ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1
DOCKET NO. 50-390

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE WBN-TS-98-016
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

I. DESCRIPTION OF THE PROPOSED CHANGE

The proposed amendment would revise the Watts Bar Nuclear Plant (WBN) TS and TS Bases to reflect a revised large break loss-of-coolant-accident (LBLOCA) analysis from TVA's NSSS Vendor, Westinghouse. This analysis is based on an NRC approved Best Estimate (BE) methodology. By incorporating this methodology, WBN gains additional margin on peak clad temperature and additional margin on various plant operating parameters (i.e., accumulator water level, steam generator tube plugging, and $F^N_{\Delta H}$).

The following provides a discussion of the applicable sections affected by this proposed amendment:

Surveillance Requirements (SR) 3.5.1.2, "Accumulator," provides assurance that accumulator water volume is maintained within limits supported by applicable safety analyses. This SR is being revised to provide an increased operating range to ≥7630 and ≤8000 gallons. This change is supported by the BE LBLOCA analysis and a TVA design calculation.

TS 5.9.5.b, "Core Operating Limits Report (COLR)," identifies the applicable references for the analytical methods used to determine the core operating limits and specifies that the references shall be reviewed and approved by the NRC. Reference b.2 identifies the Westinghouse topical report that documents the approved LBLOCA analysis methodology using the BASH Code. With the introduction of BE LBLOCA analysis methods, this reference needs to be revised to reflect the new approved WCAP (WCAP-12945-P-A). The reference for the small break LOCA is being added for completeness.

TS Bases Sections 3.2.1, "Heat Flux Hot Channel Factor (Fq(Z)" and 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," describe the basis for assurance that the emergency core cooling system (ECCS) acceptance criteria limit of 2200°F is not exceeded with respect to the power distribution limits. Consistent with 10 CFR 50.46(a)(1)(i), this section is being revised to indicate Best Estimate methodology is used and to state there is a high level of probability that the ECCS acceptance criteria

limit is not exceeded with regard to the LBLOCA analysis.

TS Bases 3.5.1, "Accumulators," references the ECCS acceptance criteria limit of 2200°F with respect to the accumulator technical specification limits. This section is being revised to state there is a high level of probability that the ECCS acceptance criteria limit is not exceeded with regard to the LBLOCA analysis. Additionally, the discussion under "Applicable Safety Analyses," is being modified to reflect the BE LBLOCA methodology and the revised technical specification values for accumulator water volume.

TS Bases 3.5.2, "ECCS Operating," references the ECCS acceptance criteria limit of 2200°F with respect to the ECCS technical specification limits. This section is being revised to state there is a high level of probability that the ECCS acceptance criteria limit is not exceeded with regard to the LBLOCA analysis. Additionally, the discussion under "Applicable Safety Analyses," is being modified to reflect the BE LBLOCA methodology.

The specific changes to the TS and Bases are noted in the marked up copies of the applicable TS pages provided in Enclosure 2.

II. BACKGROUND

10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," and 10 CFR 50, Appendix K, "ECCS Evaluation Models," specify the acceptance criteria for the performance of the ECCS, maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling in response to a LOCA. After years of extensive research that improved the understanding of ECCS performance during a postulated LOCA, the methods specified in the original rule were found to be highly conservative and were found to be restrictive in operating flexibility. The NRC amended those rules in 1988 to allow use of realistic (best estimate) models to be able to quantify the ECCS performance and Appendix K limits. The term "Best Estimate" is used to indicate that the techniques attempt to predict realistic reactor system thermal-hydraulic response.

Regulatory Guide 1.157, "Best Estimate Calculation of Emergency Core Cooling System Performance," was issued in May 1989. This regulatory guide describes models, correlations, data, model evaluation procedures, and methods that are acceptable to the NRC staff for meeting the requirements for Best Estimate calculation of ECCS performance during a LOCA.

WCAP-12945(P), "Westinghouse Code Qualification Document for Best Estimate Loss Of Coolant Analysis," was submitted to NRC in August 1992. The WCAP was subsequently found to

be acceptable in NRC's safety evaluation dated June 28, 1996 (TAC No. M83964). Since that time, incorporation of this Best Estimate LOCA methodology has been accepted by NRC for Indian Point 2 on March 31, 1997, Turkey Point Units 3 and 4 on December 20, 1997, and Diablo Canyon Nuclear Power Plant Units 1 and 2 on February 13, 1998.

