5

Attachment 1, Volume 9, Rev. 0, Page 304 of 415

A01

ITS 3.4.14

ITS

Revised by NRC Letter Dated
June 6, 1995

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once each REFUELING INTERVAL, or prior to operation after ECCS piping has been drained by verifying that the ECCS piping is full of water by venting the ECCS pump casings and discharge piping high points.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment emergency sump and cause restriction of the pump suction during LOCA conditions. This visual inspection shall be performed: — See ITS 3.5.2

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. For all areas of containment affected by an entry, at least once daily while work is ongoing and again during the final exit after completion of work (containment closeout) when CONTAINMENT INTEGRITY is established.

A04

d. At least once each REFUELING INTERVAL by:

See ITS 3.5.2

LCO 3.4.14 part 2 — 1. Verifying that the interlocks:

LA03

SR 3.4.14.4 — a) Close DH-11 and DH-12 and deenergize the pressurizer heaters, if either DH-11 or DH-12 is open and a simulated reactor coolant system pressure which is greater than the Allowable Value (<328 psig) is applied.

SR 3.4.14.3 Note, SR 3.4.14.4 Note — The interlock to close DH-11 and/or DH-12 is not required if the valve is closed and 480 V AC power is disconnected from its motor operators.

LA03

SR 3.4.14.3 — b) Prevent the opening of DH-11 and DH-12 when a simulated or actual reactor coolant system pressure which is greater than the Allowable Value (<328 psig) is applied.

Add proposed SR 3.4.14.5 — M01

2. a) A visual inspection of the containment emergency sump which verifies that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

b) Verifying that on a Borated Water Storage Tank (BWST) Low-Low Level interlock trip, with the motor operators for the BWST outlet isolation valves and the containment emergency sump recirculation valves energized, the BWST Outlet Valve HV-DH7A (HV-DH7B) automatically close in ≤75 seconds after the operator manually pushes the control switch to open the Containment Emergency Sump Valve HV-DH9A (HV-DH9B) which should be verified to open in ≤75 seconds. — See ITS 3.5.2

3. Deleted

DISCUSSION OF CHANGES
ITS 3.4.14, RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.4.6.2 Actions b and c specify the compensatory actions to take when the leakage through any RCS PIV(s) is greater than the specified limit. ITS 3.4.14 ACTIONS A and B also state the appropriate compensatory actions under the same condition; however, ITS 3.4.14 ACTIONS Note 1 has been added. ITS 3.4.14 ACTIONS Note 1 allows separate Condition entry for each RCS PIV flow path. This changes the CTS by explicitly stating that the Actions are to be taken separately for each inoperable RCS PIV flow path.

The purpose of the Note is to provide explicit instructions for proper application of the Action for Technical Specification compliance. In conjunction with proposed Specification 1.3, "Completion Times," this Note provides direction consistent with the intent of the existing Action for inoperable PIVs. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS 3.4.6.2 Actions b and c specify the compensatory actions to take when the leakage through any RCS PIV(s) is greater than the specified limit. ITS 3.4.14 ACTIONS A and B also state the appropriate compensatory actions under the same condition; however, ITS 3.4.14 ACTIONS Note 2 has been added. ITS 3.4.14 ACTIONS Note 2 states "Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable RCS PIV." This changes the CTS by explicitly stating that the Conditions and Required Actions for systems made inoperable by an inoperable RCS PIV must be entered.

The purpose of the Note is to provide explicit instructions for proper application of the ACTION for Technical Specification compliance. This Note facilitates the use and understanding of the intent to consider any system affected by inoperable RCS PIVs, which is to have its ACTIONS also apply if it is determined to be inoperable. With the addition of ITS LCO 3.0.6, this intent would not be necessarily applied. This clarification is consistent with the intent and interpretation of the existing Technical Specifications, and is therefore considered an administrative presentation preference. This change is designated as administrative because it does not result in technical changes to the CTS.

A04 CTS Table 3.3-3 requires one channel of the decay heat isolation valve interlock to be OPERABLE. This channel is the channel common to the Safety Features Actuation System (SFAS) instrumentation, and it provides a interlock signal to one of the two isolation valves. The other channel that provides an interlock signal to the decay heat isolation valve is not common to SFAS instrumentation. This channel is covered by CTS 4.5.2.d.1, which requires interlock testing for the

DISCUSSION OF CHANGES
ITS 3.4.14, RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

two decay heat isolation valves (DH-11 and DH-12). ITS 3.4.14 is combining these two requirements into a single LCO. ITS LCO 3.4.14 part 2 requires the Decay Heat Removal (DHR) System interlock function to be OPERABLE. This changes the CTS by combining the requirements for the interlock function into a single LCO.

This change is acceptable since the requirements are not being changed, except as justified in other Discussion of Changes. The requirements are simply being combined into a single LCO, consistent with NUREG-1430. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.4.6.2 Actions b and c specify the compensatory actions to take when the leakage through any RCS PIV(s) is greater than the specified limit. The compensatory action is to isolate the high pressure portion of the affected system from the low pressure portion of the affected system by use of a combination of at least two closed valves. The CTS does not include any leakage restrictions that may be used to satisfy the isolation requirement of this action. ITS 3.4.14 ACTION A is consistent with the requirement in CTS 3.4.6.2 Action c, however, a Note has been added to the Required Actions (ITS 3.4.14 Required Actions A.1 and A.2 Note) which specifies that each valve used to satisfy ITS 3.4.14 Required Actions A.1 and A.2 must have been verified to meet SR 3.4.14.2.a, the RCS PIV maximum leakage limit Surveillance Requirement, and either be in the RCS pressure boundary or the high pressure portion of the system. This changes the CTS by providing a Note which explicitly states that the valves used to satisfy Required Action must satisfy the same leakage requirements of the RCS PIVs and provides an option for them to be in the RCS pressure boundary.

The purpose of CTS 3.4.6.2 Action c is to isolate the flow path in order to minimize the leakage from the high pressure portion of the RCS to the low pressure piping. The ITS 3.4.14 Required Actions A.1 and A.2 Note requires the valves used to provide isolation between the high pressure and low pressure portions of the affected system to have been verified to meet the RCS PIV maximum leakage limits within the required Surveillance Frequency. The addition of the Note represents an additional restriction on unit operation necessary to help ensure the valves used to isolate the high pressure portion from the low pressure portion of the affected system are capable of preventing the overpressurization of the low pressure portion of the system. The ITS 3.4.14 Required Actions A.1 and A.2 Note also provides the option for the valves to be in the RCS pressure boundary. However, if it is in the RCS pressure boundary, it is in the high pressure portion of the system. This change is designated as more restrictive because it adds a new requirement to the CTS.

M02 The CTS does not require a CHANNEL CALIBRATION of the decay heat isolation valve interlock channel that is not common to SFAS instrumentation. ITS SR 3.4.14.5 requires a CHANNEL CALIBRATION every 24 months. This changes the CTS by adding a specific CHANNEL CALIBRATION requirement for this channel.

DISCUSSION OF CHANGES
ITS 3.4.14, RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

The purpose of the CHANNEL CALIBRATION is to ensure the channel can perform as required. Currently, the CTS only requires a functional test of the channel (CTS 4.5.2.d.1). The addition of the CHANNEL CALIBRATION requirement will help ensure the accuracy of the instrument string, therefore the change is acceptable. The proposed 24 month Frequency is consistent with the CHANNEL CALIBRATION Frequency for the other channel (the channel common to SFAS instrumentation) and with the Frequency of CTS 4.5.2.d.1. This change is designated as more restrictive because it adds a new requirement to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.4.6.2.f requires the leakage from each RCS PIV specified in Table 3.4-2 to be ≤ 5 gpm. CTS 4.4.6.2.2, the Surveillance which checks the RCS PIV leakage, also references Table 3.4-2. CTS Table 3.4-2 contains a list of the RCS PIVs and their associated valve numbers. ITS 3.4.14 does not contain a list of the RCS PIVs or their associated valve numbers. This changes the CTS by relocating the list of RCS PIVs and their associated valve numbers to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 3.4.14 still requires the RCS PIVs to be OPERABLE, and ITS SR 3.4.14.2 requires periodic Surveillances to determine RCS PIV leakage. It is not necessary for the list of RCS PIVs to be in the Technical Specifications in order to ensure that the RCS PIVs are OPERABLE. Other lists of components, such as containment isolation valves and equipment response time, have been relocated from the Technical Specification to licensee-controlled documents while retaining the requirements on these components in Technical Specifications. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS Table 3.4-2 is modified by Notes (b) and (c). Note (b) describes the pressure at which the RCS PIVs are to be tested. Note (c) explains an alternative method of testing the PIVs to satisfy the ALARA requirements. ITS 3.4.14 does not retain these Notes. This changes the CTS by relocating the information in the Notes to the Bases.

DISCUSSION OF CHANGES
ITS 3.4.14, RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

The removal of these details for performing Surveillance Requirements from the Technical Specification is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 3.4.14 still retains the requirements that RCS PIV leakage must be within limit and provides the appropriate Surveillance that includes the leakage limit. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA03 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Table 3.3-3 Action 13 and CTS 4.5.2.d.1 provide the specific valve numbers for the decay heat removal isolation valves. CTS Table 3.3-4 footnote * states that the Decay Heat Removal System interlock function Allowable Value is referenced to the RCS pressure instrumentation tap. ITS 3.4.14 does not include these details. This changes the CTS by moving the valve numbers and information concerning the Allowable Value reference point to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the Decay Heat Removal System interlock function to be OPERABLE, and provides Surveillances to ensure the interlock operates at the proper setpoint. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

L01 *(Category 2 – Relaxation of Applicability)* CTS 3.4.6.2.f is applicable in MODES 1, 2, 3, and 4. ITS 3.4.14 is applicable in MODES 1, 2, and 3, and in MODE 4, except valves in the decay heat removal (DHR) flow path when in, or the transition to or from, the DHR mode of operation. This changes the CTS by exempting the DHR flow path PIVs (CF-30, CF-31, DH-76, and DH-77) from the leakage requirements when in or during the transition to or from the DHR mode of operation.

The purpose of CTS 3.4.6.2.f is to ensure the RCS PIVs are within leakage limits. This change is acceptable because the LCO requirements continue to ensure that the components are maintained consistent with the safety analyses and

DISCUSSION OF CHANGES
ITS 3.4.14, RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

licensing basis. It is not necessary for the DHR PIVs to meet the leakage limits when in or during transition to or from the DHR mode of operation. These check valves cannot open until the DHR System is placed in service, which is not until RCS pressure is less than the test pressure of the DHR system. Thus overpressurization of the DHR piping is not a concern. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than are being applied in the CTS.

