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ComEd

December 22, 1999

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

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Request for Technical Specifications Change
Expanded Core Operating Limits Reports

References: (1) Westinghouse WCAP-14483, "Generic Methodology for Expanding Core Operating Limits Report," November 1995.

(2) Letter from Thomas H. Essig (NRC) to Andrew Drake (Westinghouse Owners Group), "Acceptance For Referencing Of Licensing Topical Report WCAP-14483, 'Generic Methodology For Expanded Core Operating Limits Report,'" January 19, 1999.

In accordance with 10 CFR 50.90, we request changes to Technical Specifications (TS), of Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. The proposed changes relocate Reactor Coolant System (RCS) related cycle-specific parameter limits from the TS to, and thus expanding, the Core Operating Limits Reports (COLRs) for Braidwood Station and Byron Station. The justification to implement the expansion of the COLRs is provided in Westinghouse WCAP-14483-A, "Generic Methodology for Expanding Core Operating Limits Report," (Reference 1). This justification has subsequently been approved by the NRC (Reference 2). The proposed changes will allow us the flexibility to enhance plant operating margin and core design margin without the need for cycle-specific license amendment requests.

Specifically, the proposed changes to TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and associated Bases relocate the pressurizer pressure, RCS average temperature, and RCS total flow rate values to the

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COLRs. However, as discussed in the Reference, the minimum limit for RCS total flow rate
is being retained in TS 3.4.1. The proposed changes to TS Table 3.3.1-1, "Reactor Trip System Instrumentation," relocate the Overtemperature ΔT and Overpower ΔT constant (K) values, the dynamic compensation time (τ) values, and the breakpoint and slope values for the $f(\Delta I)$ penalty functions to the COLRs. Additionally, the proposed changes to TS 2.1.1, "Reactor Core Safety Limits," and associated Bases relocate the reactor core safety limit figure to the COLRs and replace it with the more specific fuel DNB Ratio (DNBR) and peak fuel centerline temperature safety limit requirements.

NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," dated October 4, 1988, provides guidance to licensees for the removal of such cycle-dependent variables from the TS provided that the values of these variables are included in a COLR and are determined with NRC-approved methodology which is referenced in the TS. The changes we are proposing herein meet these criteria. The proposed changes will obviate the need for future revisions of the TS to change the value of those operating limits which cannot be specified to reasonably bound several operating cycles without incurring a significant loss of operating flexibility. We request approval of these proposed changes prior to July 1, 2000, to support the issuance of the expanded revised COLRs for the upcoming reload designs for refueling outages in 2000 for Braidwood Station, Unit 2, and Byron Station, Unit 1. An expanded revised COLR for each of the four units will be issued concurrent with implementation of the approved TS amendment requested herein.

This request is subdivided as follows.

1. Attachment A gives a description and safety analysis of the proposed changes.

2. Attachments B-1 and B-2 include the marked-up TS pages, and marked-up TS Bases pages for information only, with the proposed changes indicated for Braidwood Station and Byron Station, respectively. Attachments B-3 and B-4 include the associated pages with the proposed changes incorporated for Braidwood Station and Byron Station, respectively.

3. Attachment C describes our evaluation performed using the criteria in 10 CFR 50.91(a)(1), which provides information supporting a finding of no significant hazards consideration using the standards in 10 CFR 50.92(c).

4. Attachment D provides information supporting an environmental assessment.

5. Attachment E provides the information that will be relocated from the TS to the expanded COLR for each unit upon implementation of the TS amendment.

The proposed changes have been reviewed by the Braidwood Station and Byron Station Plant Operations Review Committee and the Nuclear Safety Review Board in accordance with the Quality Assurance Program.

ComEd is notifying the State of Illinois of this application request for changes to the TS by transmitting a copy of this letter and its attachments to the designated State Official.

Should you have any questions concerning this letter, please contact Ms. K. M. Root at (630) 663-7292.

Respectfully,

R. M. Krich
Vice President - Regulatory Services

Attachments:

Affidavit
Attachment A: Description and Safety Analysis for Proposed Changes
Attachment B-1: Marked-up Pages for Proposed Changes for Braidwood Station
Attachment B-2: Marked-up Pages for Proposed Changes for Byron Station
Attachment B-3: Incorporated Proposed Changes for Braidwood Station
Attachment B-4: Incorporated Proposed Changes for Byron Station
Attachment C: Information Supporting a Finding of No Significant Hazards Consideration
Attachment D: Information Supporting an Environmental Assessment
Attachment E: Expanded COLR Information for Braidwood Station and Byron Station

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Braidwood Station
NRC Senior Resident Inspector - Byron Station
Office of Nuclear Facility Safety - Illinois Department of Nuclear Safety

STATE OF ILLINOIS	)	
COUNTY OF DUPAGE	)	
IN THE MATTER OF	)	
COMMONWEALTH EDISON (COMED) COMPANY	)	**Docket Nos.**
BRAIDWOOD STATION - UNITS 1 and 2	)	**STN 50-456 and STN 50-457**
BYRON STATION - UNITS 1 and 2	)	**STN 50-454 and STN 50-455**

SUBJECT: **Request for Technical Specifications Change, Expanded Core Operating Limits Reports**

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

R. M. Krich
Vice President - Regulatory Services

Subscribed and sworn to before me, a Notary Public in and

for the State above named, this 22 day of

December, 1999.

OFFICIAL SEAL
JEFFERY A BATARA
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES:02/24/02

Notary Public

(OFFICIAL SEAL)

ATTACHMENT A

DESCRIPTION AND SAFETY ANALYSIS FOR PROPOSED CHANGES

A. SUMMARY OF PROPOSED CHANGES

Pursuant to 10 CFR 50.90, we are requesting changes to Technical Specifications (TS), of Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66 for the Braidwood Station, Units 1 and 2, and the Byron Station, Units 1 and 2, respectively. The proposed changes relocate Reactor Coolant System (RCS) related cycle-specific parameter limits from the TS to, and thus expanding, the Core Operating Limits Reports (COLRs) for Braidwood Station and Byron Station.

Specifically, the proposed changes to TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and associated Bases relocate the pressurizer pressure, RCS average temperature, and RCS total flow rate values to the COLRs. However, as discussed in References 1 and 2, the minimum limit for RCS total flow rate is being retained in TS 3.4.1. The proposed changes to TS Table 3.3.1-1, "Reactor Trip System Instrumentation," relocate the Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) constant (K) values, the dynamic compensation time (τ) values, and the breakpoint and slope values for the f(ΔI) penalty functions to the COLRs. Additionally, the proposed changes to TS 2.1.1, "Reactor Core Safety Limits," and associated Bases relocate the reactor core safety limit figure to the COLRs and replace it with the more specific fuel DNB Ratio (DNBR) and peak fuel centerline temperature safety limit requirements.

The justification to implement the expansion of the COLRs is provided in Westinghouse WCAP-14483-A, "Generic Methodology for Expanding Core Operating Limits Report," (Reference 1), as approved by the NRC in Reference 2. The proposed changes will allow us the flexibility to enhance plant operating margin and core design margin without the need for cycle-specific license amendment requests. We request approval of the proposed changes prior to July 1, 2000, to support the issuance of the expanded revised COLRs for the upcoming reload designs for Braidwood Station, Unit 2 (i.e., A2RO8), and Byron Station, Unit 1 (i.e., B1R10). An expanded revised COLR for each of the four units will be issued concurrent with implementation of the approved TS amendment requested herein.

The proposed changes are described in detail in Section E of this Attachment A. The marked-up TS pages, and marked-up TS Bases pages which are provided for information only, are shown in Attachments B-1 and B-2 for Braidwood Station and Byron Station, respectively. Attachments B-3 and B-4 include the associated pages with the proposed changes incorporated for Braidwood Station and Byron Station, respectively.

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

<u>RCS Pressure, Temperature and Flow DNB Parameters</u>

TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," specifies the RCS DNB parameter limits that shall be maintained; pressurizer pressure $\geq$ 2219 psig, RCS average temperature $\leq$ 591.2 °F, and RCS total flow rate $\geq$ 371,400 gpm.

Overtemperature ΔT and Overpower ΔT Parameters

TS Table 3.3.1-1, "Reactor Trip System Instrumentation," Note 1: OTΔT, and Note 2: OPΔT, specify the setpoint parameter constant (K) values, dynamic compensation time (τ) values, T′, T″, P′, and the breakpoint and slope values for the f(ΔI) penalty functions for the trip setpoints.

Reactor Core Safety Limits

TS 2.1.1, "Reactor Core Safety Limits," specifies that the combination of Thermal Power, RCS highest loop average temperature, and pressurizer pressure shall not exceed the safety limits specified in Figure 2.1.1-1, "Reactor Core Safety Limits."

COLR Analytical Methods

TS Section 5.6.5b, "Core Operating Limits Report (COLR)," specifies the analytical methods previously reviewed and approved by the NRC that are used to determine the core operating limits.

C. BASES FOR THE CURRENT REQUIREMENTS

RCS Pressure, Temperature and Flow DNB Parameters

The TS limits on the DNB parameters assure that pressurizer pressure, RCS average temperature, and RCS total flow rate will be maintained within the limits of steady-state operation assumed in the accident analyses. These limits are consistent with the initial full power conditions considered in the Updated Final Safety Analysis Report (UFSAR) accident analyses. For Condition I and II events for which precluding DNB is the primary criterion, the safety analyses have demonstrated that the DNB design basis is satisfied, assuming that the plant is operating in compliance with the TS DNB parameter limits prior to initiation of the event. The DNB parameter limits are also based on the initial conditions assumed for Condition III and IV events for which precluding DNB is not a criterion. Given that the DNB parameter limits ensure that the DNB design basis and other safety criteria are satisfied, continuous plant operation at less than limiting conditions would result in margin to these safety criteria.

Overtemperature ΔT and Overpower ΔT Parameters

The OTΔT and OPΔT reactor trip functions ensure that during any Condition I or II transient, there is at least a 95% probability that the peak kW/ft fuel rods will not exceed the uranium dioxide, UO_2, melting temperature. To achieve this, a fuel centerline temperature limit has been established based on the melting temperature for UO_2 of 5080 °F, decreasing by 58 °F per 10,000 MWD/MTU of burnup. For design purposes, this fuel centerline temperature limit of 4700 °F is significantly below the melting temperature to allow for fuel temperature calculation and other uncertainties. In addition, the DNB design basis is defined as the probability being at least 95% at a 95% confidence level that DNB will not occur on the limiting fuel rod(s). If DNB is precluded, adequate heat transfer is assured between the fuel cladding and the reactor coolant, and damage due to inadequate cooling is prevented.

The OTΔT reactor trip function, in conjunction with the OPΔT reactor trip function, ensures operation within the DNB design basis and within the hot-leg boiling limits. Since both of these limits are functions of the coolant temperature, pressure, and core thermal power, the OTΔT reactor trip function is correlated with the vessel ΔT, the RCS average temperature, and pressurizer pressure. A compensating term which is a function of ΔI is also factored into the OTΔT setpoint to account for the effect of changes in the axial power shape. The setpoint is scaled to be consistent with the full power operating conditions.