III. BASIS FOR THE PROPOSED CHANGE

These changes are being made to incorporate the Best Estimate approach into the licensing basis for the WBN LBLOCA analyses in accordance with 10 CFR 50.46, Regulatory Guide 1.157, and Westinghouse's WCAP-12945-P-A, Volumes I-V. This Best Estimate methodology provides a more realistic response of the plant to large break LOCA.

The revised technical specification values for accumulator water level range are supported by a TVA calculation. The calculation utilizes the additional safety analysis operating range provided in WCAP-14839, "Best Estimate Analysis of the Large Break Loss of Coolant Accident for the Watts Bar Nuclear Plant" Revision 1, dated September 1998, to determine the technical specification values after accounting for the appropriate instrument inaccuracies.

Table 1 lists the plant specific parameters used in the Watts Bar plant specific analysis and the location of the documentation of the values and ranges used for the parameters.

Table 2 presents the 50th and 95th percentile Peak Clad Temperature (PCT) for Watts Bar, maximum cladding oxidation, maximum hydrogen generation, and cooling results.

A BE LBLOCA analysis has been performed for Watts Bar using the approved Westinghouse Best Estimate methodology contained in WCAP-12945-P-A. Plant specific parameters used in the analysis are treated as described in the generic methodology. Therefore, the Watts Bar specific analysis conforms to 10 CFR 50.46 and Section II of Appendix K, and meets the intent of Regulatory Guide 1.157.

The conclusions of the analysis are that there is a high level of probability that

1) The calculated maximum fuel element cladding temperature (peak cladding temperature) will not exceed 2200°F.

2) The calculated total oxidation of the cladding (maximum cladding oxidation) will nowhere exceed 0.17 times the total cladding thickness before oxidation.

3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam (maximum hydrogen generation) will not exceed

0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4) The calculated changes in core geometry are such that the core cooling can be maintained.

5) After successful initial operation of the ECCS, the core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The changes in the technical specification values and surveillance requirements for accumulator water level have been evaluated through the design change process. Appropriate calculations and setpoint/scaling documents have been revised to incorporate the new safety analysis limits for accumulator water level.

Therefore, TVA has concluded that adopting the Best Estimate LBLOCA methodology for WBN Unit 1 and making the proposed TS changes to allow implementation of this change will not adversely affect the health and safety of the public.

IV. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The proposed amendment requests approval to apply the Westinghouse generic Best Estimate Large Break (LB) Loss-of-Coolant Accident(LOCA) Analysis methodology using the WCOBRA/TRAC computer code. TVA has concluded that operation of Watts Bar Nuclear Plant (WBN) Unit 1 in accordance with the proposed change to the technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

A. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes involve use of the Best Estimate Large Break loss-of-coolant accident (LOCA) analysis methodology and associated technical specification changes. Accumulator water level set points will be revised from ≥7717 gallons and ≤8004 gallons to ≥7630 gallons and ≤8000 gallons to provide the plant with an increased operating range. The plant conditions assumed in the analysis, including the accumulator water level instrumentation changes, are bounded by the design conditions for all equipment in the plant.

Therefore, there will be no increase in the probability of a LOCA. The consequences of a LOCA are not being increased, since it is shown that the emergency core cooling system (ECCS) is designed so that its calculated cooling performance conforms to the criteria contained in 10 CFR 50.46, Paragraph b. The small break LOCA analysis assumes only a nominal accumulator water level which is the same nominal value assumed in this analysis, therefore, the small break LOCA analysis is unaffected by the increase in the accumulator range. Also, the increased safety analysis range in accumulator water volume (+/-15 ft^3) has an insignificant effect on the containment related analyses.

