ATTACHMENT I

MARKUPS OF THE TECHNICAL SPECIFICATIONS AND BASES

I-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	-------------------NOTE------------------- With three RCPs operating, the limits are applied to the loop with the highest pressure. -- Verify RCS loop pressure is within limits specified in the COLR.	12 hours
SR 3.4.1.2	-------------------NOTE------------------- With three RCPs operating, the limits are applied to the loop with the lowest loop average temperature for the condition where there is a 0 °F ΔTc setpoint. -- Verify RCS loop average temperature is within limits specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow is within limits specified in the COLR.	12 hours
SR 3.4.1.4	-------------------NOTE------------------- Not required to be performed until 7 days after stable thermal conditions are established in the higher power range of MODE 1. -- Verify by measurement RCS total flow rate is within limit specified in the COLR.	18 months

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 For Units not analyzed in accordance with DPC-NE-3005-PA, two ~~Two~~ pressurizer safety valves shall be OPERABLE with lift settings ≥ 2475 psig and ≤ 2525 psig. For Units analyzed in accordance with DPC-NE-3005-PA, two pressurizer safety valves shall be OPERABLE with lift settings ≥ 2425 psig and ≤ 2575 psig.

APPLICABILITY: MODES 1 and 2,
MODE 3 with all RCS cold leg temperatures > 325°F.

---------------------------NOTE---------------------------
The lift settings are not required to be within the LCO limits for entry into the applicable portions of MODE 3 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 36 hours following entry into the applicable portions of MODE 3 provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1	Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
	AND		
OR Two pressurizer safety valves inoperable.	B.2	Be in MODE 3 with any RCS cold leg temperature ≤ 325°F.	18 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
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

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3.7 PLANT SYSTEMS

3.7.4 Atmospheric Dump Valve (ADV) Flow Paths

LCO 3.7.4 The ADV flow path for each steam generator shall be OPERABLE.

For Units analyzed in accordance with DPC-NE-3005-PA, MODES 1, 2, and 3, and MODE 4 when steam generator is relied upon for heat removal

APPLICABILITY: When required by Required Actions B.2 and C.2 of LCO 3.5.2, "High Pressure Injection (HPI)"

For Units not analyzed in accordance with DPC-NE-3005-PA,

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both required ADV flow path(s) inoperable. In a Unit not analyzed in accordance with DPC-NE-3005-PA	A.1 Be in MODE 3. AND A.2 Reduce RCS temperature to ≤ 350°F.	12 hours 60 hours
B. One or both ADV flow path(s) inoperable in a Unit analyzed in accordance with DPC-NE-3005-PA	B.1 Be in MODE 3. AND B.2 Be in MODE 4 without reliance upon steam generator for heat removal.	12 hours 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.4.1 Cycle the valves which comprise the ADV flow paths.	18 months

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3.9 REFUELING OPERATIONS

3.9.7 Unborated Water Source Isolation Valves

LCO 3.9.7 Each valve used to isolate unborated water sources shall be secured in the closed position.

APPLICABILITY: MODE 6.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each unborated water source isolation valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Action A.3 must be completed whenever Condition A is entered. ---------------------- One or more valves not secured in closed position.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2 Initiate actions to secure valve in closed position.	Immediately
	<u>AND</u>	
	A.3 Perform SR 3.9.1.1.	4 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.7.1	Verify each valve that isolates unborated water sources is secured in the closed position.	31 days

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5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

(1) DPC-NE-1002A, Reload Design Methodology II, ~~October 1985;~~ Rev. 1, (SER dated October 1, 1985);

(2) NFS-1001A, Reload Design Methodology, ~~April 1984;~~ Rev. 4, (SER dated July 29, 1981);

(3) DPC-NE-2003P-A, Oconee Nuclear Station Core Thermal Hydraulic Methodology Using VIPRE-01, ~~July 1989;~~ (SER dated July 19, 1989);

(4) DPC-NE-1004A, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, ~~November 1992;~~ (SER dated November 23, 1992);

(5) ~~BAW-10162P-A, TACO3 Fuel Pin Thermal Analysis Computer Code, B&W Fuel Company, November, 1989;~~

(6) ~~BAW-10183P, Fuel Rod Gas Pressure Criterion, B&W Fuel Company, as approved by SER dated February, 1994;~~

(7) DPC-NE-3000P-A, Thermal Hydraulic Transient Analysis Methodology, ~~August, 1994;~~ Rev. 2, (SER dated October 14, 1998);

(8) DPC-NE-2005P-A, Thermal Hydraulic Statistical Core Design Methodology, ~~February, 1995 and~~ Rev. 1, (SER dated November 7, 1996); and

(9) DPC-NE-3005-PA, UFSAR Chapter 15 Transient Analysis Methodology, ~~DPC, JUL97.~~ Rev. 1, (SER pending)

replace (5) and (6) with

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

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

(5) DPC-NE-2008P-A, Fuel Mechanical Reload Analysis Methodology Using TACO3 (SER dated April 3, 1995);

(6) BAW-10192-PA, BWNT LOCA - BWNT Loss of Coolant Accident Evaluation Model for Once-Through Steam Generator Plants, (SER dated February 18, 1997);

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BASES

ACTIONS

A.2.1.2 (continued)

reactivity on tripping. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour to initiate boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.

A.2.2

Reduction of THERMAL POWER to ≤ 60% ALLOWABLE THERMAL POWER ensures that local LHR increases, due to a misaligned rod, will not cause the core design criteria to be exceeded. The required Completion Time of 2 hours allows the operator sufficient time for reducing THERMAL POWER.

A.2.3

Reduction of the nuclear overpower trip setpoints, based on flux and flux/flow imbalance, to ≤ 65.5% ALLOWABLE THERMAL POWER, after THERMAL POWER has been reduced to 60% ALLOWABLE THERMAL POWER, maintains both core protection and an operating margin at reduced power similar to that at RTP. The required Completion Time of 10 hours allows the operator 8 additional hours after completion of the THERMAL POWER reduction in Required Action A.2.2.1 to adjust the trip setpoints.

A.2.4

for Units not analyzed in accordance with DPC-NE-3005-PA

The existing CONTROL ROD configuration must not cause an ejected rod to exceed the limit of 0.65% $\Delta k/k$ at RTP or 1.0% $\Delta k/k$ at zero power (Ref. 3). The existing CONTROL ROD configuration must not cause an ejected rod to exceed the limit of 0.2% $\Delta k/k$ at RTP, 0.4% $\Delta k/k$ at 80% RTP, or 0.8% $\Delta k/k$ at zero power for Units analyzed in accordance with DPC-NE-3005-PA (Ref. 3). This evaluation may require a ~~computer calculation of the~~ maximum ejected rod worth based on nonstandard configurations of the CONTROL ROD groups. The evaluation must determine the ejected rod worth for the duration of time that operation is expected to continue with a misaligned rod. Should fuel cycle conditions at some later time become more bounding than those at the time of

(continued)

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

4. RCS Low Pressure

The RCS Low Pressure trip, in conjunction with the RCS High Outlet Temperature and Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the system pressure approaches the conditions necessary for DNB. The RCS Low Pressure trip provides DNB low pressure limit for the RCS Variable Low Pressure trip.

The RCS Low Pressure setpoint Allowable Value is selected to ensure that a reactor trip occurs before RCS pressure is reduced below the lowest point at which the RCS Variable Low Pressure trip is analyzed. The RCS Low Pressure trip provides protection for primary system depressurization events and has been credited in the accident analysis calculations for small break loss of coolant accidents (LOCAs). ~~and~~ main steam line break (MSLB) accidents. Harsh RB conditions created by small break LOCAs cannot affect performance of the RCS pressure sensors and transmitters within the time frame for a reactor trip. Therefore, degraded environmental conditions are not considered in the Allowable Value determination.

In addition for Units not analyzed in accordance with DPC-NE-3005-PA, the RCS low pressure trip is credited in the

5. RCS Variable Low Pressure

The RCS Variable Low Pressure trip, in conjunction with the RCS High Outlet Temperature and RCS Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the system parameters of pressure and temperature approach the conditions necessary for DNB. The RCS Variable Low Pressure trip provides a floating low pressure trip based on the RCS High Outlet Temperature within the range specified by the RCS High Outlet Temperature and RCS Low Pressure trips.

The RCS Variable Low Pressure setpoint Allowable Value is selected to ensure that a trip occurs when temperature and pressure approach the conditions necessary for DNB while operating in a temperature pressure region constrained by the low pressure and high temperature trips. The RCS Variable Low Pressure trip is assumed for transient protection ~~in the unit~~

for Units not analyzed in accordance with DPC-NE-3005-PA

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5. RCS Variable Low Pressure (continued)

safety analysis but does not affect the limiting cases; therefore, determination of the setpoint Allowable Value does not account for errors induced by a harsh RB environment for Units not analyzed in accordance with DPC-NE-3005-PA.

¶ The RCS Variable Low Pressure trip is also assumed for transient protection in the main steam line break analysis for Units analyzed in accordance with DPC-NE-3005-PA. The setpoint allowable value does not include errors induced by the harsh environment because the trip function actuates prior to the harsh environment.

6. Reactor Building High Pressure

The Reactor Building High Pressure trip provides an early indication of a high energy line break (HELB) inside the RB. By detecting changes in the RB pressure, the RPS can provide a reactor trip before the other system parameters have varied significantly. Thus, this trip acts to minimize accident consequences. It also provides a backup for RPS trip instruments exposed to an RB HELB environment.

The Allowable Value for RB High Pressure trip is set at the lowest value consistent with avoiding spurious trips during normal operation. The electronic components of the RB High Pressure trip are located in an area that is not exposed to high temperature steam environments during HELB transients inside containment. The components are exposed to high radiation conditions. Therefore, the determination of the setpoint Allowable Value accounts for errors induced by the high radiation.

7. Reactor Coolant Pump to Power

The Reactor Coolant Pump to Power trip provides protection for changes in the reactor coolant flow due to the loss of multiple RCPs. Because the flow reduction lags loss of power indications due to the inertia of the RCPs, the trip initiates protective action earlier than a trip based on a measured flow signal.

The Reactor Coolant Pump to Power trip has been credited in the accident analysis calculations for the loss of more than two RCPs.

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(continued)

B 3.3 INSTRUMENTATION

B 3.3.15 Turbine Stop Valve (TSV) Closure

BASES

BACKGROUND

The Turbine Stop Valves (TSV) Closure function partially isolates the main steam lines from the SGs by closing the TSVs on both main steam lines following a ~~high energy line break (HELB)~~ turbine or reactor trip signal.

Two TSVs are provided for each main steam line and are located outside of containment. The TSVs are downstream from the main steam safety valves (MSSVs) and emergency feedwater pump turbine's steam supply to prevent the MSSVs and EFW pump's steam supply from being isolated from the steam generators by TSV closure. Closing the TSVs partially isolates each steam generator from the other, and isolates the turbine from the steam generators.

TSV Closure is initiated by a reactor trip. To keep from rapidly cooling down the primary plant by drawing off too much steam, the turbine is tripped when the reactor trips. Two independent and redundant "Reactor Trip Confirmed" signals in the form of contact closures from the control rod drive system will energize two independent turbine trip mechanisms. The Channel A trip circuit will close all four TSVs within a maximum of 1 second. The Channel B trip circuit will close the TSVs within a maximum of 15 seconds.

APPLICABLE SAFETY ANALYSES

The design basis of the TSV Closure function is established by the analysis for the main steam line break (MSLB) as discussed in the UFSAR, Section 15.13 (Ref. 1). TSV closure is necessary to stop steam flow to the turbine (to prevent overcooling) following all reactor trips.

The accident analysis compares several different MSLB events. The MSLB outside containment upstream of the TSV is limiting for offsite dose, although a break in this section of main steam header has a very low probability. The ~~main~~ MSLB without ICS and without operator action is the limiting case for a post trip return to power for Units not analyzed in accordance with DPC-NE-3005-PA. For Units analyzed in accordance with DPC-NE-3005-PA, the MSLB with ICS low level control and without operator action prior to ten minutes is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available and with a loss of offsite power. ~~following turbine trip.~~ With offsite power

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the High Pressure Injection (HPI) System pumps, is delayed.

The TSVs remain open during power operation. These valves close upon a reactor trip or turbine trip signal.

a. For an HELB or an MSLB inside containment, ~~the analysis assumes the TSV in the affected steam generator remains open. For this scenario,~~ steam is discharged into containment from both steam generators until closure of the TSVs ~~in the intact steam generator occurs~~. After TSV closure, steam is discharged into containment only from the affected steam generator.

b. An MSLB outside of containment and upstream from the TSVs is not a containment pressurization concern. The uncontrolled blowdown of both steam generators must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the TSVs isolates the break and limits the blowdown to a single steam generator.

c. An event such as increased steam flow through the turbine will terminate on closing the TSVs.

d. Following a steam generator tube rupture, closure of the TSVs isolates the ruptured steam generator from the intact steam generator.

The TSV Closure function satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

Two TSV Closure channels are required to be OPERABLE.

This LCO provides assurance that the TSVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 limits (Ref. 3).