L02 *(Category 3 – Relaxation of Completion Time)* CTS 3.4.6.2 Action b requires, in part, that if the RCS PIV leakage is not within limit, it must be restored within 4 hours. If RCS PIV leakage is not restored, either a unit shutdown is required or the requirements of CTS 3.4.6.2 Action c must be met. CTS 3.4.6.2 Action c states, in part, that with the integrity of any pressure isolation valve specified in Table 3.4-2 not demonstrated, power operation may continue provided at least two valves in each high pressure line that has a non-functional valve are in and remain in, the mode corresponding to the isolated condition. Therefore, the two CTS Actions result in requiring the two valves to be in the isolated condition within 4 hours. ITS 3.4.14 ACTION A contains this same requirements, but allows 4 hours to isolate the first valve and 72 hours to isolate the second valve. This changes the CTS by extending the time requirement to close the second valve from 4 hours to 72 hours.

The purpose of CTS 3.4.6.2 Actions b and c is to allow time to reduce leakage before isolating the pathway. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the allowed Completion Time. The time to close the first valve remains the same and the time to close the second valve has been changed from 4 hours to 72 hours. The 4 hour Completion Time to close the first valve ensures leakage in excess of the allowable limit is reduced. The 4 hour time allows time for these actions and restricts the time of operation with leaking valves. The 72 hours Completion Time to close the second valve considers the time required to complete the Required Action and the low probability of the first valve failing during this period. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

L03 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.4.6.2.2.c requires testing of RCS PIVs following maintenance, repair, or replacement work on the valve. ITS 3.4.14 does not include this requirement. This changes the CTS by eliminating a post-maintenance Surveillance Requirement.

This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function. Whenever, the OPERABILITY of a system or component has been affected by repair, maintenance, modification, or replacement of a component, post maintenance testing is required to

DISCUSSION OF CHANGES
ITS 3.4.14, RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

demonstrate the OPERABILITY of a system or component. This is described in the Bases for ITS SR 3.0.1 and required under SR 3.0.1. In addition, the requirements of 10 CFR 50, Appendix B, Section XI (Test Control), provide adequate controls for test programs to ensure that testing incorporates applicable acceptance criteria. Compliance with 10 CFR 50, Appendix B is required under the unit operating license. As a result, post-maintenance testing will continue to be performed and an explicit requirement in the Technical Specifications is not necessary. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

L04 *(Category 5 - Deletion of Surveillance Requirement)* CTS 4.4.6.2.3 provides additional compensatory measures to take, above those required by CTS 3.6.4.2 Action c, when leakage through an RCS PIV is not within limit. The CTS requires a daily leakage test of the remaining OPERABLE RCS PIV in the flow path or a combined leakage test of the two valves used to comply with CTS 3.6.4.2 Action c. In addition, the position of the second, non-RCS PIV valve is required to be recorded on a daily basis. ITS 3.4.14 does not include these additional compensatory measures. This changes the CTS by deleting the additional compensatory measures taken when leakage through an RCS PIV is not within limit.

The purpose of CTS 4.4.6.2.3 is to help ensure that the leakage through the valves used to isolate the penetration with an inoperable RCS PIV is minimized so that an overpressurization event of the downstream piping cannot occur. The change is acceptable since the requirements to ensure the leakage through the two closed valves is within the RCS PIV leakage limit and to ensure closure of the valves are maintained in the ITS. The RCS PIV leakage is ensured prior to using each of the valves as an isolation boundary, as required by the ITS 3.4.14 Required Actions Note. Once leakage is checked, it is not expected to change since the valve cannot be manipulated (ITS 3.4.14 ACTION A requires the valves to be isolated - thus they must remain isolated to comply with the ACTION). Manipulation of manual valves that have been closed and automatic valves that have de-activated to comply with Technical Specification Actions is a controlled evolution and the valves are not expected to be inadvertently moved from the isolated condition. Furthermore, these valves will be verified to be in the correct position when first isolated to comply with ITS 3.4.14 ACTION A. This change is designated as less restrictive because a Surveillance required by the CTS will not be required in the ITS.

L05 *(Category 3 – Relaxation of Completion Time)* CTS Table 3.3-3 Action 13.a states, in part, that with the decay heat isolation valve interlock channel inoperable, both Decay Heat Removal Isolation Valves shall be verified closed. While no specific time is provided, the term "verified closed" implies this is an immediate action. ITS 3.4.14 ACTION C states, in part, that with the Decay Heat Removal (DHR) System interlock function inoperable, isolate the affected penetration by use of two closed deactivated automatic valves within 4 hours. This changes the CTS by allowing 4 hours to complete the Required Action instead of the current immediate time.

The purpose of CTS 3.3-3 Action 13.a is to isolate the DHR isolation valves if the DHR valve interlock is inoperable. This change is acceptable because the

DISCUSSION OF CHANGES
ITS 3.4.14, RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

Completion Time is consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a overpressurization event occurring during the allowed Completion Time. The four hour Completion Time will provide the operator sufficient time to reposition the valves. This change is designated as less restrictive because the Completion Time specified in CTS has been extended in the ITS.

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

CTS

RCS PIV Leakage
3.4.14

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

3.4.6.2.f LCO 3.4.14 Leakage from each RCS PIV shall be within limits.

INSERT 1 (1)

APPLICABILITY: MODES 1, 2, and 3,
MODE 4, except valves in the decay heat removal (DHR) flow path when in, or during the transition to or from, the DHR mode of operation.

ACTIONS

-----NOTES-----

DOC A02 1. Separate Condition entry is allowed for each flow path.

DOC A03 2. Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Actions b and c — A. One or more flow paths with leakage from one or more RCS PIVs not within limit.	-----NOTE----- Each valve used to satisfy Required Action A.1 and Required Action A.2 must have been verified to meet SR 3.4.14.[1] and be on the RCS pressure boundary [or the high pressure portion of the system]. (2.a) (3) (5) (2)	
	A.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve.	4 hours
	AND	

CTS

1 **INSERT 1**

AND

Table 3.3-3 Functional Unit 5.a, 4.5.2.d.1 — The Decay Heat Removal (DHR) System interlock function shall be OPERABLE.

CTS

RCS PIV Leakage
3.4.14

ACTIONS (continued)

CTS	CONDITION	REQUIRED ACTION		COMPLETION TIME	
		A.2	[Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.	72 hours	2
			[or] Restore RCS PIV to within limits.	72 hours]	2
Action b	B. Required Action and associated Completion Time for Condition A not met.	B.1 AND B.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours	
Table 3.3-3 Action 13	C. [Decay Heat Removal (DHR) System ~~autoclosure~~ interlock function inoperable.	C.1	Isolate the affected ~~penetration~~ line by use of ~~one~~ two closed ~~manual or~~ deactivated automatic valves.	4 hours]	2 3
			INSERT 2		3

CTS

3 **INSERT 2**

Table 3.3-3 Action 13	OR		
	C.2	-----------NOTE------------ Only applicable if RCS pressure < 328 psig. -------------------------------	
		Restore the interlock function to OPERABLE status.	Prior to increasing RCS pressure ≥ 328 psig

Attachment 1, Volume 9, Rev. 0, Page 317 of 415

CTS

SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY	
Action c footnote (a), 4.4.6.2.2.d	SR 3.4.14.~~1~~ 2	-------------------------------NOTES------------------------------- ~~1.~~ Only ~~Not~~ required to be performed in MODES ~~3~~ 1 and ~~4~~ 2 ~~2. Not required to be performed on the RCS PIVs located in the DHR flow path when in the DHR mode of operation.~~ ~~3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided.~~ ---	INSERT 3	3 4 8 6 7
4.4.6.2.2.a, 4.4.6.2.2.b, Table 3.4-2		Verify ; ¶ a. leakage from each RCS PIV is equivalent to ≤ ~~0.5~~ 5.0 gpm ~~per nominal inch of valve size up to a maximum of 5 gpm~~ at an RCS pressure ~~≥ [2215] psia~~ of 2155 psig ~~and ≤ [2255] psia.~~ ; and INSERT 4	~~In accordance with the Inservice Testing Program or~~ ~~[18]~~ 24 months AND Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months ~~AND~~ ~~[Within 24 hours following valve actuation due to automatic or manual action or flow through the valve]~~	5 1 5 5 7

CTS

3 **INSERT 3**

Table 4.3-2 Functional Unit 5.a	SR 3.4.14.1	Perform CHANNEL CHECK on the DHR System interlock channel common to Safety Features Actuation System (SFAS) instrumentation.	12 hours

5 **INSERT 4**

Table 3.4-2 Note (a)

b. When current measured rate is > 1gpm, the current measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and 5.0 gpm by 50%.

CTS

RCS PIV Leakage
3.4.14

SURVEILLANCE REQUIREMENTS (continued)

CTS	SURVEILLANCE		FREQUENCY	
Table 4.3-2 Functional Unit 5.a, 4.5.2.d.1.b)	SR 3.4.14.~~2~~ 3	-----NOTE----- [ ] Not required to be met when the DHR System ~~autoclosure~~ interlock function is disabled in accordance with LCO 3.4.12. ----- Verify DHR System ~~autoclosure~~ interlock function prevents the valves from being opened with a simulated or actual RCS pressure signal ≥ [~~425~~] 328 psig.	[~~18~~] 24 months [ ]	3 2
Table 4.3-2 Functional Unit 5.a, 4.5.2.d.1.a)	SR 3.4.14.~~3~~ 4	-----NOTE----- [ ] Not required to be met when the DHR System ~~autoclosure~~ interlock function is disabled in accordance with LCO 3.4.12. ----- Verify DHR System ~~autoclosure~~ interlock function causes the valves to close automatically with a simulated or actual RCS pressure signal ≥ [~~600~~] 328 psig.	[~~18~~] 24 months [ ]	3 2

INSERT 5 3

CTS

3 **INSERT 5**

Table 4.3-2 Functional Unit 5.a	SR 3.4.14.5	Perform CHANNEL CALIBRATION on the DHR System interlock channels.	24 months

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.14, RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

1. The second part of the LCO has been added to ensure consistency between the LCO, ACTIONS, and Surveillance Requirements. The ISTS LCO, ACTIONS, and Surveillances do not match up since there is no explicit statement in the LCO requiring the DHR System interlock function to be OPERABLE. LCO 3.0.1 requires LCOs to be met during the MODES or other specified conditions in the Applicability. LCO 3.0.2 states that upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met. Currently, if the DHR System interlock function is inoperable, the LCO is still met. Thus, ACTION C is not required to be entered since the LCO is still met. Therefore, the inclusion of the second portion of the LCO ensures consistency between the LCO, ACTIONS, and Surveillance Requirements. In addition, due to the addition of the term "DHR" into the LCO statement, the use of the term "decay heat removal (DHR)" in the Applicability has been changed to "DHR."