The OPΔT reactor trip function, in conjunction with the OTΔT reactor trip function, ensures operation within the fuel temperature design basis. This is accomplished through the OPΔT reactor trip function by correlating the core thermal power with the temperature difference across the vessel (ΔT). Since the thermal power is not precisely proportional to ΔT, because of the effects of changes in coolant density and heat capacity, a compensation term, which is a function of vessel average temperature, is factored into the calculated OPΔT trip setpoint. The setpoint is set to be consistent with the nominal full power operating conditions.

Reactor Core Safety Limits

TS Figure 2.1.1-1, "Reactor Core Safety Limits," (RCSL) presents the limiting RCS average temperature conditions as a function of pressurizer pressure and fractional Rated Thermal Power. This figure is included in the TS to satisfy the requirements of 10 CFR 50.36 which states that, "safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity." The RCSL figure provides the relationship between RCS average temperature, pressurizer pressure and Rated Thermal Power level, and the DNB design basis. If a Condition I or II event were to occur, the RCSL figure could be used to determine whether or not the DNB design basis was met.

Because the RCSL figure is used in the generation of the OTΔT and OPΔT reactor trip setpoints, it contains the hot-leg boiling limits, which are not true safety limits. The hot-leg boiling limits preclude saturation conditions to ensure that the measured ΔT remains proportional to thermal power. The DNB limits of the figure are based on the DNBR safety analysis limit and assume a specific RCS flow rate and a symmetrical reference axial power shape. Based on this figure, the gains (K_1 through K_6) of the OTΔT and OPΔT reactor trip setpoints are generated. For non-symmetrical power shapes that are more limiting than the reference axial power shape, the $f(\Delta I)$ penalty functions reduce the corresponding trip setpoints. Thus, the OTΔT and OPΔT reactor trip setpoints ensure that the RCSL figure is satisfied during a Condition I or II event, and ensure that for non-symmetrical axial power shapes that the DNB design basis is satisfied. Because the OTΔT and OPΔT reactor trip setpoints are based on the RCSL figure, the only way to violate the figure is under the postulated condition where both trains of the Reactor Protection System (RPS) do not function as designed. Operation of the RPS and the Main Steam Safety Valves (MSSVs) ensure that the DNB design basis is satisfied for any Condition I or II transient, independent of the RCSL figure.

COLR Analytical Methods

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

D. NEED FOR REVISION OF THE REQUIREMENT

NRC Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," (Reference 3), provides guidance to licensees for the removal of such cycle-dependent variables from the TS provided that the values of these variables are included in a COLR and are determined in accordance with NRC-approved methodology which is referenced in the TS. The changes we are proposing herein meet these criteria. The proposed changes will obviate the need for frequent future revisions of the TS to change the values of those operating limits which cannot be specified to reasonably bound several operating cycles without incurring a significant loss of operating flexibility.

RCS Pressure, Temperature and Flow DNB Parameters

The planned cycle-specific operating configuration for pressurizer pressure, RCS average temperature, and RCS total flow rate is assumed in the core reload design process described in Reference 4, which demonstrates that these reload related parameters assumed in the safety analyses are valid for the cycle in question. This ensures that the safety analyses remain bounding and the conclusions of the UFSAR remain valid. To minimize any licensing impacts associated with cycle-to-cycle changes in these DNB parameters, significant margin is allocated to support the current TS. The disadvantage is that the margin allocated to support the conservative TS limits cannot be easily utilized. To better utilize the margin currently allocated to support the existing DNB parameter limit values, it is proposed that these parameters be relocated to the COLRs. This will ensure that available margins are not unnecessarily allocated and thus unavailable on a cycle-specific basis just to support overly conservative TS limits, and is consistent with the NRC's guidance provided in Reference 3.

Overtemperature ΔT and Overpower ΔT Parameters

Relocating the OTΔT and OPΔT setpoint parameter values (K_1 through K_6, τ_1 through τ_7, T', T'', P', and $f(\Delta I)$ functions) to the COLRs is primarily based on the fact that the OTΔT and OPΔT setpoints are based on several important reload design parameters. Changes in these reload-related parameters can impact the OTΔT and OPΔT reactor trip setpoints on a cycle-specific basis. As the setpoints are based on core design parameters which are verified on a cycle-specific basis, which can be used on a cycle-specific basis to verify fuel design criteria, and which have significant margin which cannot be fully utilized, it is proposed that these parameters be relocated to the COLRs. This relocation would minimize the chance that a reload-related parameter change would necessitate a TS change. In addition, margin that is currently utilized in the setpoints could be made available to provide enhanced setpoints.

Reactor Core Safety Limits

Because there are limitations associated with the RCSL figure, replacing the figure will eliminate the potential for drawing an incorrect conclusion with respect to the DNB design basis. The violation of a safety limit could only result if the RPS was not functioning as designed, in which case, the use of the RCSL figure has limitations in determining whether or not the DNB design basis had been violated. As the hot-leg boiling limits are not true safety limits, a violation of the hot-leg boiling limits of the RCSL figure does not necessarily mean a safety limit has been violated. As the RCSL figure assumes all Reactor Coolant Pumps are operating, if a partial or complete loss of flow transient occurs, the figure is not valid under the reduced RCS flow conditions. As the DNBR limit lines presented in the RCSL figure are based on a reference axial power shape, if the axial power shape during a transient were to become more limiting, or less limiting, than that used to define the DNBR limits of the RCSL figure, the figure could lead to an invalid conclusion as to whether or not the DNB design basis is satisfied. As the RCSL figure has built-in DNBR margin, a violation of the safety limits does not necessarily indicate that the plant has violated the licensed DNB design basis, which is the true safety limit. The

RPS and the MSSVs ensure all safety limits will be met, independent of the RCSL figure. Therefore, the use of the figure to determine whether or not a safety limit had been violated is not critical. In the event of a Condition I or II transient, verification that the RPS and the MSSVs are functioning as designed will ensure that all safety limits are met.

COLR Analytical Methods

Although the Figure 2.1.1-1 RCSL is being relocated to the COLR, the "requirement" for the RCSL figure is being retained in the TS to ensure that the limits upon important process variables continue to meet the requirements of 10 CFR 50.36. This requirement is maintained by referencing the analytical methods previously reviewed and approved by the NRC that are used to derive the parameters in the RCSL figure in the Reporting Requirements section of the TS. The Reference 4 methodology used to calculate the RCSL figure is specifically required by TS Section 5.6.5b, as one of the analytical methods to determine the core operating limits.

As the OTΔT and OPΔT setpoint parameter values are being relocated to the COLR, the applicable NRC reviewed and approved setpoint methodology, Westinghouse WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," (Reference 5), is being added to the administrative reporting requirements, TS Section 5.6.5b, "Core Operating Limits Report (COLR)."

E. DESCRIPTION OF THE PROPOSED CHANGES

RCS Pressure, Temperature and Flow DNB Parameters

The TS 3.4.1 limits specified for the RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate are being relocated to the COLR for each of the four units.

Because the RCS total flow rate is being relocated to the COLRs, the minimum limit for RCS total flow rate of ≥ 371,400 gpm, based on NRC approved analysis, is being retained in TS 3.4.1. As documented in the NRC Request for Additional Information (RAI), (Reference 6), a change in RCS flow from cycle-to-cycle is an indication of a physical change to the plant that should be reviewed by the NRC. To comply with this recommendation and the Westinghouse Owners Group (WOG) response, (Reference 7), the minimum limit for RCS total flow rate is being retained in TS 3.4.1 to assure that a lower flow rate than reviewed by the NRC will not be used.

Overtemperature ΔT and Overpower ΔT Parameters

The TS Table 3.3.1-1, Note 1: OTΔT, and Note 2: OPΔT, setpoint parameter constant (K_1 through K_6) values, dynamic compensation time (τ_1 through τ_7) values, T′, T″, P′, and the breakpoint and slope values for the f(ΔI) penalty functions for the trip setpoints are being relocated to the COLR for each of the four units.

Reactor Core Safety Limits

TS Figure 2.1.1-1, "Reactor Core Safety Limits," is being relocated to the COLR for each of the four units, and is being replaced with more specific fuel DNBR and peak fuel centerline temperature safety limits.

As documented in the WOG response (Reference 7) to the NRC RAI (Reference 6) associated with Reference 1, because the RCSL figure is being relocated to the COLR, the "requirement" for this RCSL figure will be retained in the TS. The methodology used to calculate the RCSL figure is contained in Westinghouse WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," (Reference 4). The existing TS Section 5.6.5b, COLR Reporting Requirements, specifically requires Reference 4 to be used as one of the analytical methods to determine the core operating limits. Therefore, the NRC's request (Reference 6), that the NRC-approved methodology used to derive the parameters in the figure should be referenced in the Reporting Requirements section of the TS, is currently in-place.

COLR Analytical Methods

An additional reference is being added to the listing of NRC reviewed and approved analytical methods used to determine the core operating limits provided in TS Section 5.6.5b, "Core Operating Limits Report (COLR)." Specifically, Westinghouse WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," (Reference 5), is being added.

F. SAFETY ANALYSIS OF THE PROPOSED CHANGES

The justification to implement the expansion of the COLRs is provided in Westinghouse WCAP-14483-A, "Generic Methodology for Expanding Core Operating Limits Report," (Reference 1), as approved in the NRC Safety Evaluation Report transmitted by T. H. Essig (NRC) to A. Drake (WOG), dated January 19, 1999, (Reference 2).

RCS Pressure, Temperature and Flow DNB Parameters

Relocating the DNB parameters limit values to the COLRs would allow the flexibility to utilize the available margins to increase cycle operating margins and improve core reload designs without the requirement of cycle-specific license amendments. The relocation of the DNB parameters to the COLRs would result in more complete COLRs containing not only cycle-specific core reload-related parameters, but also cycle-specific operating condition parameters. Thus the safety analyses could credit the actual cycle-specific operating condition in the same way that the core reload designs currently do. The COLRs and safety analyses would more closely reflect the cycle-specific conditions that the plant control and protection systems are set to for a given cycle.

Overtemperature ΔT and Overpower ΔT Parameters

Relocation of the OTΔT and OPΔT setpoint parameter values to the COLRs would minimize the chance that a reload-related parameter change would necessitate a TS change. In addition, significant DNB and operating margin currently utilized in the setpoints that is unnecessarily allocated and thus unavailable could be utilized to enhance plant operating margins, enhance the OTΔT and OPΔT setpoints, and increase the flexibility of the core designs without any reduction in the margin of safety.

Reactor Core Safety Limits

Relocating the RCSL figure to the COLRs would eliminate the possibility of reaching an incorrect conclusion concerning whether or not a safety limit has been violated for a Condition I or II event. Additionally, removal of the RCSL figure would prevent the possibility of misusing the RCSL figure to define an "acceptable" operating configuration that could result in the plant being placed in an unanalyzed condition.

It is proposed that the RCSL figure be replaced with the DNB design basis limit and the fuel centerline melting limit as these limits are criteria that must be satisfied for all Condition I and II transients. Confirming that the RPS and the MSSVs are functioning as designed will ensure that both the DNB design basis and fuel centerline melting criteria are satisfied for any Condition I or II event. With this approach, the chance of reaching an incorrect conclusion with respect to the safety limits would be greatly reduced, if not eliminated.