BASES

LCO (continued)

operates within the limits assumed for the plant safety analyses. Operating within these limits will result in meeting DNBR criteria in the event of a DNB limited transient.

The pressure limits are applied to the loop with the highest pressure. The temperature limits are applied to the loop with the lowest loop average temperature for the condition in which there is a 0°F ΔT_c setpoint.

APPLICABILITY

In MODE 1 during steady state operation, the limits on RCS loop pressure, RCS loop average temperature, and RCS flow rate must be maintained 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. Steady state operation, for the purposes of this specification, is defined as operation within a 4% (e.g., 88% - 92% RTP) power band for ≥ 4 hours.

ACTIONS

A.1

Loop pressure and loop average coolant temperature are controllable and measurable parameters. With one or both of these parameters not within the LCO limits, action must be taken to restore the parameters. RCS flow rate is not a controllable parameter and is not expected to vary during steady state four pump or three pump operation. However, if the flow rate is below the LCO limit, the parameter must be restored to within limits or power must be reduced as required in Required Action B.1, to restore DNBR margin and eliminate the potential for violation of the accident analysis bounds. The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust unit parameters, determine the cause for the off normal condition, and restore the readings within limits. The Completion Time is based on plant operating experience.

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(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.2 (continued)

the 12 hour Surveillance Frequency for loop average 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. A Note has been added to indicate the temperature limits for three pumps operating are applied to the loop with the lowest loop average temperature for the condition in which there is a 0 °F ΔT_c setpoint.

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 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 specified in the COLR.

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. The Surveillance is modified by a Note that indicates the SR does not need to be performed until 7 days after stable

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(continued)

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. For the remainder of MODE 3, MODE 4, 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."

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 setpoint of the pressurizer code safety valves is in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1967. ~~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.

INSERT A

~~The upper and lower pressure limits are 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. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

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.

Insert A

For units analyzed in accordance with DPC-NE-3005-PA, the required lift pressure is 2500 psig +/- 3%. For units not analyzed in accordance with DPC-NE-3005-PA, the required lift pressure is 2500 psig +/- 1%. The upper and lower pressure limits are based on the requirements of ASME Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1967, which limit the rise in pressure within the vessels which they protect to 10% above the design pressure.

BASES (continued)

APPLICABLE SAFETY ANALYSES

All accident analyses in the UFSAR that require safety valve actuation assume operation of both pressurizer safety valves to limit increasing reactor coolant pressure. The overpressure protection analysis is also based on operation of both safety valves and assumes that the valves open at the high range of the setting (2500 psig system design pressure plus ~~1%)~~. These valves must accommodate pressurizer insurges that could occur during a startup, rod withdrawal, ejected rod, or loss of main feedwater. The startup accident establishes the minimum safety valve capacity. The startup accident is assumed to occur at < 15% power. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this Specification is required to ensure that the accident analysis and design basis calculations remain valid.

INSERT B

Pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

The two pressurizer safety valves are set to open at the RCS design pressure (2500 psig) and within the ASME specified tolerance to avoid exceeding the maximum RCS design pressure SL, to maintain accident analysis assumptions and to comply with ASME Code requirements. ~~The upper and lower pressure tolerance limits are based on the ±1% tolerance requirements (Ref. 1) for lifting pressures above 1000 psig. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure.~~ Inoperability of one or both valves could result in exceeding the SL if a transient were to occur.

INSERT C

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.

(continued)

Insert B

3% for units analyzed in accordance with DPC-NE-3005-PA, and 2500 psig plus 1% for units not analyzed in accordance with DPC-NE-3005-PA).

Insert C

The valves will be tested per ASME Section XI requirements and returned to service with as-left setpoints of 2500 psig +/- 1%. The upper and lower pressure tolerance limits are based on the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1967, which limits the rise in pressure within the vessel which they protect, to 10% above the design pressure.

BASES

ACTIONS

B.1 and B.2 (continued)

18 hours. The 12 hours allowed is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. Similarly, the 18 hours allowed is reasonable, based on operating experience, to reach MODE 3 with any RCS cold leg temperature ≤ 325°F without challenging unit systems. With any RCS cold leg temperature at or below 325°F, overpressure protection is provided by LTOP. Reducing the RCS temperature to ≤ 325°F reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by two pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 2), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valves setpoint is ± 3% for OPERABILITY for units analyzed in accordance with DPC-NE-3005-PA; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. ASME, Boiler and Pressure Vessel Code, Section XI.
3. 10 CFR 50.36.

for Units not analyzed in accordance with DPC-NE-3005-PA

BASES

Insert A

APPLICABLE SAFETY ANALYSES (continued)

return to power. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the High Pressure Injection (HPI) System pumps, is delayed.

The TSVs remain open during power operation. These valves close upon a turbine trip or reactor trip signal.

a. For an HELB or an SLB inside containment, ~~the analysis assumes the TSV for the affected steam generator remains open~~. ~~For this scenario,~~ steam is discharged into containment from both steam generators until closure of the TSVs. ~~in the intact steam generator occurs~~. After TSV closure, steam is discharged into containment only from the affected steam generator.

Insert A

For Units analyzed in accordance with DPC-NE-3005-PA, the MSLB with ICS low level control and no operator action prior to ten minutes is the limiting case for a post trip return to power.

b. An MSLB outside of containment and upstream from the TSVs is not a containment pressurization concern. The uncontrolled blowdown of both steam generators must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the TSVs isolates the break and limits the blowdown to a single steam generator.

c. Steam flow to the turbine if not controlled by the turbine control valves will terminate on closing the TSVs.

d. Following a steam generator tube rupture, closure of the TSVs isolates the ruptured steam generator from the intact steam generator.

The TSVs satisfy Criterion 3 of 10 CFR 50.36, (Ref. 3).

LCO

This LCO requires that the two TSVs in each steam line be OPERABLE. The TSVs are considered OPERABLE when the isolation times are within limits and they close on an isolation actuation signal.

This LCO provides assurance that the TSVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 limits (Ref. 4).

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B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Dump Valve (ADV) Flow Paths

BASES

INSERT A

BACKGROUND

The ADV flow path for each steam generator is credited as a compensatory measure in Actions B and C of LCO 3.5.2, "High Pressure Injection (HPI)," to permit operation to continue with THERMAL POWER ≤ 75% RTP: a) for 30 days with an HPI pump or HPI discharge crossover valve(s) inoperable; and b) for 72 hours with an HPI train inoperable.

During these periods of time, the ADV flow path for one steam generator is credited to depressurize the steam generator and enhance primary-to-secondary heat transfer during certain small break loss of coolant accidents (LOCAs). ~~This is done in conjunction with the secondary cooling water from the Emergency Feedwater (EFW) System. The preferred heat sink via the Turbine Bypass System to the condenser may not be available following a small break LOCA.~~

INSERT B

For each steam generator, the ADV flow path is comprised of the atmospheric dump block valve bypass (1" bypass), the atmospheric vent valve (a 12" block valve), the atmospheric dump control valve (i.e., throttle valve), and the atmospheric vent block valve (i.e., isolation valve). The throttle valve and the isolation valve are in parallel and are located downstream of the atmospheric vent valve.

The atmospheric vent valve should be opened prior to opening the throttle valve or isolation valve. This is accomplished by first opening the atmospheric dump block valve bypass. This equalizes the differential pressure across the atmospheric vent valve. Once the atmospheric vent valve is opened, the cool down rate is controlled using the throttle valve. If additional relief capacity is needed, the isolation valve can be opened. The capacity of the throttle or isolation valve exceeds decay heat loads and is sufficient to cool down the plant.

(continued)

Insert A

The ADV flow paths provide a method for cooling the unit to decay heat removal (DHR) entry conditions, should the preferred heat sink via the Turbine Bypass System to the condenser not be available, as discussed in the UFSAR (Ref. 2). This is done in conjunction with the secondary cooling water from the Emergency Feedwater (EFW) System.

Insert B

Additionally, for Units that have been analyzed in accordance with DPC-NE-3005-PA (Ref. 6), the steam generator tube rupture (SGTR) analysis (Ref. 3) credits operator action to depressurize the steam generators by opening each of the ADV flow paths.

BASES (continued)

APPLICABLE SAFETY ANALYSES

INSERT C

If enhanced steam generator cooling is not credited in the small break LOCA analysis, two HPI trains are required to mitigate specific small break LOCAs. However, if equipment not qualified as QA-1 (i.e., an ADV flow path for a steam generator) is credited for enhanced steam generator cooling, the safety analyses have determined that the capacity of one HPI train is sufficient to mitigate a small break LOCA on the discharge of the reactor coolant pumps if THERMAL POWER is ≤ 75% RTP.

The analysis for Action C of LCO 3.5.2, "High Pressure Injection (HPI)," credits an ADV flow path for one steam generator as a compensatory measure in the event an HPI train is inoperable and THERMAL POWER is ≤ 75% RTP. During this situation, the ADV flow path for one steam generator is credited during certain small break LOCAs to depressurize the steam generator and enhance primary-to-secondary heat transfer. This is done in conjunction with the EFW System providing cooling water to the steam generator. The ADV flow path is comprised of manual valves. Operator action is credited within 25 minutes of an Engineered Safeguards Protective System (ESPS) signal to open them.

Additionally, the ADV flow path for each steam generator is credited as a compensatory measure in the analysis for Action B of LCO 3.5.2, "High Pressure Injection (HPI)," to permit an HPI pump or HPI discharge crossover valve(s) to be inoperable for 30 days with the THERMAL POWER ≤ 75% RTP. Typically, single failures are not considered once the plant has entered a condition defined in the Technical Specifications. However, the Completion Time permitted by Required Actions B.3 and B.4 of LCO 3.5.2, "High Pressure Injection (HPI)," is an extended period of time (i.e., 30 days). In the event an accident occurred during this 30-day Completion Time and a single failure were to occur in the degraded HPI System, the ability of a plant to mitigate the consequences of specific small break LOCAs continues to be assured by the ADV flow path for one steam generator.

INSERT D

The ADV flow paths satisf~~ies~~y Criterion 3 of 10 CFR 50.36 (Ref.1).

Insert C

Operator action to depressurize a steam generator via its ADV flow path is credited in the analysis of certain small break LOCAs with THERMAL POWER ≤ 75% RTP and the plant operated with a degraded HPI System in accordance with Condition B or Condition C of ITS 3.5.2, "High Pressure Injection (HPI) System" (Ref. 4). This event credits operator action to open one ADV flow path within 25 minutes of an ESPS actuation.

Insert D

For Units that have been analyzed in accordance with DPC-NE-3005-PA (Ref. 6), the SGTR analysis credits operator action to depressurize the steam generators by opening both ADV flow paths (i.e., the ADV flow path for each steam generator) within 40 minutes of identifying the ruptured steam generator. Within this 40-minute time period, the operators are only required to open the bypass valve, the block valve, and the throttle valve. However, later in the event, the analysis also assumes that the operators will open the isolation valves in each ADV flow path.

BASES (continued)

LCO

The ADV flow path for each steam generator is required to be OPERABLE. Failure to meet the LCO can result in the inability to depressurize a steam generator following a small break LOCA. This function is required to support operation with a degraded HPI System when THERMAL POWER is ≤ 75% RTP.

INSERT E

An ADV flow path is considered OPERABLE when it is capable of providing a controlled relief of the main steam flow, and each valve which comprises the ADV flow path is capable of opening and closing.

APPLICABILITY

~~When required by Required Actions B.2 and C.2 of LCO 3.5.2, "High Pressure Injection (HPI)," the ADV flow path for each steam generator is required to be OPERABLE.~~

INSERT F

~~For all other conditions, an ADV flow path is not credited for mitigating a small break LOCA to satisfy the conditions of 10 CFR 50.46.~~

ACTIONS

A.1 and A.2

With one or both of the required ADV flow path(s) inoperable in a Unit that has not been analyzed in accordance with DPC-NE-3005-PA, the unit must be placed in a condition in which the LCO does not apply. The ADV flow path for each steam generator is required to support operation with a degraded HPI System. Thus, the unit must be placed in a condition outside the Applicability of LCO 3.5.2, "High Pressure Injection (HPI)." To achieve this status, the unit must be placed in at least MODE 3 within 12 hours, and RCS temperature reduced to ≤ 350°F within 60 hours. The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. They are consistent with the Completion Times provided in Required Actions G.1 and G.2 of LCO 3.5.2, "High Pressure Injection (HPI)."

INSERT G

Insert E

Additionally, for Units analyzed in accordance with DPC-NE-3005-PA (Ref. 6), failure to meet the LCO can result in the inability to depressurize the steam generators following a SGTR.

Insert F

For Units analyzed in accordance with DPC-NE-3005-PA (Ref. 6), the ADV flow path for each steam generator is required to be OPERABLE in MODES 1, 2, and 3, and in MODE 4, when a steam generator is being relied upon for heat removal. In MODE 4, steam generators are relied upon for heat removal whenever an RCS loop is required to be OPERABLE or operating to satisfy LCO 3.4.5, "RCS Loops - MODE 4" or available to transfer decay heat to satisfy LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled." The steam generators do not contain a significant amount of energy in MODE 4 when the unit is not relying upon a steam generator for heat transfer, and MODES 5 and 6; therefore, the ADV flow paths are not required to be OPERABLE in these MODES and condition.

