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SOUTHERN COMPANY
Energy to Serve Your World

May 3, 2001

NEL-01-0059

Docket Nos.: 50-348
50-364

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Joseph M. Farley Nuclear Plant
Submittal of License Amendment Request,
Correction to Technical Specification 5.6.5,
Addition of Elbow Tap Flow Measurement Methodology
and
Relocation of Cycle-specific Parameters

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, Southern Nuclear Operating Company (SNC) hereby requests an amendment to the Farley Nuclear Plant (FNP) Unit 1 Operating License (NPF-2) and Unit 2 Operating License (NPF-8) by incorporating the attached changes into the FNP Units 1 and 2 Technical Specifications. These changes apply equally to each unit.

SNC proposes to remove several cycle-specific parameter limits from the Technical Specifications (TS). These parameter limits will be added to the Core Operating Limits Report (COLR). The relocation of the limits allows for the available operating and analytical margins to be used in the most efficient manner. In addition, the core safety limit curves will be replaced with safety limits more directly applicable to the fuel and fuel cladding fission product barriers. Historically, SNC has re-analyzed the reactor coolant system (RCS) flow, overtemperature and overpower reactor trip setpoints and applicable changes for the safety limit curves for each cycle of operation and, as appropriate, requested changes to the Technical Specifications on a cycle-specific basis. In lieu of cycle-specific changes for Unit 2 Cycle 15 operation, SNC is requesting approval for removal of these cycle-specific parameters in accordance with the NRC approved methodology described in WCAP 14483-A "Generic Methodology for Expanded Core Operating Limits Report". The proposed change will result in resource savings for SNC and the NRC by eliminating license amendment requests now required to change the values of these parameters. The proposed change is consistent with the intent of Generic Letter (GL) 88-16, which provides guidelines for the removal of cycle-specific parameter limits from the Technical Specifications. TXU Electric Company, South Texas Project, and Commonwealth Edison Company submitted similar license amendment requests on May 24, 1999, June 7, 1999 and December 22, 1999 respectively.

SNC also proposes to add a RCS minimum flow limit based upon the elbow tap flow measurement methodology of WCAP-14750-P-A Revision 1, "RCS Flow Verification Using Elbow Taps at Westinghouse 3-Loop PWRs". This methodology was approved via an NRC letter dated July 7, 1999. This change will provide separate acceptance criteria for use of elbow tap RCS flow measurement and the precision heat balance methodologies.

A001

In addition to the proposed changes associated with WCAP-14483-A and WCAP-14750-P-A Revision 1, a change is proposed to reference the refueling boron concentration in TS section 5.6.5 list of contents of the COLR. The refueling boron concentration is contained in the COLR but was not referenced as such by TS 5.6.5. This addition will correct a previous oversight and does not impact the content of the COLR.

Attachment 1 provides a discussion and evaluation of the proposed changes. Attachment 2 provides a determination that the proposed changes do not involve a significant hazard consideration. Markups and clean copies of the proposed changes are provided in attachments 3 and 4. Attachments 5 and 6 contain markups and clean copies of the associated bases changes. These are included for information, as changes to these will be made under the FNP Bases Change Program. Examples of the proposed added COLR pages are provided in attachment 7 for information.

SNC has reviewed the proposed amendment pursuant to 10 CFR 50.92 and determined that it does not involve a significant hazard consideration. In addition, there is no significant increase in the amounts of effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure. Consequently, the proposed amendment satisfies the criteria of 10 CFR 51.22 for categorical exclusion from the requirements for an environmental assessment.

SNC requests an approval date of December 31, 2001 for this proposed license amendment.

In accordance with 10 CFR 50.91(b), SNC is providing the State of Alabama with a copy of this proposed amendment.

Should you have any questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

D. Morey

Dave Morey

Sworn to and subscribed before me this 3rd *day of* May *2001*

Martha Gayle Dow
Notary Public

My Commission Expires: November 1, 2001

EWC/maf:R1COLRdiscussion.doc
Attachments:

1. Discussion and Evaluation of the Proposed Amendment
2. No Significant Hazards Consideration Determination
3. Mark-up of the Technical Specifications
4. Clean Typed Changed Technical Specifications
5. Mark-up of the Technical Specification Bases
6. Clean Typed Changed Technical Specification Bases Pages
7. Example COLR Pages

cc: Southern Nuclear Operating Company
Mr. L. M. Stinson, General Manager - Farley

U. S. Nuclear Regulatory Commission, Washington, D. C.
Mr. F. Rinaldi, Licensing Project Manager – Farley

U. S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. T. P. Johnson, Senior Resident Inspector – Farley

Alabama Department of Public Health
Dr. D. E. Williamson, State Health Officer

cc: Southern Nuclear Operating Company
Mr. L. M. Stinson, General Manager - Farley

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Alabama Department of Public Health
Dr. D. E. Williamson, State Health Officer

Attachment 1

Discussion and Evaluation of the Proposed Amendment

BACKGROUND

SNC proposes to remove several cycle-specific parameter limits from the Technical Specification, (TS) and to add these limits to the Core Operating Limits Report (COLR). Generic Letter (GL) 88-16 "Removal of Cycle-specific Parameter Limits from Technical Specifications" provides guidance on removal of cycle-specific parameters from the TS to reduce an unnecessary burden on licensees and on the NRC. Generally, the methodology for determining cycle-specific parameter limits is documented in a NRC-approved Topical Report or in a plant-specific submittal. As a consequence, the NRC review of proposed changes to TS for these limits is primarily limited to confirmation that the updated limits are calculated using an NRC-approved methodology and consistent with all applicable limits of the safety analysis. This alternative allows continued trending of these limits by the NRC staff without the necessity of prior review and approval since the COLR is routinely submitted to the NRC.

In response to GL 88-16 the Westinghouse Owners Group (WOG) developed a generic COLR license amendment request package (WCAP-14483-A), "Generic Methodology for Expanded Core Operating Limits Report". This WCAP was accepted by the NRC as documented in a letter dated January 19, 1999 to the Westinghouse Owners Group. The proposes changes to the TS are consistent with the WCAP, however, SNC has chosen to retain some information in the TS bases that was removed in the examples provided in the WCAP and is providing additional references for approved methodologies.