The post-LOCA containment sump boron calculation assumes a minimum accumulator volume which bounds (is smaller than) the 1005 ft^3 (7518 gallons) value supported by the Best Estimate Large Break LOCA analysis. Also, the hot leg switchover calculation models a maximum accumulator volume which is not bounded by the 1095 ft^3 (8191 gallons) maximum value supported by the Best Estimate Large Break LOCA analysis. However, an evaluation concludes that the Watts Bar hot leg switchover time is unaffected by the difference in maximum volumes.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

B. **The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

No new modes of plant operation are being introduced by the new analysis or by the changes in instrumentation setpoints for accumulator water level. The parameters assumed in the analysis are within the design limits of existing plant equipment. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

C. **The proposed amendment does not involve a significant reduction in a margin of safety.**

It has been shown that the analytic technique used in the analysis realistically describes the expected behavior of the WBN Unit 1 reactor system during a postulated loss of coolant accident. Uncertainties have been accounted for as required by 10 CFR 50.46. The physical setpoint changes to accumulator water level instrumentation are bounded by the uncertainty evaluation addressing accumulator water level. A sufficient number of loss of coolant accidents with different break sizes, different locations, and other

variations in properties have been considered to provide assurance that the most severe postulated LOCAs were evaluated. It has been shown by the analysis that there is a high level of probability that all criteria contained in 10 CFR 50.46, Paragraph b are met.

D. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The incorporation of these changes: a) will not increase the probability of the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Updated Final Safety Analysis Report (UFSAR); b) will not create the possibility for an accident or malfunction of a new or different kind from any evaluated previously in the UFSAR; and c) will not reduce the margin of safety as defined in the bases for any technical specification. Based upon the above, TVA concludes that the changes proposed by the amendment request satisfy the no significant hazards consideration standards of 10 CFR 50.92(c), and accordingly a no significant hazards finding is justified.

V. ENVIRONMENTAL IMPACT CONSIDERATION

The proposed change does not involve a significant hazards consideration, a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

TABLE 1

Major Plant Parameter Assumptions Used in the BE LOCA Analysis. for Watts Bar and Where They Will Be Documented.

Parameter	Documentation
Plant Physical Description	UFSAR 15.4.1
Plant Initial Operating Conditions	
Reactor Power	UFSAR 15.4.1
Peaking Factors	COLR Section 2
Axial Power Distribution	UFSAR 15.4.1
Fluids Conditions	
Tavg	UFSAR 15.4.1
Pressurizer Pressure	UFSAR 15.4.1
Reactor Coolant Flow	UFSAR 15.4.1
Accumulator Temperature	UFSAR 15.4.1
Accumulator Pressure	UFSAR 15.4.1
Accumulator Volume	UFSAR 15.4.1
Accident Boundary Conditions	
Single Failure Assumptions	UFSAR 15.4.1
Safety Injection Flow	UFSAR 15.4.1
Safety Injection Temperature	UFSAR 15.4.1
Safety Injection Initiation Delay Time	UFSAR 15.4.1
Containment Pressure	UFSAR 15.4.1

TABLE 2

WATTS BAR BEST ESTIMATE LBLOCA RESULTS

	Value	Criteria
50th Percentile PCT (°F)[1]	1564	N/A
95th Percentile PCT (°F)	1892[2]	≤2200
Maximum Cladding Oxidation (%)[1]	15	≤17
Maximum Hydrogen Generation (%)[1]	0.61	≤1
Coolable Geometry	Core Remains Coolable	Core Remains Coolable
Long Term Cooling	Core Remains Cool in Long Term	Core Remains Cool in Long Term

(1) Documented in WCAP-14839, Revision 1 and calculated using the methodology in the following reference:

WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best-Estimate Loss-of-Coolant Accident Analysis," March 1998 (Westinghouse Proprietary).

(2) Due to an error discovered in the Best Estimate Large Break LOCA analysis which was reported by Westinghouse to TVA in the 10 CFR 50.46 Notification for 1998, the revised analysis adjusted the PCT by a -4°F. Therefore, the PCT of 1892°F which is documented in WCAP-14839, Revision 1, is changed to 1888°F. The discussion of this change is documented in the Attachment to this enclosure (Enclosure 1).