For Units not analyzed in accordance with DPC-NE-3005-PA, the ADV flow path for each steam generator is only required to be OPERABLE when required by Required Actions B.2 and C.2 of LCO 3.5.2, "High Pressure Injection (HPI)." For all other conditions, the ADV flow paths for these Units are not credited in the analyses of any accident.

Insert G

B.1 and B.2

With one or both of the ADV flow path(s) inoperable in a Unit analyzed in accordance with DPC-NE-3005-PA, the Unit must be placed in a condition in which the LCO does not apply. To achieve this status, the Unit must be placed in at least MODE 3 within 12 hours, and at least MODE 4 without reliance on a steam generator for heat removal within 24 hours. The Completion Times are reasonable, based on operating experience, to reach the required Unit conditions from full power conditions in an orderly manner and without challenging Unit systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a controlled cool down of the RCS, the valves which comprise the ADV flow path for each steam generator must be able to perform the following functions:

a) the atmospheric dump block valve bypass and the atmospheric vent valve must be capable of being opened and closed; and

b) the atmospheric dump control valve and atmospheric vent block valve must be capable of being opened and throttled through their full range.

This SR ensures that the valves which comprise the ADV flow path for each steam generator are tested through a full control cycle at least once per 18 months. Performance of inservice testing or use of an ADV flow path during a unit cool down may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50.36.

INSERT I

Insert I

2. UFSAR, Section 10.3.

3. UFSAR, Section 15.9.

4. UFSAR, Section 15.14.

5. UFSAR, Section 15.12.

6. DPC-NE-3005-PA, "UFSAR Chapter 15 Transient Analysis Methodology," Duke Power Company, Oconee Nuclear Station, January 1999.

B 3.9 REFUELING OPERATIONS

B 3.9.7 Unborated Water Source Isolation Valves

BASES

BACKGROUND	During MODE 6 operations, all isolation valves for reactor makeup water sources containing unborated water that are connected to the Reactor Coolant System (RCS) must be closed to prevent unplanned boron dilution of the reactor coolant. The isolation valves must be secured in the closed position. The Coolant Storage System is capable of supplying borated and unborated water to the RCS through various flow paths. Since a positive reactivity addition made by reducing the boron concentration is inappropriate during MODE 6, isolation of all unborated water sources prevents an unplanned boron dilution.
APPLICABLE SAFETY ANALYSES	The possibility of an inadvertent boron dilution event (Ref. 1) occurring during MODE 6 refueling operations is precluded by adherence to this LCO, which requires that potential dilution sources be isolated. Closing the required valves during refueling operations prevents the flow of unborated water to the filled portion of the RCS. The valves are used to isolate unborated water sources. These valves have the potential to indirectly allow dilution of the RCS boron concentration in MODE 6. By isolating unborated water sources when in MODE 6, a boron dilution event as analyzed in the UFSAR is prevented. The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2).
LCO	This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent unplanned boron dilution during MODE 6 and thus avoid a reduction in SDM.

(continued)

BASES (continued)

APPLICABILITY

In MODE 6, this LCO is applicable to prevent an inadvertent boron dilution event by ensuring isolation of all sources of unborated water to the RCS.

For all other applicable MODES, the boron dilution accident was analyzed and was found to be capable of being mitigated. The boron dilution event is applicable in MODES 1 and 6.

ACTIONS

The ACTIONS table has been modified by a Note that allows separate Condition entry for each unborated water source isolation valve.

A.1

Continuation of CORE ALTERATIONS is contingent upon maintaining the unit in compliance with this LCO. With any valve used to isolate unborated water sources not secured in the closed position, all operations involving CORE ALTERATIONS must be suspended immediately. The Completion Time of "immediately" for performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

Condition A has been modified by a Note to require that Required Action A.3 be completed whenever Condition A is entered.

A.2

Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the unborated water isolation valves secured closed. Securing the valves in the closed position ensures that the valves cannot be inadvertently opened. The Completion Time of "immediately" requires an operator to initiate actions to close an open valve and secure the isolation valve in the closed position immediately. Once actions are initiated, they must be continued until the valves are secured in the closed position.

(continued)

BASES

ACTIONS (continued)

A.3

Due to the potential of having diluted the boron concentration of the reactor coolant, SR 3.9.1.1 (verification of boron concentration) must be performed whenever Condition A is entered to demonstrate that the required boron concentration exists. The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

These valves are to be secured closed to isolate possible dilution paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote due to the large mass of borated water in the fuel transfer canal and the fact that all unborated water sources are isolated, precluding a dilution. The boron concentration is checked every 72 hours during MODE 6 under SR 3.9.1.1. This Surveillance demonstrates that the valves are closed through a system walkdown. The 31 day Frequency is based on engineering judgment and is considered reasonable in view of other administrative controls that will ensure that the valve opening is an unlikely possibility.

REFERENCES

1. UFSAR, Section 15.4.1.
2. 10 CFR 50.36.

ATTACHMENT II

RETYPED TECHNICAL SPECIFICATIONS AND BASES

INSTRUCTIONS FOR UPDATING THE
TECHNICAL SPECIFICATIONS AND BASES

Reprinted Technical Specification Pages

Instructions for Updating

the Technical Specifications and Bases

Remove Page	**Insert Page**
Technical Specifications, Table of Contents, page iv	Technical Specifications, Table of Contents, page iv
3.4.1-2	3.4.1-2
3.4.10-1 - 3.4.10-2	3.4.10-1 - 3.4.10-2
3.7.4-1	3.7.4-1
	3.9.7-1 - 3.9.7-2
5.0-32	5.0-32
Bases, Table of Contents, page iv	Bases, Table of Contents, page iv
B 3.1.4-6 - B 3.1.4-10	B 3.1.4-6 - B 3.1.4-10
B 3.3.1-15 - B 3.3.3-28	B 3.3.1-15 - B 3.3.1-28
B 3.3.15-1 - B 3.3.15-4	B 3.3.15-1 - B 3.3.15-4
B 3.4.1-3 - B 3.4.1-6	B 3.4.1-3 - B 3.4.1-5
B 3.4.10-1 - B 3.4-10-4	B 3.4.10-1 - B 3.4.10-5
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	B 3.9.7-1 - B 3.9.7-3

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	-------------------NOTE------------------- With three RCPs operating, the limits are applied to the loop with the highest pressure. -- Verify RCS loop pressure is within limits specified in the COLR.	12 hours
SR 3.4.1.2	-------------------NOTE------------------- With three RCPs operating, the limits are applied to the loop with the lowest loop average temperature for the condition where there is a 0°F ΔTc setpoint. -- Verify RCS loop average temperature is within limits specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow is within limits specified in the COLR.	12 hours
SR 3.4.1.4	-------------------NOTE------------------- Not required to be performed until 7 days after stable thermal conditions are established in the higher power range of MODE 1. -- Verify by measurement RCS total flow rate is within limit specified in the COLR.	18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 For Units not analyzed in accordance with DPC-NE-3005-PA, two pressurizer safety valves shall be OPERABLE with lift settings ≥ 2475 psig and ≤ 2525 psig.

For Units analyzed in accordance with DPC-NE-3005-PA, two pressurizer safety valves shall be OPERABLE with lift settings ≥ 2425 psig and ≤ 2575 psig.

APPLICABILITY: MODES 1 and 2,
MODE 3 with all RCS cold leg temperatures > 325°F.

---------------------------NOTE---------------------------
The lift settings are not required to be within the LCO limits for entry into the applicable portions of MODE 3 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 36 hours following entry into the applicable portions of MODE 3 provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 3 with any RCS cold leg temperature ≤ 325°F.	12 hours 18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
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

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Dump Valve (ADV) Flow Paths

LCO 3.7.4 The ADV flow path for each steam generator shall be OPERABLE.

APPLICABILITY: For Units analyzed in accordance with DPC-NE-3005-PA, MODES 1, 2, and 3, and MODE 4 when steam generator is relied upon for heat removal

For Units not analyzed in accordance with DPC-NE-3005-PA, when required by Required Actions B.2 and C.2 of LCO 3.5.2, "High Pressure Injection (HPI)."

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both required ADV flow path(s) inoperable in a Unit not analyzed in accordance with DPC-NE-3005-PA.	A.1 Be in MODE 3. AND A.2 Reduce RCS temperature to $\leq$ 350°F.	12 hours 60 hours
B. One or both ADV flow path(s) inoperable in a Unit analyzed in accordance with DPC-NE-3005-PA.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 without reliance upon steam generator for heat removal.	12 hours 24 hours

3.9 REFUELING OPERATIONS

3.9.7 Unborated Water Source Isolation Valves

LCO 3.9.7 Each valve used to isolate unborated water sources shall be secured in the closed position.

APPLICABILITY: MODE 6.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each unborated water source isolation valve.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ---------NOTE--------- Required Action A.3 must be completed whenever Condition A is entered. --------------------- One or more valves not secured in closed position.	A.1	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	A.2	Initiate actions to secure valve in closed position.	Immediately
	<u>AND</u>		
	A.3	Perform SR 3.9.1.1.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.7.1	Verify each valve that isolates unborated water sources is secured in the closed position.	31 days

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

(1) DPC-NE-1002A, Reload Design Methodology II, Rev. 1, (SER dated October 1, 1985);

(2) NFS-1001A, Reload Design Methodology, Rev. 4, (SER dated July 29, 1981);

(3) DPC-NE-2003P-A, Oconee Nuclear Station Core Thermal Hydraulic Methodology Using VIPRE-01, (SER dated July 19, 1989);

(4) DPC-NE-1004P-A, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, (SER dated November 23, 1992);

(5) DPC-NE-2008P-A, Fuel Mechanical Reload Analysis Methodology Using TACO3, (SER dated April 3, 1995);

(6) BAW-10192-PA, BWNT LOCA - BWNT Loss of Coolant Accident Evaluation Model for Once-Through Steam Generator Plants, (SER dated February 18, 1997);

(7) DPC-NE-3000P-A, Thermal Hydraulic Transient Analysis Methodology, Rev. 2, (SER dated October 14, 1998);

(8) DPC-NE-2005P-A, Thermal Hydraulic Statistical Core Design Methodology, Rev. 1 (SER dated November 7, 1996); and

(9) DPC-NE-3005-PA, UFSAR Chapter 15 Transient Analysis Methodology, Rev. 1, (SER Pending).

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

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

(continued)

BASES

ACTIONS

A.2.1.2 (continued)

reactivity on tripping. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour to initiate boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.

A.2.2

Reduction of THERMAL POWER to ≤ 60% ALLOWABLE THERMAL POWER ensures that local LHR increases, due to a misaligned rod, will not cause the core design criteria to be exceeded. The required Completion Time of 2 hours allows the operator sufficient time for reducing THERMAL POWER.

A.2.3

Reduction of the nuclear overpower trip setpoints, based on flux and flux/flow imbalance, to ≤ 65.5% ALLOWABLE THERMAL POWER, after THERMAL POWER has been reduced to 60% ALLOWABLE THERMAL POWER, maintains both core protection and an operating margin at reduced power similar to that at RTP. The required Completion Time of 10 hours allows the operator 8 additional hours after completion of the THERMAL POWER reduction in Required Action A.2.2.1 to adjust the trip setpoints.

A.2.4

The existing CONTROL ROD configuration must not cause an ejected rod to exceed the limit of 0.65% Δk/k at RTP or 1.0% Δk/k at zero power for Units not analyzed in accordance with DPC-NE-3005-PA (Ref. 3). The existing CONTROL ROD configuration must not cause an ejected rod to exceed the limit of 0.2% Δk/k at RPT, 0.4% Δk/k at 80% RPT, or 0.8% Δk/k at zero power for Units analyzed in accordance with DPC-NE-3005-PA (Ref. 3). This evaluation may require a computer calculation of the maximum ejected rod worth based

(continued)

BASES

ACTIONS

A.2.4 (continued)

on nonstandard configurations of the CONTROL ROD groups. The evaluation must determine the ejected rod worth for the duration of time that operation is expected to continue with a misaligned rod. Should fuel cycle conditions at some later time become more bounding than those at the time of the rod misalignment, additional evaluation will be required to verify the continued acceptability of operation. The required Completion Time of 72 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and sufficient time is provided to perform the required evaluation.

B.1

If the Required Actions and associated Completion Times for Condition A are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems.

C.1.1

More than one trippable CONTROL ROD becoming inoperable or misaligned, or both inoperable but trippable and misaligned from their group average position, is not expected and may violate the minimum SDM requirement. Therefore, SDM must be evaluated. Ensuring the SDM meets the minimum requirement within 1 hour allows the operator adequate time to determine the SDM.

C.1.2

If the SDM is less than the limit, then the restoration of the required SDM requires increasing the RCS boron concentration to provide negative reactivity. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident

(continued)

BASES

ACTIONS

C.1.2 (continued)

occurring, and the steps required to complete the action. This allows the operator sufficient time for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.
If more than one trippable CONTROL ROD is inoperable or misaligned from their group average position, continued operation of the reactor may cause the misalignment to increase, as the regulating rods insert or withdraw to control reactivity. If the CONTROL ROD misalignment increases, local power peaking may also increase, and local LHRs will also increase if the reactor continues operation at THERMAL POWER. The SDM is decreased when one or more CONTROL RODS become inoperable at a given THERMAL POWER level, or if one or more CONTROL RODS become misaligned by insertion from the group average position.