COLR Analytical Methods

NRC GL 88-16 allows removal of cycle-dependent variables from the TS provided that the values of these variables are included in a COLR and are determined with NRC reviewed and approved methodology which is referenced in the TS. Safety limits, however, may not be placed in the COLR. Therefore, the "requirement" for the RCSL figure is being retained in the TS by the current reference to the NRC reviewed and approved methodology used to calculate the RCSL figure in TS Section 5.6.5b, (Reference 4). The applicable NRC reviewed and approved setpoint methodology for the OTΔT and OPΔT setpoint parameter values being relocated to the COLR, (Reference 5), is being added as a referenced analytical method in TS Section 5.6.5b.

G. IMPACT ON PREVIOUS SUBMITTALS

We have reviewed the proposed changes regarding their impact on any previous submittals and have determined that there is no impact on any previous submittals.

H. SCHEDULE REQUIREMENTS

We request approval of the proposed changes prior to July 1, 2000, to support the issuance of the expanded revised COLRs for the upcoming reload designs for Braidwood Station, Unit 2 (i.e., A2RO8), and Byron Station, Unit 1 (i.e., B1R10). An expanded revised COLR for each of the four units will be issued concurrent with implementation of the approved TS amendment requested herein.

I. REFERENCES

1. Westinghouse WCAP-14483-A, "Generic Methodology for Expanding Core Operating Limits Report," November 1995.

2. T. H. Essig (NRC) Letter to A. P. Drake (WOG), "Acceptance For Referencing Of Licensing Topical Report WCAP-14483, 'Generic Methodology for Expanded Core Operating Limits Report,'" January 19, 1999.

3. NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," October 4, 1988.

4. Westinghouse WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

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

6. P. C. Wen (NRC) Request for Additional Information to A. P. Drake (WOG), September 2, 1998.

7. L. F. Liberatori Jr. (WOG) Response to Request for Additional Information to NRC Document Control Desk, November 25, 1998.

ATTACHMENT B-1

MARKED-UP PAGES FOR PROPOSED CHANGES FOR BRAIDWOOD STATION

REVISED PAGES

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 **(T_{avg})**, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ~~≥ 2219 psig~~ **within the limit specified in the COLR**;

b. RCS average temperature (T_{avg}) ~~≤ 591.2°F~~ **within the limit specified in the COLR**; and

c. RCS total flow rate ≥ 371,400 gpm **and within the limit specified in the COLR**.

---------------------------NOTE---------------------------

Pressurizer pressure limit does not apply during:

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

b. THERMAL POWER step > 10% RTP.

APPLICABILITY: MODE 1.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is ~~≥ 2219 psig~~ **within the limit specified in the COLR**.	12 hours
SR 3.4.1.2	Verify RCS average temperature (T_{avg}) is ~~≤ 591.2°F~~ **within the limit specified in the COLR**.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ 371,400 gpm **and within the limit specified in the COLR**.	12 hours
SR 3.4.1.4	-------------------NOTE------------------- Not required to be performed until 7 days after ≥ 90% RTP. --	
	Verify by precision heat balance that RCS total flow rate is ≥ 371,400 gpm **and within the limit specified in the COLR**.	18 months

BASES

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNBR criterion ~~of ≥ 1.4~~. 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 DNB 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)."

Safety Analyses assumed a value of 2207 psia (2192.3 psig). This value is bounded by the ~~LCO value of 2219 psig~~ **limit specified in the COLR** assuming a measurement accuracy of less than 26.7 psi.

Safety Analyses assumed a value of 588.4°F for the vessel average temperature. In addition, the analyses assumed the calculated error (including the ± 4°F dead band for the rod control system) for the temperature is 8.74°F (2σ random error of 7.6°F plus the 1.14°F bias error). The value assumed in the non-Revised Thermal Design Procedure (non-RTDP) analyses is -8°F, +9.5°F. For the RTDP analyses, a value of ± 7.6°F with a bias of +1.5°F is assumed.

Safety Analyses assumed a total RCS flow rate of 358,800 gpm. This value is bounded by the LCO value of 371,400 gpm **and the limit specified in the COLR** assuming a flow measurement uncertainty of 3.5%.

This 3.5% flow measurement uncertainty assumed in the analyses included errors from all sources including fouling ~~in~~ **of** the **feedwater** venturi. The use of 3.5% flow error is acceptable if actual uncertainty is unknown. At the time analyses were performed, the flow accuracy was unavailable. Subsequent calculations determined the error to be less than 3.5%.

Any fouling that might bias the flow rate measurement greater than ~~0.1%~~ **the penalty for undetected fouling of the feedwater venturi** can be detected by monitoring and trending various plant performance parameters. If detected, 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

APPLICABLE SAFETY ANALYSES (continued)

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 (T_{avg}), and RCS total flow rate–to ensure the core operates within the limits assumed in the safety analyses. **These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, based on maximum analyzed steam generator tube plugging, is retained in the LCO to assure that a lower flow rate than reviewed by the NRC will not be used.** Operating within these limits will result in meeting the DNB design criterion in the event of a DNB limited transient.

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

The DNBR limit ~~Another set of limits on DNB related parameters~~ is provided in SL 2.1.1, "Reactor Core SLs." LCO 3.4.1 represents the initial conditions of the safety analysis which are far more restrictive than the ~~Safety Limit (SL)~~ **conditions which define the DNBR limit**. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS total flow rate must be maintained during steady state operation in order to ensure DNB design 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.

Table 3.3.1-1 (page 5 of 6)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.04% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left(\frac{1}{1+\tau_3 s}\right) \le \Delta T_O \left\{ K_1 - K_2 \frac{(1+\tau_4 s)}{(1+\tau_5 s)}\left[T \frac{1}{(1+\tau_6 s)} - T' \right] + K_3 (P - P') - f_1(\Delta I) \right\}$$

Where: ΔT is measured Reactor Coolant System (RCS) ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T′ is the nominal T_{avg} at RTP, ≤ ~~588.4°F~~ *.

P is the measured pressurizer pressure, psig.
P′ is the nominal RCS operating pressure, = ~~2235 psig~~ *.

K_1 = ~~1.325~~ *	K_2 = ~~0.0297/°F~~ *	K_3 = ~~0.00181/psig~~ *
τ_1 = ~~8 sec~~ *	τ_2 = ~~3 sec~~ *	τ_3 ≤ ~~2 sec~~ *
τ_4 = ~~33 sec~~ *	τ_5 = ~~4 sec~~ *	τ_6 ≤ ~~2 sec~~ *

$f_1(\Delta I)$ = ~~-3.35~~ * {~~24~~ * + (q_t - q_b)} when q_t - q_b < ~~-24%~~ * RTP
0% of RTP when ~~-24%~~ * RTP ≤ q_t - q_b ≤ ~~10%~~ * RTP
~~4.11~~ * {(q_t - q_b) - ~~10~~ * } when q_t - q_b > ~~10%~~ * 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 6 of 6)
Reactor Trip System Instrumentation

<u>Note 2: Overpower ΔT</u>

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 3.60% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left(\frac{1}{1+\tau_3 s}\right) \le \Delta T_0 \left\{ K_4 - K_5 \frac{\tau_7 s}{1+\tau_7 s}\left(\frac{1}{1+\tau_6 s}\right) T - K_6 \left[T \frac{1}{1+\tau_6 s} - T'' \right] - f_2(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T″ is the nominal T_{avg} at RTP, ≤ ~~588.4°F~~ *.

K_4 = ~~1.072~~ *

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

K_6 = ~~0.00245/°F~~ * when T > T"
~~0/°F~~ * when T ≤ T"

τ_1 = ~~8 sec~~ *
τ_2 = ~~3 sec~~ *
τ_3 ≤ ~~2 sec~~ *
τ_6 ≤ ~~2 sec~~ *
τ_7 = ~~10 sec~~ *

$f_2(\Delta I)$ = ~~0 for all ΔI.~~ *

*** As specified in the COLR.**

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.~~

2.1.1.1 In MODE 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained ≥ 1.25 for the WRB-2 DNB correlation.

2.1.1.2 In MODE 2, the DNBR shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.30 for the W-3 DNB correlation.

2.1.1.3 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained ≤ 4700 °F.

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.

DELETE FIGURE

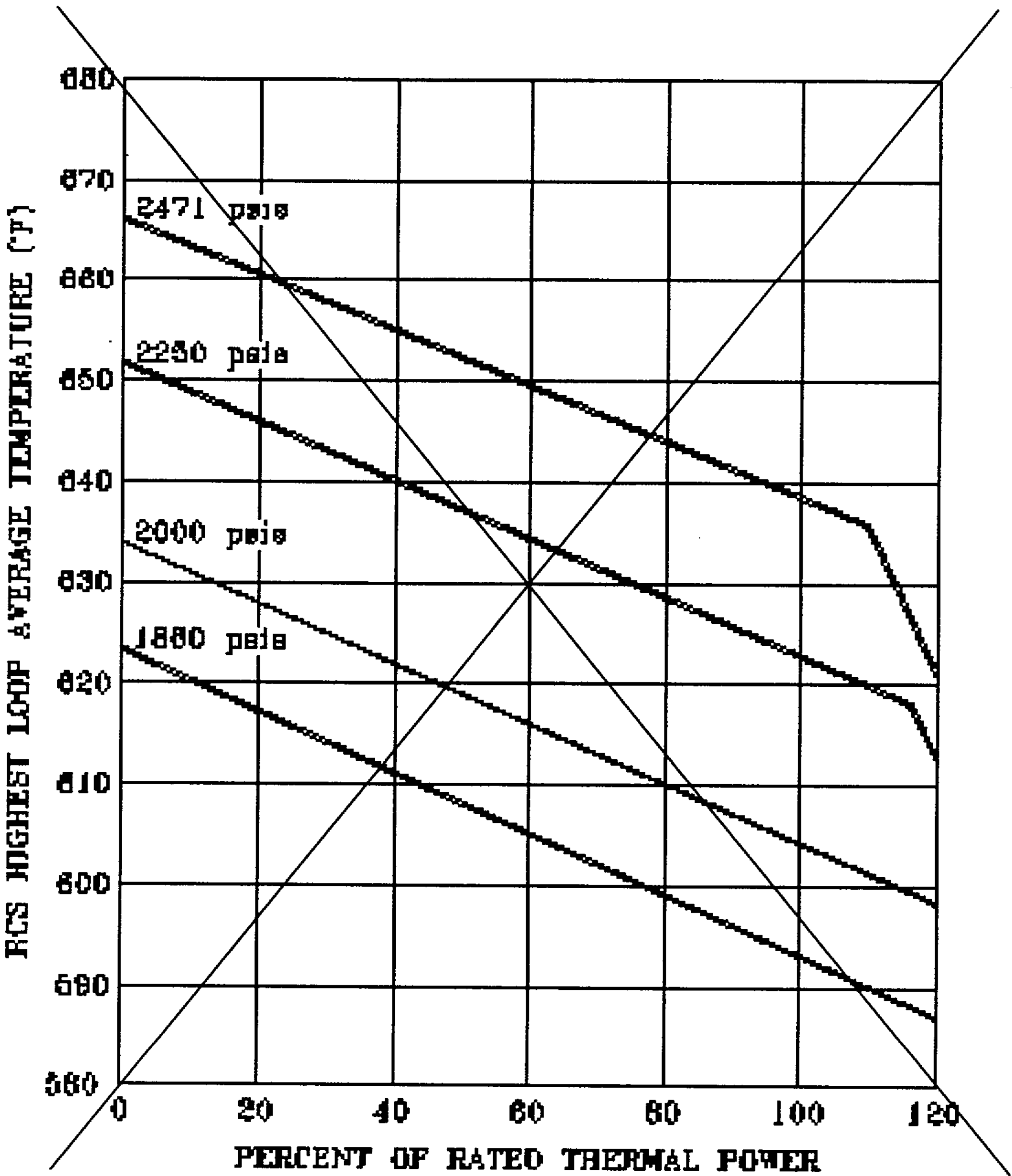

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits

BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) and Main Steam Safety Valves (MSSVs) prevents violation of the reactor core SLs.