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. ISTS 3.4.14 has been modified to reflect the Davis-Besse current licensing basis requirements for the DHR System interlock function, with the exception of the Completion Time provided in ISTS 3.4.14 Required Action C.1. The Current Technical Specifications (CTS) requires the DHR System line to be isolated by closing both of the automatic valves in the flow path. This is reflected in ITS 3.4.14 Required Action C.1. The CTS also allows the interlock function to be inoperable with RCS pressure below 328 psig provided the interlock function is restored to OPERABLE status prior to increasing RCS pressure to $\geq$ 328 psig. This is reflected in ITS 3.4.14 Required Action C.2. In addition, both a 12 hour CHANNEL CHECK and a 24 month CHANNEL CALIBRATION are required by the CTS. These Surveillances are reflected in ITS SR 3.4.14.1 and ITS SR 3.4.14.5. Due to the addition of these Surveillances, the remaining Surveillances have been renumbered.

4. Editorial changes have been made to be consistent with the Writers Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01, Section 4.1.7.g.

5. The Davis-Besse RCS PIV leakage limits have been provided, consistent with current licensing basis. In addition, since ITS SR 3.4.14.2 includes two limits, only the first limit (a maximum limit) is applicable for the Required Actions A.1 and A.2 Note.

6. Note 2 to ISTS SR 3.4.14.1 has been deleted since it is not necessary. The ISTS 3.4.14 Applicability does not require leakage to be met for DHR valves in the flow path when in MODE 4 and when in, or during the transition to or from, the DHR mode of operation.

7. The third Frequency of ISTS SR 3.4.14.1 has been deleted since it is not required by the current licensing basis. The first two Frequencies are adequate to ensure the RCS PIV leakage is within the limit. In addition, due to this deletion, Note 3 has also been deleted.

8. Due to the deletion of ISTS SR 3.4.14.1 Notes 2 and 3, the remaining Note has not been numbered and the word "NOTES" has been changed to "NOTE."

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

discuss reactor coolant pressure boundary valves, which are

INSERT 1

10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3), define RCS ~~PIVs~~ as any two normally closed valves in series within the R~~C~~S pressure boundary that separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

reactor coolant

1

The PIV leakage limit applies to each individual valve. Leakage through both series PIVs in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 3.4.13.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leakage rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident that could degrade the ability for low pressure injection.

~~The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 4) that identified potential intersystem LOCAs as a significant contributor to the risk of core melt.~~ 1

~~A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs.~~

PIVs are provided to isolate the RCS from the ~~following typically connected systems:~~ 1

Decay Heat Removal (DHR) System.

1 **INSERT 1**

The 1975 Reactor Safety Study, WASH-1400, (Ref. 4) identified intersystem loss of coolant accidents (LOCAs) as a significant contributor to the risk of core melt. The study considered designs containing two in-series check valves and two check valves in series with a motor operated valve that isolated the high pressure RCS from the low pressure safety injection system. The scenario considered is a failure of the two check valves leading to overpressurization and rupture of the low pressure injection piping which results in a LOCA that bypasses containment. A letter was issued (Ref. 5) by the NRC requiring plants to describe the PIV configuration of the plant. On April 20, 1981, the NRC issued an Order modifying the Davis-Besse Technical Specifications to *include testing requirements on PIVs and to specify the PIVs to be tested* (Ref. 6).

BASES

BACKGROUND (continued)

a. Decay Heat ~~Removal~~ (DHR) System,

b. Emergency Core ~~Cooling~~ System (ECCS), and

c. Makeup and ~~Purification~~ System.

1

CF-30, CF-31, DH-76, and DH-77

The PIVs are listed in [FSAR ~~section~~] Reference 6.

2

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier. INSERT 1A

3

APPLICABLE SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the DHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the reactor coolant pressure boundary (RCPB), and the subsequent pressurization of the DHR System downstream of the PIVs from the RCS. Because ~~the low pressure portion of~~ the DHR System is ~~typically~~ designed ~~for 600 psig,~~ not to handle normal RCS pressures, overpressurization failure of the DHR low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

1

~~Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.~~

1

RCS PIV leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is ~~0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.~~ INSERT 2

3

Attachment 1, Volume 9, Rev. 0, Page 326 of 415

3 **INSERT 1A**

Two motor operated valves (which are not PIVs) are included in series in the suction piping of the DHR System to isolate the high pressure RCS from the low pressure piping of the DHR System when the RCS pressure is above the design pressure of the DHR System piping and components. Ensuring the DHR System interlock function that closes the valves and prevents the valves from being opened is OPERABLE ensures that RCS pressure will not pressurize the DHR System beyond its test pressure.

3 **INSERT 2**

≤ 5.0 gpm. However, when the current measured rate is > 1.0 gpm, the current measured rate shall not exceed the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate (5.0 gpm) by 50%.

BASES

LCO (continued)

Reference 7 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power. 3

INSERT 3

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the DHR flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the DHR mode of operation.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 is added to provide clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

A.1 and A.2

If the leakage from one or more RCS PIVs is not within limit, the

~~The~~ flow path must be isolated by two valves. Required Actions A.1 and A.2 are modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be on the RCS pressure boundary [or the high pressure portion of the system]. 5 2

Required Action A.1 requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced. The 4 hours allows the actions and restricts the operation with leaking isolation valves.

3 **INSERT 3**

Ensuring the DHR System interlock function that closes the valves and prevents the valves from being opened is OPERABLE ensures that RCS pressure will not pressurize the DHR System beyond its test pressure.

BASES

ACTIONS (continued)

[Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation or restoring one leaking PIV.] The 72 hour time after exceeding the limit considers the time required to complete the Action and the low probability of a second valve failing during this time period. 2 2

or The 72 hour time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This timeframe considers the time required to complete this Action and the low probability of a second valve failing during this period.]

2

-------------------------------REVIEWER'S NOTE------------------------------ Two options are provided for Required Action A.2. The second option (72 hour restoration) is appropriate if isolation of a second valve would place the unit in an unanalyzed condition. --

4

B.1 and B.2

If leakage cannot be reduced, [the system isolated,] or other Required Actions accomplished, 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 MODE 3 within 6 hours and to MODE 5 within 36 hours. This Required Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. 2

C.1 and C.2

system function

The inoperability of the DHR ~~autoclosure~~ interlock renders the DHR suction isolation valves incapable of isolating in response to a high pressure condition and preventing inadvertent opening of the valves at RCS pressures in excess of the DHR systems design pressure. If the DHR ~~autoclosure~~ interlock is inoperable, operation may continue as long as the DHR suction ~~penetration~~ line is ~~closed~~ isolated by ~~at least one~~ two closed ~~manual~~ and ~~or~~ deactivated automatic valves within 4 hours. This action accomplishes the purpose of the ~~autoclosure~~ interlock function. 3 5

INSERT 4

3 **INSERT 4**

Alternately, if the RCS pressure is < 328 psig, isolating the associated DHR penetration is not required. In this case, the DHR System interlock function must be restored to OPERABLE status prior to increasing RCS pressure ≥ 328 psig. Since RCS pressure is below the setpoint, there is no need to isolate the associated penetration.

Attachment 1, Volume 9, Rev. 0, Page 331 of 415

BASES

SURVEILLANCE REQUIREMENTS

INSERT 5

SR 3.4.14.[1] 2

3

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 or A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

INSERT 6

INSERT 7

3

1

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

24

Testing is to be performed every [18] months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The [18 month] Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code (Ref. 7), and is based on the need to perform such surveillances under conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the plant at power.

24 2

[In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.]

3

performed

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

5

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complimentary to the Frequency of prior to entry into MODE 2 whenever the unit has been

3

INSERT 5

SR 3.4.14.1

SR 3.4.14.1 is the performance of the CHANNEL CHECK of the decay heat isolation valve interlock channel that is common to the Safety Features Actuation System (SFAS) instrumentation. The check provides reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

3

INSERT 6

The RCS PIV leakage limit is ≤ 5.0 gpm. However, RCS PIV leakage is also limited when the current measured rate is > 1.0 gpm, such that the current measured rate shall not exceed the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and 5.0 gpm by 50%.

1

INSERT 7

Valves CF-30 and CF-31 will be tested with the RCS pressure > 1200 psig and valves DH-76 and DH-77 will be tested at > 575 psig (i.e., the normal core flooding tank pressure). Minimum differential test pressure across each valve shall be > 150 psid. Additionally, 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. In addition, this Surveillance is not required to be performed on the DHR System when the DHR System is aligned to the RCS in the decay heat removal mode of operation. PIVs contained in the DHR flow path must be leakage rate tested after DHR is secured and stable unit conditions and the necessary differential pressures are established. 3

-----REVIEWER'S NOTE-----
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 earlier 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. Based upon the information available to the Staff at the time, the content of those Orders was considered acceptable. Since 1980, the NRC Staff has determined an additional PIV leakage rate determination is required within 24 hours following actuation of the valve and flow through the valve. This is necessary in order to ensure the PIV's ability to support the integrity of the reactor coolant pressure boundary. The Revised Standard Technical Specifications include the "24 hours..." Frequency to reflect current NRC Staff position on the need to include this test requirement within Technical Specifications. 4

-----

3 4

[SR 3.4.14.2 and SR 3.4.14.3 2 3

430 psig, the pressure at which this section of DHR piping was tested

at the RCS pressure instrumentation tap

allows DH-11 and DH-12 to be opened by the operator prior to the point where net positive suction pressure is lost to the reactor coolant pumps

328

24

24

Verifying that the DHR ~~autoclosure~~ interlocks are OPERABLE ensures that RCS pressure will not pressurize the DHR system beyond ~~125% of its design pressure of [600] psig~~. The interlock setpoint that prevents the valves from being opened is set so the actual RCS pressure must be < [425] psig to open the valves. This setpoint ~~ensures the DHR design pressure will not be exceeded and the DHR relief valves will not lift~~. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance was 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. 3 1 2 1 3 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

System interlock

These SRs are modified by Notes allowing the DHR ~~autoclosure~~ function to be disabled when using the DHR System suction relief valve for cold overpressure protection in accordance with LCO 3.4.12. [] INSERT 8 3 2 1 3

INSERT 9

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 55a(c).
3. 10 CFR 50, Appendix A, Section V, GDC 55.
4. NUREG-75/014, Appendix V, October 1975.
5. ~~NUREG-0677, NRC, May 1980.~~ 1
6. ~~[Document containing list of PIVs.]~~ 1
7. ASME Code for Operation and Maintenance of Nuclear Power Plants[] , 1995 Edition with 1996 Addenda. 1
8. 10 CFR 50.55a(g). 1

5. Letter from D.G. Eisenhut, NRC, to all LWR Licenses, LWR Primary Coolant System Pressure Isolation Valves, February 23, 1980.