Therefore, it is prudent to place the reactor in MODE 3. LCO 3.1.4 does not apply in MODE 3 since excessive power peaking cannot occur. The allowed Completion Time of 12 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems.

D.1.1 and D.1.2

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

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

D.2

If the untrippable rod(s) cannot be restored to OPERABLE status, the unit must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve

(continued)

BASES

ACTIONS

D.2 (continued)

this status, the unit must be brought to at least MODE 3 within 12 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual CONTROL RODS are aligned within 6.5% of their group average height limits at a 12 hour Frequency allows the operator to detect a rod that is beginning to deviate from its expected position. The specified Frequency takes into account other CONTROL ROD position information that is continuously available to the operator in the control room, so that during actual CONTROL ROD motion, deviations can immediately be detected.

SR 3.1.4.2

Verifying each CONTROL ROD is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each CONTROL ROD could result in radial tilts. Exercising each individual CONTROL ROD every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each CONTROL ROD by an amount in any direction sufficient to demonstrate the absence of thermal binding will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between required performances of SR 3.1.4.2 (determination of CONTROL ROD OPERABILITY by movement), if a CONTROL ROD(S) is discovered to be immovable, but is determined to be trippable and aligned, the CONTROL ROD(S) is considered to be OPERABLE. At any time, if a CONTROL ROD(S) is immovable, a determination of the trippability

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.2 (continued)

(OPERABILITY) of the CONTROL ROD(S) must be made, and appropriate action taken.

SR 3.1.4.3

Verification of CONTROL ROD drop time allows the operator to determine that the maximum CONTROL ROD drop time permitted is consistent with the assumed CONTROL ROD drop time used in the safety analysis. The rod drop time given in the safety analysis is 1.66 seconds at reactor coolant full flow conditions and ≤ 1.40 seconds at no flow conditions to ¾ insertion (Ref. 5). The zone reference lights will activate at ¾ insertion to give an indication of the CONTROL ROD drop time and CONTROL ROD location. Measuring CONTROL ROD drop times, prior to reactor criticality after reactor vessel head removal, ensures that the reactor internals and CRDM will not interfere with CONTROL ROD motion or CONTROL ROD drop time. This Surveillance is performed during a unit outage, due to the unit conditions needed to perform the SR and the potential for an unplanned unit transient if the Surveillance were performed with the reactor at power.

This testing is normally performed with all reactor coolant pumps operating to simulate a reactor trip under actual conditions.

REFERENCES

1. UFSAR, Section 3.1.
2. 10 CFR 50.46.
3. UFSAR, Chapter 15.
4. 10 CFR 50.36.
5. UFSAR, Section 15.7.3.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

4. RCS Low Pressure

The RCS Low Pressure trip, in conjunction with the RCS High Outlet Temperature and Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the system pressure approaches the conditions necessary for DNB. The RCS Low Pressure trip provides DNB low pressure limit for the RCS Variable Low Pressure trip.

The RCS Low Pressure setpoint Allowable Value is selected to ensure that a reactor trip occurs before RCS pressure is reduced below the lowest point at which the RCS Variable Low Pressure trip is analyzed. The RCS Low Pressure trip provides protection for primary system depressurization events and has been credited in the accident analysis calculations for small break loss of coolant accidents (LOCAs). In addition, for Units not analyzed in accordance with DPC-NE-3005-PA, the RCS low pressure trip is credited in the main steam line break (MSLB) accidents. Harsh RB conditions created by small break LOCAs cannot affect performance of the RCS pressure sensors and transmitters within the time frame for a reactor trip. Therefore, degraded environmental conditions are not considered in the Allowable Value determination.

5. RCS Variable Low Pressure

The RCS Variable Low Pressure trip, in conjunction with the RCS High Outlet Temperature and RCS Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the system parameters of pressure and temperature approach the conditions necessary for DNB. The RCS Variable Low Pressure trip provides a floating low pressure trip based on the RCS High Outlet Temperature within the range specified by the RCS High Outlet Temperature and RCS Low Pressure trips.

The RCS Variable Low Pressure setpoint Allowable Value is selected to ensure that a trip occurs when temperature and pressure approach the conditions necessary for DNB while operating in a temperature pressure region constrained by the low pressure and

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5. RCS Variable Low Pressure (continued)

high temperature trips. The RCS Variable Low Pressure trip is assumed for transient protection for Units not analyzed in accordance with DPC-NE-3005-PA safety analysis but does not affect the limiting cases; therefore, determination of the setpoint Allowable Value does not account for errors induced by a harsh RB environment for Units not analyzed in accordance with DPC-NE-3005-PA.

The RCS Variable Low Pressure trip is also assumed for transient protection in the main steam line break analysis for Units analyzed in accordance with DPC-NE-3005-PA. The setpoint allowable value does not include errors induced by the harsh environment because the trip function actuates prior to the harsh environment.

6. Reactor Building High Pressure

The Reactor Building High Pressure trip provides an early indication of a high energy line break (HELB) inside the RB. By detecting changes in the RB pressure, the RPS can provide a reactor trip before the other system parameters have varied significantly. Thus, this trip acts to minimize accident consequences. It also provides a backup for RPS trip instruments exposed to an RB HELB environment.

The Allowable Value for RB High Pressure trip is set at the lowest value consistent with avoiding spurious trips during normal operation. The electronic components of the RB High Pressure trip are located in an area that is not exposed to high temperature steam environments during HELB transients inside containment. The components are exposed to high radiation conditions. Therefore, the determination of the setpoint Allowable Value accounts for errors induced by the high radiation.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

7. Reactor Coolant Pump to Power

The Reactor Coolant Pump to Power trip provides protection for changes in the reactor coolant flow due to the loss of multiple RCPs. Because the flow reduction lags loss of power indications due to the inertia of the RCPs, the trip initiates protective action earlier than a trip based on a measured flow signal.

The Reactor Coolant Pump to Power trip has been credited in the accident analysis calculations for the loss of more than two RCPs. The Allowable Value for the Reactor Coolant Pump to Power trip setpoint is selected to prevent normal power operation unless at least three RCPs are operating. RCP status is monitored by power transducers on each pump. These relays indicate a loss of an RCP on underpower. The underpower setpoint is selected to reliably trip on loss of voltage to the RCPs. Neither the reactor power nor the pump power setpoint account for instrumentation errors caused by harsh environments because the trip Function is not required to respond to events that could create harsh environments around the equipment.

8. Nuclear Overpower Flux/Flow Imbalance

The Nuclear Overpower Flux/Flow Imbalance trip provides steady state protection for the power imbalance SLs. A reactor trip is initiated prior to the core power, AXIAL POWER IMBALANCE, and reactor coolant flow conditions exceeding the DNB or fuel centerline temperature limits.

This trip supplements the protection provided by the Reactor Coolant Pump to Power trip, through the power to flow ratio, for loss of reactor coolant flow events. The power to flow ratio provides direct protection for the DNBR SL for the loss of one or more RCPs and for locked RCP rotor accidents.

The power to flow ratio of the Nuclear Overpower Flux/Flow Imbalance trip also provides steady state protection to prevent reactor power from exceeding the

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Nuclear Overpower Flux/Flow Imbalance (continued)

allowable power when the primary system flow rate is less than full four pump flow. Thus, the power to flow ratio prevents overpower conditions similar to the Nuclear Overpower trip. This protection ensures that during reduced flow conditions the core power is maintained below that required to begin DNB.

The Allowable Value is selected to ensure that a trip occurs when the core power, axial power peaking, and reactor coolant flow conditions indicate an approach to DNB or fuel centerline temperature limits. By measuring reactor coolant flow and by tripping only when conditions approach an SL, the unit can operate with the loss of one pump from a four pump initial condition at power levels at least as low as approximately 80% RTP. The Allowable Value for the Function is given in the unit COLR because the cycle specific core peaking changes affect the Allowable Value.

9. Main Turbine Trip (Hydraulic Fluid Pressure)

The Main Turbine Trip Function trips the reactor when the main turbine is lost at high power levels. The Main Turbine Trip Function provides an early reactor trip in anticipation of the loss of heat sink associated with a turbine trip. The Main Turbine Trip Function was added to the B&W designed units in accordance with NUREG-0737 (Ref. 5) following the Three Mile Island Unit 2 accident. The trip lowers the probability of an RCS power operated relief valve (PORV) actuation for turbine trip cases. This trip is activated at higher power levels, thereby limiting the range through which the Integrated Control System must provide an automatic runback on a turbine trip.

Each of the four turbine hydraulic fluid pressure switches feeds one protective channel through buffers that continuously monitor the status of the contacts.

For the Main Turbine Trip (Hydraulic Fluid Pressure) bistable, the Allowable Value of 800 psig is selected to provide a trip whenever main turbine hydraulic

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Main Turbine Trip (Hydraulic Fluid Pressure) (continued)

fluid pressure drops below the normal operating range. To ensure that the trip is enabled as required by the LCO, the reactor power bypass is set with an Allowable Value of 30% RTP. The turbine trip is not required to protect against events that can create a harsh environment in the turbine building. Therefore, errors induced by harsh environments are not included in the determination of the setpoint Allowable Value.

10. Loss of Main Feedwater Pumps (Hydraulic Oil Pressure)

The Loss of Main Feedwater Pumps (Hydraulic Oil Pressure) trip provides a reactor trip at high power levels when both MFW pumps are lost. The trip provides an early reactor trip in anticipation of the loss of heat sink associated with the LOMF. This trip was added in accordance with NUREG-0737 (Ref. 5) following the Three Mile Island Unit 2 accident. This trip provides a reactor trip at high power levels for a LOMF to minimize challenges to the PORV.

For the feedwater pump hydraulic oil pressure bistables, the Allowable Value of 75 psig is selected to provide a trip whenever feedwater pump hydraulic oil pressure drops below the normal operating range. To ensure that the trip is enabled as required by the LCO, the reactor power bypass is set with an Allowable Value of 2% RTP. The Loss of Main Feedwater Pumps (Hydraulic Oil Pressure) trip is not required to protect against events that can create a harsh environment in the turbine building. Therefore, errors caused by harsh environments are not included in the determination of the setpoint Allowable Value.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

11. Shutdown Bypass RCS High Pressure

The RPS Shutdown Bypass RCS High Pressure is provided to allow for withdrawing the CONTROL RODS prior to reaching the normal RCS Low Pressure trip setpoint. The shutdown bypass provides trip protection during deboration and RCS heatup by allowing the operator to at least partially withdraw the safety groups of CONTROL RODS. This makes their negative reactivity available to terminate inadvertent reactivity excursions. Use of the shutdown bypass trip requires that the neutron power trip setpoint be reduced to 5% of full power or less. The Shutdown Bypass RCS High Pressure trip forces a reactor trip to occur whenever the unit switches from power operation to shutdown bypass or vice versa. This ensures that the CONTROL RODS are all inserted before power operation can begin. The operator is required to remove the shutdown bypass, reset the Nuclear Overpower – High Power trip setpoint, and again withdraw the safety group rods before proceeding with startup.

Accidents analyzed in the UFSAR, Chapter 15 (Ref. 2), do not describe events that occur during shutdown bypass operation, because the consequences of these events are enveloped by the events presented in the UFSAR.

During shutdown bypass operation with the Shutdown Bypass RCS High Pressure trip active with a setpoint of $\leq$ 1720 psig and the Nuclear Overpower – Low Setpoint set at or below 5% RTP, the trips listed below can be bypassed. Under these conditions, the Shutdown Bypass RCS High Pressure trip and the Nuclear Overpower – Low Setpoint trip act to prevent unit conditions from reaching a point where actuation of these Functions is necessary.

1.a Nuclear Overpower – High Setpoint;

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

11. Shutdown Bypass RCS High Pressure (continued)

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower Flux/Flow Imbalance.

The Shutdown Bypass RCS High Pressure Function's Allowable Value is selected to ensure a trip occurs before producing THERMAL POWER.

General Discussion

The RPS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 8).

In MODES 1 and 2, the following trips shall be OPERABLE because the reactor can be critical in these MODES. These trips are designed to take the reactor subcritical to maintain the SLs during anticipated transients and to assist the ESPS in providing acceptable consequences during accidents.

1a. Nuclear Overpower – High Setpoint;

2. RCS High Outlet Temperature;

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

6. Reactor Building High Pressure;

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower Flux/Flow Imbalance.

Functions 1, 3, 4, 5, 7, and 8 just listed may be bypassed in MODE 2 when RCS pressure is below 1720 psig, provided the Shutdown Bypass RCS High Pressure and the Nuclear Overpower – Low setpoint trip are placed in operation. Under these conditions, the Shutdown Bypass RCS High Pressure trip and the Nuclear Overpower – Low setpoint trip act to prevent unit conditions from reaching a point where actuation of these Functions is necessary.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

General Discussion (continued)

The Main Turbine Trip (Hydraulic Fluid Pressure) Function is required to be OPERABLE in MODE 1 at ≥ 30% RTP. The Loss of Main Feedwater Pumps (Hydraulic Oil Pressure) Function is required to be OPERABLE in MODE 1 and in MODE 2 at ≥ 2% RTP. Analyses presented in BAW-1893 (Ref. 6) have shown that for operation below these power levels, these trips are not necessary to minimize challenges to the PORVs as required by NUREG-0737 (Ref. 5).