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 that** the hot fuel pellet in the core must not experience centerline fuel melting; and

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

The Reactor Trip System setpoints (Ref. 2) specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation," in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) highest loop average temperature, pressurizer pressure, **RCS flow, Axial Flux Difference (AFD),** 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.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Automatic ~~enforcement~~ **preservation** of these reactor core SLs is provided by the ~~following functions~~ **appropriate operation of the RPS and the MSSVs** (Ref. 2)~~:~~**.**

~~a. Pressurizer Pressure - high trip;~~

~~b. Pressurizer Pressure - low trip;~~

~~c. Overtemperature ΔT trip;~~

~~d. Overpower ΔT trip;~~

~~e. Power Range Neutron Flux trip;~~

~~f. Low reactor coolant flow trip; and~~

~~g. Main steam safety valves.~~

~~Additional trip functions are provided to backup these functions for specific abnormal conditions.~~

~~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.~~

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

~~Figure B 2.1.1-1 shows an example of the reactor core safety limits of THERMAL POWER, RCS 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 or equal to the enthalpy of saturated liquid, or that the core exit quality is within the limits defined by the DNBR correlation.~~

BASES

SAFETY LIMITS (continued)

~~The curves are typically derived based on enthalpy hot channel factor limits such as those provided in the COLR. The dashed line of Figure B 2.1.1-1 shows an example of a limit curve at 2235 psig. In addition, it illustrates the various RPS functions that are designed to prevent the unit from reaching the limit.~~

~~The SL is higher than the limit calculated when the Axial Flux Difference (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).~~

⇒ **INSERT 1 HERE**

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 MSSVs 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. In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMITS 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.

INSERT 1

BASES

SAFETY LIMITS (continued)

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

a. There must be at least 95% probability that the hot fuel pellet in the core must not experience centerline fuel melting; and

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

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 AOOs. To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature ΔT 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 that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS and the MSSVs ensure that for variations in the RCS average temperature, pressurizer pressure, RCS flow, AFD, and THERMAL POWER that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. UFSAR, Section 7.2.
3. ~~WCAP-8746-A, March 1977.~~
4. ~~WCAP-9273-NP-A, July 1985.~~

DELETE FIGURE

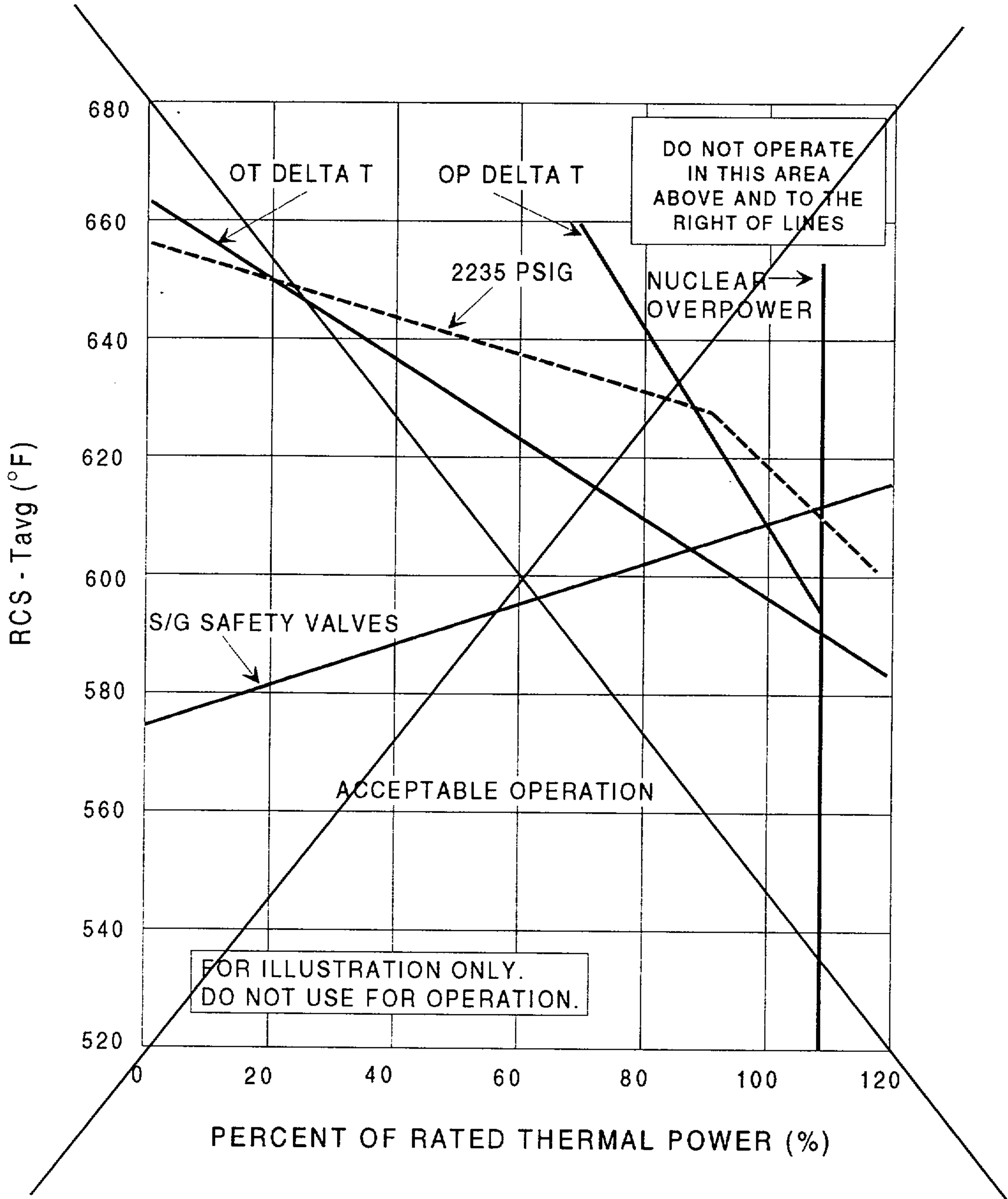

Figure B 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits vs. Boundary of Protection

5.6 Reporting Requirements

5.6.5 Core Operating Limits Report (COLR) (continued)

6. WCAP-9220-P-A, "Westinghouse ECCS Evaluation Model-1981 Version," February 1982.

7. WCAP-9561-P-A, Add. 3, "BART A-1: a Computer Code for Best Estimate Analysis of Reflood Transients - Special Report: Thimble Modeling in Westinghouse ECCS Evaluation Model," July 1986.

8. WCAP-10266-P-A, "The 1981 Version of Westinghouse Evaluation Model using BASH Code," March 1987, including Addendum 1 "Power Shape Sensitivity Studies," Revision 2-P-A, dated December 15, 1987, and Addendum 2 "BASH Methodology Improvements and Reliability Enhancements," Revision 2, Dated May 1988.

9. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.

10. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using NOTRUMP Code," August 1985.

11. WCAP-10216-A, Revision 1, "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification," February 1994.

12. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986;

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; and

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

ATTACHMENT B-2

MARKED-UP PAGES FOR PROPOSED CHANGES FOR BYRON STATION

REVISED PAGES

RCS DNB Parameters	Revised TS Page	Page 3.4.1-1
RCS DNB Parameters	Revised TS Page	Page 3.4.1-2
RCS DNB Parameters	Revised TS Bases Page	Page B 3.4.1-2
RCS DNB Parameters	Revised TS Bases Page	Page B 3.4.1-3
RTS OTΔT	Revised TS Page	Page 3.3.1-18
RTS OPΔT	Revised TS Page	Page 3.3.1-19
Reactor Core Safety Limits	Revised TS Page	Page 2.0-1
Reactor Core Safety Limits	Revised TS Page	Page 2.0-2
Reactor Core Safety Limits	Revised TS Bases Page	Page B 2.1.1-2
Reactor Core Safety Limits	Revised TS Bases Page	Page B 2.1.1-3
Reactor Core Safety Limits	Revised TS Bases Page	Page B 2.1.1-4
Reactor Core Safety Limits	(Insert Number One)	Page B 2.1.1-4
Reactor Core Safety Limits	Revised TS Bases Page	Page B 2.1.1-5
Reactor Core Safety Limits	Revised TS Bases Page	Page B 2.1.1-6
COLR Analytical Methods	Revised TS Page	Page 5.6-4

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 **(T_{avg})**, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ~~≥ 2219 psig~~ **within the limit specified in the COLR**;

b. RCS average temperature (T_{avg}) ~~≤ 591.2ºF~~ **within the limit specified in the COLR**; and

c. RCS total flow rate ≥ 371,400 gpm **and within the limit specified in the COLR**.

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

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

b. THERMAL POWER step > 10% RTP.

APPLICABILITY: MODE 1.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is ~~≥ 2219 psig~~ **within the limit specified in the COLR**.	12 hours
SR 3.4.1.2	Verify RCS average temperature (T_{avg}) is ~~≤ 591.2°F~~ **within the limit specified in the COLR**.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ 371,400 gpm **and within the limit specified in the COLR**.	12 hours
SR 3.4.1.4	------------------NOTE------------------ Not required to be performed until 7 days after ≥ 90% RTP. --- Verify by precision heat balance that RCS total flow rate is ≥ 371,400 gpm **and within the limit specified in the COLR**.	 18 months

BASES

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNBR criterion ~~of ≥ 1.4~~. 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 DNB 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)."

Safety Analyses assumed a value of 2207 psia (2192.3 psig). This value is bounded by the ~~LCO value of 2219 psig~~ **limit specified in the COLR** assuming a measurement accuracy of less than 26.7 psi.

Safety Analyses assumed a value of 588.4°F for the vessel average temperature. In addition, the analyses assumed the calculated error (including the ± 4°F dead band for the rod control system) for the temperature is 8.74°F (2σ random error of 7.6°F plus the 1.14°F bias error). The value assumed in the non-Revised Thermal Design Procedure (non-RTDP) analyses is -8°F, +9.5°F. For the RTDP analyses, a value of ± 7.6°F with a bias of +1.5°F is assumed.

Safety Analyses assumed a total RCS flow rate of 358,800 gpm. This value is bounded by the LCO value of 371,400 gpm **and the limit specified in the COLR** assuming a flow measurement uncertainty of 3.5%.