ATTACHMENT

WATTS BAR NUCLEAR PLANT
EMERGENCY CORE COOLING EVALUATION MODEL CHANGES
FOR 1998
BEST ESTIMATE LARGE BREAK LOCA

ATTACHMENT

Westinghouse LOCA Peak Clad Temperature Summary For Best Estimate Large Break

Plant Name: Watts Bar Unit 1

Utility Name: Tennessee Valley Authority

Revision Date: 3/3/99

Future

Analysis Information

EM:	WCOBRATRAC	Analysis Date:	08/98	Limiting Break Size: Guillotine
FQ:	2.5	FdH:	1.65	
Fuel:	Vantage+	SGTP(%)	10	
Notes:	Mixed Core - Vantage+/Performance -			

	Clad Temp (F)	Ref. Notes
LICENSING BASIS		
Analysis-Of-Record PCT	1892	1,2
MARGIN ALLOCATIONS (Delta PCT)		
A. PRIOR PERMANENT ECCS MODEL ASSESSMENTS		
1. Total	0	
B. 10 CFR 50.59 SAFETY EVALUATIONS		
1. None		
C. 1998 10 CFR 50.46 MODEL ASSESSMENTS (Permanent Assessment of PCT Margin)		
1. Vessel Channel DX Error	-4	
D. TEMPORARY ECCS MODEL ISSUES*		
1. None	0	
E. OTHER		
1. None	0	
LICENSING BASIS PCT + MARGIN ALLOCATIONS	PCT = 1888	

* It is recommended that these temporary PCT allocations which address current LOCA model issues not be considered with respect to 10 CFR 50.46 reporting requirements.

References:

1. WCAP- 14839. Rev. 1. "Best Estimate Analysis of the Large Break Loss of Coolant Accident for the Watts Bar Nuclear Plant." August 1998.

2. WAT-D- 10499, "Tennessee Valley Authority Watts Bar Nuclear Plant Units 1 and 2. 10 CFR 50.46 Annual Notification and Reporting for 1997," February 27, 1998.

Notes:

None

ATTACHMENT

VESSEL CHANNEL DX ERROR (INCLUDING INVESTIGATION OF CODE UNCERTAINTIES)

Background:

Incorrect cell height is used in calculating gap flow wall friction and interfacial drag coefficients at gap level J. The error is that only cell 1 DX value is used, rather than cell heights specific to each level, 1 through J.

This was determined to be a Non-discretionary change as described in Section 4.1.2 of WCAP-13451.

Affected Evaluation Models

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

Estimated Effect

The impact of a WCOBRA/TRAC code error potentially has two effects, one on plant calculation results and one on the calculated code uncertainties. It was determined that this code error does not require any modifications to the uncertainty distributions used to propagate global or local model uncertainties. The estimated effects on the final 95% probability PCT for the plant calculations are shown in Table 1. For the purpose of this evaluation, plants were classified into two types. "Early reflood" plants are those which exhibit no significant core heatup after downcomer boiling begins. "Late reflood" plants are those which have at least some transients with late core heatup resulting from loss of reflood driving head after downcomer boiling begins.

These results will be incorporated on a plant specific basis. The licensing basis PCT (i.e., the limiting $PCT^{95\%}$) will be reported.

The SER requirement to verify the normality of the code bias uncertainty distribution has been satisfied.

This error was previously assessed for the SECY UPI LBLOCA Evaluation model in 1997 50.46 report (NSD-NRC-98-5575) and has no impact for 1998.

Table 1

Effect of NR-97-120 on 95% Probability PCT

	$\Delta PCT^{95\%}$		
Plant Type	Blowdown PCT (°F)	First Reflood PCT (°F)	Second Reflood PCT (°F)
Plants with early Reflood PCT	N/A[a]	+0	N/A[b]
Plants with late Reflood PCT	N/A[a]	+56	-4
AP600	+11	+15	N/A[c]

a) Adder for blowdown PCT was-not assessed since the blowdown PCT is far below the limiting reflood PCT.
b) First reflood PCT was confirmed to remain limiting for all early reflood PCT plants.
c) There is no second reflood in AP600 LBLOCA.

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE WBN-TS-98-016
MARKED PAGES

I. AFFECTED PAGE LIST

Technical Specifications

3.5-2
5.0-32

Technical Specification Bases

B 3.2-2
B 3.2-4
B 3.2-13
B 3.2-14
B 3.5-2
B 3.5-3
B 3.5-4
B 3.5-12
B 3.5-13

II. MARKED PAGES

See attached.