Because the safety function of the RPS is to trip the CONTROL RODS, the RPS is not required to be OPERABLE in MODE 3, 4, or 5 if either the reactor trip breakers are open, or the CRD System is incapable of rod withdrawal. Similarly, the RPS is not required to be OPERABLE in MODE 6 because the CONTROL RODS are normally decoupled from the CRDs.

However, in MODE 2, 3, 4, or 5, the Shutdown Bypass RCS High Pressure and Nuclear Overpower – Low setpoint trips are required to be OPERABLE if the CRD trip breakers are closed and the CRD System is capable of rod withdrawal. Under these conditions, the Shutdown Bypass RCS High Pressure and Nuclear Overpower – Low setpoint trips are sufficient to prevent an approach to conditions that could challenge SLs.

ACTIONS

Conditions A and B are applicable to all RPS protective Functions. If a channel's trip setpoint is found nonconservative with respect to the required Allowable Value in Table 3.3.1-1, or the transmitter, instrument loop, signal processing electronics or bistable is found inoperable, the channel must be declared inoperable and Condition A entered immediately.

A.1

For Required Action A.1, if one or more Functions in a required protective channel becomes inoperable, the affected protective channel must be placed in trip. This Required Action places all RPS Functions in a one-out-of-two logic configuration. The "non-required" channel is placed in bypass when the required inoperable

(continued)

BASES

ACTIONS

A.1 (continued)

channel is placed in trip to prevent bypass of a second required channel. In this configuration, the RPS can still perform its safety functions in the presence of a random failure of any single Channel. The 1 hour Completion Time is sufficient time to perform Required Action A.1.

B.1

Required Action B.1 directs entry into the appropriate Condition referenced in Table 3.3.1-1. The applicable Condition referenced in the table is Function dependent. If the Required Action and the associated Completion Time of Condition A are not met or if more than two channels are inoperable, Condition B is entered to provide for transfer to the appropriate subsequent Condition.

C.1 and C.2

If the Required Action and associated Completion Time of Condition A are not met and Table 3.3.1-1 directs entry into Condition C, the unit must be brought to a MODE in which the specified RPS trip Functions are not required to be OPERABLE. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and to open all CRD trip breakers without challenging unit systems.

D.1

If the Required Action and associated Completion Time of Condition A are not met and Table 3.3.1-1 directs entry into Condition D, the unit must be brought to a MODE in which the specified RPS trip Functions are not required to be OPERABLE. To achieve this status, all CRD trip breakers must be opened. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open CRD trip breakers without challenging unit systems.

(continued)

BASES

ACTIONS (continued)

E.1

If the Required Action and associated Completion Time of Condition A are not met and Table 3.3.1-1 directs entry into Condition E, the unit must be brought to a MODE in which the specified RPS trip Function is not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced < 30% RTP. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach 30% RTP from full power conditions in an orderly manner without challenging unit systems.

F.1

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

SURVEILLANCE REQUIREMENTS

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

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

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1 (continued)

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

Agreement criteria are determined based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

The Frequency, equivalent to once every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal but more frequent checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels.

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

SR 3.3.1.2

This SR is the performance of a heat balance calibration for the power range channels every 24 hours when reactor power is > 15% RTP. The heat balance calibration consists of a comparison of the results of the calorimetric with the power range channel output. The outputs of the power range

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued)

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

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

SR 3.3.1.3

A comparison of power range nuclear instrumentation channels against incore detectors shall be performed at a 31 day Frequency when reactor power is ≥ 15% RTP. A Note clarifies that 24 hours is allowed for performing the first Surveillance after reaching 15% RTP. If the absolute difference between the power range and incore measurements is ≥ 2% RTP, the power range channel is not inoperable, but an adjustment of the measured imbalance to agree with the incore measurements is necessary. If the power range channel cannot be properly recalibrated, the channel is declared inoperable. The calculation of the Allowable Value envelope assumes a difference in out of core to incore measurements of 2.5%. Additional inaccuracies beyond those that are measured are also included in the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.3 (continued)

setpoint envelope calculation. The 31 day Frequency is adequate, considering that long term drift of the excore linear amplifiers is small and burnup of the detectors is slow. Also, the excore readings are a strong function of the power produced in the peripheral fuel bundles, and do not represent an integrated reading across the core. The slow changes in neutron flux during the fuel cycle can also be detected at this interval.

SR 3.3.1.4

A CHANNEL FUNCTIONAL TEST is performed on each required RPS channel to ensure that the entire channel will perform the intended function. Setpoints must be found within the Allowable Values specified in Table 3.3.1-1. Any setpoint adjustment shall be consistent with the assumptions of the current setpoint analysis.

The as found and as left values must also be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis. The requirements for this review are outlined in BAW-10167 (Ref. 7).

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

SR 3.3.1.5

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

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.5 (continued)

ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature detectors (RTD)sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

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

REFERENCES

1. UFSAR, Chapter 7.
2. UFSAR, Chapter 15.
3. 10 CFR 50.49.
4. EDM-102, "Instrument Setpoint/Uncertainty Calculations."
5. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1979.
6. BAW-1893, "Basis for Raising Arming Threshold for Anticipating Reactor Trip on Turbine Trip," October 1985.
7. BAW-10167, May 1986.
8. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.15 Turbine Stop Valve (TSV) Closure

BASES

BACKGROUND

The Turbine Stop Valves (TSV) Closure function partially isolates the main steam lines from the SGs by closing the TSVs on both main steam lines following a turbine or reactor trip signal.

Two TSVs are provided for each main steam line and are located outside of containment. The TSVs are downstream from the main steam safety valves (MSSVs) and emergency feedwater pump turbine's steam supply to prevent the MSSVs and EFW pump's steam supply from being isolated from the steam generators by TSV closure. Closing the TSVs partially isolates each steam generator from the other, and isolates the turbine from the steam generators.

TSV Closure is initiated by a reactor trip. To keep from rapidly cooling down the primary plant by drawing off too much steam, the turbine is tripped when the reactor trips. Two independent and redundant "Reactor Trip Confirmed" signals in the form of contact closures from the control rod drive system will energize two independent turbine trip mechanisms. The Channel A trip circuit will close all four TSVs within a maximum of 1 second. The Channel B trip circuit will close the TSVs within a maximum of 15 seconds.

APPLICABLE SAFETY ANALYSES

The design basis of the TSV Closure function is established by the analysis for the main steam line break (MSLB) as discussed in the UFSAR, Section 15.13 (Ref. 1). TSV closure is necessary to stop steam flow to the turbine (to prevent overcooling) following all reactor trips.

The accident analysis compares several different MSLB events. The MSLB outside containment upstream of the TSV is limiting for offsite dose, although a break in this section of main steam header has a very low probability. The MSLB without ICS and without operator action is the limiting case for a post trip return to power for Units not analyzed in accordance with DPC-NE-3005-PA. For Units analyzed in

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

accordance with DPC-NE-3005-PA, the MSLB with ICS low level control and without operator action prior to ten minutes is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available and with a loss of offsite power. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the High Pressure Injection (HPI) System pumps, is delayed.

The TSVs remain open during power operation. These valves close upon a reactor trip or turbine trip signal.

a. For an HELB or an MSLB inside containment, steam is discharged into containment from both steam generators until closure of the TSVs. After TSV closure, steam is discharged into containment only from the affected steam generator.

b. An MSLB outside of containment and upstream from the TSVs is not a containment pressurization concern. The uncontrolled blowdown of both steam generators must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the TSVs isolates the break and limits the blowdown to a single steam generator.

c. An event such as increased steam flow through the turbine will terminate on closing the TSVs.

d. Following a steam generator tube rupture, closure of the TSVs isolates the ruptured steam generator from the intact steam generator.

The TSV Closure function satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

(continued)

BASES (continued)

LCO

Two TSV Closure channels are required to be OPERABLE.

This LCO provides assurance that the TSVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 limits (Ref. 3).

APPLICABILITY

Both TSV Closure channels must be OPERABLE in MODES 1, 2 and 3 with any TSVs open. In these conditions when there is significant mass and energy in the RCS and steam generators, the TSV Closure function must be OPERABLE or the TSVs closed. When the TSVs are closed, they are already performing the safety function.

In MODE 4, the steam generator energy is low. Therefore, the TSV Closure channels are not required to be OPERABLE. In MODES 5 and 6, the steam generators do not contain a significant amount of energy because their temperature is below the boiling point of water; therefore, the TSV Closure channels are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one or more TSV Closure channels inoperable, all TSVs must be declared inoperable. A Completion Time of 1 hour is provided to return the TSV Closure channels to OPERABLE status. The 1 hour Completion Time is sufficient time to correct minor problems.

SURVEILLANCE REQUIREMENTS

SR 3.3.15.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST to ensure that the channels can perform their intended function. This test verifies the TSV Closure automatic actuation channels are functional. This test simulates the required inputs to the logic circuit and verifies successful operation of the automatic actuation logic channels. The

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.3.15.1 (continued) test need not include actuation of the end device. This is due to the risk of a unit transient caused by the closure of TSVs during testing at power. The Frequency of 31 days is based on engineering judgment and operating experience, which determined the interval provided adequate confidence that the TSV Closure channels are available to perform their safety function, while the risks of testing at operation are avoided.
REFERENCES	1. UFSAR, Section 15.13. 2. 10 CFR 50.36. 3. 10 CFR 100.

BASES

LCO (continued)

operates within the limits assumed for the plant safety analyses. Operating within these limits will result in meeting DNBR criteria in the event of a DNB limited transient.

The pressure limits are applied to the loop with the highest pressure. The temperature limits are applied to the loop with the lowest loop average temperature for the condition in which there is a 0°F ΔTc setpoint.

APPLICABILITY

In MODE 1 during steady state operation, the limits on RCS loop pressure, RCS loop average temperature, and RCS flow rate must be maintained 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. Steady state operation, for the purposes of this specification, is defined as operation within a 4% (e.g., 88% - 92% RTP) power band for ≥ 4 hours.

ACTIONS

A.1

Loop pressure and loop average coolant temperature are controllable and measurable parameters. With one or both of these parameters not within the LCO limits, action must be taken to restore the parameters. RCS flow rate is not a controllable parameter and is not expected to vary during steady state four pump or three pump operation. However, if the flow rate is below the LCO limit, the parameter must be restored to within limits or power must be reduced as required in Required Action B.1, to restore DNBR margin and eliminate the potential for violation of the accident analysis bounds. The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust unit parameters, determine the cause for the off normal condition, and restore the readings within limits. The Completion Time is based on plant operating experience.

(continued)

BASES

ACTIONS (continued)

B.1

If the Required Action and associated Completion Time are not met, 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 2 within 12 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds.

The 12 hour Completion Time is reasonable, based on operating experience, to reduce power in an orderly manner.

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 in the COLR 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 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 for three pumps operating is applied to the loop with the highest pressure.

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 loop average 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. A Note has been added to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.2 (continued)

indicate the temperature limits for three pumps operating are applied to the loop with the lowest loop average temperature for the condition in which there is a 0°F ΔTc setpoint.

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 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 specified in the COLR.

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. The Surveillance is modified by a Note that indicates the SR does not need to be performed until 7 days after 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 may be too small to provide meaningful test results.

REFERENCES

1. UFSAR, Chapter 15.
2. 10 CFR 50.36

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. For the remainder of MODE 3, MODE 4, 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."

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 setpoint of the pressurizer code safety valves is in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1967. 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.

For Units analyzed in accordance with DPC-NE-3005-PA, the required lift pressure is 2500 psig ± 3%. For Units not analyzed in accordance with DPC-NE-3005-PA, the required lift pressure is 2500 psig ± 1%. The upper and lower pressure limits are based on the requirements of ASME Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1967, which limit the rise in pressure within the vessels which they protect to 10% above the design pressure. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

(continued)

BASES

BACKGROUND (continued)	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 UFSAR that require safety valve actuation assume operation of both pressurizer safety valves to limit increasing reactor coolant pressure. The overpressure protection analysis is also based on operation of both safety valves and assumes that the valves open at the high range of the setting (2500 psig system design pressure plus 3% for Units analyzed in accordance with DPC-NE-3005-PA, and 2500 psig plus 1% for Units not analyzed in accordance with DPC-NE-3005-PA). These valves must accommodate pressurizer insurges that could occur during a startup, rod withdrawal, ejected rod, or loss of main feedwater. The startup accident establishes the minimum safety valve capacity. The startup accident is assumed to occur at < 15% power. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this Specification is required to ensure that the accident analysis and design basis calculations remain valid. Pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).
LCO	The two pressurizer safety valves are set to open at the RCS design pressure (2500 psig) and within the ASME specified tolerance to avoid exceeding the maximum RCS design pressure SL, to maintain accident analysis assumptions and to comply with ASME Code requirements. The valves will be tested per ASME Section XI requirements and returned to service with as-left setpoints of 2500 psig ± 1%. The upper and lower pressure tolerance limits are based on the requirements of the ASME Boiler and Pressure Vessel Code, Section III,

(continued)

BASES

LCO (continued)

Article 9, Summer 1967, which limit the rise in pressure within the vessel which they protect, to 10% above the design pressure. Inoperability of one or both valves could result in exceeding the SL if a transient were to occur.

The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and portions of MODE 3 above the LTOP cut in temperature, OPERABILITY of two valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. Portions of MODE 3 are conservatively included, although the listed accidents may not require both safety valves for protection.

The LCO is not applicable in MODE 3 when any RCS cold leg temperature is ≤ 325°F, MODE 4 and MODE 5 because LTOP protection is provided. Overpressure protection is not required in MODE 6 with the reactor vessel head detensioned.

The Note allows entry into MODE 3 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 36 hour exception is based on an 18 hour outage time for each of the two valves. The 18 hour period is derived from operating experience that hot testing can be performed in this time frame.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS

(continued)

BASES

ACTIONS

A.1 (continued)

overpressure protection system. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the RCPB.

B.1 and B.2

If the Required Action cannot be met within the required Completion Time or if both pressurizer safety valves are inoperable, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and to MODE 3 with any RCS cold leg temperature ≤ 325°F within 18 hours. The 12 hours allowed is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. Similarly, the 18 hours allowed is reasonable, based on operating experience, to reach MODE 3 with any RCS cold leg temperature ≤ 325°F without challenging unit systems. With any RCS cold leg temperature at or below 325°F, overpressure protection is provided by LTOP. Reducing the RCS temperature to ≤ 325°F reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by two pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 2), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valves setpoint is ± 3% for OPERABILITY for Units analyzed in accordance with DPC-NE-3005-PA; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

(continued)

BASES (continued)

REFERENCES	1.	ASME, Boiler and Pressure Vessel Code, Section III.
	2.	ASME, Boiler and Pressure Vessel Code, Section XI.
	3.	10 CFR 50.36.

BASES

APPLICABLE SAFETY ANALYSES (continued)

return to power for Units not analyzed in accordance with DPC-NE-3005-PA. For Units analyzed in accordance with DPC-NE-3005-PA, the MSLB with ICS low level control and no operator action prior to ten minutes is the limiting case for a post rip return to power. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the High Pressure Injection (HPI) System pumps, is delayed.

The TSVs remain open during power operation. These valves close upon a turbine trip or reactor trip signal.

a. For an HELB or an SLB inside containment, steam is discharged into containment from both steam generators until closure of the TSVs. After TSV closure, steam is discharged into containment only from the affected steam generator.

b. An MSLB outside of containment and upstream from the TSVs is not a containment pressurization concern. The uncontrolled blowdown of both steam generators must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the TSVs isolates the break and limits the blowdown to a single steam generator.

c. Steam flow to the turbine if not controlled by the turbine control valves will terminate on closing the TSVs.

d. Following a steam generator tube rupture, closure of the TSVs isolates the ruptured steam generator from the intact steam generator.

The TSVs satisfy Criterion 3 of 10 CFR 50.36, (Ref. 3).

LCO

This LCO requires that the two TSVs in each steam line be OPERABLE. The TSVs are considered OPERABLE when the isolation times are within limits and they close on an isolation actuation signal.

This LCO provides assurance that the TSVs will perform their design safety function to mitigate the consequences of

(continued)

BASES

LCO (continued)	accidents that could result in offsite exposures comparable to the 10 CFR 100 limits (Ref. 4).
APPLICABILITY	The TSVs must be OPERABLE in MODES 1, 2 and 3 with any TSVs open. In these conditions when there is significant mass and energy in the RCS and steam generators, the TSVs must be OPERABLE or closed. When the TSVs are closed, they are already performing the safety function. In MODE 4, the steam generator energy is low. Therefore, the TSVs are not required to be OPERABLE. In MODES 5 and 6, the steam generators do not contain a significant amount of energy because their temperature is below the boiling point of water; therefore, the TSVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.
ACTIONS	<u>A.1</u> With one or both TSVs for one main steam line inoperable in MODE 1, action must be taken to restore the components to OPERABLE status within 8 hours. Some repairs can be made to the TSV with the unit hot. The 8 hour Completion Time is reasonable, considering the probability of an accident that would require actuation of the TSVs occurring during this time interval. The turbine control valves may be available to provide the isolation for the postulated accidents although control valve response is not as rapid. The Completion Time is reasonable because the TSVs isolate a closed system which provides an additional barrier against releases. <u>B.1</u> If the TSVs cannot be restored to OPERABLE status within 12 hours, the unit must be placed in MODE 2 and the inoperable TSVs closed within the next 6 hours. The Completion Time is reasonable, based on operating experience, to reach MODE 2.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Dump Valve (ADV) Flow Paths

BASES

BACKGROUND

The ADV flow paths provide a method for cooling the unit to decay heat removal (DHR) entry conditions, should the preferred heat sink via the Turbine Bypass System to the condenser not be available, as discussed in the UFSAR (Ref. 2). This is done in conjunction with the secondary cooling water from the Emergency Feedwater (EFW) System.

The ADV flow path for each steam generator is credited as a compensatory measure in Actions B and C of LCO 3.5.2, "High Pressure Injection (HPI)," to permit operation to continue with THERMAL POWER ≤ 75% RTP: a) for 30 days with an HPI pump or HPI discharge crossover valve(s) inoperable; and b) for 72 hours with an HPI train inoperable. During these periods of time, the ADV flow path for one steam generator is credited to depressurize the steam generator and enhance primary-to-secondary heat transfer during certain small break loss of coolant accidents (LOCAs).

Additionally, for Units that have been analyzed in accordance with DPC-NE-3005-PA (Ref. 6), the steam generator tube rupture (SGTR) analysis (Ref. 3) credits operator action to depressurize the steam generators by opening each of the ADV flow paths.

For each steam generator, the ADV flow path is comprised of the atmospheric dump block valve bypass (1″ bypass), the atmospheric vent valve (a 12″ block valve), the atmospheric dump control valve (i.e., throttle valve), and the atmospheric vent block valve (i.e., isolation valve). The throttle valve and the isolation valve are in parallel and are located downstream of the atmospheric vent valve.

The atmospheric vent valve should be opened prior to opening the throttle valve or isolation valve. This is accomplished by first opening the atmospheric dump block valve bypass.

(continued)

BASES

BACKGROUND (continued)

This equalizes the differential pressure across the atmospheric vent valve. Once the atmospheric vent valve is opened, the cool down rate is controlled using the throttle valve. If additional relief capacity is needed, the isolation valve can be opened. The capacity of the throttle or isolation valve exceeds decay heat loads and is sufficient to cool down the plant.

APPLICABLE SAFETY ANALYSIS

Operator action to depressurize a steam generator via its ADV flow path is credited in the analysis of certain small break LOCAs with THERMAL POWER ≤ 75% RTP and the plant operated with a degraded HPI System in accordance with Condition B or Condition C of ITS 3.5.2, "High Pressure Injection (HPI) System" (Ref. 4). This event credits operator action to open one ADV flow path within 25 minutes of an ESPS actuation.

If enhanced steam generator cooling is not credited in the small break LOCA analysis, two HPI trains are required to mitigate specific small break LOCAs. However, if equipment not qualified as QA-1 (i.e., an ADV flow path for a steam generator) is credited for enhanced steam generator cooling, the safety analyses have determined that the capacity of one HPI train is sufficient to mitigate a small break LOCA on the discharge of the reactor coolant pumps if THERMAL POWER is ≤ 75% RTP.

The analysis for Action C of LCO 3.5.2, "High Pressure Injection (HPI)," credits an ADV flow path for one steam generator as a compensatory measure in the event an HPI train is inoperable and THERMAL POWER is ≤ 75% RTP. During this situation, the ADV flow path for one steam generator is credited during certain small break LOCAs to depressurize the steam generator and enhance primary-to-secondary heat transfer. This is done in conjunction with the EFW System providing cooling water to the steam generator. The ADV flow path is comprised of manual valves. Operator action is credited within 25 minutes of an Engineered Safeguards Protective System (ESPS) signal to open them.

Additionally, the ADV flow path for each steam generator is credited as a compensatory measure in the analysis for Action B of LCO 3.5.2, "High Pressure Injection (HPI)," to permit an HPI pump or HPI discharge crossover valve(s) to be

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

inoperable for 30 days with the THERMAL POWER ≤ 75% RTP. Typically, single failures are not considered once the plant has entered a condition defined in the Technical Specifications. However, the Completion Time permitted by Required Actions B.3 and B.4 of LCO 3.5.2, "High Pressure Injection (HPI)," is an extended period of time (i.e., 30 days). In the event an accident occurred during this 30-day Completion Time and a single failure were to occur in the degraded HPI System, the ability of a plant to mitigate the consequences of specific small break LOCAs continues to be assured by the ADV flow path for one steam generator.

For Units that have been analyzed in accordance with DPC-NE-3005-PA (Ref. 6), the SGTR analysis credits operator action to depressurize the steam generators by opening both ADV flow paths (i.e., the ADV flow path for each steam generator) within 40 minutes of identifying the ruptured steam generator. Within this 40-minute time period, the operators are only required to open the bypass valve, the block valve, and the throttle valve. However, later in the event, the analysis also assumes that the operators will open the isolation valves in each ADV flow path.

The ADV flow paths satisfy Criterion 3 of 10 CFR 50.36 (Ref.1).

LCO

The ADV flow path for each steam generator is required to be OPERABLE. Failure to meet the LCO can result in the inability to depressurize a steam generator following a small break LOCA. This function is required to support operation with a degraded HPI System when THERMAL POWER is ≤ 75% RTP.

Additionally, for Units analyzed in accordance with DPC-NE-3005-PA (Ref. 6), failure to meet the LCO can result in the inability to depressurize the steam generators following a SGTR.

(continued)

BASES

LCO (continued)

An ADV flow path is considered OPERABLE when it is capable of providing a controlled relief of the main steam flow, and each valve which comprises the ADV flow path is capable of opening and closing.

APPLICABILITY

For Units analyzed in accordance with DPC-NE-3005-PA (Ref. 6), the ADV flow path for each steam generator is required to be OPERABLE in MODES 1, 2, and 3, and in MODE 4, when a steam generator is being relied upon for heat removal. In MODE 4, steam generators are relied upon for heat removal whenever an RCS loop is required to be OPERABLE or operating to satisfy LCO 3.4.5, "RCS Loops - MODE 4" or available to transfer decay heat to satisfy LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled." The steam generators do not contain a significant amount of energy in MODE 4 when the unit is not relying upon a steam generator for heat transfer, and MODES 5 and 6; therefore, the ADV flow paths are not required to be OPERABLE in these MODES and condition.

For Units not analyzed in accordance with DPC-NE-3005-PA, the ADV flow path for each steam generator is only required to be OPERABLE when required by Required Actions B.2 and C.2 of LCO 3.5.2, "High Pressure Injection (HPI)." For all other conditions, the ADV flow paths for these Units are not credited in the analyses of any accident.

ACTIONS

A.1 and A.2

With one or both of the required ADV flow path(s) inoperable in a Unit that has not been analyzed in accordance with DPC-NE-3005-PA, the unit must be placed in a condition in which the LCO does not apply. The ADV flow path for each steam generator is required to support operation with a degraded HPI System. Thus, the unit must be placed in a condition outside the Applicability of LCO 3.5.2, "High Pressure Injection (HPI)." To achieve this status, the unit must be placed in at least MODE 3 within 12 hours, and RCS temperature reduced to ≤ 350°F within 60 hours. The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

challenging unit systems. They are consistent with the Completion Times provided in Required Actions G.1 and G.2 of LCO 3.5.2, "High Pressure Injection (HPI)."

B.1 and B.2

With one or both of the ADV flow path(s) inoperable in a Unit analyzed in accordance with DPC-NE-3005-PA, the Unit must be placed in a condition in which the LCO does not apply. To achieve this status, the Unit must be paced in at least MODE 3 within 12 hours, and at least MODE 4 without reliance on a steam generator for heat removal within 24 hours. The Completion Times are reasonable, based on operating experience, to reach the required Unit conditions from full power conditions in an orderly manner and without challenging Unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a controlled cool down of the RCS, the valves which comprise the ADV flow path for each steam generator must be able to perform the following functions:

a) the atmospheric dump block valve bypass and the atmospheric vent valve must be capable of being opened and closed; and

b) the atmospheric dump control valve and atmospheric vent block valve must be capable of being opened and throttled through their full range.

This SR ensures that the valves which comprise the ADV flow path for each steam generator are tested through a full control cycle at least once per 18 months. Performance of inservice testing or use of an ADV flow path during a unit cool down may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50.36.
2. UFSAR, Section 10.3.
3. UFSAR, Section 15.9.
4. UFSAR, Section 15.14.
5. UFSAR, Section 15.12.
6. DPC-NE-3005-PA, "UFSAR Chapter 15 Transient Analysis Methodology," Duke Power Company, Oconee Nuclear Station, January 1999.

B 3.9 REFUELING OPERATIONS

B 3.9.7 Unborated Water Source Isolation Valves

BASES

BACKGROUND

During MODE 6 operations, all isolation valves for reactor makeup water sources containing unborated water that are connected to the Reactor Coolant System (RCS) must be closed to prevent unplanned boron dilution of the reactor coolant. The isolation valves must be secured in the closed position.

The Coolant Storage System is capable of supplying borated and unborated water to the RCS through various flow paths. Since a positive reactivity addition made by reducing the boron concentration is inappropriate during MODE 6, isolation of all unborated water sources prevents an unplanned boron dilution.

APPLICABLE SAFETY ANALYSES

The possibility of an inadvertent boron dilution event (Ref. 1) occurring during MODE 6 refueling operations is precluded by adherence to this LCO, which requires that potential dilution sources be isolated. Closing the required valves during refueling operations prevents the flow of unborated water to the filled portion of the RCS. The valves are used to isolate unborated water sources. These valves have the potential to indirectly allow dilution of the RCS boron concentration in MODE 6. By isolating unborated water sources when in MODE 6, a boron dilution event as analyzed in the UFSAR is prevented.

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2).

LCO

This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent unplanned boron dilution during MODE 6 and thus avoid a reduction in SDM.

(continued)

BASES (continued)

APPLICABILITY

In MODE 6, this LCO is applicable to prevent an inadvertent boron dilution event by ensuring isolation of all sources of unborated water to the RCS.

For all other applicable MODES, the boron dilution accident was analyzed and was found to be capable of being mitigated. The boron dilution event is applicable in MODES 1 and 6.

ACTIONS

The ACTIONS table has been modified by a Note that allows separate Condition entry for each unborated water source isolation valve.

A.1

Continuation of CORE ALTERATIONS is contingent upon maintaining the unit in compliance with this LCO. With any valve used to isolate unborated water sources not secured in the closed position, all operations involving CORE ALTERATIONS must be suspended immediately. The Completion Time of "immediately" for performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

Condition A has been modified by a Note to require that Required Action A.3 be completed whenever Condition A is entered.

A.2

Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the unborated water isolation valves secured closed. Securing the valves in the closed position ensures that the valves cannot be inadvertently opened. The Completion Time of "immediately" requires an operator to initiate actions to close an open valve and secure the isolation valve in the closed position immediately. Once actions are initiated, they must be continued until the valves are secured in the closed position.

(continued)

BASES

ACTIONS (continued)

A.3

Due to the potential of having diluted the boron concentration of the reactor coolant, SR 3.9.1.1 (verification of boron concentration) must be performed whenever Condition A is entered to demonstrate that the required boron concentration exists. The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

These valves are to be secured closed to isolate possible dilution paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote due to the large mass of borated water in the fuel transfer canal and the fact that all unborated water sources are isolated, precluding a dilution. The boron concentration is checked every 72 hours during MODE 6 under SR 3.9.1.1. This Surveillance demonstrates that the valves are closed through a system walkdown. The 31 day Frequency is based on engineering judgment and is considered reasonable in view of other administrative controls that will ensure that the valve opening is an unlikely possibility.

REFERENCES

1. UFSAR, Section 15.4.1.

2. 10 CFR 50.36.

ATTACHMENT III

DESCRIPTION OF PROPOSED CHANGES AND JUSTIFICATION FOR PROPOSED CHANGES

ATTACHMENT III

DESCRIPTION OF THE PROPOSED CHANGES AND TECHNICAL JUSTIFICATIONS FOR THE PROPOSED CHANGES

On July 30, 1997, Duke submitted Topical Report DPC-NE-3005-P, "UFSAR Chapter 15 Transient Analysis Methodology," to the NRC for review and approval. The topical report describes the new methodology for analyzing the UFSAR Chapter 15 non-LOCA transients and accidents. Its objective was to provide a modern analysis to replace the non-LOCA analyses presented in Chapter 15 of the Oconee UFSAR. The NRC approved the topical report, with some exceptions, in a Safety Evaluation Report issued on October 1, 1998. To resolve these exceptions, Duke submitted Revision 1 to DPC-NE-3005-P to the NRC on February 1, 1999.

In addition, on December 23, 1997, Duke submitted Revision 2 of Topical Report DPC-NE-3000-P, "Thermal-Hydraulic Transient Analysis Methodology," to the NRC for review and approval. The analytical methodology of Topical Report DPC-NE-3005-P required minor revisions to the simulation models previously approved by the NRC in Revision 1 of DPC-NE-3000-PA. The revisions to the simulation models were required to reflect the new Mk-B11 fuel assembly design, application of the new critical heat flux correlation (i.e., BWU-Z with the Mk-B11V multiplier), and several RETRAN model improvements. The NRC approved the topical report in a Safety Evaluation Report issued on October 14, 1998.

Currently, the methodology provided in DPC-NE-3005-P has only been utilized in the accident analyses for Oconee Unit 2. For Units 1 and 3, the revisions to the accident analyses are planned to be complete by November 15, 1999. However, to provide flexibility, the proposed Technical Specifications have been developed assuming that the re-analyses are only complete for one or two units. Drafting the Technical Specifications in this manner will minimize the number of license amendments required to implement DPC-NE-3005-P. Additionally, in the event the revised accident analyses are completed for each unit prior to this LAR being approved by the NRC, Duke will submit revised Technical Specifications to remove the material germane to the previous accident analyses.

This License Amendment Request proposes changes to the Oconee Technical Specifications, associated Bases, UFSAR, and Core Operating Limits Report (COLR) to implement Topical Report DPC-NE-3005-PA. These proposed changes are dependent upon NRC approval of Revision 1 to DPC-NE-3005-P.

Proposed Changes to Technical Specifications and Bases

Mark-ups of the proposed changes to the Oconee Technical Specifications and Bases are provided in Attachment I. Retyped pages of the Oconee Technical Specifications and Bases are provided in Attachment II.

<u>Proposed Changes to ITS 3.4.1 and Associated Bases</u>

Duke proposes to modify the Note to SR 3.4.1.2 by identifying that the delta-Tcold (ΔTc) limits are applied to the loop with the lowest average temperature when there is a 0°F ΔTc setpoint dialed into the ICS. In addition, Duke proposes to revise the Bases discussions for LCO 3.4.1 and SR 3.4.1.2 to reflect the change.

When operating with three reactor coolant pumps, the NOTE to SR 3.4.1.2 allows the use of the lowest loop average temperature when comparing against the Tavg limit. The proposed change clarifies that the NOTE is valid only for operation with a ΔTc setpoint of 0°F dialed into the ICS. With units analyzed in accordance with DPC-NE-3005-PA, ΔTc operation of up to 5°F will be allowed. Analyses are underway to justify a 5°F ΔTc provided the maximum loop average temperature remains below the limits specified in the COLR (the values presently listed are typical). To account for a 5°F ΔTc, the limiting DNB peaking limits have been appropriately penalized for units analyzed in accordance with DPC-NE-3005-PA.

Proposed Changes to ITS 3.4.10 and Associated Bases

Duke proposes to modify ITS 3.4.10 by:

1. increasing the range of lift settings for the pressurizer safety valves (PSV), for units analyzed in accordance with DPC-NE-3005-PA, from ±1% (i.e., 2525 psig ≥ PSV lift settings ≥ 2475 psig) to ±3% (i.e., 2575 psig ≥ PSV lift settings ≥ 2425 psig);

2. denoting that for units not analyzed in accordance with DPC-NE-3005-PA the acceptance criteria for PSV lift settings remain at ±1% (i.e., 2525 psig ≥ PSV lift settings ≥ 2475 psig); and

3. adding a requirement to SR 3.4.10.1 to ensure that the PSV lift settings are within ±1% following testing.

In addition, corresponding changes are proposed to be made to the Bases for ITS 3.4.10.

The pressurizer code safeties are not tested in place but are removed and shipped to a testing facility. The safety concerns for removal and replacement of these valves are 1) difficult access to the work area, 2) difficulty in handling the lifting device rigging for valve removal/replacement, and 3) valve transport to/from the pressurizer. This work is performed in a radiological environment, thus, the work activities are complicated by anti-contamination clothing. For a conservative approach, both safety valves are currently removed each outage for testing. The setpoint drift seen during testing would again fall under the proposed setpoint variance change. The change would possibly reduce work in the reactor building by requiring only one valve to be tested per outage. In summary, safety benefits would be gained by less work in a radiological environment; this is consistent with ALARA principles.

The larger allowable deviation from the nominal lift setting is consistent with the proposed licensing basis analyses for units analyzed in accordance with DPC-NE-3005-

PA. Increasing the pressurizer safety valve lift setpoint impacts the peak Reactor Coolant System pressure calculated for pressure increase transients. A pressure increase is the result of a heatup in the Reactor Coolant System due to a mismatch between the heat generated in the reactor core and the heat removed by the secondary system. The startup accident and rod ejection transients generate the most severe pressure increases. These two limiting events have been analyzed assuming a lift setpoint of 3 percent above the nominal value. These analyses show that the peak Reactor Coolant System pressure criterion is met for each event.

The safety valve lift setpoint is allowed by the revised Technical Specifications to drift downward to -3%; this ensures that safety valve lift cannot preclude reactor trip on high Reactor Coolant System pressure. The nominal high pressure reactor trip setpoint is 2355 psig. This is well below the proposed lower lift setting of 2425 psig. For DNB transients in which a high pressure reactor trip does not prevent the lifting of the safety valves, the effect of this reduced setpoint on the transient DNBR is evaluated. Events satisfying this condition include the uncontrolled rod withdrawal from power and rod ejection transients. The analyses show that the DNB acceptance criteria are satisfied for both of these events.

In addition, SR 3.4.10.1 has been changed to require the PSV lift settings to be within ±1% following testing. This requirement allows for drift.

Proposed Changes to ITS 3.7.4 and Associated Bases

Duke proposed to add ITS 3.7.4, "Atmospheric Dump Valve (ADV) Flow Paths," in a License Amendment Request submitted to the NRC on December 16, 1998. Duke proposes the following changes to ITS 3.7.4:

1. Expand the Applicability for ITS 3.7.4 to Modes 1, 2, and 3, and Mode 4 when a steam generator is relied upon for heat removal for units that have been analyzed in accordance with DPC-NE-3005-PA. Previously, ITS 3.7.4

was only applicable when required by Required Actions B.2 and C.2 of LCO 3.5.2;

2. Denote the Applicability for ITS 3.7.4 remains only when required by Required Actions B.2 and C.2 of LCO 3.5.2 for units that have not been analyzed in accordance with DPC-NE-3005-PA;

3. Add Action B to require the plant be placed in Mode 3 within 12 hours and in Mode 4 without reliance upon a steam generator for heat removal within 24 hours when one or both ADV flow path(s) are inoperable in a unit analyzed in accordance with DPC-NE-3005-PA; and

4. Modify Condition A to identify that it only pertains to units not analyzed in accordance with DPC-NE-3005-PA.

Corresponding changes were made to the Bases for ITS 3.7.4.

For Units that have been analyzed in accordance with DPC-NE-3005-PA, the steam generator tube rupture (SGTR) analysis credits operator action to depressurize the steam generators by opening both ADV flow paths (i.e., the ADV flow path for each steam generator). These proposed changes reflect the fact that both ADV flow paths are credited during a plant cooldown following a SGTR event for units analyzed in accordance with DPC-NE-3005-PA. Crediting both ADV flow paths for the SGTR event speeds the rate of plant cool down and limits the offsite dose consequences, as documented in DPC-NE-3005-PA.

With one or both of the ADV flow path(s) inoperable in a Unit analyzed in accordance with DPC-NE-3005-PA, the unit must be placed in a condition in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours, and at least MODE 4 without reliance on a steam generator for heat removal within 24 hours. The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

Proposed Changes to ITS 3.9.7

Duke proposes to add ITS 3.9.7, "Unborated Water Source Isolation Valves." This specification will require each valve that is used to isolate unborated water sources to be secured in the closed position while in Mode 6 to prevent unplanned boron dilution during MODE 6 and thus avoid a reduction in shutdown margin. SR 3.9.7.1 ensures that the requirements of the LCO are met by verifying that the subject valves are secured in position every 31 days. The following actions are required in the event one or more of these valves are discovered not secured in the closed position: 1) core alterations suspended immediately; 2) actions initiated to secure the valve in the closed position immediately; and 3) SR 3.9.1.1 performed within 4 hours.

Proposed Changes to ITS 5.6.5b

Duke proposes to update the COLR References in ITS 5.6.5b. These changes reflect the most up-to-date analytical methods used to determine the core operating limits. The referenced topical reports are NRC approved or pending approval. Where approved, a reference to the NRC SER has been added.

Proposed Changes to the Bases for ITS 3.1.4

The proposed change revises the ejected rod worths in the Bases discussion for Required Action A.2.4 of ITS 3.1.4.

The ejected rod worths are made consistent with the rod ejection analysis described in the proposed UFSAR Section 15.12 for units analyzed in accordance with DPC-NE-3005-PA.

Proposed Changes to the Bases for ITS 3.3.1

Duke proposes to change the "Applicable Safety Analyses, LCO and Applicability" section of the Bases of ITS 3.3.1 to:

1. identify that the RCS variable low pressure trip is assumed for transient protection in the main steam line

break (MSLB) analysis for units analyzed in accordance with DPC-NE-3005-PA;

2. identify that, for units analyzed in accordance with DPC-NE-3005-PA, the RCS variable low pressure trip setpoint allowable value does not include errors induced by the harsh environment because the trip function actuates prior to the harsh environment; and

3. clarify that the RCS low pressure reactor trip function remains the primary reactor trip function credited in the MSLB analysis for units not analyzed in accordance with DPC-NE-3005-PA.

The current analysis credits the RCS low pressure trip as the primary trip function in the MSLB analysis. The methodology approved by the NRC in DPC-NE-3005-PA credits the RCS variable low pressure reactor trip function as the primary reactor trip function in the MSLB analysis. Thus, the proposed changes to the Bases for ITS 3.3.1 were revised to reflect this revised methodology.

<u>Proposed Changes to the Bases for ITS 3.3.15</u>

Duke proposes to revise the Bases for ITS 3.3.15 to:

1. define in the Background and Applicable Safety Analyses sections that the turbine stop valves (TSVs) on both main steam lines are closed following a turbine or reactor trip signal;

2. define in the Applicable Safety Analyses section, for units analyzed in accordance with DPC-NE-3005-PA, the MSLB with ICS low level control and without operator action prior to ten minutes is the limiting case for a post-trip return to power;

3. clarify in the Applicable Safety Analyses section, for units not analyzed in accordance with DPC-NE-3005-PA, that the MSLB without ICS and without operator actions remains the limiting case for a post-trip return to power; and

4. clarify that the turbine stop valves on both steam generators close for an HELB or an MSLB inside containment.

For an HELB or an MSLB inside containment, the turbine stop valves in both steam lines are assumed to close in the analyses for all units. The TSVs do not close on the break, but instead close when a turbine or reactor trip signal is generated. After closure of the TSVs, steam continues to be discharged into the containment only from the affected steam generator.

Currently, the MSLB without operator action and without ICS is the limiting return to power case. For units analyzed in accordance with DPC-NE-3005-PA, the limiting case for a post-trip return to power is a MSLB with ICS low level control and no operator action prior to 10 minutes. Thus, the proposed changes to the Bases for ITS 3.3.15 were revised to reflect this methodology.

Proposed Changes to the Bases for ITS 3.7.2

Duke proposes to revise the Bases for ITS 3.7.2 to:

1. define in the Applicable Safety Analyses section that the TSVs on both main steam lines are closed following a turbine or reactor trip signal;

2. define in the Applicable Safety Analyses section, for units analyzed in accordance with DPC-NE-3005-PA, the MSLB with ICS low level control and without operator action prior to ten minutes is the limiting case for a post-trip return to power;

3. clarify in the Applicable Safety Analyses section, for units not analyzed in accordance with DPC-NE-3005-PA, that the MSLB without ICS and without operator actions remains the limiting case for a post-trip return to power; and

4. clarify that the turbine stop valves on both steam generators close for an HELB or an MSLB inside containment.

For an HELB or an MSLB inside containment, the turbine stop valves in both steam lines are assumed to close in the analyses for all units. The TSVs do not close on the break, but instead close when a turbine or reactor trip signal is generated. After closure of the TSVs, steam continues to be discharged into the containment only from the affected steam generator.

Currently, the MSLB without operator action and without ICS is the limiting return to power case. For units analyzed in accordance with DPC-NE-3005-PA, the limiting case for a post-trip return to power is a MSLB with ICS low level control and no operator action prior to 10 minutes. Thus, the proposed changes to the Bases for ITS 3.3.15 were revised to reflect this methodology.

These changes are consistent with the changes proposed to be made to the Bases for ITS 3.3.15.

Proposed Changes to the UFSAR

Mark-ups of the UFSAR are provided in Attachment VII. Oconee Unit 2 Cycle 18 will implement the revised UFSAR Chapter 15 non-LOCA analysis methodology of topical report DPC-NE-3005-P, "UFSAR Chapter 15 Transient Analysis Methodology." Revision 0 of this topical report was conditionally approved by the NRC Safety Evaluation Report (SER) dated October 1, 1998. Revision 1 was submitted on February 1, 1999 to respond to the conditions in the SER. The UFSAR Chapter 15 revisions are based on the Revision 1 methodology. Since the methodology is being separately reviewed, and since the analysis results presented follow the methodology, the UFSAR revisions are the implementation of an approved methodology once Revision 1 is approved by the NRC. Many of the results in the UFSAR revisions were already presented in Revisions 0 and 1 of DPC-NE-3005-P.

Note that the environmental consequences (offsite dose analysis) content of this revision to Chapter 15 has not been completed and will be submitted for NRC review at a

later date. Each applicable section to be submitted later includes a notation to that effect.

Several sections of UFSAR Chapter 15 are not revised since not all of the analysis methodology for the licensing basis events was modernized in DPC-NE-3005-P. The following events were not reanalyzed and so no UFSAR revisions are included for these events:

Section 15.10 - Waste Gas Tank Rupture Accident
Section 15.11 - Fuel Handling Accidents
Section 15.14 - Loss of Coolant Accidents
Section 15.15 - Maximum Hypothetical Accident
Section 15.16 - Post-Accident Hydrogen Control

The following sections of UFSAR Chapter 15 are essentially replaced in their entirety to reflect the new DPC-NE-3005-P methodology and the deletion of the 1970-vintage original licensing basis analyses:

Section 15.1 - Methodology (replaces "Uncompensated Operating Reactivity Changes")
Section 15.2 - Startup Accident
Section 15.3 - Rod Withdrawal at Power Accident (replaces "Rod Withdrawal at Rated Power Accident")
Section 15.4 - Moderator Dilution Accidents
Section 15.5 - Cold Water Accident
Section 15.6 - Loss of Coolant Flow Accidents (Pump Coastdown and Locked Rotor)
Section 15.7 - Control Rod Misalignment Accidents (Dropped Rod and Misaligned Rod)
Section 15.8 - Turbine Trip Accident (replaces "Loss of Electric Load Accidents")
Section 15.9 - Steam Generator Tube Rupture Accident
Section 15.12 - Rod Ejection Accident
Section 15.13 - Steam Line Break
Section 15.17 - Small Steam Line Break (new licensing basis event)

The following revisions are noted as being more than simply replacing old methodology and analysis results with the DPC-NE-3005-P methodology and results:

1) The new Section 15.1 provides an overview and summary of the methodologies, computer codes, and analysis assumptions. This will expand on the level of detail in the current Chapter 15, and will assist the reader in understanding and using it.

2) The original Section 15.1 has been deleted since experience has shown that this section did not add any value to the UFSAR.

3) The original Section 15.12.4 has been deleted. This section discusses the potential for a rod ejection accident to cause an explosive type of energy release and evaluates the stress on the reactor vessel. This type of evaluation is not typical of modern UFSARs and is not required by the Standard Review Plan. Furthermore, the limits placed on ejected rod worth by the new analysis methodology precludes approaching the worth which has the potential to overstress the reactor vessel. Therefore, this section has been deleted in the proposed revision.

4) The original Section 15.13.4 is being relocated to UFSAR Chapter 5 by a separate licensing submittal. Since this section describes stress analyses on the steam generator tubes, it is judged to be more appropriately located with the other stress analyses in Chapter 5 of the UFSAR.

Proposed Changes to the COLR

The Oconee Unit 2 Cycle 18 COLR is included as Attachment VI. This LAR proposes the following changes to the COLR to reflect Topical Report DPC-NE-3005-PA.

1) The moderator temperature coefficient limits were revised as follows:

Old Limits		New Limits	
MTC X 10-4	%FP	MTC X 10-4	%FP

Δρ/°F		Δρ/°F	
0.90	0	0.700	0
0.00	95	0.030	15
0.00	100	-0.281	95
		-0.300	100
		-0.375	120

2) The departure from nuclear boiling parameter for the RCS loop pressure was revised as follows

Old Limits	New Limits
4 RCP: measured hot leg pressure ≥ 2070 psig	4 RCP: measured hot leg pressure ≥ 2125 psig
3 RCP: measured hot leg pressure ≥ 2100 psig	3 RCP: measured hot leg pressure ≥ 2125 psig

3) The DNB parameter for RCS loop average temperature was changed from 581 °F at a ΔTc of 0 °F to a range of values dependent upon the ΔTc. These values are presently being validated. Typical (i.e., expected) values are:

Maximum Loop Tavg Including 2°F Uncertainty	ΔTc, °F
582.20	5
582.00	4
581.75	3
581.50	2
581.25	1
581.00	0

4) The DNB parameter for RCS loop total flow was revised as follows:

Old Limits	New Limits
4 RCP: measured hot leg pressure ≥ 109.5% df	4 RCP: measured hot leg pressure ≥ 107.5% df
3 RCP: measured hot leg pressure ≥ 74.7% of 4 RCP minimum flow	3 RCP: measured hot leg pressure ≥ 74.7% of 4 RCP minimum flow

ATTACHMENT IV

NO SIGNIFICANT HAZARDS CONSIDERATION

No Significant Hazards

ATTACHMENT IV

NO SIGNIFICANT HAZARDS CONSIDERATION

Pursuant to 10 CFR 50.91, Duke has made the determination that this License Amendment Request involves a No Significant Hazards by applying the standards established by the NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated?

   No. The proposed changes to the Technical Specifications, Bases, Updated Final Safety Analysis Report (UFSAR), and Core Operating Limits Report (COLR) incorporate the accident analyses established in Topical Report DPC-NE-3005-P, "UFSAR Chapter 15 Transient Analysis Methodology." On July 30, 1997, Duke submitted Topical Report DPC-NE-3005-P to the NRC for approval. The NRC found DPC-NE-3005-P acceptable, with noted exceptions, in a Safety Evaluation issued on October 1, 1998. To resolve the noted NRC exceptions, Duke submitted Revision 1 of DPC-NE-3005-P to the NRC for review on February 1, 1999. This LAR is dependent upon the NRC approval of Revision 1 of DPC-NE-3005-P.

   The analyzed events are initiated by the failure of specific plant structures, systems or components. These proposed changes do not impact the condition or performance of those structures, systems or components.

   The revised accident analyses in DPC-NE-3005-P demonstrate that the applicable acceptance criteria are met. In addition, the preliminary calculations show that the applicable radiological and environmental acceptance criteria will continue to be met. The results of the associated dose analyses, along with the proposed changes, will be submitted at a later date.

   Based on the above, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. <u>Create the possibility of a new or different kind of accident from any accident previously evaluated?</u>

   No. The proposed changes do not involve a physical alteration of the plant. No new or different equipment is being installed, and no installed equipment is being operated in a new or different manner. Where setpoints and operating limits have been revised, the revised accident analyses demonstrate that the applicable acceptance criteria are met. As a result, no new failure modes are being introduced.

   Based on the above, the proposed changes do not create the possibility of any new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety?

   No. The margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to a event. The proposed changes do not involve a physical alteration of the plant. No new or different equipment is being installed, and no installed equipment is being operated in a new or different manner. Where setpoints and operating limits have been revised, the revised accident analyses in DPC-NE-3005-P demonstrate that the applicable acceptance criteria are met.

   Based on the above, the proposed changes do not involve a significant reduction in a margin of safety.

<u>Conclusion</u>

Based upon the preceding evaluation, performed pursuant to 10CFR50.92, Duke Energy Corporation has concluded that the proposed changes to the Oconee Nuclear Station Technical Specifications, Bases, UFSAR, and O2C18 COLR will not involve a significant hazards consideration.

ATTACHMENT V

ENVIRONMENTAL ASSESSMENT

Enviromental Assessment

ATTACHMENT V

ENVIRONMENTAL ASSESSMENT

Pursuant to 10CFR51.22(b), an evaluation of this LAR is being performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10CFR51.22(c)(9) or 10CFR51.22(c)(10) of the regulations.

This LAR for the Oconee Technical Specifications proposes changes to allow implementation of reactor fuel cycle 18 on Unit 2 (O2C18). The reload design for O2C18 was accomplished using DPC-NE-3005-P. The NRC approved the topical report, with some exceptions, in a Safety Evaluation Report issued on October 1, 1998. To resolve these exceptions, Duke submitted Revision 1 to DPC-NE-3005-P to the NRC on February 1, 1999.

If it can be determined there are:

1) No significant hazards considerations;

2) No significant change in the types, or significant increase in the amounts, of any effluents that may be released offsite; and

3) No significant increase in individual or cumulative occupational radiation exposures involved;

then this LAR will qualify for categorical exclusion from the requirement to perform an environmental assessment/impact statement.

For Item 1, as listed above, Attachment IV of this reload submittal package documents the determination of no significant hazards considerations.

The final determination for Items 2 and 3, as listed above, is dependent upon the completion of the dose analysis for O2C18. This effort is currently in progress. The completed dose analysis will be provided to the NRC in a subsequent Duke submittal, as stated in the cover letter of this submittal package. The preliminary calculations show that the O2C18 reload will continue to meet the applicable radiological and environmental acceptance criteria; thus,

this LAR will meet the criteria for categorical exclusion from performing an environmental assessment/impact statement.