SNC also proposes to add a RCS minimum flow limit to TS 3.4.1 based on the elbow tap flow measurement methodology (ETFMM) of WCAP-14750-P-A Revision 1, "RCS Flow Verification Using Elbow Taps at Westinghouse 3-Loop PWRs". This methodology was approved via an NRC letter dated July 7, 1999. This change will provide separate acceptance criteria for the elbow tap and the precision heat balance methodologies. The ETFMM was developed on a generic basis for 3 loop Westinghouse Plants and was submitted by SNC on behalf of the Westinghouse Owners Group (WOG) via a letter dated November 26, 1996. In the NRC safety evaluation transmitted July 7, 1999 the staff states. "The staff will not repeat its review of the matters described in the subject report, when the report appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved." This change conforms to the WCAP-14750-P-A Revision 1, and applies to FNP units 1 and 2. FNP was the lead WOG plant for submittal of the WCAP. For development of WCAP-14750-P-A Revision 1, FNP data was used as the technical basis. Therefore, additional justification for addition of the ETFMM is not provided.

A change is proposed to TS 5.6.5 to add a reference to the Refueling Boron Concentration to the list of contents of the COLR. Since this addition corrects the reference only, it has no impact on the content of the COLR and affects no TS limit or requirement.

DESCRIPTION OF PROPOSED CHANGE

This amendment request proposes that the cycle-specific reactor coolant system (RCS) related TS parameter limits be relocated to the COLR. This change will allow SNC the flexibility to optimize plant operating and design margins without the need for cycle-specific license amendment requests. In addition to the cycle-specific parameter relocation, a reference to the Refueling Operations Boron Concentration is added to TS 5.6.5. This is done to correct an omission. The TS changes include:

Revising TS 2.1, "Safety Limits," and the associated bases to replace figure 2.1.1-1 with more specific requirements regarding the safety limits.

Revising TS 3.3.1, "Reactor Trip System Instrumentation," and the associated bases to relocate the Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) setpoint parameter values to the COLR.

Revising TS 3.4.1 "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits" to relocate the pressurizer pressure, RCS average temperature and RCS flow values to the COLR. Minimum

RCS flow limit will be retained in TS 3.4.1 as specified by the NRC safety evaluation provided regarding WCAP-14483-A. In addition, a separate minimum RCS flow rate is provided for the elbow tap methodology described in WCAP-14750-P-A and for the precision heat balance method.

Revising TS 5.6.5 "Core Operating Limits Report (COLR) to reflect the relocation of the parameter limits discussed by WCAP 14483-A to the COLR. In addition TS 5.6.5 is revised to reference the limit for refueling boron concentration currently located in the COLR. This change is done to correct a previous omission.

SAFETY EVALUATION

Relocation of the Cycle-Specific Parameters to the COLR

The methodology for calculating these parameters have been approved by the NRC (Safety Evaluation relating to Topical Report WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report") and are consistent with the applicable limits in the Final Safety Analysis Report (FSAR). The WCAP was approved by the NRC staff on January 19, 1999. The current TS method for controlling reactor physics parameters to assure conformance to 10 CRF 50.36 is to specify the value determined to be within the acceptance criteria using an approved calculation methodology. If approved, the proposed amendment will not alter the input parameters or the methodologies for calculating these limits.

Removal of cycle-dependent parameters from the TS has no impact upon plant operation or safety. No plant equipment, safety function or plant operation will be altered as a result of this proposed change. The core safety limits imposed in TS 2.1 are consistent with the values stated in the FNP FSAR. Because applicable FSAR limits will be maintained and TS will continue to require operation within the core operational limits calculated by approved methodologies, this proposed change is administrative in nature and does not impact the effectiveness of the associated TS. Actions to be taken if the limits are violated will remain in the TS.

If approved, this proposed change will control the cycle-specific parameters within the acceptance criteria and conform to 10 CFR 50.36 by using the approved methodology instead of specifying TS values. The COLR will document the specific parameter limits resulting from NRC approved calculation methods including mid-cycle or other revisions to parameter values.

Any changes to the COLR will be made in accordance with the requirements of 10 CRF 50.59 with a copy of the revised COLR sent to the NRC as required in TS section 5.6.5. The COLR will be revised such that appropriate core operating limits will apply.

The relocation of selected RCS related TS limits to the COLR will allow FNP to present cycle-specific operating condition parameters in the COLR. This will allow FNP to credit the cycle-specific operating configuration in the safety analyses as is currently done for the core reload designs. This approach is supported by NRC approved reload methodologies such as the Westinghouse reload methodology (WCAP-9272-P-A). This methodology specifies examination of the RCS DNB parameters and the overpressure delta temperature (OPΔT) and overtemperature delta temperature (OTΔT) setpoints and the supporting bases for the setpoints on a cycle-specific basis.

Addition of Elbow Tap Flow Methodology RCS Flow Limit

SNC proposes to add a RCS minimum flow limit based up the elbow tap flow measurement methodology of WCAP-14750-P-A Revision 1, "RCS Flow Verification Using Elbow Taps at Westinghouse 3-Loop PWRs". This methodology was approved via a safety evaluation provided in a NRC letter dated July 7, 1999. This change will provide acceptance separate criteria for use of elbow tap RCS flow and the precision heat balance methodologies. The use of this methodology has received NRC approval. The approved methods apply to FNP Units 1 and 2, therefore, no additional justification is provided for this proposed change.

Correction to TS 5.6.5

The Refueling Boron concentration was previously removed from the TS and relocated to the COLR. When that change was made the list of contents of the COLR in TS 5.6.5 was not updated. This proposed change will correct that oversight and does not change any limits or controls.