6. Letter from J.F. Stoltz, NRC, to R.P. Crouse, Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves, April 20, 1981. 1

1 **INSERT 8**

This allowance is necessary since opening and removing control power to the DHR System isolation valves (as required by LCO 3.4.12) disables the interlock.

3 **INSERT 9**

SR 3.4.14.5

SR 3.4.14.5 requires the performance of a CHANNEL CALIBRATION of the DHR System interlock channels (both the channel common to the SFAS instrumentation and the channel not common to the SFAS instrumentation). The calibration verifies the accuracy of the instrument string. The Frequency of 24 months is a typical refueling cycle and considers channel reliability. Operating experience has proven this Frequency is acceptable.

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.14 BASES, RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

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

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. Changes are made to reflect changes made to the Specification.

4. The Reviewer's Note is deleted because it is not intended to be included in the plant specific ITS submittal.

5. Changes made to be consistent with the Specification.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.14, RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

There are no specific NSHC discussions for this Specification.

ATTACHMENT 15

ITS 3.4.15, RCS LEAKAGE DETECTION INSTRUMENTATION

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

ITS

A01

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

LCO 3.4.15 3.4.6.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. The containment sump ~~level and flow~~ monitoring system, and LA01

b. One containment atmosphere radioactivity monitor (gaseous or particulate).

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Add proposed Required Action A.1 Note L01

ACTION A a. With the required containment sump ~~level and flow~~ monitoring system inoperable, operation may continue up to 30 days provided Surveillance Requirement 4.4.6.2.1.d is performed at least once per 24 hours. LA01

ACTION B b. With the required containment atmosphere radioactivity monitor inoperable, operation may continue up to 30 days provided:

1. Containment atmosphere grab samples are obtained and analyzed at least once per 24 hours, or
2. Surveillance Requirement 4.4.6.2.1.d is performed at least once per 24 hours.

Add proposed Required Action B.1.2 Note L01

ACTION C c. With the above required ACTION and associated completion time not met, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION D d. With the required containment atmosphere radioactivity monitor and the containment sump level and flow monitoring system inoperable, enter TS 3.0.3 immediately.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

SR 3.4.15.1, SR 3.4.15.2, SR 3.4.15.3 a. Containment atmosphere particulate monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3.

ITS

A01

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.4.15.4 b. Containment sump level and flow monitoring system-performance of CHANNEL CALIBRATION at least once each REFUELING INTERVAL. LA01

SR 3.4.15.1, SR 3.4.15.2, SR 3.4.15.3 c. Containment atmosphere gaseous monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3.

ITS

A01

ITS 3.4.15

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION	
	1. AREA MONITORS						
	a. Fuel Storage Pool Area Emergency Ventilation System Actuation	1	**	≤ 2 × background	0.1 - 10^7 mr/hr	22	See ITS 3.3.14
	2. PROCESS MONITORS						
	a. Containment						
LCO 3.4.15.b	i. Gaseous Activity RCS Leakage Detection	1*	1, 2, 3, & 4	Not Applicable	~~10 - 10^6 cpm~~	21 B	LA02
LCO 3.4.15.b	ii. Particulate Activity RCS Leakage Detection	1*	1, 2, 3, & 4	Not Applicable	~~10 - 10^6 cpm~~	21 B	

* As required by Specification 3.4.6.1.
**With fuel in the storage pool or building — See ITS 3.3.14

Attachment 1, Volume 9, Rev. 0, Page 344 of 415

A01

ITS

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION B **ACTION 21** - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.

See ITS 3.3.14

ITS

A01

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	SR 3.4.15.1 CHANNEL CHECK	SR 3.4.15.3 CHANNEL CALIBRATION	SR 3.4.15.2 CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Fuel Storage Pool Area Emergency Ventilation System Actuation	S	E	M	**
2. PROCESS MONITORS				
a. Containment				
i. Gaseous Activity RCS Leakage Detection*	S -1	E -3	M -2	1, 2, 3 & 4
ii. Particulate Activity RCS Leakage Detection*	S -1	E -3	M -2	1, 2, 3 & 4

See ITS 3.3.14

LCO 3.4.15 * If required by Specification 3.4.6.1 to be OPERABLE.

**With fuel in the storage pool or building

See ITS 3.3.14

Attachment 1, Volume 9, Rev. 0, Page 346 of 415

DISCUSSION OF CHANGES
ITS 3.4.15, RCS LEAKAGE DETECTION INSTRUMENTATION

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.4.6.1.a states that the containment sump monitoring system includes both "level and flow." In addition, CTS 3.4.6.1 Action a and CTS 4.4.6.1.b both include "level and flow" when referring to the containment sump monitoring system. ITS 3.4.15 requires the containment sump monitor to be OPERABLE, but the details of what constitutes an OPERABLE monitor are moved to the Bases. This changes the CTS by moving the details of what constitutes an OPERABLE containment sump monitor to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the CTS.

LA02 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Table 3.3-6 provides the measurement range for the gaseous and particulate containment atmosphere radioactivity monitors. ITS 3.4.15 requires either the gaseous or particulate containment atmosphere radioactivity monitor to be OPERABLE, but the details concerning their measurement range are not included. This changes the CTS by moving the

DISCUSSION OF CHANGES
ITS 3.4.15, RCS LEAKAGE DETECTION INSTRUMENTATION

details of the measurement ranges for the gaseous and particulate containment atmosphere radioactivity monitors to the UFSAR, where it currently exists.

The removal of these details, which are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. Changes to the UFSAR are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the CTS.

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.4.6.1 Actions a and b.2 do not include an exclusion allowing a delay in performing an RCS water inventory balance. ITS 3.4.15 Required Action A.1 and Required Action B.1.2 include a Note that states "Not required until 12 hours after establishment of steady state operation." This changes the CTS by allowing 12 hours after establishment of steady state operation before the RCS water inventory balance must be performed.

The purpose of CTS 3.4.6.1 Actions a and b.2 to perform an RCS water inventory balance is to provide another means of leakage detection. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity of remaining features, a reasonable time for repairs or replacement of required feature, and the low probability of a DBA occurring during the repair period. The RCS water inventory balance is still performed, but the delay in performing it allows unit conditions to provide an accurate indication. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

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

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

3.4.6.1, Table 3.3-6 Instruments 2.a and 2.b

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

a. One containment sump monitor ; and 1

b. One containment atmosphere radioactivity monitor (gaseous or particulate).

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

ACTIONS

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a	A. Required containment sump monitor inoperable.	A.1 -------------NOTE------------- Not required until 12 hours after establishment of steady state operation. ------------------------------------	
		Perform SR 3.4.13.1.	Once per 24 hours
		AND	
		A.2 Restore required containment sump monitor to OPERABLE status.	30 days
Action b	B. Required containment atmosphere radioactivity monitor inoperable.	B.1.1 Analyze grab samples of the containment atmosphere.	Once per 24 hours
		OR	

CTS

ACTIONS (continued)

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME	
		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	
		AND		
		B.2 Restore required containment atmosphere radioactivity monitor to OPERABLE status.	30 days	
Action c	C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3.	6 hours	
		AND		
		C.2 Be in MODE 5.	36 hours	2
Action d	D. Both required monitors inoperable.	D.1 Enter LCO 3.0.3.	Immediately	

SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY	
4.4.6.1.a, 4.4.6.1.c	SR 3.4.15.1	Perform CHANNEL CHECK of required containment atmosphere radioactivity monitor.	12 hours	
4.4.6.1.a, 4.4.6.1.c	SR 3.4.15.2	Perform CHANNEL FUNCTIONAL TEST of required containment atmosphere radioactivity monitor.	~~92~~ 31 days	3

SURVEILLANCE REQUIREMENTS (continued)

CTS	SURVEILLANCE		FREQUENCY	
4.4.6.1.b	SR 3.4.15.~~3~~ 4	Perform CHANNEL CALIBRATION of required containment sump monitor.	[18] 24 months	4
4.4.6.1.a, 4.4.6.1.c, Table 4.3-3 Instruments 2.a.i and 2.a.ii	SR 3.4.15.~~4~~ 3	Perform CHANNEL CALIBRATION of required containment atmosphere radioactivity monitor.	[18] months	4

Attachment 1, Volume 9, Rev. 0, Page 352 of 415

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.15, RCS LEAKAGE DETECTION INSTRUMENTATION

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

2. The specific Conditions the ACTION applies to have been added, since there is one ACTION it does not apply to (ACTION D). This is consistent with the Writers Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01, Section 4.1.6.i.5.ii.

3. The CHANNEL FUNCTIONAL TEST Frequency has been changed to be consistent with the Davis-Besse current licensing basis.

4. The brackets have been removed and the proper plant specific information/value is provided. Also, the Surveillances have been put in the correct order based on the Frequency.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

Although not committed to

GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems. , it 2

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can readily be detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment sump used to collect unidentified LEAKAGE is instrumented to ~~alarm for~~ allow detecting increases of 0.5 to 1.0 gpm in the normal flow rates. This sensitivity is acceptable for detecting increases in unidentified LEAKAGE. 2

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivities of 10^{-9} µCi/cc radioactivity for particulate monitoring and of 10^{-6} µCi/cc radioactivity for gaseous monitoring are practical for these leakage detection systems. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS LEAKAGE.

~~An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an indicator of potential RCS LEAKAGE. A 1°F increase in dew point is well within the sensitivity range of available instruments.~~ 2

BASES

BACKGROUND (continued)

~~Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment sump [and condensate flow from air coolers]. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required for this LCO.~~ 2

Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS LEAKAGE into the containment. The relevance of temperature and pressure measurements are affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.