8000

7630

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ ~~7717~~ gallons and ≤ ~~8004~~ gallons.	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 610 psig and ≤ 660 psig.	12 hours
SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ 2400 ppm and ≤ 2700 ppm.	31 days AND -----NOTE------ Only required to be performed for affected accumulators ---------- Once within 6 hours after each solution volume increase of ≥ 75 gallons, that is not the result of addition from the refueling water storage tank

(continued)

5.9 Reporting Requirements (continued)

5.9.5 CORE OPERATING LIMITS REPORT (COLR)

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

LCO 3.1.4 Moderator Temperature Coefficient
LCO 3.1.6 Shutdown Bank Insertion Limit
LCO 3.1.7 Control Bank Insertion Limits
LCO 3.2.1 Heat Flux Hot Channel Factor
LCO 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor
LCO 3.2.3 Axial Flux Difference
LCO 3.9.1 Boron Concentration

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. WCAP-9272-P-A, WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary). (Methodology for Specifications 3.1.4 - Moderator Temperature Coefficient, 3.1.6 - Shutdown Bank Insertion Limit, 3.1.7 - Control Bank Insertion Limits, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalphy Rise Hot Channel Factor, 3.2.3 - Axial Flux Difference, and 3.9.1 - Boron Concentration.

Insert A

2. ~~WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor).~~

3. WCAP-10216-P-A, Revision 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F(Q) SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary).(Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor (W(Z) Surveillance Requirements For F(Q) Methodology) and 3.2.3 - Axial Flux Difference (Relaxed Axial Offset Control).)

4. WCAP-12610-P-A, "VANTAGE + FUEL ASSEMBLY REFERENCE CORE REPORT," April 1995. (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor).

(continued)

Page 5.0-32

Insert A:

2a. WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best-Estimate Loss of Coolant Analysis," March 1998 (W Proprietary). (Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor, and 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

2b. WCAP-10054-P-A, "Small Break ECCS Evaluation Model Using NOTRUMP Code," August 1985. Addendum 2, Rev. 1: "Addendum to the Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997. (W Proprietary). (Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor, and 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

Page B 3.2-2

Insert B:

During a loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref.1).

Page B 3.2-4

Insert C:

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during a small break LOCA, and assures with a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref.1).

Page B 3.2-13

Insert D:

During a loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 3).

Page B 3.2.14

Insert E:

The LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref.3) model $F^N_{\Delta H}$ as well as the Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$).

Page B 3.5-2

Insert F:

The assumption of loss of offsite power is also considered to determine if it yields limiting results. The loss of offsite power assumption imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence.

Insert G: The limiting large break LOCA is a double ended guillotine break in the cold leg.

Page B 3.5-4

Insert H: The safety analyses support a range of 7518 gallons to 8191 gallons. To allow for instrument inaccuracy, values of 7630 gallons and 8000 gallons are specified.

Insert I: The small break LOCA analysis is performed at the minimum nitrogen cover pressure, since generic sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak cladding temperature benefit.

Insert J: The LOCA analyses support a range of 585 to 690 psig.

Page B 3.5-13

Insert K: A large break LOCA event, with or without loss of offsite power and with a single failure disabling one ECCS train (in the containment pressure analysis, both EDG trains are conservatively assumed to operate due to requirements for modeling full active containment heat removal system operations); and

BASES

BACKGROUND (continued)

the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

Insert B

a. ~~During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1);~~

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

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

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of the NRC Policy Statement.

BASES

LCO (continued)

The expression for $F^W_Q(Z)$ is:

$$F^W_Q(Z) = F^C_Q(Z)\ W(Z)$$

where W(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR.

Insert C

~~The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.~~

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ produces unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by $\geq$ 1% RTP for each 1% by which $F^C_Q(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F^C_Q(Z)$ is $F^M_Q(Z)$ multiplied by a factor accounting for manufacturing tolerances and measurement uncertainties. $F^M_Q(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the

(continued)

BASES

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

Insert D

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

b. ~~During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F;~~

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, $F^N_{\Delta H}$ is a significant core parameter. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum local DNB heat flux ratio to a value which satisfies the 95/95 criterion for the DNB correlation used. Refer to the Bases for the Reactor Core Safety Limits, B 2.1.1, for a discussion of the applicable DNBR limits. The W-3 Correlation with a DNBR limit of 1.3 is applied in the heated region below the first mixing vane grid. In addition, the W-3 DNB correlation is applied in the analysis of accident conditions where the system pressure is below the range of the WRB-1 correlation. For system pressures in the range of 500 to 1000 psia, the W-3 correlation DNBR limit is 1.45 instead of 1.3.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Application of these criteria provides assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

Insert E

~~The LOCA safety analysis indirectly models $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor (FQ(Z)) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 3).~~

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.7, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor (FQ(Z))."