CONCLUSION

Relocation these cycle-specific parameters to the COLR will allow FNP the flexibility to utilize available margins to increase cycle operating margins and /or improve core reload designs. The use of appropriate methodologies to examine these limits will ensure that these core-specific parameters will remain acceptable. Relocation will also provide consistency to the TS, the safety analyses and the protection systems for each cycle. These proposed changes are in accordance with WCAP-14483-A, which was approved by the NRC via a letter dated January 19, 1999. In addition, resources will be conserved for both SNC and the NRC by reducing the number of license amendment requests.

The proposed change to TS 3.4.1 to provide separate minimum RCS flow limits derived for the elbow tap and the precision heat balance methods applies a previously NRC approved methodology in accordance with WCAP-14750-P-A.

The proposed change to TS 5.6.5 corrects a previous oversight in the list of content of the COLR. No limits or conditions are impacted.

IMPLEMENTATION

SNC requests that this amendment be approved by December 31, 2001.

Attachment 2

No Significant Hazards Consideration Determination

No Significant Hazards Consideration Determination

Pursuant to 10 CFR 50.91, this analysis provides a determination that the proposed change to the Technical Specification (TS) does not involve any significant hazard consideration as defined in 10 CFR 50.92.

Southern Nuclear Operating Company (SNC) proposes to remove several cycle-specific parameter limits from the TS. These parameter limits will be added to the Core Operating Limits Report (COLR). This proposed change utilizes the guidance of WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report." SNC also proposes to add a Reactor Coolant System (RCS) minimum flow limit based up the elbow tap flow measurement methodology of WCAP-14750-P-A Revision 1, "RCS Flow Verification Using Elbow Taps at Westinghouse 3-Loop PWRs." In addition, a change is proposed to reference the refueling boron concentration in the TS section 5.6.5 list of contents of the COLR. This addition will correct a previous oversight and does not impact the content of the COLR.

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

The proposed amendment is a programmatic and administrative change that does not physically alter plant systems, nor does it impact the performance of their functions. No new equipment is added nor is installed equipment being changed or operated in a different manner. Because the design of the facility and system operating parameters are not being changed, the proposed amendment does not involve an increase in the probability or consequences of any accident previously evaluated.

The cycle-specific limits in the Core Operating Limits Report (COLR) will continue to be controlled by the Farley Nuclear Plant (FNP) programs and procedures. Each accident analysis addressed in the Final Safety Analysis Report (FSAR) will be examined with respect to changes in the cycle dependent parameters, which are obtained from the use of Nuclear Regulatory Commission (NRC) approved reload design methodologies, to ensure that the transient evaluation of new reloads are bounded by previously accepted analyses. This examination, which will be conducted per the requirements of 10 CFR 50.59, will ensure that future reloads will not involve a significant increase in the probability or consequences of an accident previously evaluated. The safety limits imposed in Technical Specification (TS) 2.1 are consistent with the values stated in the FNP FSAR.

This change does not involve an increase in the probability or consequences of any accident previously evaluated.

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

Relocation of the cycle-specific parameters has no influence or impact on, nor does it contribute in any way to the probability or consequences of an accident. No plant equipment, function or plant operation will be altered as a result of this proposed change. The cycle-specific parameters are calculated using the NRC approved methods and submitted to the NRC to allow the staff to continue to trend the values of these limits. The TS will continue to require operation within the core operating limits and appropriate actions will be required if these limits are exceeded. The safety limits are maintained in the COLR and appropriate actions will be required if these limits are exceeded. In addition, the minimum limit for Reactor Coolant System flow will be retained in the TS. The safety limits imposed in TS 2.1 are consistent with the values stated in the FNP FSAR.

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

The Proposed change does not involve a significant reduction in a margin of safety

The margin of safety is not affected by the removal of cycle-specific core operating limits from the TS. The margin of safety presently provided by current TS limits remains unchanged. Appropriate measures exist to control the values of these cycle-specific limits. The proposed amendment continues to require operation within the core limits as obtained from NRC approved reload design methodologies and the actions to be taken if a limit is exceeded remain unchanged.

The development of the limits for future reloads will continue to conform to those methods described in NRC approved documentation. In addition, each future reload will involve a 10 CFR 50.59 evaluation to assure that operation of the unit within the cycle-specific limits will not involve a significant reduction in the margin of safety.

The proposed changes to relocate cycle specific parameter limits to the COLRs will not affect plant design or system operating parameters, there is no detrimental impact on any equipment design parameters, and the plant will continue to operate within prescribed limits. The safety limits imposed in TS 2.1 are consistent with the values stated in the FNP FSAR.

This proposed change does not involve a significant reduction in the margin of safety.

Conclusion

Based on the information presented above the three standards of 10 CFR 50.92 are satisfied; therefore the proposed change does not involve a significant hazards consideration.

Attachment 3

Mark-up of the Technical Specifications

files: R1COLRS2_0MU, R1COLRS3_3_1MU, R1COLRS3_4_1MU, R1COLRS5_6MU

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

~~In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the SLs specified in Figure 2.1.1-1.~~

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

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

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

2.1.1.1 In MODES 1 and 2, the departure from nucleate boiling ratio (DNBR) shall be maintained within the 95/95 DNB criterion correlation specified in the COLR.

2.1.1.2 In MODES 1 and 2, the peak fuel centerline temperature shall be Maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

~~In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the SLs specified in Figure 2.1.1-1.~~

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2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

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

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

2.1.1.1 In MODES 1 and 2, the departure from nucleate boiling ratio (DNBR) shall be maintained within the 95/95 DNB criterion correlation specified in the COLR.

2.1.1.2 In MODES 1 and 2, the peak fuel centerline temperature shall be Maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU.

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In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

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2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

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

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

RELOCATE TO COLR

Editorial Note:
DELETE PAGE

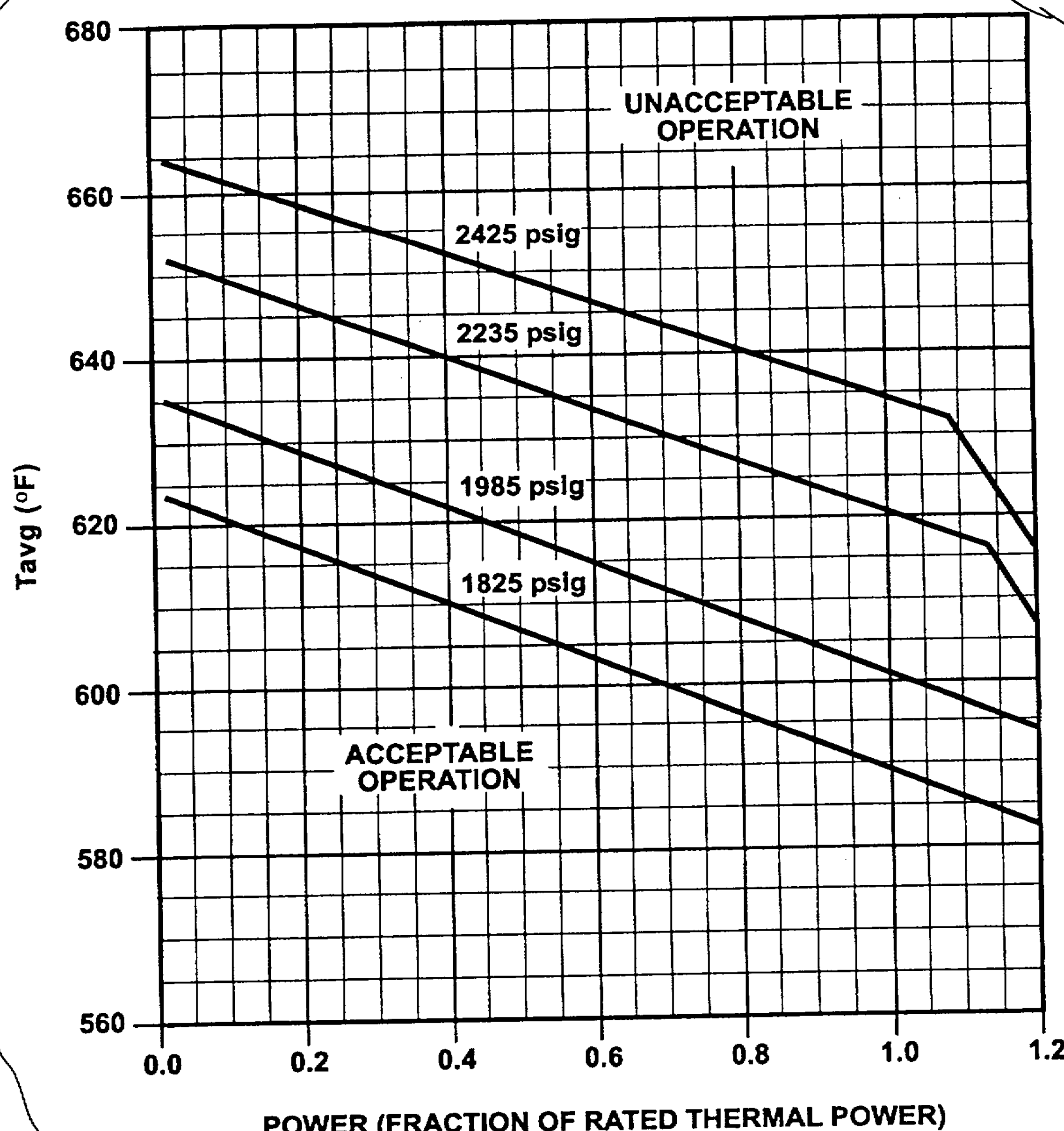

Figure 2.1.1-1
Reactor Core Safety Limits

Table 3.3.1-1 (page 7 of 8)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 0.4% of ΔT span.

$$\Delta T\frac{(1+\tau_4 s)}{(1+\tau_5 s)} \leq \Delta T_O\left\{K_1 - K_2\frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left[T\frac{1}{(1+\tau_6 s)} - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where: ΔT is measured loop ΔT, °F.
ΔT_O is the indicated loop ΔT at RTP and reference T_{avg}, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured loop average temperature, °F.
T' is the reference T_{avg} at RTP, ≤ ~~577.2~~°F.

P is the measured pressurizer pressure, psig.
P' is the nominal pressurizer operating pressure = ~~2235~~ psig.

K_1 = ~~1.17~~	K_2 = ~~0.017~~/°F	K_3 = ~~0.000825~~/psi
$\tau_1 \geq$ ~~30~~ sec	$\tau_2 \leq$ ~~4~~ sec	
τ_4 = ~~0~~ sec	$\tau_5 \leq$ ~~6~~ sec	$\tau_6 \leq$ ~~6~~ sec

$f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

$f_1(\Delta I)$ =	~~2.48~~{~~23~~ + $(q_t - q_b)$}	when $(q_t - q_b) \leq$ - ~~23~~% RTP
	~~0~~%of RTP	when -~~23~~% RTP < $(q_t - q_b) \leq$ ~~15~~% RTP
	~~2.05~~{$(q_t - q_b)$ - ~~15~~}	when $(q_t - q_b)$ > ~~15~~% RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

*** as specified in the COLR**

Editorial Note:
-Deleted values will be replaced by asterisks and presented in the COLR.

Table 3.3.1-1 (page 8 of 8)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 0.4% of ΔT span.

$$\Delta T \frac{(1+\tau_4 s)}{(1+\tau_5 s)} \leq \Delta T_O \left\{ K_4 - K_5 \frac{\tau_3 s}{1+\tau_3 s}\left(\frac{1}{1+\tau_6 s}\right) T - K_6 \left[T \frac{1}{1+\tau_6 s} - T'' \right] - f_2(\Delta I) \right\}$$

Where: ΔT is measured loop ΔT, °F.
ΔT_O is the indicated loop ΔT at RTP and reference T_{avg}, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured loop average temperature, °F.
T" is the reference T_{avg} at RTP, ≤ ~~577.2~~°F.

K_4 = ~~1.10~~

K_5 = ~~0.02~~/°F for increasing T_{avg}
K_5 = ~~0~~/°F for decreasing T_{avg}

K_6 = ~~0.00109~~/°F when T > T"
K_6 = ~~0~~/°F when T ≤ T"

$\tau_3 \geq$ ~~10~~ sec

τ_4 = ~~0~~ sec

$\tau_5 \leq$ ~~6~~ sec

$\tau_6 \leq$ ~~6~~ sec

$f_2(\Delta I)$ = ~~0~~% RTP for all ΔI.

<u>* as specified in the COLR</u>

Editorial Note:
-Deleted values will be replaced by asterisks and presented in the COLR.

3.4 REACTOR COOLANT SYSTEM (RCS)

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

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified ~~below:~~ in the COLR. The minimum RCS total flow rate shall be ≥ 263,400 GPM when using the precision heat balance method, ≥ 264,200 GPM when using the elbow tap method, and ≥ the limit specified in the COLR.

~~a. Pressurizer pressure ≥ 2209 psig;~~

~~b. RCS average temperature ≤ 580.3°F; and~~

~~c. RCS total flow rate ≥ 264,200 gpm.~~

APPLICABILITY: MODE 1.

-----------------------------------NOTE-----------------------------------
Pressurizer pressure limit does not apply during:

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

b. THERMAL POWER step > 10% RTP.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is ~~≥ 2209 psig~~ **within the limit specified in the COLR**.	12 hours
SR 3.4.1.2	Verify RCS average temperature is ~~≤ 580.3°F~~ **within the limit specified in the COLR.**	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ~~≥ 264,200 gpm~~ **within the limits.**	12 hours
SR 3.4.1.4	-----------------------------NOTE----------------------- Not required to be performed until 7 days after ≥ 90% RTP. -- Verify by measurement that RCS total flow rate is ~~≥ 264,200 gpm~~ **within the limits.**	18 months

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

2 ~~1~~. SHUTDOWN MARGIN limit for MODES 2 (with $k_{eff} < 1$), 3, 4, and 5 for LCO 3.1.1,

Insert 1

3 ~~2~~. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm and 100 ppm surveillance limits for LCO 3.1.3,

4 ~~3~~. Shutdown Bank Insertion Limits for LCO 3.1.5,

5 ~~4~~. Control Bank Insertion Limit for LCO 3.1.6,

8 ~~5~~. Axial Flux Limits for LCO 3.2.3.

6 . Heat Flux Hot Channel Factor F_Q^{RTP} limits, K(Z) figure, W(Z) values, and $F_Q(Z)$ Penalty Factors for LCO 3.2.1,

7 . Nuclear Enthalpy Rise Hot Channel Factor limits, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$, for LCO 3.2.2,

Insert 2

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 LCOs 3.1.1 - SHUTDOWN MARGIN, 3.1.3 - Moderator Temperature Coefficient, 3.1.5 - Shutdown Bank Insertion Limit, 3.1.6 - Control Bank Insertion Limits, 3.2.3 - Axial Flux Difference, 3.2.1 - Heat Flux Hot Channel Factor, and 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor.)

3.9.1 - Boron Concentration

2. WCAP-10216-P-A, Rev.1A, "Relaxation of Constant Axial Offset Control / F_Q Surveillance Technical Specification," February 1994 (W Proprietary).

   (Methodology for LCOs 3.2.3 - Axial Flux Difference and 3.2.1 - Heat Flux Hot Channel Factor.)

3a. WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, "Code Qualification Document for Best Estimate LOCA Analysis," March 1998 (W Proprietary).

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3b. WCAP-12610-P-A, "Vantage+ Fuel Assembly Reference Core Report," April 1995 (W Proprietary).

(Methodology for LCO 3.2.1 - Heat Flux Hot Channel Factor**, LCO 3.4.1- RCS Pressure, Temperature and Flow Departure from Nucleate Boiling Limits.)**

Insert 3

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 Systems (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.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. The reactor coolant system pressure and temperature limits, including heatup and cooldown rates, shall be established and documented in the PTLR for LCO 3.4.3.

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the NRC letters dated March 31, 1998 and April 3, 1998.

c. The PTLR shall be provided to the NRC upon issuance for each reactor fluence period and for any revision or supplement thereto.

5.6.7 EDG Failure Report

If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures shall be reported within 30 days. Reports on EDG failures shall include a description of the failures, underlying causes, and corrective actions taken per the Emergency Diesel Generator Reliability Monitoring Program.

INSERT 1

1. Reactor Core Safety Limits for THERMAL POWER, Reactor Coolant System highest loop average temperature and pressurizer pressure for Safety Limit 2.1.1,

INSERT 2

9. Reactor Trip System Instrumentation Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) setpoint parameter values for Table 3.3.1-1,
10. Reactor Coolant System pressure, temperature, and flow in LCO 3.4.1,
11. Refueling Operations Boron Concentration for LCO 3.9.1.

INSERT 3

4. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986 (Westinghouse Proprietary)

   (Methodology for Overpower ΔT and Thermal Overtemperature ΔT Trip Functions)

5. WCAP-14750-P-A Revision 1, "RCS Flow Verification Using Elbow Taps at Westinghouse 3-Loop PWRs." (Westinghouse Proprietary)

   (Methodology for minimum RCS flow determination using the elbow tap measurement)

6. WCAP-11596-P-A, "Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988

   (Methodology for LCO 3.9.1 – Boron Concentration)

7. WCAP-11397-P-A "Revised Thermal Design Procedure," April 1989

   (Methodology for LCO 2.1.1-Reactor Core Safety Limits, LCO 3.4.1- RCS Pressure, Temperature and Flow Departure from Nucleate Boiling Limits.)