APPLICABLE SAFETY ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. ~~The system response times and sensitivities are described in the FSAR (Ref. 3).~~ Multiple instrument locations are utilized, if needed, to ensure the transport delay time of the leakage from its source to an instrument location yields an acceptable overall response time. 2

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. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leak occur detrimental to the safety of the unit and the public.

Refer to the Bases of LCO 3.4.13, "RCS Operational LEAKAGE," for further information regarding RCS LEAKAGE. 2

RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

One method of protecting against large RCS LEAKAGE derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition when RCS LEAKAGE indicates possible RCPB degradation.

The LCO requirements are satisfied when monitors of diverse measurement means are available. Thus, the containment sump monitor (both the level and flow portions) in combination with a particulate or gaseous radioactivity monitor, provides an acceptable minimum. 2

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature is ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation is much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

A.1 and A.2

With the required containment sump monitor (i.e., either level or flow or both) inoperable, no other form of sampling can provide the equivalent information. 2

However, the containment atmosphere ~~activity~~ radioactivity monitor will provide indications of changes in leakage. Together with the atmosphere monitor, the periodic surveillance for RCS inventory balance, SR 3.4.13.1, water inventory balance, must be performed at an increased frequency of 24 hours to provide 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 establishing steady state operation (stable temperature, power level, and pressurizer ~~and makeup tank~~ level~~s, 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. 3 2 1 3

Restoration of the required containment sump monitor to OPERABLE status is required to regain the function in a Completion Time of 30 days after the monitor's failure. This time is acceptable considering the frequency and adequacy of the RCS water inventory balance required by Required Action A.1.

BASES

ACTIONS (continued)

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.

The 24 hour interval 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 establishing steady state operation (stable temperature, power level, and 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. 2 1

C.1 and C.2

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 experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

With both required monitors inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

BASES

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 monitor. ~~A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.~~ The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of ~~[92]~~ 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation. 2 4

SR 3.4.15.3 and SR 3.4.15.4

These SRs require the performance of a CHANNEL CALIBRATION for each of the required RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of [18] months or 24 months, as applicable, ~~is a typical refueling cycle and~~ considers channel reliability ~~. Again,~~ and operating experience has proven this Frequency is acceptable. 4 1 4

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.
2. Regulatory Guide 1.45.
3. ~~FSAR, Section [ ].~~ 1

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.15 BASES, RCS LEAKAGE DETECTION INSTRUMENTATION

1. The brackets have been removed and the proper plant specific information/value has been provided.
2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.
3. Changes made to be consistent with the Specification.
4. Changes made to be consistent with changes made to the Specification.

Attachment 1, Volume 9, Rev. 0, Page 360 of 415

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.15, RCS LEAKAGE DETECTION INSTRUMENTATION

There are no specific NSHC discussions for this Specification.

ATTACHMENT 16

ITS 3.4.16, RCS SPECIFIC ACTIVITY

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

ITS 3.4.16

ITS

A01

REACTOR COOLANT SYSTEM

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

LCO 3.4.16 — 3.4.8 The specific activity of the primary coolant shall be limited to:

SR 3.4.16.2 — a. ≤ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

SR 3.4.16.1 — b. ≤ 100/Ē μCi/gram

APPLICABILITY: MODES 1, 2, 3, ~~4 and 5~~. L01

ACTION:

MODES 1, 2 and 3*. Add proposed ACTION A Note L02

ACTION A / ACTION B — a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} < 530°F within 6 hours.

ACTION B — b. With the specific activity of the primary coolant > 100/Ē μCi/gram, be in at least HOT STANDBY with T_{avg} < 530°F within 6 hours.

MODES 1, 2, 3, ~~4 and 5~~: L01

ACTION A — a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 ~~or > 100/Ē μCi/gram~~, perform the sampling and analysis requirements of item 4 a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits. For reporting requirements refer to Section 6.9.1.5.c, Annual Operating Report. L03 A02 A03

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1, SR 3.4.16.2, SR 3.4.16.3 — 4.4.8 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

Applicability — *With T_{avg} ≥ 530°F.

ITS

A01

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE

AND ANALYSIS PROGRAM

ITS	TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED	
SR 3.4.16.1	1. Gross Activity Determination	At least once each ~~72 hours~~ 7 days	1, 2, 3, ~~4~~	L04, L01
SR 3.4.16.2	2. ~~Isotopic Analysis~~ for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1	LA01
SR 3.4.16.3	3. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1	A02
Required Action A.1	4. ~~Isotopic Analysis for Iodine Including I-131, I-133, and I-135~~	a) Once per 4 hours, whenever the specific activity exceeds 1.0 μCi/gram DOSE EQUIVALENT I-131 ~~or 100/$\bar{E}$ μCi/gram~~, and	1#, 2#, 3#, ~~4#, 5#~~	L01, L03
SR 3.4.16.2		b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3	L01

~~#Until the specific activity of the primary coolant system is restored within its limits.~~ A02

SR 3.4.16.3 Note — *Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since the reactor was last subcritical for 48 hours or longer. L05

DAVIS-BESSE, UNIT 1 3/4 4-22

ITS 3.4.16

A01

ITS

Figure 3.4.16-1

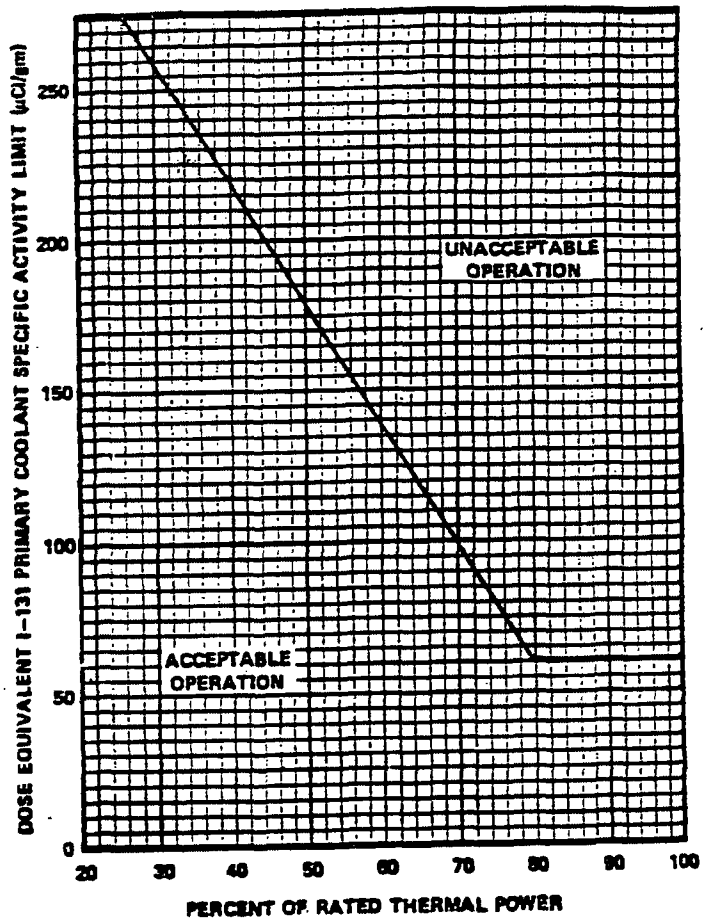


FIGURE 3.4-1

DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0 µCi/gram DOSE EQUIVALENT I-131

DAVIS-BESSE, UNIT 1 3/4 4-23 Amendment No. 135

Page 3 of 3

DISCUSSION OF CHANGES
ITS 3.4.16, RCS SPECIFIC ACTIVITY

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.4.8 Action a (MODES 1, 2, 3, 4, and 5) and CTS Table 4.4-4, Footnote #, require the isotopic analysis for iodine to be performed until the specific activity of the primary coolant system is restored to within limits. ITS 3.4.16 Required Action A.1 requires this same analysis, however the explicit statement to perform the isotopic analysis for iodine until the limits are met has been deleted. This changes the CTS by deleting the explicit statement to perform the isotopic analysis for iodine until the limits are met.

The purpose of the CTS 3.4.8 Action a (MODES 1, 2, 3, 4, and 5) and CTS Table 4.4-4 is to ensure the Surveillance is performed to determine whether the specific activity is met. This statement is not necessary in the ITS, because ITS LCO 3.0.2 requires the Required Actions of the associated Conditions to be met upon discovery of failure to meet an LCO. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated. This change is acceptable since ITS LCO 3.0.4 will require the Required Action to be performed until the LCO is met. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS 3.4.8 Action a (MODES 1, 2, 3, 4, and 5) provides a cross-reference to CTS 6.9.1.5.c, the Annual Operating Report. ITS 3.4.16 does not contain this cross-reference. This changes the CTS by deleting a cross-reference to another CTS requirement.

The purpose of the reference is to alert the user that a report may need to be generated due to the specific activity being outside the limit. However, CTS 6.9.1.5.c has not been included in the Davis-Besse ITS. Therefore, the cross-reference is not needed. Furthermore, it is an ITS convention to not include these types of cross-references. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

DISCUSSION OF CHANGES
ITS 3.4.16, RCS SPECIFIC ACTIVITY

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS Table 4.4-4 Item 2 requires an isotopic analysis to determine whether DOSE EQUIVALENT I-131 concentration is within limit. CTS Table 4.4-4 Item 4 requires an isotopic analysis for iodine including I-131, I-133, and I-135. ITS SR 3.4.16.2 requires the verification that reactor coolant DOSE EQUIVALENT I-131 specific activity is within limit. ITS 3.4.16 Required Action A.1 requires the verification that DOSE EQUIVALENT I-131 is within the acceptable region of Figure 3.4.16-1. This changes the CTS by moving the detail that an "Isotopic Analysis" or "Isotopic Analysis for Iodine Including I-131, I-133, and I-135" must be performed to satisfy the requirements of the Surveillances to the Bases.

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS SR 3.4.16.2 and ITS 3.4.16 Required Action A.1 still retain the requirements to verify reactor coolant DOSE EQUIVALENT I-131 is within limit. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 2 – Relaxation of Applicability)* CTS 3.4.8 is applicable in MODES 1, 2, 3, 4, and 5. In addition, the testing for gross activity determination in CTS Table 4.4-4 Item 1 is required in MODES 1, 2, 3, and 4, and the isotopic analysis for iodine requirement in CTS Table 4.4-4 Item 4.a and 4.b is required periodically in MODES 1, 2, 3, 4, and 5 and after a 15% RTP change in MODES 1, 2, and 3, respectively. ITS 3.4.16, including the Surveillances, is applicable in MODES 1 and 2, and MODE 3 with RCS $T_{avg} \geq 530°F$. This changes the CTS by reducing the MODES in which the LCO is applicable, including the Surveillances, to only MODES 1 and 2, and MODES 3 with RCS $T_{avg} \geq 530°F$.