This 3.5% flow measurement uncertainty assumed in the analyses included errors from all sources including fouling ~~in~~ **of** the **feedwater** venturi. The use of 3.5% flow error is acceptable if actual uncertainty is unknown. At the time analyses were performed, the flow accuracy was unavailable. Subsequent calculations determined the error to be less than 3.5%.

Any fouling that might bias the flow rate measurement greater than ~~0.1%~~ **the penalty for undetected fouling of the feedwater venturi** can be detected by monitoring and trending various plant performance parameters. If detected, 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

APPLICABLE SAFETY ANALYSES (continued)

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 (T_{avg}), and RCS total flow rate–to ensure the core operates within the limits assumed in the safety analyses. **These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, based on maximum analyzed steam generator tube plugging, is retained in the LCO to assure that a lower flow rate than reviewed by the NRC will not be used.** Operating within these limits will result in meeting the DNB design criterion in the event of a DNB limited transient.

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

The DNBR limit ~~Another set of limits on DNB related parameters~~ is provided in SL 2.1.1, "Reactor Core SLs." LCO 3.4.1 represents the initial conditions of the safety analysis which are far more restrictive than the ~~Safety Limit (SL)~~ **conditions which define the DNBR limit**. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS total flow rate must be maintained during steady state operation in order to ensure DNB design 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.

Table 3.3.1-1 (page 5 of 6)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.04% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left(\frac{1}{1+\tau_3 s}\right) \leq \Delta T_0 \left\{ K_1 - K_2 \frac{(1+\tau_4 s)}{(1+\tau_5 s)}\left[T \frac{1}{(1+\tau_6 s)} - T' \right] + K_3 (P - P') - f_1(\Delta I) \right\}$$

Where: ΔT is measured Reactor Coolant System (RCS) ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T′ is the nominal T_{avg} at RTP, ≤ ~~588.4°F~~ *.

P is the measured pressurizer pressure, psig.
P′ is the nominal RCS operating pressure, = ~~2235 psig~~ *.

K_1 = ~~1.325~~ *	K_2 = ~~0.0297/°F~~ *	K_3 = ~~0.00181/psig~~ *
τ_1 = ~~8 sec~~ *	τ_2 = ~~3 sec~~ *	τ_3 ≤ ~~2 sec~~ *
τ_4 = ~~33 sec~~ *	τ_5 = ~~4 sec~~ *	τ_6 ≤ ~~2 sec~~ *

$f_1(\Delta I)$ = ~~-3.35~~ * {~~24~~ * + (q_t - q_b)} when q_t - q_b < ~~-24%~~ * RTP
0% of RTP when ~~-24%~~ * RTP ≤ q_t - q_b ≤ ~~10%~~ * RTP
~~4.11~~ * {(q_t - q_b) - ~~10~~ * } when q_t - q_b > ~~10%~~ * 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 6 of 6)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 3.60% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left(\frac{1}{1+\tau_3 s}\right) \leq \Delta T_0 \left\{ K_4 - K_5 \frac{\tau_7 s}{1+\tau_7 s}\left(\frac{1}{1+\tau_6 s}\right) T - K_6\left[T\frac{1}{1+\tau_6 s} - T''\right] - f_2(\Delta I)\right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T″ is the nominal T_{avg} at RTP, ≤ ~~588.4°F~~ *.

K_4 = ~~1.072~~ *

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

K_6 = ~~0.00245/°F~~ * when T > T"
~~0/°F~~ * when T ≤ T"

τ_1 = ~~8 sec~~ *
τ_2 = ~~3 sec~~ *
τ_3 ≤ ~~2 sec~~ *
τ_6 ≤ ~~2 sec~~ *
τ_7 = ~~10 sec~~ *

$f_2(\Delta I)$ = ~~0 for all ΔI.~~ *

* **As specified in the COLR.**

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.~~

2.1.1.1 In MODE 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained ≥ 1.25 for the WRB-2 DNB correlation.

2.1.1.2 In MODE 2, the DNBR shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.30 for the W-3 DNB correlation.

2.1.1.3 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained ≤ 4700 °F.

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.

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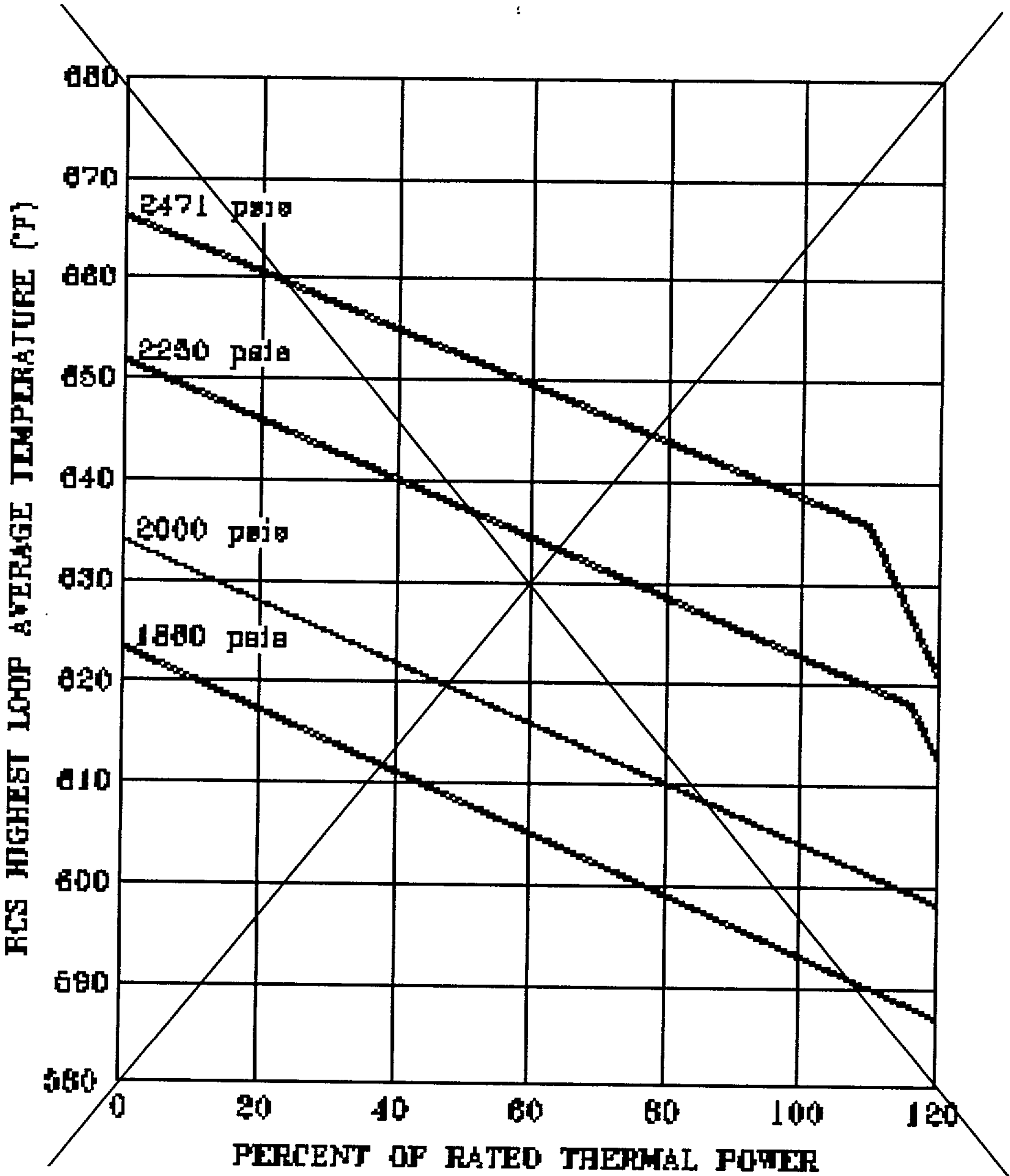

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits

BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) and Main Steam Safety Valves (MSSVs) prevents violation of the reactor core SLs.

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 that** the hot fuel pellet in the core must not experience centerline fuel melting; and

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

The Reactor Trip System setpoints (Ref. 2) specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation," in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) highest loop average temperature, pressurizer pressure, **RCS flow, Axial Flux Difference (AFD),** 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.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Automatic ~~enforcement~~ **preservation** of these reactor core SLs is provided by the ~~following functions~~ **appropriate operation of the RPS and the MSSVs** (Ref. 2)~~:~~**.**

~~a. Pressurizer Pressure - high trip;~~

~~b. Pressurizer Pressure - low trip;~~

~~c. Overtemperature ΔT trip;~~

~~d. Overpower ΔT trip;~~

~~e. Power Range Neutron Flux trip;~~

~~f. Low reactor coolant flow trip; and~~

~~g. Main steam safety valves.~~

~~Additional trip functions are provided to backup these functions for specific abnormal conditions.~~

~~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.~~

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

~~Figure B 2.1.1-1 shows an example of the reactor core safety limits of THERMAL POWER, RCS 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 or equal to the enthalpy of saturated liquid, or that the core exit quality is within the limits defined by the DNBR correlation.~~

BASES

SAFETY LIMITS (continued)

~~The curves are typically derived based on enthalpy hot channel factor limits such as those provided in the COLR. The dashed line of Figure B 2.1.1-1 shows an example of a limit curve at 2235 psig. In addition, it illustrates the various RPS functions that are designed to prevent the unit from reaching the limit.~~

~~The SL is higher than the limit calculated when the Axial Flux Difference (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).~~

⇒ **INSERT 1 HERE**

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 MSSVs 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. In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMITS 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.