Attachment 4

Clean Typed Changed Technical Specifications

Pages
2.0-1
3.3.1-20
3.3.1-21
3.4.1-1
3.4.1-2
5.6.2 (rollback)
5.6-3
5.6-4
5.6-5 (rollover)
5.6-6 (rollover)

files: R1COLRS2_0CT, R1COLRS3_3_1CT, R1COLRS3_4_1CT, R1COLRS5_6CT

Table 3.3.1-1 (page 7 of 8)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 0.4% of ΔT span.

$$\Delta T\frac{(1+\tau_4 s)}{(1+\tau_5 s)} \le \Delta T_O\left\{K_1 - K_2\frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left[T\frac{1}{(1+\tau_6 s)} - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where: ΔT is measured loop ΔT, °F.
ΔT_O is the indicated loop ΔT at RTP and reference T_{avg}, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured loop average temperature, °F.
T' is the reference T_{avg} at RTP, ≤ * °F.

P is the measured pressurizer pressure, psig.
P' is the nominal pressurizer operating pressure = * psig.

K_1 = *	K_2 = * /°F	K_3 = * /psi
$\tau_1 \ge$ * sec	$\tau_2 \le$ * sec	
τ_4 = * sec	$\tau_5 \le$ * sec	$\tau_6 \le$ * sec

$f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

$f_1(\Delta I)$ =	*{* + $(q_t - q_b)$}	when $(q_t - q_b) \le$ * % RTP
	* % of RTP	when * % RTP < $(q_t - q_b) \le$ * % RTP
	*{$(q_t - q_b)$ - *}	when $(q_t - q_b)$ > * % RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

* as specified in the COLR

Table 3.3.1-1 (page 8 of 8)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 0.4% of ΔT span.

$$\Delta T \frac{(1+\tau_4 s)}{(1+\tau_5 s)} \leq \Delta T_O \left\{ K_4 - K_5 \frac{\tau_3 s}{1+\tau_3 s}\left(\frac{1}{1+\tau_6 s}\right) T - K_6 \left[T \frac{1}{1+\tau_6 s} - T'' \right] - f_2(\Delta I) \right\}$$

Where: ΔT is measured loop ΔT, °F.
ΔT_O is the indicated loop ΔT at RTP and reference T_{avg}, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured loop average temperature, °F.
T" is the reference T_{avg} at RTP, ≤ * °F.

K_4 = *

K_5 = */°F for increasing T_{avg}
K_5 = */°F for decreasing T_{avg}

K_6 = */°F when T > T"
K_6 = */°F when T ≤ T"

$\tau_3 \geq$ * sec

$\tau_4 =$ * sec

$\tau_5 \leq$ * sec

$\tau_6 \leq$ * sec

$f_2(\Delta I)$ = *% RTP for all ΔI.

* as specified in the COLR

3.4 REACTOR COOLANT SYSTEM (RCS)

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

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified in the COLR. The minimum RCS total flow rate shall be ≥ 263,400 GPM when using the precision heat balance method, ≥ 264,200 GPM when using the elbow tap method, and ≥ the limit specified in the COLR.

APPLICABILITY: MODE 1.

-----NOTE-----

Pressurizer pressure limit does not apply during:

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

b. THERMAL POWER step > 10% RTP.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is within the limit specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average temperature is within the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is within the limits.	12 hours
SR 3.4.1.4	-------------------------------NOTE------------------------- Not required to be performed until 7 days after ≥ 90% RTP. -- Verify by measurement that RCS total flow rate is within the limits.	18 months

5.6 Reporting Requirements

5.6.3 Radioactive Effluent Release Report

-----NOTE-----
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report. In the event a RHR relief valve or a RCS vent is used to mitigate a RCS pressure transient, the monthly operating report shall describe the circumstances initiating the transient, the effect of the RHR relief valves or vent on the transient, and any corrective action necessary to prevent recurrence.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

   1. Reactor Core Safety Limits for THERMAL POWER, Reactor Coolant System highest loop average temperature and pressurizer pressure for Safety Limit 2.1.1,

   2. SHUTDOWN MARGIN limit for MODES 2 (with $k_{eff} < 1$), 3, 4, and 5 for LCO 3.1.1,

   3. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm and 100 ppm surveillance limits for LCO 3.1.3,

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. Shutdown Bank Insertion Limits for LCO 3.1.5,

5. Control Bank Insertion Limit for LCO 3.1.6,

6. Heat Flux Hot Channel Factor F_Q^{RTP} limits, K(Z) figure, W(Z) values, and $F_Q(Z)$ Penalty Factors for LCO 3.2.1,

7. Nuclear Enthalpy Rise Hot Channel Factor limits, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$, for LCO 3.2.2,

8. Axial Flux Limits for LCO 3.2.3,

9. Reactor Trip System Instrumentation Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) setpoint parameter values for Table 3.3.1-1,

10 Reactor Coolant System pressure, temperature, and flow in LCO 3.4.1,

11. Refueling Operations Boron Concentration for LCO 3.9.1.

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 LCOs 3.1.1 - SHUTDOWN MARGIN, 3.1.3 - Moderator Temperature Coefficient, 3.1.5 - Shutdown Bank Insertion Limit, 3.1.6 - Control Bank Insertion Limits, 3.2.3 - Axial Flux Difference, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor and 3.9.1 – Boron Concentration.)

2. WCAP-10216-P-A, Rev.1A, "Relaxation of Constant Axial Offset Control / F_Q Surveillance Technical Specification," February 1994 (W Proprietary).

   (Methodology for LCOs 3.2.3 - Axial Flux Difference and 3.2.1 - Heat Flux Hot Channel Factor.)

(continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3a. WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, "Code Qualification Document for Best Estimate LOCA Analysis," March 1998 (W Proprietary).

3b. WCAP-12610-P-A, "Vantage+ Fuel Assembly Reference Core Report," April 1995 (W Proprietary).

(Methodology for LCO 3.2.1 - Heat Flux Hot Channel Factor and LCO 3.4.1- RCS Pressure, Temperature and Flow Departure from Nucleate Boiling Limits.)

4. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986 (Westinghouse Proprietary)

(Methodology for Overpower ΔT and Thermal Overtemperature ΔT Trip Functions)

5. WCAP-14750-P-A Revision 1, "RCS Flow Verification Using Elbow Taps at Westinghouse 3-Loop PWRs. (Westinghouse Proprietary)

(Methodology for minimum RCS flow determination using the elbow tap measurement)

6. WCAP-11596-P-A, "Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988

(Methodology for LCO 3.9.1 – Boron Concentration)

7. WCAP-11397-P-A "Revised Thermal Design Procedure," April 1989

(Methodology for LCO 2.1.1-Reactor Core Safety Limits, LCO 3.4.1-RCS Pressure, Temperature and Flow Departure from Nucleate Boiling Limits.)

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 Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. The reactor coolant system pressure and temperature limits, including heatup and cooldown rates, shall be established and documented in the PTLR for LCO 3.4.3.

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the NRC letters dated March 31, 1998 and April 3, 1998.

c. The PTLR shall be provided to the NRC upon issuance for each reactor fluence period and for any revision or supplement thereto.

5.6.7 EDG Failure Report

If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures shall be reported within 30 days. Reports on EDG failures shall include a description of the failures, underlying causes, and corrective actions taken per the Emergency Diesel Generator Reliability Monitoring Program.

5.6.8 PAM Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.9 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance

(continued)

5.6 Reporting Requirements

5.6.9 Tendon Surveillance Report (continued)

Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.10 Steam Generator Tube Inspector Report

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days of the completion of the plugging effort.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission within 12 months following the completion of the inspection. This Report shall include:

   1. Number and extent of tubes inspected.

   2. Location and percent of wall-thickness penetration for each indication of an imperfection.

   3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be considered a Reportable Event and shall be reported pursuant to 10 CFR 50.73 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence.

5.6.11 Alternate AC (AAC) Source Out of Service Report

The NRC shall be notified if the AAC source is out of service for greater than 10 days.

Attachment 5

Mark-up of the Technical Specifications Bases

Pages
B 2.1.1-2
B 2.1.1-3
B 2.1.1-4
B 3.4.1-2
B 3.4.1-3
Insert

BASES

APPLICABLE SAFETY ANALYSES

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

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

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

In meeting the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and computer codes must be considered. As described in the FSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty to determine design limit DNBR values that satisfy the DNB design criterion.

Additional DNBR margin is maintained by performing the safety analyses to a higher DNB limit. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility.

The Reactor Trip System Functions (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions (i.e., resulting from a Condition I or II event) for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

flow, ΔI

Automatic enforcement of these reactor core SLs is provided by ~~the following functions:~~ **appropriate operation of the RPS and the steam generator safety valves.**

~~a. High pressurizer pressure trip;~~

~~b. Low pressurizer pressure trip;~~

~~c. Overtemperature ΔT trip;~~

O/

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~d. Overpower ΔT trip;~~

~~e. Reactor Coolant Flow-Low trip;~~

~~f. Power Range Neutron Flux trip; and~~

~~g. Main steam safety valves.~~

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

9

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

~~The curves provided in Figure 2.1.1-1 show the reactor core safety limits for a range of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation (not a limiting criterion).~~

~~The curves are based on enthalpy hot channel factor limits provided in the COLR.~~

~~The SL is higher than the limit calculated when the AFD is within the limits of the $F_1(\Delta I)$ function of the overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature ΔT reactor trips will reduce the setpoints to provide protection consistent with the reactor core SLs (Refs. 3 and 4).~~

9

INSERT 1

BASES

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The main steam safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. FSAR, Section 7.2.

3. ~~WCAP-8746-A, March 1977~~.

4. ~~WCAP-9273-NP-A, July 1985~~.

Insert 1

The reactor core SLs are established to preclude violation of the following fuel design criteria:

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

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

The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operations, normal operational transients, and AOOs.

BASES

APPLICABLE SAFETY ANALYSES (continued)

each analyzed transient. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit ~~of 2200 psig~~ and the RCS average temperature limit **specified in the COLR** ~~of 580.3°F~~ correspond to analytical limits ~~of 2185 psig and 583.2°F~~ used in the safety analyses, with allowance for measurement uncertainty.

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

LCO

This LCO specifies limits on the monitored process variables — pressurizer pressure, RCS average temperature, and RCS total flow rate — to ensure the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNB design criterion in the event of a DNB limited transient.

RCS total flow rate is based on two elbow tap measurements from each loop and contains a measurement error of 2.3% based on Δp measurements from the cold leg elbow taps, which are correlated to past precision heat balance measurements or performing a precision heat balance at the beginning of the current cycle. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi raises the nominal flow measurement allowance to 2.4%.

Any fouling that might bias the flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

BASES

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure 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 that DNB is not a concern.

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp > 5% RTP per minute or a THERMAL POWER step > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

~~Another set of limits~~ **The DNBR limit** on DNB related parameters is provided in SL 2.1.1, "Reactor Core SLs." ~~Those limits~~ **The conditions that define the DNBR limit** are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, as required by Required Action B.1, to restore DNB margin and eliminate the potential for violation of the accident analysis bounds.

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

(continued)

Attachment 6

Clean Typed Technical Specification Changed Bases Pages

Pages
B 2.1.1-2
B 2.1.1-3
B 2.1.1-4
B 3.4.1-2
B 3.4.1-3

files: R1COLRB2-1-1, R1COLR3_4_1CT

BASES

APPLICABLE SAFETY ANALYSES

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

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

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

In meeting the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and computer codes must be considered. As described in the FSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty to determine design limit DNBR values that satisfy the DNB design criterion.

Additional DNBR margin is maintained by performing the safety analyses to a higher DNB limit. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility.

The Reactor Trip System Functions (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions (i.e., resulting from a Condition I or II event) for Reactor Coolant System (RCS) temperature, pressure, flow, ΔI and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by appropriate operation of the RPS and the steam generator safety valves.

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses provide more restrictive limits to ensure that the SLs are not exceeded.

BASES

SAFETY LIMITS	The reactor core SLs are established to preclude violation of the following fuel design criteria: a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience DNB; and b. The hottest fuel pellet in the core must not experience centerline fuel melting. The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operations, normal operational transients, and AOOs.
APPLICABILITY	SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The main steam safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.
SAFETY LIMIT VIOLATIONS	If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable. The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. FSAR, Section 7.2.