The purpose of CTS 3.4.8 is to ensure that the specific activity of the RCS is within the assumptions of the Steam Generator Tube Rupture (SGTR) analysis. This change is acceptable because the requirements continue to ensure that the process variables are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. During operation in MODE 3 with RCS $T_{avg} < 530°F$, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely because the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves. Furthermore, the CTS Actions for when the limits are not met only

DISCUSSION OF CHANGES
ITS 3.4.16, RCS SPECIFIC ACTIVITY

require the unit to be shutdown to MODE 3 with RCS T_{avg} < 530°F. This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

L02 *(Category 9 - Addition of LCO 3.0.4 Exception)* CTS 3.4.8 does not allow the unit to change MODES when the RCS specific activity is not within limits. ITS 3.4.16 ACTION A Note specifies that LCO 3.0.4.c is applicable. This changes the CTS by allowing the unit to change MODES or other specified conditions in the Applicability when the specific activity for DOSE EQUIVALENT I-131 is > 1.0 μCi/gm.

The purpose of CTS 3/4.4.8 is to ensure appropriate limitations are placed on reactor coolant activity. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering that the DOSE EQUIVALENT I-131 is still within the limits of ITS Figure 3.4.16-1. This includes the low probability of a DBA occurring during the restoration time period. This change allows the unit to change MODES or other specified conditions in the Applicability when the specific activity for DOSE EQUIVALENT I-131 is > 1.0 μCi/gm. However, after entering the Applicability the unit must enter ACTION A and verify DOSE EQUIVALENT I-131 is within the acceptable region of Figure 3.4.16-1 every 4 hours. This verification will ensure that a steam generator tube rupture will not lead to a site boundary dose that exceeds the 10 CFR 100 dose guideline limits. Therefore, this change is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the unit remains at, or proceeds to power operation. In addition, ITS 3.4.16 ACTION A requires DOSE EQUIVALENT I-131 to be within limit in 48 hours. This change is designated as less restrictive because the Required Action Note allows entry into the MODE of Applicability when the specific activity for DOSE EQUIVALENT I-131 is > 1.0 μCi/gm.

L03 *(Category 4 – Relaxation of Required Action)* CTS 3.4.8 Action a (MODES 1, 2, 3, 4, and 5) and CTS Table 4.4-4 Item 4.a require isotopic analysis for iodine once per 4 hours when the specific activity exceeds $100/\bar{E}$ μCi/gm. The ITS does not contain this Action. This changes the CTS by eliminating a conditionally performed Surveillance when gross activity exceeds $100/\bar{E}$ μCi/gm.

The purpose of CTS 3.4.8 Action a (MODES 1, 2, 3, 4, and 5) and CTS Table 4.4-4 Item 4.a is to monitor iodine activity when the specific activity limits are exceeded. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering that DOSE EQUIVALENT I-131 is still being monitored and the low probability of a DBA occurring during the restoration time period. When specific

DISCUSSION OF CHANGES
ITS 3.4.16, RCS SPECIFIC ACTIVITY

activity exceeds $100/\bar{E}$ μCi/gm, ITS 3.4.16 Required Action B.1 and CTS 3.4.8 Action b (MODES 1, 2, and 3*) require the plant to be in MODE 3 with T_{avg} < 530°F within 6 hours. Monitoring of $\bar{E}$ is required in order to determine if the LCO is met and the ACTION can be exited. Furthermore, if the Condition is entered and the unit is in MODE 2 in 4 hours or less, the Required Action is in conflict with the Note of ITS SR 3.4.16.2, which states that this SR is only required in MODE 1. Finally, this action is an unnecessary burden as the unit is required to be in MODE 3 with T_{avg} < 530°F within 6 hours, exiting the Applicability. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L04 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS Table 4.4-4 Item 1 requires gross activity to be determined at least once per 72 hours. ITS SR 3.4.16.1 requires verification that the reactor coolant gross specific activity is $\leq 100/\bar{E}$ μCi/gm every 7 days. This changes the CTS by reducing the Frequency from at least once per 72 hours to 7 days.

The purpose of CTS Table 4.4-4 Item 1 is to obtain a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, which provides an indication of increases in gross specific activity. This change is acceptable because the new Surveillance Frequency ensures that it provides an acceptable level of monitoring. A Frequency of 7 days provides sufficient information to trend the results in order to detect gross fuel failure, while considering the low probability of a gross fuel failure between performances. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L05 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS Table 4.4-4 Item 3 requires radiochemical determination of $\bar{E}$ once per 6 months. Footnote * states that the sample is to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since the reactor was last subcritical for 48 hours or longer. ITS SR 3.4.16.3 requires $\bar{E}$ to be determined from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. ITS SR 3.4.16.3 is further modified by a Note which states, "Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours." This changes the CTS by putting a limit, 31 days, on when the Surveillance must be performed after the requisite conditions are met.

The purpose of CTS Table 4.4-4 Item 3 is to determine the value of $\bar{E}$ when the isotopic concentrations in the core are stable. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of monitoring. Circumstances could arise in which the 6 month Frequency for performance of the SR has passed but the operating conditions for performance of the test have not been met. In this circumstance, the Surveillance would be immediately past due as soon as the operating conditions are met. The ITS SR 3.4.16.3 Note allows 31 days to perform the

DISCUSSION OF CHANGES
ITS 3.4.16, RCS SPECIFIC ACTIVITY

Surveillance after the operating conditions are met. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

Attachment 1, Volume 9, Rev. 0, Page 372 of 415

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

CTS

RCS Specific Activity
3.4.16

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.8

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

530

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ $\boxed{500}$°F. 1

ACTIONS

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a (MODES 1, 2, and 3* and MODES 1, 2, 3, 4, and 5), Table 4.4-4 Item 4.a	A. DOSE EQUIVALENT I-131 > 1.0 µCi/gm.	-------------------NOTE------------------- LCO 3.0.4.c is applicable. ---	
		A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1.	Once per 4 hours
		AND	
		A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
Action a (MODES 1, 2, and 3*)	B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3 with T_{avg} < $\boxed{500}$°F. 530	6 hours 1
	OR		
Action a (MODES 1, 2, 3, 4, and 5)	DOSE EQUIVALENT I-131 in unacceptable region of Figure 3.4.16-1.		

Action b (MODES 1, 2, and 3*)

OR

Gross specific activity of the reactor coolant not within limit. 2

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Gross specific activity of the coolant not within limit.	C.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

2

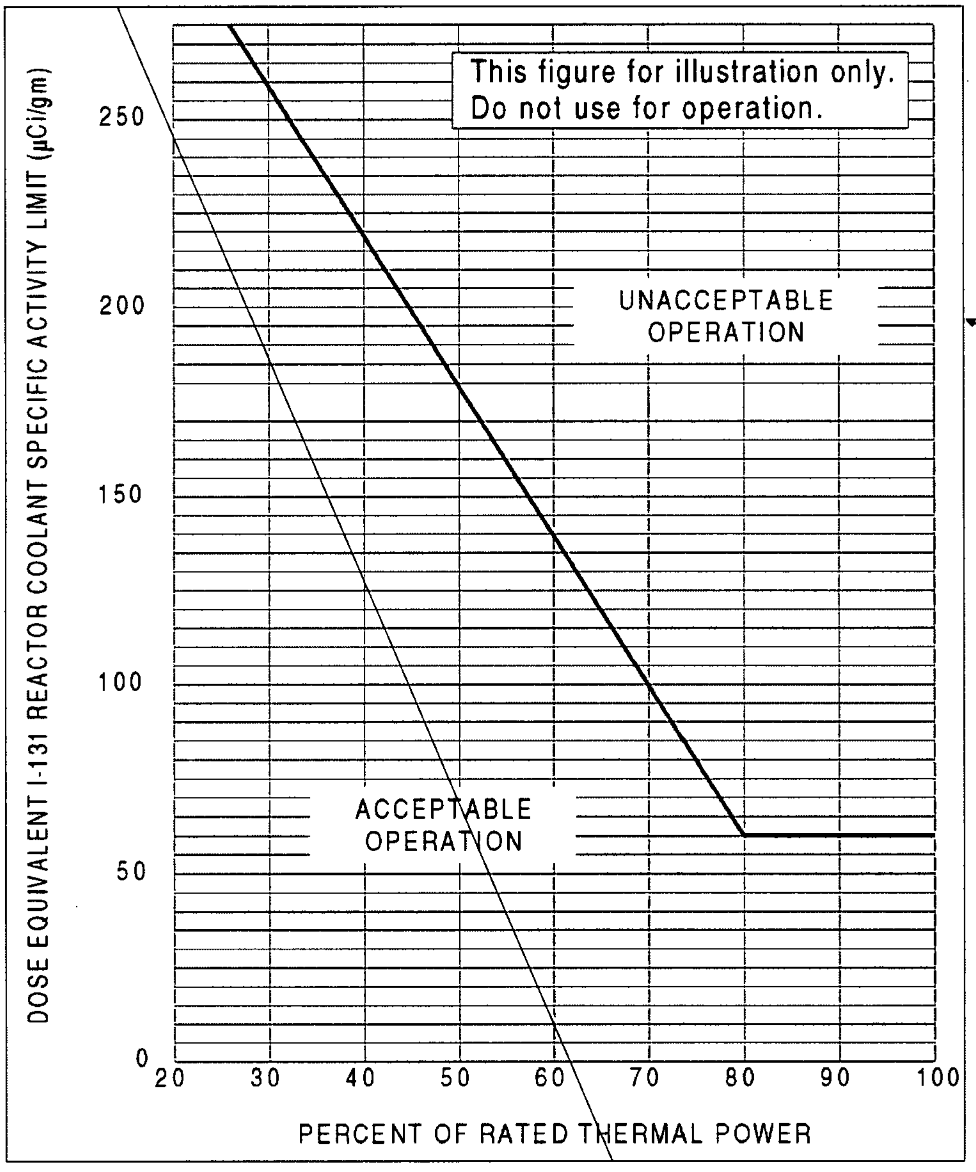
SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY
LCO 3.4.8.b, Table 4.4-4 Item 1	SR 3.4.16.1	Verify reactor coolant gross specific activity ≤ 100/Ē μCi/gm.	7 days
LCO 3.4.8.a, Table 4.4-4 Item 2 and Item 4.b	SR 3.4.16.2	------------------------------NOTE------------------------------ Only required to be performed in MODE 1. -- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	14 days AND Between 2 and 6 hours after THERMAL POWER change of ≥ 15% RTP within a 1 hour period
Table 4.4-4 Item 3	SR 3.4.16.3	------------------------------NOTE------------------------------ Not required to be performed until 31 days after a minimum of 2 EFPD and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. -- Determine Ē.	184 days