INSERT 1

BASES

SAFETY LIMITS (continued)

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

a. **There must be at least 95% probability that the hot fuel pellet in the core must not experience centerline fuel melting; and**

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

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 AOOs. To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature ΔT 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 that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS and the MSSVs ensure that for variations in the RCS average temperature, pressurizer pressure, RCS flow, AFD, and THERMAL POWER that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. UFSAR, Section 7.2.
3. ~~WCAP 8746 A, March 1977.~~
4. ~~WCAP 9273 NP A, July 1985.~~

DELETE FIGURE

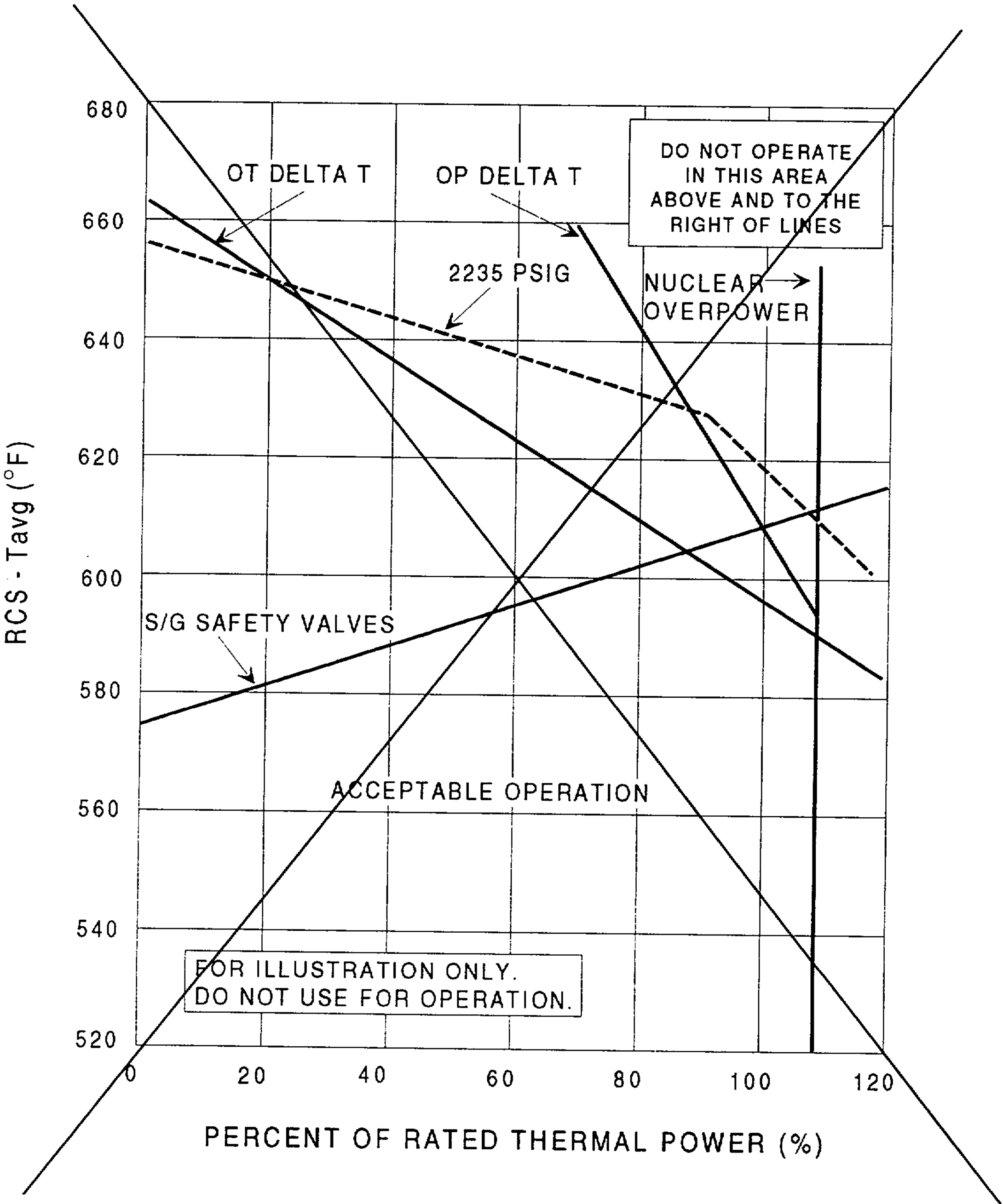

Figure B 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits vs. Boundary of Protection

5.6 Reporting Requirements

5.6.5 Core Operating Limits Report (COLR) (continued)

6. WCAP-9220-P-A, "Westinghouse ECCS Evaluation Model-1981 Version," February 1982.

7. WCAP-9561-P-A, Add. 3, "BART A-1: a Computer Code for Best Estimate Analysis of Reflood Transients - Special Report: Thimble Modeling in Westinghouse ECCS Evaluation Model," July 1986.

8. WCAP-10266-P-A, "The 1981 Version of Westinghouse Evaluation Model using BASH Code," March 1987, including Addendum 1 "Power Shape Sensitivity Studies," Revision 2-P-A, dated December 15, 1987, and Addendum 2 "BASH Methodology Improvements and Reliability Enhancements," Revision 2, Dated May 1988.

9. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.

10. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using NOTRUMP Code," August 1985.

11. WCAP-10216-A, Revision 1, "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification," February 1994.

12. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986;

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; and

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

ATTACHMENT B-3

INCORPORATED PROPOSED CHANGES FOR BRAIDWOOD STATION

REVISED PAGES

RCS DNB Parameters	Revised TS Page	Page 3.4.1-1
RCS DNB Parameters	Revised TS Page	Page 3.4.1-2
RCS DNB Parameters	Revised TS Bases Page	Page B 3.4.1-2
RCS DNB Parameters	Revised TS Bases Page	Page B 3.4.1-3
RCS DNB Parameters	Revised TS Bases Page	Page B 3.4.1-4
RCS DNB Parameters	Revised TS Bases Page	Page B 3.4.1-5
RCS DNB Parameters	Revised TS Bases Page	Page B 3.4.1-6
RTS OTΔT	Revised TS Page	Page 3.3.1-18
RTS OPΔT	Revised TS Page	Page 3.3.1-19
Reactor Core Safety Limits	Revised TS Page	Page 2.0-1
Reactor Core Safety Limits	Revised TS Bases Page	Page B 2.1.1-2
Reactor Core Safety Limits	Revised TS Bases Page	Page B 2.1.1-3
Reactor Core Safety Limits	Revised TS Bases Page	Page B 2.1.1-4
COLR Analytical Methods	Revised TS Page	Page 5.6-4

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 (T_{avg}), and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure within the limit specified in the COLR;

b. RCS average temperature (T_{avg}) within the limit specified in the COLR; and

c. RCS total flow rate ≥ 371,400 gpm and within the limit specified in the COLR.

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

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

b. THERMAL POWER step > 10% RTP.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	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 (T_{avg}) is within the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ 371,400 gpm and within the limit specified in the COLR.	12 hours
SR 3.4.1.4	-------------------NOTE------------------- Not required to be performed until 7 days after ≥ 90% RTP. -- Verify by precision heat balance that RCS total flow rate is ≥ 371,400 gpm and within the limit specified in the COLR.	18 months

B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the departure from nucleate boiling (DNB) will be met for each of the transients analyzed.

The RCS pressure limit is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits.

The RCS coolant average temperature (T_{avg}) limit is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNB limits.

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. Flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

BASES

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNBR criterion. 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 DNB 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)."

Safety Analyses assumed a value of 2207 psia (2192.3 psig). This value is bounded by the limit specified in the COLR assuming a measurement accuracy of less than 26.7 psi.

Safety Analyses assumed a value of 588.4°F for the vessel average temperature. In addition, the analyses assumed the calculated error (including the ± 4°F dead band for the rod control system) for the temperature is 8.74°F (2σ random error of 7.6°F plus the 1.14°F bias error). The value assumed in the non-Revised Thermal Design Procedure (non-RTDP) analyses is -8°F, +9.5°F. For the RTDP analyses, a value of ± 7.6°F with a bias of +1.5°F is assumed.

Safety Analyses assumed a total RCS flow rate of 358,800 gpm. This value is bounded by the LCO value of 371,400 gpm and the limit specified in the COLR assuming a flow measurement uncertainty of 3.5%.

This 3.5% flow measurement uncertainty assumed in the analyses included errors from all sources including fouling of the feedwater venturi. The use of 3.5% flow error is acceptable if actual uncertainty is unknown. At the time analyses were performed, the flow accuracy was unavailable. Subsequent calculations determined the error to be less than 3.5%.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Any fouling that might bias the flow rate measurement greater than the penalty for undetected fouling of the feedwater venturi can be detected by monitoring and trending various plant performance parameters. If detected, 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.

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 (T_{avg}), and RCS total flow rate – to ensure the core operates within the limits assumed in the safety analyses. These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, based on maximum analyzed steam generator tube plugging, is retained in the LCO to assure that a lower flow rate than reviewed by the NRC will not be used. Operating within these limits will result in meeting the DNB design criterion in the event of a DNB limited transient.

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

The DNBR limit is provided in SL 2.1.1, "Reactor Core SLs." LCO 3.4.1 represents the initial conditions of the safety analysis which are far more restrictive than the conditions which define the DNBR limit. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

BASES

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS total flow rate must be maintained during steady state operation in order to ensure DNB design 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.

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 unit parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience.

B.1

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

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

SR 3.4.1.2

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

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The required minimum RCS flow rate is met with ≥ 95% indicated flow rate. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.4

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

The Frequency of 18 months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance.

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until 7 days after ≥ 90% RTP. This exception is appropriate since the heat balance requires the unit to be at a minimum of 90% RTP to obtain the stated RCS flow accuracies. The Surveillance shall be performed within 7 days after reaching 90% RTP.

REFERENCES

1. UFSAR, Chapter 15.

Table 3.3.1-1 (page 5 of 6)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.04% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[\frac{1}{1+\tau_3 s}\right] \le \Delta T_0 \left\{K_1 - K_2 \frac{(1+\tau_4 s)}{(1+\tau_5 s)} \left[T \frac{1}{(1+\tau_6 s)} - T'\right] + K_3 (P - P') - f_1(\Delta I)\right\}$$

Where: ΔT is measured Reactor Coolant System (RCS) ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T' is the nominal T_{avg} at RTP, ≤ *.

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

K_1 = *	K_2 = *	K_3 = *
τ_1 = *	τ_2 = *	τ_3 ≤ *
τ_4 = *	τ_5 = *	τ_6 ≤ *

$f_1(\Delta I)$ =	$*\{* + (q_t - q_b)\}$	when $q_t - q_b < *$ RTP
	0% of RTP	when * RTP ≤ $q_t - q_b$ ≤ * 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 6 of 6)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 3.60% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[\frac{1}{1+\tau_3 s}\right] \leq \Delta T_0 \left\{K_4 - K_5 \frac{\tau_7 s}{1+\tau_7 s} \left[\frac{1}{1+\tau_6 s}\right] T - K_6 \left[T \frac{1}{1+\tau_6 s} - T''\right] - f_2(\Delta I)\right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T″ is the nominal T_{avg} at RTP, ≤ *.

K_4 = *

K_5 = * for increasing T_{avg}
* for decreasing T_{avg}

K_6 = * when T > T"
* when T ≤ T"

τ_1 = *
τ_2 = *
τ_3 ≤ *
τ_6 ≤ *
τ_7 = *

$f_2(\Delta I)$ = *

* As specified in the COLR.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODE 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained ≥ 1.25 for the WRB-2 DNB correlation.

2.1.1.2 In MODE 2, the DNBR shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.30 for the W-3 DNB correlation.

2.1.1.3 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained ≤ 4700°F.

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.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and Anticipated Operational Occurrences (AOOs). This is accomplished by having a Departure from Nucleate Boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak Linear Heat Rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) and Main Steam Safety Valves (MSSVs) prevents violation of the reactor core SLs.

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 that the hot fuel pellet in the core must not experience centerline fuel melting; and

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

The Reactor Trip System setpoints (Ref. 2) specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation," in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) highest loop average temperature, pressurizer pressure, RCS flow, Axial Flux Difference (AFD), 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.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Automatic preservation of these reactor core SLs is provided by the appropriate operation of the RPS and the MSSVs (Ref. 2).

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

a. There must be at least 95% probability that the hot fuel pellet in the core must not experience centerline fuel melting; and

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

The reactor core SLs are used to defined the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and AOOs. To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature ΔT 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 that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS and the MSSVs ensure that for variations in the RCS average temperature, pressurizer pressure, RCS flow, AFD, and THERMAL POWER that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

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 MSSVs 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. In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMITS 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. UFSAR, Section 7.2.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. WCAP-9220-P-A, "Westinghouse ECCS Evaluation Model-1981 Version," February 1982.