$F^N_{\Delta H}$ and FQ(Z) are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of the NRC Policy Statement.

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat

(continued)

BASES

BACKGROUND (continued)

This interlock also prevents inadvertent closure of the valves during normal operation prior to an accident. Although not required for accident mitigation, the valves will automatically open as a result of an SI signal. These features ensure that the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) for "operating bypasses" and that the accumulators will be available for injection without reliance on operator action.

The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 2). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. ~~The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence.~~ In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break.

Insert F

Insert G

~~The limiting large break LOCA is a double ended guillotine break at the discharge of the reactor coolant pump.~~ During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting and the pumps being loaded and delivering full flow. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and centrifugal charging pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the centrifugal charging pumps become solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 3) will be met following a LOCA:

Paragraph b

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. ~~For small breaks, an increase in water volume is a peak clad temperature penalty. For large breaks, an increase in water volume can be either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core reflooding portion of the transient. The analysis makes a conservative assumption with respect to ignoring or taking credit for line water volume from the accumulator to the check valve. The safety analysis assumes values of 7627 gallons and 8082 gallons. To allow for instrument inaccuracy, values of 7717 gallons and 8004 gallons are specified.~~

Insert H

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

Insert I

~~The large and small break LOCA analyses are performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit.~~ The maximum nitrogen cover pressure analysis limit of 690 psig prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity. ~~The LOCA analysis assumes a value of 585 psig.~~ To account for the accumulator tank design pressure rating, and to allow for instrument accuracy values of ≥610 psig and ≤660 psig are specified for the pressure indicator in the main control room.

Insert J

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 2 and 4).

(continued)

BASES

BACKGROUND (continued)

The centrifugal charging subsystem of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Cold Overpressure Mitigation System (COMS)," for the basis of these requirements.

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence for a loss of offsite power. If offsite power is available, the safeguard loads start immediately. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA: Paragraph b

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Maximum hydrogen generation from a zirconium water reaction is ≤0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post trip return to power following an MSLB event and ensures that containment temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The centrifugal charging pumps and SI pumps are credited in a small break LOCA event. This event establishes the flow and discharge head at the design point for the centrifugal charging pumps. The SGTR and MSLB events also credit the centrifugal charging pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

Insert K

a. ~~A large break LOCA event, with loss of offsite power and a single failure disabling one RHR pump (both EDG trains are assumed to operate due to requirements for modeling full active containment heat removal system operation); and~~

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

(continued)

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE WBN-TS-98-016
REVISED PAGES

I. AFFECTED PAGE LIST

Technical Specifications

3.5-2
5.0-32

Technical Specification Bases

B 3.2-2
B 3.2-4
B 3.2-13
B 3.2-14
B 3.5-2
B 3.5-3
B 3.5-4
B 3.5-12
B 3.5-13

II. REVISED PAGES

See attached.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ 7630 gallons and ≤ 8000 gallons.	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 610 psig and ≤ 660 psig.	12 hours
SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ 2400 ppm and ≤ 2700 ppm.	31 days AND -----NOTE------ Only required to be performed for affected accumulators ---------- Once within 6 hours after each solution volume increase of ≥ 75 gallons, that is not the result of addition from the refueling water storage tank

(continued)

5.9 Reporting Requirements (continued)

5.9.5 CORE OPERATING LIMITS REPORT (COLR)

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

LCO 3.1.4 Moderator Temperature Coefficient
LCO 3.1.6 Shutdown Bank Insertion Limit
LCO 3.1.7 Control Bank Insertion Limits
LCO 3.2.1 Heat Flux Hot Channel Factor
LCO 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor
LCO 3.2.3 Axial Flux Difference
LCO 3.9.1 Boron Concentration

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. WCAP-9272-P-A, WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary). (Methodology for Specifications 3.1.4 - Moderator Temperature Coefficient, 3.1.6 - Shutdown Bank Insertion Limit, 3.1.7 - Control Bank Insertion Limits, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalphy Rise Hot Channel Factor, 3.2.3 - Axial Flux Difference, and 3.9.1 - Boron Concentration.