CTS

RCS Specific Activity
3.4.16

Figure 3.4-1



INSERT 1

3

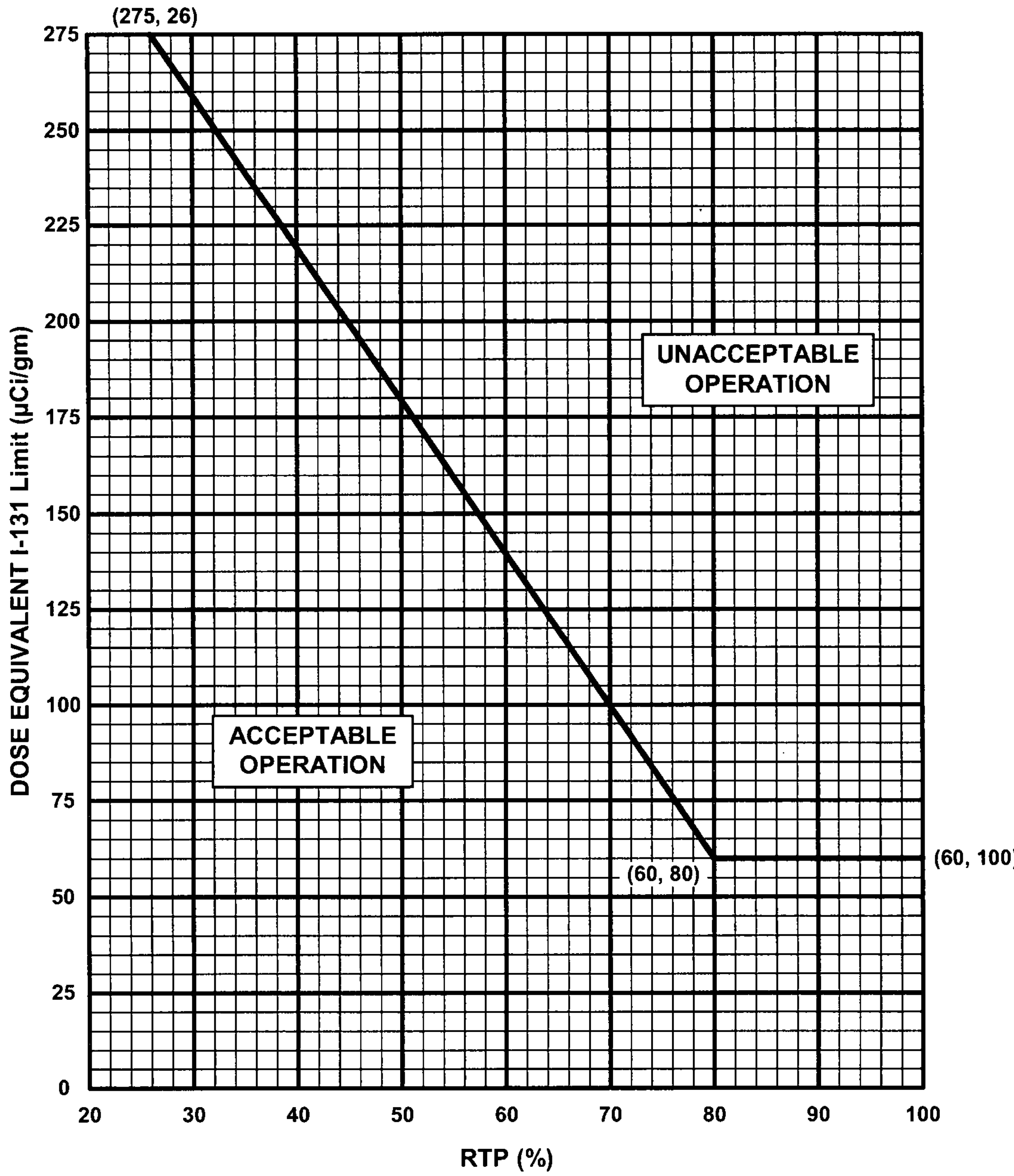
Figure 3.4.16-1 (page 1 of 1)
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity Limit Versus Percent of RATED THERMAL POWER With Reactor Coolant Specific Activity >1.0 μCi/gm DOSE EQUIVALENT I-131

3.4.16

CTS

3 INSERT 1

Figure 3.4-1



JUSTIFICATION FOR DEVIATIONS
ITS 3.4.16, RCS SPECIFIC ACTIVITY

1. The MODE 3 Applicability for this Specification has been changed from 500°F to 530°F, consistent with current licensing basis. The Davis-Besse temperature limit is 530°F, since at this temperature the saturation pressure of the primary coolant is below the lift pressure of the main steam safety valves.

2. ISTS 3.4.16 ACTION C has been deleted and incorporated in ISTS 3.4.16 ACTION B because the Required Actions are identical (be in MODE 3 with T_{avg} < 500°F). In NUREG-1430, Rev. 1, ISTS 3.4.16 ACTION C contained an additional Required Action. This Required Action was deleted in NUREG-1430, Rev. 2, as a result of approved TSTF-28. The entire ACTION C should have been deleted as a result of the application of TSTF-28, but was not. This changes the ISTS to be consistent with other Specifications where ACTION Conditions are combined when the same Required Actions apply.

3. The Davis-Besse reactor coolant DOSE EQUIVALENT I-131 specific power limit verses percent of RATED THERMAL POWER curve is substituted for the curve provided for illustration in the ISTS.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 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 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 a small fraction of 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.

The parametric evaluations showed the potential offsite dose levels for an SGTR accident were an appropriately small fraction of 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.

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 SGTR accident. The SGTR safety analysis (Ref. 2) assumes ~~the~~ a specific activity ~~of the reactor coolant at the LCO limits~~ value equivalent to 1% failed fuel and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm. The analysis also assumes a reactor trip and a turbine trip ~~at the same time as the~~ following a SGTR ~~event~~.

The analysis for the SGTR accident establishes 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. The assumed RCS specific activity in the SGTR analysis bounds the LCO limit for RCS specific activity.

3

BASES

APPLICABLE SAFETY ANALYSES (continued)

The rise in pressure in the ruptured SG causes radioactively contaminated steam to discharge to the atmosphere through the ~~atmospheric dump~~ valves or the main steam safety valves. The atmospheric discharge stops when the turbine bypass to the condenser removes the excess energy to rapidly reduce the RCS pressure and close the valves. The unaffected SG removes core decay heat by venting steam until the cooldown ends. 3

The safety analysis shows the radiological consequences of an SGTR accident are within a small fraction of the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the ~~applicable~~ Specification, for more than 48 hours. 5

The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of an SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.

RCS Specific Activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the primary coolant is limited to the number of μCi/gm equal to 100 divided by Ē (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the Design Basis Accident (DBA) will be a small fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.

The SGTR accident analysis (Ref. 2) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO such that the RCS specific activity is greater than the analysis assumptions, may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits. 3

BASES

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ ~~500~~ 530 °F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values. 1

For operation in MODE 3 with RCS average temperature < ~~500~~ 530 °F, and in MODES 4 and 5, the release of radioactivity in the event of an SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the ~~atmospheric dump valves and~~ main steam safety valves. 1 3

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to ~~demonstrate~~ verify the limits of Figure 3.4.16-1 are not exceeded. An isotopic analysis of a reactor coolant sample must be performed for at least I-131, I-133, and I-135. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling must continue for trending. 2 3

The DOSE EQUIVALENT I-131 must be restored to limits within 48 hours. The Completion Time of 48 hours is required, if the limit violation resulted from normal iodine spiking.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1

If a Required Action and associated Completion Time of Condition A are not met ~~or~~ , if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.16-1, or if the gross specific activity is not within limit, the reactor must be brought to MODE 3 with RCS average temperature < ~~500~~ 530 °F within 6 hours. The Completion Time of 6 hours is required to get to MODE 3 below ~~500~~ 530 °F without challenging reactor emergency systems. 1

BASES

ACTIONS (continued)

C.1

~~With the gross specific activity in excess of the allowed limit, the unit must be placed in a MODE in which the requirement does not apply.~~

~~The allowed Completion Time of 6 hours to reach MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves, and prevents venting the SG to the environment in an SGTR event. The Completion Time of 6 hours is required to reach MODE 3 from full power conditions in an orderly manner and without challenging reactor emergency systems.~~ 1

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once per 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with RCS average temperature at least ~~500~~ 530°F. The 7 day Frequency considers the unlikelihood of a gross fuel failure during that time period. 1

SR 3.4.16.2

This Surveillance requires the verification that the reactor coolant DOSE EQUIVALENT I-131 specific activity is within limit. This Surveillance is accomplished by performing an isotopic analysis of a reactor coolant sample.

This Surveillance is performed in MODE 1 only to ensure the iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level considering gross specific activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change of ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results. 2 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.3

SR 3.4.16.3 requires radiochemical analysis for Ē determination every 184 days ~~(6 months)~~ with the plant operating in MODE 1 equilibrium conditions. The Ē determination directly relates to the LCO and is required to verify plant operation within the specific gross activity LCO limit. The analysis for Ē is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Frequency of 184 days recognizes Ē does not change rapidly. 2

states is not required

This SR has been modified by a Note that ~~requires~~ sampling to be performed 31 days after a minimum of 2 EFPD and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures the radioactive materials are at equilibrium so the analysis for Ē is representative and not skewed by a crud burst or other similar abnormal event. 2

until

REFERENCES

1. 10 CFR 100.11.
2. U FSAR, Section ~~[15.6.3]~~ 15.4.2. 3 4

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.16 BASES, RCS SPECIFIC ACTIVITY

1. Changes are made to be consistent with changes made to the Specification.

2. Changes are made to be consistent with the Specification.

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

4. The brackets have been removed and the proper plant specific information/value has been provided.

5. Editorial change with no change in intent.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.16, RCS SPECIFIC ACTIVITY

There are no specific NSHC discussions for this Specification.

Attachment 1, Volume 9, Rev. 0, Page 387 of 415

ATTACHMENT 17

ITS 3.4.17, STEAM GENERATOR (SG) TUBE INTEGRITY

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

A01

ITS 3.4.17

ITS

REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

LCO 3.4.17 3.4.5 a. SG tube integrity shall be maintained, and

b. All SG tubes satisfying the tube repair criteria shall be plugged or repaired in accordance with the Steam Generator Program.

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

ACTION:

ACTIONS NOTE Note: These ACTIONS may be entered separately for each SG tube.

ACTION A a. With one or more SG tubes satisfying the tube repair criteria and not plugged or repaired in accordance with the Steam Generator Program,

ACTION A 1. Within 7 days, verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection, or be in HOT STANDBY within the

ACTION B next 6 hours and in COLD SHUTDOWN within the following 30 hours, and

ACTION A 2. Plug or repair the affected tube(s) in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following the next refueling outage or SG tube inspection.

ACTION B b. With SG tube integrity not maintained, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1 4.4.5.1 Verify SG tube integrity in accordance with the Steam Generator Program.

SR 3.4.17.2 4.4.5.2 Verify that each inspected SG tube that satisfies the tube repair criteria is plugged or repaired in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following a SG tube inspection.

DISCUSSION OF CHANGES
ITS 3.4.17, STEAM GENERATOR (SG) TUBE INTEGRITY

ADMINISTRATIVE CHANGES

A01 In the conversion of the Davis-Besse Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1430, Rev. 3.1, "Standard Technical Specifications-Babcock and Wilcox Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Attachment 1, Volume 9, Rev. 0, Page 391 of 415

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

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

3.4.5 LCO 3.4.17 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged [or repaired] in accordance with the Steam Generator Program. 1

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

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each SG tube.

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME	
Actions a.1 and a.2	A. One or more SG tubes satisfying the tube repair criteria and not plugged [or repaired] in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days	1
		AND		
		A.2 Plug [or repair] the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection	1
Actions a.1 and b.	B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3.	6 hours	
		AND		
		B.2 Be in MODE 5.	36 hours	

CTS

SURVEILLANCE REQUIREMENTS

CTS		SURVEILLANCE	FREQUENCY	
4.4.5.1	SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program	
4.4.5.2	SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged [or repaired] in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection	1

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.17, STEAM GENERATOR (SG) TUBE INTEGRITY

1. The brackets have been removed and the proper plant specific information/value is provided.

Attachment 1, Volume 9, Rev. 0, Page 395 of 415

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.4, "RCS Loops – MODES 1 and 2," LCO 3.4.5, "RCS Loops – MODE 3," LCO 3.4.6, "RCS Loops – MODE 4," and LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 5.5.9, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.9, tube integrity is
8 maintained when the SG performance criteria are met. There are three 5
SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.9. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is ~~only briefly~~ released to the atmosphere via main steam safety valves ~~and the majority is discharged to the main condenser~~. 2

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of [1 gallon per minute] ~~or is assumed to increase to [1 gallon per minute] as a result of accident induced conditions~~. ~~For accidents that do not involve fuel damage, the primary coolant activity level of~~ DOSE EQUIVALENT I-131 is assumed to be equivalent to 1% failed fuel in the accident analyses ~~equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel~~. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits). 1 2 2

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged [or repaired] in accordance with the Steam Generator Program. 1

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is [repaired or] removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged [or repaired] the tube may still have tube integrity. 1 1

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall [and any repairs made to it], between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube. 1

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.[9] 8 ~~"Steam Generator Program,"~~ and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria. 5 4

BASES

LCO (continued)

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed [1 gpm per SG~~, except for specific types of degradation at specific locations where the NRC has approved greater accident induced leakage~~.] The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident. 1

BASES

LCO (continued)

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

<u>A.1 and A.2</u>

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged [or repaired] in accordance with the Steam Generator Program as required by SR 3.4.17.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged [or repaired] has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity 1

BASES

ACTIONS (continued)

determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged [or repaired] prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment. 1

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.[9] contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections. 8 5

SR 3.4.17.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is [repaired or] removed from service by plugging. The tube repair criteria delineated in Specification 5.5.[9] are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria. 1 8 5

[Steam generator tube repairs are only performed using approved repair methods as described in the Steam Generator Program.] 1

The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged [or repaired] prior to subjecting the SG tubes to significant primary to secondary pressure differential. 1

BASES

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
2. 10 CFR 50 Appendix A, GDC 19.
3. 10 CFR 100.
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."
7. Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," July 1975.

3

Attachment 1, Volume 9, Rev. 0, Page 403 of 415

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.17 BASES, STEAM GENERATOR (SG) TUBE INTEGRITY

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

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

3. Updated references with valid and committed Regulatory Guide.

4. Editorial change. The title of the SG Program has already been defined in the Bases.

5. The correct LCO number has been provided.

Attachment 1, Volume 9, Rev. 0, Page 404 of 415

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.17, STEAM GENERATOR (SG) TUBE INTEGRITY

There are no specific NSHC discussions for this Specification.

Attachment 1, Volume 9, Rev. 0, Page 406 of 415

ATTACHMENT 18

Relocated Current Technical Specifications

CTS 3/4.4.10.1, ASME CODE CLASS 1, 2, AND 3 COMPONENTS

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

REACTOR COOLANT SYSTEM

3.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

APPLICABILITY: All MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50 °F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200 °F. R01

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10.1 In addition to the requirements of Specification 4.0.5:

a. Inservice inspection of each reactor coolant pump flywheel shall be performed at least once every 10 years. The inservice inspection shall be either an ultrasonic examination of the volume from the inner bore of the flywheel to the circle of one-half the outer radius, or a surface examination of exposed surfaces of the disassembled flywheel. The recommendations delineated in Regulatory Guide 1.14, Revision 1, August 1975, Positions 3, 4 and 5 of Section C.4.b shall apply. See ITS 5.5

CTS 3/4.4.10.1

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

b. Each internals vent valve shall be demonstrated OPERABLE at least once per 24 months* during shutdown by:

1. Verifying through visual inspection that the valve body and valve disc exhibit no abnormal degradation,

2. Verifying the valve is not stuck in an open position, and

3. Verifying through manual actuation that the valve is fully open when a force of ≤ 400 lbs. is applied vertically upward.

See ITS 5.5

* An exception applies for the interval following the March 2003 verification completed during the Thirteenth Refueling Outage. Under this exception, the next performance of this surveillance requirement may be delayed until March 25, 2006.

Attachment 1, Volume 9, Rev. 0, Page 411 of 415

DISCUSSION OF CHANGES
CTS 3/4.4.10.1, ASME CODE CLASS 1, 2, AND 3 COMPONENTS

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

R01 CTS 3/4.4.10.1 provides requirements for the ASME Code Class 1, 2 and 3 components to ensure their structural integrity. The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained throughout the life of the components. ASME Code Class 1, 2, and 3 components are monitored so that the possibility of component structural failure does not degrade the safety function of the system. The monitoring activity is of a preventive nature rather than a mitigative action. Other Technical Specifications require important systems to be OPERABLE (for example, Emergency Core Cooling Systems) and in a ready state for mitigative action. This Technical Specification is more directed toward prevention of component degradation and continued long term maintenance of acceptable structural conditions. Hence, it is not necessary to retain this Specification to ensure immediate OPERABILITY of safety systems. Further, this Technical Specification prescribes inspection requirements that are performed during plant shutdown. It is, therefore, not directly important for responding to design basis accidents. This LCO does not meet the criteria for retention in the ITS; therefore, it will be retained in the Technical Requirements Manual (TRM).

This change is acceptable because CTS 3/4.4.10.1 does not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion into the ITS.

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. The programmatic inspections stipulated by this Specification are not installed instrumentation used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary during operations prior to a design basis accident (DBA). The ASME Code Class 1, 2 and 3 Components Specification does not satisfy criterion 1.

2. The programmatic inspections stipulated by this Specification are not a process variable, design feature, or operating restriction that is an initial assumption in a DBA or transient. The ASME Code Class 1, 2 and 3 Components Specification does not satisfy criterion 2.

3. The ASME Code Class 1, 2, and 3 components inspected per this Specification are assumed to function to mitigate a DBA. Their capability to perform this function is addressed by other Technical Specifications. This Technical Specification only specifies programmatic inspection

DISCUSSION OF CHANGES
CTS 3/4.4.10.1, ASME CODE CLASS 1, 2, AND 3 COMPONENTS

requirements for these components, and these inspections can only be performed when the plant is shutdown. Therefore, criterion 3 is not satisfied.

4. As discussed in B&W Owners Group Technical Report 47-1170689-00 (Appendix A pages A-63 and A-64), the assurance of operability of the entire system as verified in the system operability Specification dominates the risk contribution of the system. The lack of a long term assurance of structural integrity as stipulated by this Specification was found to be non-significant risk contributor to core damage frequency and offsite releases. Davis-Besse has reviewed this evaluation, considers it applicable to Davis-Besse Nuclear Power Station, and concurs with the assessment. The ASME Code Class 1, 2 and 3 Components Specification does not meet criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria have not been met, the ASME Code Class 1, 2 and 3 Components LCO and associated Surveillances may be relocated out of the Technical Specifications. The ASME Code Class 1, 2 and 3 Components Specification will be relocated to the TRM. The TRM is currently incorporated by reference into the UFSAR, thus any changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. In addition, Surveillances, except for the reactor coolant pump (RCP) flywheel inspection and the internal vent valve requirements, are already required by regulations in 10 CFR 50.55a to be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda. The RCP flywheel inspection requirement and the internal vent valve requirements are not covered by other regulatory requirements and are needed for safe operation of the plant; therefore, these requirements will be maintained in the Davis-Besse Improved Technical Specifications. Chapter 5.0 of the Davis-Besse Improved Technical Specifications will contain a section which provides a programmatic approach to the requirements relating to the structural integrity of ASME Code Class 1, 2, and 3 components. This change is designated as relocation because the Specification did not meet the criteria in 10 CFR 50.36(c)(2)(ii) and has been relocated to the TRM.

<u>REMOVED DETAIL CHANGES</u>

None

<u>LESS RESTRICTIVE CHANGES</u>

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.4.10.1, ASME CODE CLASS 1, 2, AND 3 COMPONENTS

There are no specific NSHC discussions for this Specification.

Attachment 1, Volume 9, Rev. 0, Page 415 of 415