7. WCAP-9561-P-A, Add. 3, "BART A-1: a Computer Code for Best Estimate Analysis of Reflood Transients - Special Report: Thimble Modeling in Westinghouse ECCS Evaluation Model," July 1986.

8. WCAP-10266-P-A, "The 1981 Version of Westinghouse Evaluation Model using BASH Code," March 1987, including Addendum 1 "Power Shape Sensitivity Studies," Revision 2-P-A, dated December 15, 1987, and Addendum 2 "BASH Methodology Improvements and Reliability Enhancements," Revision 2, Dated May 1988.

9. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.

10. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using NOTRUMP Code," August 1985.

11. WCAP-10216-A, Revision 1, "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification," February 1994.

12. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986;

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; and

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

ATTACHMENT B-4

INCORPORATED PROPOSED CHANGES FOR BYRON STATION

REVISED PAGES

RCS DNB Parameters	Revised TS Page	Page 3.4.1-1
RCS DNB Parameters	Revised TS Page	Page 3.4.1-2
RCS DNB Parameters	Revised TS Bases Page	Page B 3.4.1-2
RCS DNB Parameters	Revised TS Bases Page	Page B 3.4.1-3
RCS DNB Parameters	Revised TS Bases Page	Page B 3.4.1-4
RCS DNB Parameters	Revised TS Bases Page	Page B 3.4.1-5
RCS DNB Parameters	Revised TS Bases Page	Page B 3.4.1-6
RTS OTΔT	Revised TS Page	Page 3.3.1-18
RTS OPΔT	Revised TS Page	Page 3.3.1-19
Reactor Core Safety Limits	Revised TS Page	Page 2.0-1
Reactor Core Safety Limits	Revised TS Bases Page	Page B 2.1.1-2
Reactor Core Safety Limits	Revised TS Bases Page	Page B 2.1.1-3
Reactor Core Safety Limits	Revised TS Bases Page	Page B 2.1.1-4
COLR Analytical Methods	Revised TS Page	Page 5.6-4

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 (T_{avg}), and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure within the limit specified in the COLR;

b. RCS average temperature (T_{avg}) within the limit specified in the COLR; and

c. RCS total flow rate ≥ 371,400 gpm and within the limit specified in the COLR.

---------------------------NOTE---------------------------

Pressurizer pressure limit does not apply during:

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

b. THERMAL POWER step > 10% RTP.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	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 (T_{avg}) is within the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ 371,400 gpm and within the limit specified in the COLR.	12 hours
SR 3.4.1.4	-------------------NOTE------------------- Not required to be performed until 7 days after ≥ 90% RTP. -- Verify by precision heat balance that RCS total flow rate is ≥ 371,400 gpm and within the limit specified in the COLR.	18 months

B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the departure from nucleate boiling (DNB) will be met for each of the transients analyzed.

The RCS pressure limit is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits.

The RCS coolant average temperature (T_{avg}) limit is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNB limits.

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. Flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

BASES

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNBR criterion. 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 DNB 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)."

Safety Analyses assumed a value of 2207 psia (2192.3 psig). This value is bounded by the limit specified in the COLR assuming a measurement accuracy of less than 26.7 psi.

Safety Analyses assumed a value of 588.4°F for the vessel average temperature. In addition, the analyses assumed the calculated error (including the ± 4°F dead band for the rod control system) for the temperature is 8.74°F (2σ random error of 7.6°F plus the 1.14°F bias error). The value assumed in the non-Revised Thermal Design Procedure (non-RTDP) analyses is -8°F, +9.5°F. For the RTDP analyses, a value of ± 7.6°F with a bias of +1.5°F is assumed.

Safety Analyses assumed a total RCS flow rate of 358,800 gpm. This value is bounded by the LCO value of 371,400 gpm and the limit specified in the COLR assuming a flow measurement uncertainty of 3.5%.

This 3.5% flow measurement uncertainty assumed in the analyses included errors from all sources including fouling of the feedwater venturi. The use of 3.5% flow error is acceptable if actual uncertainty is unknown. At the time analyses were performed, the flow accuracy was unavailable. Subsequent calculations determined the error to be less than 3.5%.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Any fouling that might bias the flow rate measurement greater than the penalty for undetected fouling of the feedwater venturi can be detected by monitoring and trending various plant performance parameters. If detected, 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.

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 (T_{avg}), and RCS total flow rate – to ensure the core operates within the limits assumed in the safety analyses. These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, based on maximum analyzed steam generator tube plugging, is retained in the LCO to assure that a lower flow rate than reviewed by the NRC will not be used. Operating within these limits will result in meeting the DNB design criterion in the event of a DNB limited transient.

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

The DNBR limit is provided in SL 2.1.1, "Reactor Core SLs." LCO 3.4.1 represents the initial conditions of the safety analysis which are far more restrictive than the conditions which define the DNBR limit. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

BASES

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS total flow rate must be maintained during steady state operation in order to ensure DNB design 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.

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 unit parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience.

B.1

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

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

SR 3.4.1.2

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

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. For Unit 1, the required minimum RCS flow rate is met with ≥ 95% indicated flow rate. For Unit 2, the required minimum RCS flow rate is met with ≥ 92% indicated flow rate. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.4

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

The Frequency of 18 months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance.

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until 7 days after ≥ 90% RTP. This exception is appropriate since the heat balance requires the unit to be at a minimum of 90% RTP to obtain the stated RCS flow accuracies. The Surveillance shall be performed within 7 days after reaching 90% RTP.

REFERENCES

1. UFSAR, Chapter 15.

Table 3.3.1-1 (page 5 of 6)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.04% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[\frac{1}{1+\tau_3 s}\right] \le \Delta T_0 \left\{K_1 - K_2 \frac{(1+\tau_4 s)}{(1+\tau_5 s)} \left[T \frac{1}{(1+\tau_6 s)} - T'\right] + K_3 (P - P') - f_1(\Delta I)\right\}$$

Where: ΔT is measured Reactor Coolant System (RCS) ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T′ is the nominal T_{avg} at RTP, ≤ *.

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

K_1 = *	K_2 = *	K_3 = *
τ_1 = *	τ_2 = *	τ_3 ≤ *
τ_4 = *	τ_5 = *	τ_6 ≤ *

$f_1(\Delta I)$ =	*{* + ($q_t - q_b$)}	when $q_t - q_b$ < * RTP
	0% of RTP	when * RTP ≤ $q_t - q_b$ ≤ * 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 6 of 6)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 3.60% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[\frac{1}{1+\tau_3 s}\right] \le \Delta T_0 \left\{K_4 - K_5 \frac{\tau_7 s}{1+\tau_7 s} \left[\frac{1}{1+\tau_6 s}\right] T - K_6 \left[T \frac{1}{1+\tau_6 s} - T''\right] - f_2(\Delta I)\right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T″ is the nominal T_{avg} at RTP, ≤ *.

K_4 = * K_5 = * for increasing T_{avg}, * for decreasing T_{avg} K_6 = * when T > T", * when T ≤ T"

τ_1 = * τ_2 = * τ_3 ≤ *

τ_6 ≤ * τ_7 = *

$f_2(\Delta I)$ = *

* As specified in the COLR.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODE 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained ≥ 1.25 for the WRB-2 DNB correlation.

2.1.1.2 In MODE 2, the DNBR shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.30 for the W-3 DNB correlation.

2.1.1.3 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained ≤ 4700°F.

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.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and Anticipated Operational Occurrences (AOOs). This is accomplished by having a Departure from Nucleate Boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak Linear Heat Rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) and Main Steam Safety Valves (MSSVs) prevents violation of the reactor core SLs.

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 that the hot fuel pellet in the core must not experience centerline fuel melting; and

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

The Reactor Trip System setpoints (Ref. 2) specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation," in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) highest loop average temperature, pressurizer pressure, RCS flow, Axial Flux Difference (AFD), 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.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Automatic preservation of these reactor core SLs is provided by the appropriate operation of the RPS and the MSSVs (Ref. 2).

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

a. There must be at least 95% probability that the hot fuel pellet in the core must not experience centerline fuel melting; and

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

The reactor core SLs are used to defined the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and AOOs. To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature ΔT 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 that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS and the MSSVs ensure that for variations in the RCS average temperature, pressurizer pressure, RCS flow, AFD, and THERMAL POWER that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

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 MSSVs 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. In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMITS 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. UFSAR, Section 7.2.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. WCAP-9220-P-A, "Westinghouse ECCS Evaluation Model-1981 Version," February 1982.

7. WCAP-9561-P-A, Add. 3, "BART A-1: a Computer Code for Best Estimate Analysis of Reflood Transients - Special Report: Thimble Modeling in Westinghouse ECCS Evaluation Model," July 1986.

8. WCAP-10266-P-A, "The 1981 Version of Westinghouse Evaluation Model using BASH Code," March 1987, including Addendum 1 "Power Shape Sensitivity Studies," Revision 2-P-A, dated December 15, 1987, and Addendum 2 "BASH Methodology Improvements and Reliability Enhancements," Revision 2, Dated May 1988.

9. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.

10. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using NOTRUMP Code," August 1985.

11. WCAP-10216-A, Revision 1, "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification," February 1994.

12. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986;

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; and

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

ATTACHMENT C

INFORMATION SUPPORTING A FINDING OF NO SIGNIFICANT HAZARDS CONSIDERATION

According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

Commonwealth Edison (ComEd) Company is proposing changes to the Technical Specifications (TS), of Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66 for the Braidwood Station, Units 1 and 2, and the Byron Station, Units 1 and 2, respectively. The proposed changes relocate Reactor Coolant System (RCS) related cycle-specific parameter limits from the TS to, and thus expanding, the Core Operating Limits Reports (COLRs) for Braidwood Station and Byron Station. Specifically, the proposed changes to TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and associated Bases relocate the pressurizer pressure, RCS average temperature, and RCS total flow rate values to the COLRs. However, the minimum limit for RCS total flow rate will be retained in TS 3.4.1. The proposed changes to TS Table 3.3.1-1, "Reactor Trip System Instrumentation," relocate the Overtemperature ΔT and Overpower ΔT constant (K) values, dynamic compensation time (τ) values, and the breakpoint and slope values for the $f(\Delta I)$ penalty functions to the COLRs. Additionally, the proposed changes to TS 2.1.1, "Reactor Core Safety Limits," and associated Bases relocate the reactor core safety limit figure to the COLRs and replace it with the more specific fuel DNB Ratio (DNBR) and peak fuel centerline temperature safety limit requirements. The proposed changes will allow us the flexibility to enhance plant operating margin and core design margin without the need for cycle-specific license amendment requests.

Information supporting the determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below.