2a. WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best-Estimate Loss of Coolant Analysis," March 1998 (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor, and 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

b. WCAP-10054-P-A, "Small Break ECCS Evaluation Model Using NOTRUMP Code," August 1985. Addendum 2, Rev. 1: "Addendum to the Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997. (W Proprietary). (Methodology for Specifications 3.2.1 - Heat FTux Hot Channel Factor, and 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

3. WCAP-10216-P-A, Revision 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F(Q) SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary).(Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor (W(Z) Surveillance Requirements For F(Q) Methodology) and 3.2.3 - Axial Flux Difference (Relaxed Axial Offset Control).)

4. WCAP-12610-P-A, "VANTAGE + FUEL ASSEMBLY REFERENCE CORE REPORT," April 1995. (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor).

(continued)

BASES

BACKGROUND (continued)

the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2000°F for large breaks (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

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

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of the NRC Policy Statement.

BASES

LCO (continued)

The expression for $F^W_Q(Z)$ is:

$$F^W_Q(Z) = F^C_Q(Z)\ W(Z)$$

where W(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during a small break LOCA, and assures with a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 1).

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ produces unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F^C_Q(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F^C_Q(Z)$ is $F^M_Q(Z)$ multiplied by a factor accounting for manufacturing tolerances and measurement uncertainties. $F^M_Q(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the

(continued)

BASES

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

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

b. During a loss of coolant accident (LOCA), the peak cladding temperature (PCT) must not exceed 2200°F for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 3);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, $F^N_{\Delta H}$ is a significant core parameter. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum local DNB heat flux ratio to a value which satisfies the 95/95 criterion for the DNB correlation used. Refer to the Bases for the Reactor Core Safety Limits, B 2.1.1, for a discussion of the applicable DNBR limits. The W-3 Correlation with a DNBR limit of 1.3 is applied in the heated region below the first mixing vane grid. In addition, the W-3 DNB correlation is applied in the analysis of accident conditions where the system pressure is below the range of the WRB-1 correlation. For system pressures in the range of 500 to 1000 psia, the W-3 correlation DNBR limit is 1.45 instead of 1.3.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Application of these criteria provides assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

The LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 3), model $F^N_{\Delta H}$ as well as the Nuclear Heat Flux Hot Channel Factor (FQ(Z)).

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.7, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor (FQ(Z))."

$F^N_{\Delta H}$ and FQ(Z) are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of the NRC Policy Statement.

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat

(continued)

BASES

BACKGROUND (continued)

This interlock also prevents inadvertent closure of the valves during normal operation prior to an accident. Although not required for accident mitigation, the valves will automatically open as a result of an SI signal. These features ensure that the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) for "operating bypasses" and that the accumulators will be available for injection without reliance on operator action.

The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 2). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is also considered to determine if it yields limiting results. The loss of offsite power assumption imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break.

The limiting large break LOCA is a double ended guillotine break in the cold leg. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting and the pumps being loaded and delivering full flow. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and centrifugal charging pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the centrifugal charging pumps become solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46, Paragraph b (Ref. 3) will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. The safety analyses support a range of 7518 gallons to 8191 gallons. To allow for instrument inaccuracy, values of 7630 gallons and 8000 gallons are specified.

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

The small break LOCA analysis is performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. The maximum nitrogen cover pressure analysis limit of 690 psig prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity. The LOCA analyses support a range of 585 to 690 psig. To account for the accumulator tank design pressure rating, and to allow for instrument accuracy values of ≥ 610 psig and ≤ 660 psig are specified for the pressure indicator in the main control room.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 2 and 4).

(continued)

BASES

BACKGROUND (continued)

The centrifugal charging subsystem of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Cold Overpressure Mitigation System (COMS)," for the basis of these requirements.

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence for a loss of offsite power. If offsite power is available, the safeguard loads start immediately. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46, Paragraph b (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post trip return to power following an MSLB event and ensures that containment temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The centrifugal charging pumps and SI pumps are credited in a small break LOCA event. This event establishes the flow and discharge head at the design point for the centrifugal charging pumps. The SGTR and MSLB events also credit the centrifugal charging pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large break LOCA event, with or without loss of offsite power and with a single failure disabling one ECCS train (in the containment pressure analysis, both EDG trains are conservatively assumed to operate due to requirements for modeling full active containment heat removal system operation); and

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.