BASES

APPLICABLE SAFETY ANALYSES (continued)	each analyzed transient. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)." The pressurizer pressure limit and the RCS average temperature limit specified in the COLR correspond to analytical limits used in the safety analyses, with allowance for measurement uncertainty. The RCS DNB parameters satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	This LCO specifies limits on the monitored process variables—pressurizer pressure, RCS average temperature, and RCS total flow rate—to ensure the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNB design criterion in the event of a DNB limited transient. RCS total flow rate is based on two elbow tap measurements from each loop and contains a measurement error of 2.3% based on Δp measurements from the cold leg elbow taps, which are correlated to past precision heat balance measurements or performing a precision heat balance at the beginning of the current cycle. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi raises the nominal flow measurement allowance to 2.4%. Any fouling that might bias the flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

BASES

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure 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 that DNB is not a concern.

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp > 5% RTP per minute or a THERMAL POWER step > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

The DNBR limit on DNB related parameters is provided in SL 2.1.1, "Reactor Core SLs." The conditions that define the DNBR limit are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, as required by Required Action B.1, to restore DNB margin and eliminate the potential for violation of the accident analysis bounds.

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

(continued)

Attachment 7

Example COLR Pages

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for FNP UNIT 1 CYCLE 17 has been prepared in accordance with the requirements of Technical Specification 5.6.5.

The Technical Requirement affected by this report is listed below:

13.1.1 SHUTDOWN MARGIN - MODES 1 and 2 (with $k_{eff} \geq 1$)

The Technical Specifications affected by this report are listed below:

2.1.1 Reactor Core Safety Limits
3.1.1 SHUTDOWN MARGIN - MODES 2 (with $k_{eff} < 1$), 3, 4 and 5
3.1.3 Moderator Temperature Coefficient
3.1.5 Shutdown Bank Insertion Limits
3.1.6 Control Bank Insertion Limits
3.2.1 Heat Flux Hot Channel Factor - $F_Q(Z)$
3.2.2 Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}^N$
3.2.3 Axial Flux Difference
3.3.1 Reactor Trip System Instrumentation (OTΔT and OPΔT setpoints)
3.4.1 RCS Pressure, Temperature and Flow Departure from Nucleate Boiling Limits
3.9.1 Boron Concentration

2.7 Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}^{N}$ (Specification 3.2.2)

2.7.1 $F_{\Delta H}^{N} \leq F_{\Delta H}^{RTP} * \left(1 + PF_{\Delta H} * (1 - P)\right)$

where: $P = \dfrac{THERMAL\ POWER}{RATED\ THERMAL\ POWER}$

2.7.2 $F_{\Delta H}^{RTP} = 1.70$

2.7.3 $PF_{\Delta H} = 0.3$

2.8 Axial Flux Difference (Specification 3.2.3)

2.8.1 The Axial Flux Difference (AFD) acceptable operation limits are provided in Figure 3.

2.9 Boron Concentration (Specification 3.9.1)

2.9.1 The boron concentration shall be greater than or equal to 2000 ppm.[1]

2.10 Reactor Core Safety Limits for THERMAL POWER (Specification 2.1.1)

2.10.1 In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature , and pressurizer pressure shall not exceed the SLs specified in Figure 8.

2.11 Reactor Trip System Instrumentation Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) setpoint parameter values for Table 3.3.1-1 (Specification 3.3.1)

2.11.1 The Reactor Trip System Instrumentation Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) setpoint parameter values for TS Table 3.3.1-1 are listed in COLR Tables 2 and 3.

2.12 RCS DNB parameters for pressurizer pressure, RCS average temperature and RCS total flow rate shall be within the limits specified below: (Specification 3.4.1)

a. Pressurizer pressure ≥ 2209 psig;

b. RCS average temperature ≤ 580.3°F, and

c. The minimum RCS total flow rate shall be ≥ 263,400 GPM when using the precision heat balance method and ≥ 264,200 GPM when using the elbow tap method

1 This concentration bounds the condition of $k_{eff} \leq 0.95$ (all rods in less the most reactive rod) and subcriticality (all rods out) over the entire cycle. This concentration includes additional boron to address uncertainties and B^{10} depletion.

Table 2

The Reactor Trip System Instrumentation Overtemperature ΔT (OTΔT) Setpoints used with Technical Specification Table 3.3.1-1

$T' \leq 577.2\ °F$	$P' = 2235$ psig	
$K_1 = 1.17$	$K_2 = 0.017/°F$	$K_3 = 0.000825/psi$
$\tau_1 \geq 30$ sec	$\tau_2 \leq 4$ sec	
$\tau_4 = 0$ sec	$\tau_5 \leq 6$ sec	$\tau_6 \leq 6$ sec

$$f_1(\Delta I) = \begin{cases} -2.48\{23 + (q_t - q_b)\} & \text{when } (q_t - q_b) \leq -23\% \text{ RTP} \\ 0\% \text{ of RTP} & \text{when } -23\% \text{ RTP} < (q_t - q_b) \leq 15\% \text{ RTP} \\ 2.05\{(q_t - q_b) - 15\} & \text{when } (q_t - q_b) > 15\% \text{ RTP} \end{cases}$$

Table 3

The Reactor Trip System Instrumentation Overpower ΔT (OPΔT) Setpoints used with Technical Specification Table 3.3.1-1

$T'' \leq 577.2$

$K_4 = 1.10$	$K_5 = 0.02/°F$ for increasing T_{avg} $K_5 = 0/°F$ for decreasing T_{avg}	$K_6 = 0.00109/°F$ when $T > T''$ $K_6 = 0/°F$ when $T \leq T''$

$\tau_3 \geq 10$ sec

$\tau_4 = 0$ sec

$\tau_5 \leq 6$ sec

$\tau_6 \leq 6$ sec

$f_2(\Delta I) = 0\%$ RTP for all ΔI.

Figure 8
Core Safety Limits