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

The proposed changes are programmatic and administrative in nature which do not physically alter safety-related systems, nor affect the way in which safety-related systems perform their functions. The proposed changes remove cycle-specific parameter limits from TS 3.4.1 and relocate them to the COLRs which do not change plant design or affect system operating parameters. In addition, the minimum limit for RCS total flow rate is being retained in TS 3.4.1 to assure that a lower flow rate than reviewed by the NRC will not be used. The proposed changes do not, by themselves, alter any of the parameter limits. The removal of the cycle-specific parameter limits from the TS does not eliminate existing requirements to comply with the parameter limits. The existing TS Section 5.6.5b, COLR Reporting Requirements, continues to ensure that the analytical methods used to determine the core operating limits meet NRC reviewed and approved methodologies. The existing TS Section 5.6.5c, COLR Reporting Requirements, continues to ensure that applicable limits of the safety analyses are met. Further, more specific requirements regarding the safety limits (i.e., DNBR limit and peak fuel centerline temperature limit) are being imposed in TS 2.1.1, "Reactor Core Safety Limits," replacing the Reactor Core Safety Limits (RCSL) figure which are consistent with the values stated in the Updated Final Safety Analysis Report (UFSAR).

Although the relocation of the cycle-specific parameter limits to the COLRs would allow revision of the affected parameter limits without prior NRC approval, there is no significant effect on the probability or consequences of an accident previously evaluated. Future changes to the COLR parameter limits could result in event consequences which are either slightly less or slightly more severe than the consequences for the same event using the present parameter limits. The differences would not be significant and would be bounded by the existing requirement of TS Section 5.6.5c to meet the applicable limits of the safety analyses.

The cycle-specific parameter limits being transferred from the TS to the COLRs will continue to be controlled under existing programs and procedures. The UFSAR accident analyses will continue to be examined with respect to changes in the cycle-dependent parameters obtained using NRC reviewed and approved reload design methodologies, ensuring that the transient evaluation of new reload designs are bounded by previously accepted analyses. This examination will continue to be performed pursuant to 10 CFR 50.59 requirements ensuring that future reload designs will not involve a significant increase in the probability or consequences of an accident previously evaluated. Additionally, the proposed changes do not allow for an increase in plant power levels, do not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. Therefore, the proposed changes do not change the types or increase the amounts of any effluents released offsite.

Therefore, the proposed changes do not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes that retain the minimum limit for RCS total flow rate in the TS, and that relocate certain cycle-specific parameter limits from the TS to the COLR, thus removing the requirement for prior NRC approval of revisions to those parameters, do not involve a physical change to the plant. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There is no change being made to the parameters within which the plant is operated, other than their relocation to the COLRs. There are no setpoints affected by the proposed changes at which protective or mitigative actions are initiated. The proposed changes will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alteration in the procedures which ensure the plant remains within analyzed limits is being proposed, and no change is being made to the procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced.

Relocation of cycle-specific parameter limits has no influence or impact on, nor does it contribute in any way to the possibility of a new or different kind of accident. The relocated cycle-specific parameter limits will continue to be calculated using the NRC reviewed and approved methodology. The proposed changes do not alter assumptions made in the safety analysis and operation within the core operating limits will continue.

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

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

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed changes do not physically alter safety-related systems, nor does it effect the way in which safety-related systems perform their functions. The setpoints at which protective actions are initiated are not altered by the proposed changes. Therefore, sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. As 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 parameter, and the plant will continue to operate within prescribed limits.

The development of cycle-specific parameter limits for future reload designs will continue to conform to NRC reviewed and approved methodologies, and will be performed pursuant to 10 CFR 50.59 to assure that plant operation within cycle-specific parameter limits will not involve a significant reduction in the margin of safety.

Therefore, the proposed changes do not involve a reduction in a margin of safety.

ATTACHMENT D

INFORMATION SUPPORTING AN ENVIRONMENTAL ASSESSMENT

Commonwealth Edison (ComEd) Company has evaluated the proposed changes against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. ComEd has determined that the proposed changes meet the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50, which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or which changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

(i) The amendment involves no significant hazards consideration.

As demonstrated in Attachment C, the proposed changes do not involve any significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed changes are limited to expanding the Core Operating Limits Reports. The proposed changes do not allow for an increase in the unit power level, do not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. Therefore, the proposed changes do not affect actual unit effluents. As documented in Attachment C, there will be no change in the types or increase in the amounts of any effluents released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from the proposed changes.

ATTACHMENT E

EXPANDED CORE OPERATING LIMITS REPORTS INFORMATION

1. **<u>Reactor Coolant System DNB Parameters</u>**
   RCS DNB Parameter Limits (from TS 3.4.1)

Expanded COLR Parameters	**Current Value**
Pressurizer pressure	**≥ 2219 psig**
RCS average temperature (T_{avg})	**≤ 591.2 °F**
RCS total flow rate	**≥ 371,400 gpm**

2. **Reactor Trip System OTΔT**

Overtemperature ΔT Setpoint Parameter Values (from TS 3.3.1-1)

Expanded COLR Parameters	**Current Value**
Overtemperature ΔT reactor trip setpoint	$K_1 = 1.325$
Overtemperature ΔT reactor trip setpoint T_{avg} coefficient	$K_2 = 0.0297$ / °F
Overtemperature ΔT reactor trip setpoint pressure coefficient	$K_3 = 0.00181$ / psig
Nominal T_{avg} at RTP (indicated)	$T' \leq 588.4$ °F
Nominal RCS operating pressure (indicated)	$P' = 2235$ psig
Measured reactor vessel ΔT lead/lag time constants	$\tau_1 = 8$ sec $\tau_2 = 3$ sec
Measured reactor vessel ΔT lag time constant	$\tau_3 \leq 2$ sec
Measured reactor vessel average temperature lead/lag time constants	$\tau_4 = 33$ sec $\tau_5 = 4$ sec
Measured reactor vessel average temperature lag time constant	$\tau_6 \leq 2$ sec
f_1 (ΔI) "positive" breakpoint	+10% ΔI
f_1 (ΔI) "negative" breakpoint	- 24% ΔI
f_1 (ΔI) "positive" slope	+4.11% / % ΔI
f_1 (ΔI) "negative" slope	- 3.35% / % ΔI

3. Reactor Trip System OPΔT

Overpower ΔT Setpoint Parameter Values (from TS 3.3.1-1)

Expanded COLR Parameters	**Current Value**
Overpower ΔT reactor trip setpoint	$K_4 = 1.072$
Overpower ΔT reactor trip setpoint T_{avg} rate/lag coefficient	$K_5 = 0.02$ / °F for increasing T_{avg} $K_5 = 0$ / °F for decreasing T_{avg}
Overpower ΔT reactor trip setpoint T_{avg} heatup coefficient	$K_6 = 0.00245$ / °F when $T > T''$ $K_6 = 0$ / °F when $T \leq T''$
Nominal T_{avg} at RTP (indicated)	$T'' \leq 588.4$ °F
Measured reactor vessel ΔT lead/lag time constants	$\tau_1 = 8$ sec $\tau_2 = 3$ sec
Measured reactor vessel ΔT lag time constant	$\tau_3 \leq 2$ sec
Measured reactor vessel average temperature lag time constant	$\tau_6 \leq 2$ sec
Measured reactor vessel average temperature rate/lag time constant	$\tau_7 = 10$ sec
$f_2(\Delta I)$ "positive" breakpoint	0 for all ΔI
$f_2(\Delta I)$ "negative" breakpoint	0 for all ΔI
$f_2(\Delta I)$ "positive" slope	0 for all ΔI
$f_2(\Delta I)$ "negative" slope	0 for all ΔI

4. **Reactor Core Safety Limits**
Reactor Core Safety Limits (from TS Figure 2.1.1-1)

Expanded COLR Parameters	**Current Value**
Reactor Core Safety Limits	Relocated figure

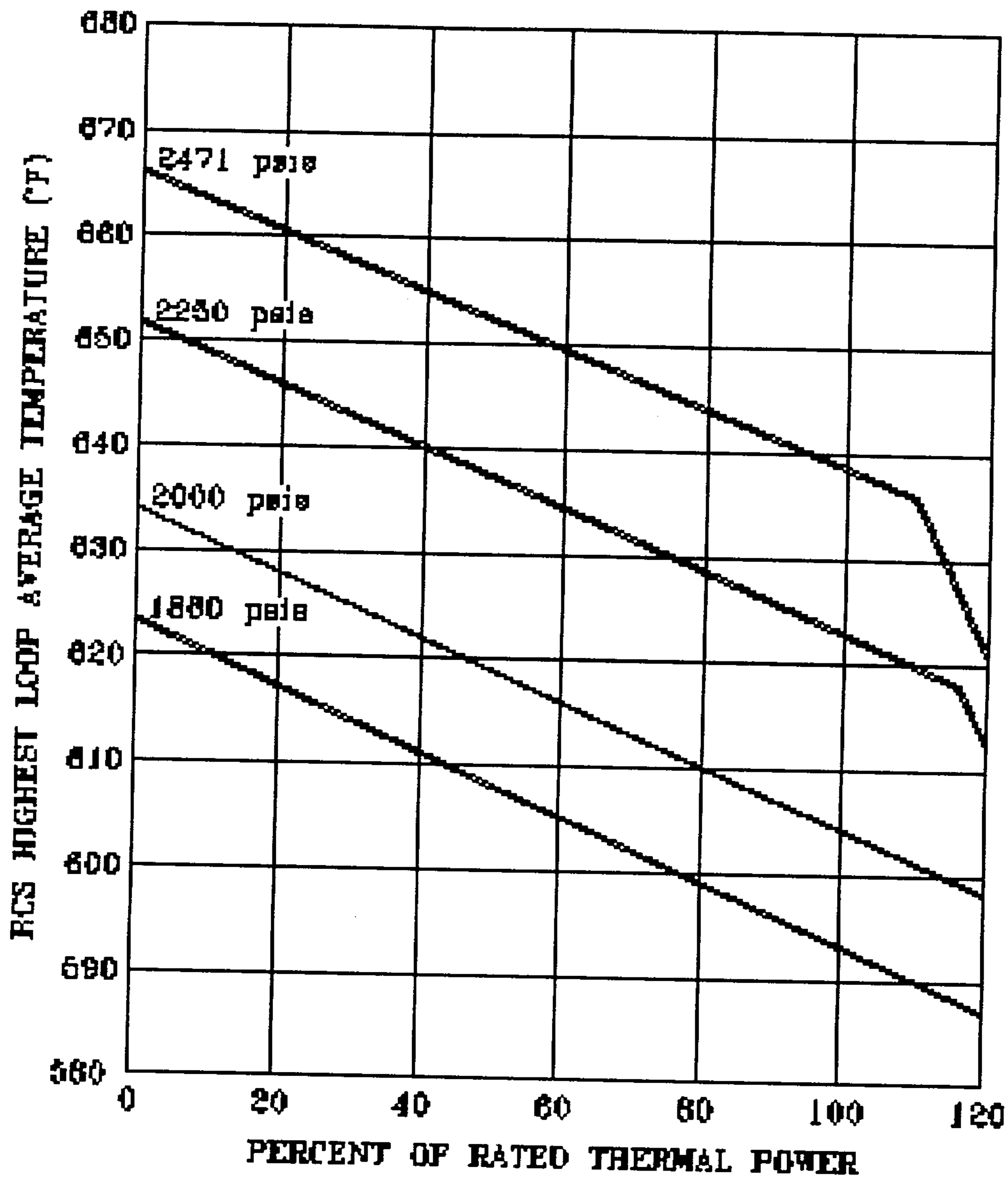

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits