EXHIBIT A

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

License Amendment Request dated March 25, 2003
Safety Analyses Transition

Description of Proposed Changes, Reasons for Requesting the Changes, Supporting Safety Evaluation and proposed Determination of No Significant Hazards Consideration

Pursuant to 10 CFR Part 50, Sections 50.59 and 50.90, the holders of Operating Licenses DPR-42 and DPR-60 hereby propose the following changes to the Technical Specifications contained in Appendix A of the Facility Operating Licenses and associated Bases changes:

BACKGROUND

The Nuclear Management Company (NMC) has historically performed many of the safety analyses that support operation of the Prairie Island Nuclear Generating Plant (PINGP) with internal resources. In the future, NMC plans to have most of these safety analyses performed by Westinghouse. In the process of transitioning to Westinghouse safety analyses, NMC is making changes in plant operations and supporting documentation that involves Technical Specification changes. This license amendment request proposes TS changes which includes implementation of relaxed axial offset control (RAOC) of the reactor cores, implementation of Westinghouse methodology for determining selected core operating parameter values, relocation of selected operating parameters to the Core Operating Limits Report (COLR) and revision of the Pressurizer Pressure-Low Allowable Value.

NMC has elected to implement the RAOC operating strategy for Prairie Island Units 1 and 2 in order to increase plant availability and operating flexibility. The NRC approved Westinghouse methodology for RAOC will be used. Application of the RAOC methodology requires revision of the Technical Specifications (TS) and Bases. The axial flux difference (AFD) limits, which currently reside in the COLR, will be determined during the cycle specific reload evaluation process.

NMC has also elected to relocate the reactor core safety limits (SLs) and the overtemperature delta-T (OTΔT) and overpower delta-T (OPΔT) parameter values to the COLR. This is being done to avoid revisions to the TS when these parameters change. The proposed revision of the TS and Bases to support this change is generally consistent with the guidance of NUREG-1431, Revision 2, "Standard Technical Specifications Westinghouse Plants". NRC approved Westinghouse methodologies will be used for the determination of the SLs and the OTΔT and OPΔT parameter values. These will be determined during the cycle specific reload evaluation process.

In addition, NMC proposes to increase the Allowable Value for the Pressurizer Pressure-Low reactor trip Function. This change is in the conservative direction and the revised value is calculated consistent with the NRC approved Westinghouse methodology for determining the OTΔT and OPΔT parameter values and the PINGP setpoint methodology (Reference 8) which was accepted by the NRC.

PROPOSED CHANGES AND JUSTIFICATION OF CHANGES

A brief description of the proposed changes is provided below along with a discussion of the justification for each change. The proposed changes are presented in three groups. The three groups are:

GROUP 1 – Implementation of relaxed axial offset control
GROUP 2 – Relocation of TS SL Figure and OTΔT and OPΔT parameter values to the COLR, and miscellaneous administrative changes
GROUP 3 – Revision of Pressurizer Pressure-Low reactor trip Allowable Value.

The specific wording changes to the TS and Bases are provided in Exhibits B and C.

GROUP 1 – Implementation of relaxed axial offset control

A. TS 3.2.1, Heat Flux Hot Channel Factor - $F_Q(Z)$ and Bases: Modification of Required Actions and Completion Time if F^W_Q (Z) is not within its limit and update Bases.

Specification 3.2.1 Condition B Required Actions (RA) B.1, B.2 and B.3 were revised, consistent with the requirements of RAOC methodology, to require reduction of AFD limits and reduction of trip set points in accordance with the allowable power level of the AFD limit reduction. The Completion Time for RA B.4 was also revised to correlate to the allowable power level of the reduced AFD limits.

Bases changes were made in support of these Specification changes which conform the 3.2.1 Bases to the RAOC methodology and NUREG-1431. Additional changes have been made based on transition to Westinghouse performance of PINGP analyses which include: reduction of the energy deposition to the fuel from 280 cal/gm to 200 cal/gm; removal of the statement that K(Z) is based on the small break loss of coolant accident (LOCA) and exclusion of the top and bottom 15% of the core from F^W_Q (Z) evaluations.

The limit on $F_Q(Z)$ is in place to prevent core power distributions that would violate design criteria associated with the local (i.e., pellet) peak power density. $F^W_Q(Z)$ is determined by multiplying the measured steady state value of $F_Q(Z)$ by 1.0815 and by a factor W(z) which accounts for plant maneuvers. $F^W_Q(Z)$ is then compared to the limit on $F_Q(Z)$. Currently, if $F^W_Q(Z)$ exceeds the $F_Q(Z)$ limit, thermal power is reduced proportionately. The proposed changes require reducing the AFD limits proportionately if $F^{W}_{Q}(Z)$ exceeds the $F_Q(Z)$ limit.

The use of RAOC methodology can minimize and/or smooth the boron system use relative to constant axial offset control (CAOC) operations. Also, it can reduce rod motion corrections and hence operator actions required to maintain conformance with power distribution control Technical Specifications. Finally, it can significantly improve the ability to return to power after a plant trip.

These changes are acceptable because they are consistent with NUREG-1431 and with the RAOC and F_Q Surveillance methodology as described in Reference 1 which has been approved by the NRC.

B. **TS 3.2.3, Axial Flux Difference (AFD) and Bases:** Modification of Limiting Condition for Operation (LCO), Actions and Surveillance Requirements and revision of the Bases.

Current Specification 3.2.3 and Surveillances have been completely replaced with the RAOC Specification and Surveillance for AFD control which conforms to the guidance of NUREG-1431. The Bases have also been revised extensively to support the RAOC methodology and conform to the guidance of NUREG-1431.

The AFD is a measure of axial power distribution skewing to the top or bottom half of the core. The limits on AFD assure that $F_Q(Z)$ is not exceeded during normal operation including allowed operational transients. The current TS refers to a target AFD band located in the COLR and allows deviation outside of the target band for certain periods of time depending on the power level and accrued penalty time. The proposed TS would replace the current TS in its entirety. The new TS refer to AFD limits and operational space that are located in the COLR. If the AFD is not within its limits, within 30 minutes AFD must be brought into compliance or power must be reduced to less than 50% rated thermal power (RTP). The AFD limits will be determined during the cycle specific reload evaluation process. Requirements for establishing and operating within a target AFD band will be addressed in operating procedures. The changes are acceptable because they are consistent with NUREG-1431 and with the intent of RAOC and F_Q Surveillance methodology as described in Reference 1 which has been approved by the NRC.

GROUP 2 - Relocation of TS SL Figure and OTΔT and OPΔT parameter values to the COLR, and miscellaneous administrative changes

A. **TS 2.1.1, "Reactor Core SLs" and Bases:** Relocate the SL Figure to the COLR, update TS 2.1.1 and Bases.

Currently, TS 2.1.1, "Reactor Core SLs," refers to Figure 2.1.1-1, "Reactor Core Safety Limits." Based on the use of the NRC approved methods documented in Reference 2, this figure is being relocated to the COLR. The appropriate additional requirements (departure from nucleate boiling ratio (DNBR) limit and peak centerline fuel temperature) discussed in Reference 2 are added to TS 2.1.1. The 2.1.1 Bases have been revised to support these changes and conform to the guidance of NUREG-1431. The proposed SLs and Bases changes are provided in Exhibits B and C and are acceptable because they are consistent with the guidance of Reference 2 which has been approved by the NRC. The SLs will be determined during the cycle specific reload evaluation.

B. **TS 3.3.1, Table 3.3.1-1 (Pages 2, 7 and 8), "Reactor Trip System Instrumentation", Overpower Delta-T Trip Function, and Overtemperature Delta-T and Overpower Delta-T parameter values:** Delete SR 3.3.1.3, SR 3.3.1.6 and remove f(ΔI) from Overpower Delta-T Trip Function, relocate OTΔT and OPΔT parameter values and revise the Bases.

The ΔI function input to the Overpower Delta-T Trip Function will be removed and accordingly f(ΔI) will be removed from the OPΔT equation in Table 3.3.1-1 (page 8). SR 3.3.1.3 and SR 3.3.1.6 will be removed from Function 7, OPΔT in Table 3.3.1-1 (page 2). Corresponding changes to the Bases for Table 3.3.1-1 Function 7 and SRs 3.3.1.3 and 3.3.1.6 have also been made. These TS changes are provided in Exhibits B and C and are acceptable because they are consistent with the guidance of NUREG-1431, which has been approved by the NRC.

Currently TS 3.3.1, "Reactor Trip System (RTS) Instrumentation", contains OTΔT and OPΔT parameter values. These values are located in Table 3.3.1-1 (pages 7 and 8). Based on the use of the NRC approved methods documented in Reference 2, these values are being relocated to the COLR. The TS changes are provided in Exhibits B and C and are acceptable because they are consistent with the guidance of Reference 2, which has been approved by the NRC. The OTΔT and OPΔT parameter values will be determined during the cycle specific reload evaluation.

C. **TS 5.6.5, CORE OPERATING LIMITS REPORT (COLR):** Additions to document TS with limits in the COLR and the analytical methods used to determine the values for relocated SLs and OTΔT and OPΔT parameters and miscellaneous administrative changes.

Specification 5.6.5.a provides a list of Specifications for which the core operating limits are documented in the COLR. This license amendment request proposes to add the necessary TS sections to document the relocation of the SLs and OTΔT and OPΔT parameter values to the COLR. Current Specification 5.6.5.b Items 11 and 12 include an incorrect date in the reference to NRC SE dated July 30, 2002. In accordance with the guidance of NUREG-1431, this incorrectly dated reference is deleted. Additionally, closing quotation marks were added for the title of LCO 3.2.2, a space was added in 5.6.5.b Item 2 and the margins were revised in Specification 5.6.8. These changes are acceptable since they are administrative changes.

Specification 5.6.5.b provides a list of the analytical methods used to determine the core operating limits. This license amendment request proposes to update the references to add the NRC approved Westinghouse methodologies that are applicable. These additional references are:

- WCAP 10216-PA-Revision 1A, "Relaxation Of Constant Axial Offset Control/ F_Q Surveillance Technical Specification";
- WCAP 8745-PA, "Design Basis for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions";
- WCAP 11397-PA, "Revised Thermal Design Procedure"; and
- WCAP 14483-A, "Generic Methodology for Expanded Core Operating Limits Report".

This change is acceptable since this is an administrative change.

GROUP 3 – Revision of Pressurizer Pressure-Low reactor trip Allowable Value

TS 3.3.1, Table 3.3.1-1 (Page 2), "Reactor Trip System Instrumentation", Function 8.a, Pressurizer Pressure-Low: Increase Pressurizer Pressure-Low Allowable Value.

The Allowable Value currently defined in Table 3.3.1-1 for the Pressurizer Pressure Low reactor trip (Function 8.a) is ≥ 1760 psig. Based on the use of the NRC approved methods documented in Reference 7, this value is being increased to ≥ 1845 psig to limit the range of pressures over which the OTΔT and OPΔT reactor trip functions are required to provide protection. The change is in the conservative direction, that is, the Pressurizer Pressure-Low Allowable Value is at a higher pressure which will be reached sooner. This change is acceptable because it is consistent with NRC approved Westinghouse methodology in Reference 7 and in accordance with the PINGP setpoint methodology (Reference 8) which was accepted by the NRC.

SAFETY EVALUATION

GROUP 1 – Implementation of relaxed axial offset control

A. **TS 3.2.1, Heat Flux Hot Channel Factor - $F_Q(Z)$ and Bases:** Modification of Required Actions and Completion Time if F^W_Q (Z) is not within its limit and update Bases.

B. **TS 3.2.3, Axial Flux Difference (AFD) and Bases:** Modification of LCO, Actions and Surveillance Requirements and revision of the Bases.

Axial power distribution control at the PINGP is currently achieved by use of the CAOC methodology. This method assures peaking factors and DNBR remain below the accident analysis limits by maintaining the axial power distribution within a band of delta-I around a measured target value during normal plant operation (including power change maneuvers). By controlling the axial power distribution in this manner, the possible skewing of the axial xenon distribution is limited, thus minimizing xenon oscillations and their effects on the power distribution.

Future reload designs for Prairie Island Units 1 and 2 will incorporate the RAOC and F_Q Surveillance methodology, as documented in Reference 1. The NRC has previously approved the implementation of RAOC and F_Q Surveillance methodology described in Reference 1. This strategy was developed to provide wider control bands and more operating flexibility than with CAOC. The wider control bands are achieved particularly at reduced power by effectively utilizing some of the available core margin to the peaking factor limits specified in the COLR.

The heat flux hot channel factor, $F_Q(Z)$, is the maximum local heat flux on the surface of a fuel rod at core elevation z, divided by the average fuel rod heat flux. For plants using RAOC and F_Q Surveillance methodology during normal operation, $F_Q(Z)$ is shown to be within its limits by performing periodic measurements of the steady state $F_Q(Z)$. The measured value is then increased by a factor of 1.0815, which accounts for manufacturing tolerances and measurement uncertainty. A multiplication factor, W(z), is then applied which accounts for plant maneuvers within the restriction on AFD and rod insertion permitted by the Technical Specifications. The product of the measured $F_Q(Z)$, 1.0815, and the analytically determined W(z), denoted as $F^W{}_Q(Z)$, is then compared to the $F_Q(Z)$ limit. If $F^W{}_Q(Z)$ exceeds its limit, the AFD limits are reduced proportionately.

The AFD is a measure of axial power distribution skewing to the top or bottom half of the core. It is sensitive to core-related parameters such as control bank position, core power level, axial burnup, and axial xenon distribution. The limits on AFD assure that the limit on heat flux hot channel factor, $F_Q(Z)$, is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The AFD limits are used in the nuclear design process and assumed in the safety analyses as a boundary of possible initial condition axial power shapes. Operation outside these limits during Condition I operation influences the possible power shapes which could result from Condition II transients. Condition II transients, assumed to begin from within the AFD limits, are used to confirm the adequacy of OTΔT and OPΔT Allowable Values.

The AFD must be maintained within the allowed operations band as a function of power. The allowed operating space, which is defined in the COLR, becomes the Technical Specification. If these limits are exceeded, within a 30 minute grace period the AFD must be returned within the limits or power must be reduced to less than 50% RTP.

The AFD bands will be determined during the cycle specific reload evaluation process (Reference 3) performed for Prairie Island Units 1 and 2 in accordance with the NRC approved Westinghouse methodology (Reference 1). Since the AFD limits are provided in the COLR, WCAP-10216 (Reference 1) is referenced in accordance with the reporting requirement of TS 5.6.5 which specifies that the applicable NRC-approved methodologies used to determine core operating limits should be referenced.

> *WCAP-10216-P-A, Rev 1A, Relaxation of Constant Axial Offset Control/ F_Q Surveillance Technical Specification*
>
> The RAOC methodology, including the associated F_Q Surveillance methods (Reference 1), has previously been approved by the NRC for use as an acceptable method for power distribution control in Westinghouse designed pressurized water reactors. Prairie Island Units 1 and 2 are both Westinghouse designed two-loop pressurized water reactors. Therefore, the methodology is directly applicable.

Application of the RAOC methodology requires revision of the Technical Specifications. These changes to the TS are consistent with the intent of NRC approved methodology described in Reference 1 and, with these changes, operation of the PINGP within the TS limits will continue to protect the health and safety of the public.

GROUP 2 - Relocation of TS SL Figure and OTΔT and OPΔT parameter values to the COLR, and miscellaneous administrative changes

A. **TS 2.1.1, "Reactor Core SLs" and Bases:** Relocate the SL Figure to the COLR, update TS 2.1.1 and Bases.

B. **TS 3.3.1, Table 3.3.1-1 (Pages 2, 7 and 8), "Reactor Trip System Instrumentation", Overpower Delta-T Trip Function, and Overtemperature Delta-T and Overpower Delta-T parameter values:** Delete SR 3.3.1.3, SR 3.3.1.6, and remove f(ΔI) from Overpower Delta-T Trip Function, relocate OTΔT and OPΔT parameter values and revise the Bases.

C. **TS 5.6.5, CORE OPERATING LIMITS REPORT (COLR):** Additions to document TS with limits in the COLR and the analytical methods used to determine the values for relocated SLs and OTΔT and OPΔT parameters and miscellaneous administrative changes.

NRC Generic Letter 88-16 (Reference 4) allows licensees to remove cycle dependent variables from the TS provided that the values of these variables are included in a COLR. Also, these cycle dependent variables are to be determined in accordance with NRC approved methods. These cycle dependent variables are moved to the COLR to avoid the need for revisions to the TS whenever the values change. Prairie Island has previously gained NRC approval for use of a COLR and has relocated other cycle dependent variables. This proposed license amendment expands the COLR to include SLs and OTΔT and OPΔT parameter values as prescribed in Reference 2. The NRC approved the expanded COLR methodology which supports these relocations via Reference 5.

Based on the use of the NRC approved method documented in Reference 2, the relocation of SLs and OTΔT and OPΔT parameter values to the COLR is acceptable. The appropriate additional requirements (DNBR limit and peak centerline fuel temperature) discussed in the NRC Staff safety evaluation report (SER) for Reference 2 are being added to the TS. The SLs and OTΔT and OPΔT parameter values that are being relocated to the COLR will be determined during the cycle specific reload evaluation process (Reference 3) using NRC approved analytical methods (References 6 and 7) which have been added to Specification 5.6.5.b. The Allowable Values will be calculated from the safety analysis limit values determined for K_1 and K_4 using PINGP setpoint methodology (Reference 8) previously accepted by the NRC.

The Westinghouse Overtemperature Delta-T and Overpower Delta-T methodology (Reference 7) does not require a ΔI function input to the Overpower Delta-T (OPΔT) Trip Function. The ΔI function input to the Overpower Delta-T Trip Function will be removed and accordingly f(ΔI) has been removed from the OPΔT equation in Table 3.3.1-1 (page 8). SR 3.3.1.3 and SR 3.3.1.6 have been removed from Function 7, OPΔT in Table 3.3.1-1 (page 2) since these SRs will not be required to verify input to the OPΔT Trip Function. Corresponding changes to the Bases for Table 3.3.1-1 Function 7 and SRs 3.3.1.3 and 3.3.1.6 have also been made.

The methods (References 6 and 7) for determining core operating limits and their applicability to Prairie Island Units 1 and 2 are discussed below.

WCAP 11397-P-A, Revised Thermal Design Procedure

WCAP-11397-P-A (Reference 6) is referenced in accordance with the reporting requirement of TS 5.6.5 which specifies that the applicable NRC-approved methodologies used to determine core operating limits should be referenced. The core operating limits in this case are Safety Limits. The NRC Staff reviewed Reference 6 and concluded in a Staff SER that the generic topical report was an acceptable reference to support plant specific applications for use of the Revised Thermal Design Procedure (RTDP), provided seven conditions identified in the SER are addressed by the licensees. Each of the seven conditions is addressed below:

Condition 1 Sensitivity factors for a particular plant and their ranges of applicability should be included in the Safety Analysis Report or reload submittal.

Response Sensitivity factors were evaluated using the WRB-1 departure from nucleate boiling (DNB) correlation and the VIPRE code for parameter values applicable to the 14x14 OFA fuel in Prairie Island Units 1 and 2. These sensitivity factors were used to determine the maximum Design Limit DNBR for the OFA fuel. The resultant Design Limit DNBR will be included in the Prairie Island Unit 1 and 2 Updated Safety Analysis Report.

Condition 2 Any changes in DNB correlation, THINC-IV correlations, or parameter values listed in Table 3-1 of WCAP-11397 outside of previously demonstrated acceptable ranges require re-evaluation of the sensitivity factors and of the use of Equation (2-3) of the topical report.

Response See Response to Condition 1 for the justification of WRB-1 correlation and the VIPRE-01 code for the Prairie Island Unit 1 and 2 application.

Condition 3 If the sensitivity factors are changed as a result of correlation changes or changes in the application or use of the THINC code, then the use of an uncertainty allowance for application of Equation (2-3) must be re-evaluated and the linearity assumption made to obtain Equation (2-17) of the topical report must be validated.

Response Equation (2-3) of WCAP-11397-P-A and the linearity approximation made to obtain Equation (2-17) have been shown to be valid for the combination of WRB-1 and the VIPRE code which was used for the application of RTDP to the OFA fuel in Prairie Island Units 1 and 2. The sensitivity factors, operating parameters, and the VIPRE model used in this application do not differ significantly from those used in WCAP-11397-P-A.

Condition 4 Variances and distributions for input parameters must be justified on a plant-by-plant basis until generic approval is obtained.

Response PINGP-specific RTDP input uncertainties (variances and distributions) are justified in the calculations included in Exhibit D of this LAR. These uncertainty calculations are developed using the PINGP setpoint methodology previously accepted by the NRC (Reference 8).

Condition 5 Nominal initial condition assumptions apply only to DNBR analyses using RTDP. Other analyses, such as overpressure calculations, require the appropriate conservative initial condition assumptions.

Response Nominal initial conditions are only applied to DNBR analyses which use RTDP.

Condition 6 Nominal conditions chosen for use in analyses should bound all permitted methods of plant operation.

Response Bounding nominal conditions are used in the DNBR analyses that are based on RTDP.

Condition 7 The code uncertainties specified in Table 3-1 (± 4 percent for THINC-IV and ± 1 percent for transients) must be included in the DNBR analyses using RTDP.

Response The code uncertainties specified in Table 3-1 of WCAP-11397-P-A are included in the DNBR analyses using RTDP.

WCAP-8745-P-A, Design Basis for the Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Functions

WCAP-8745 (Reference 7) is referenced in accordance with the reporting requirement of TS 5.6.5 which specifies that the applicable NRC-approved methodologies used to determine core operating limits should be referenced. The core operating limits in this case are the OTΔT and OPΔT parameter values. Reference 7 describes the bases for the OTΔT and OPΔT trip functions in Westinghouse reactors, and the analytical methods used to derive the limiting safety system settings for the trips. In accordance with the NRC Staff SER, Reference 7 is acceptable for referencing by licensees in licensing documents. Although the SER specifies the acceptance for plants operating under CAOC, the use of RAOC does not invalidate the applicability of Reference 7. As stated in Section 3.2 of the SER, "...the basic design philosophy described in WCAP-8745 is not invalidated by changes in DNB analysis methodology, fuel design, and plant operating procedure...". In addition the SER states, "The adequacy of the standard power shapes in establishing the core DNB protection system must be evaluated whenever changes are introduced that could potentially affect the core power distribution." The OTΔT and OPΔT parameter values to be included in the Prairie Island COLR will account for the use of RAOC, as well as the specific fuel type and DNB analysis methodology applicable to Prairie Island.

The TS changes described above are consistent with the guidance of Reference 2 which has been approved by the NRC. The Safety Limit and OTΔT and OPΔT parameter values will be determined as part of the cycle specific reload evaluation process (Reference 3) performed for Prairie Island Units 1 and 2 in accordance with NRC approved Westinghouse methodology (References 6 and 7). The Prairie Island TS with these changes will continue to protect the health and safety of the public.

GROUP 3 – Revision of Pressurizer Pressure-Low reactor trip Allowable Value

TS 3.3.1, Table 3.3.1-1 (Page 2), "Reactor Trip System Instrumentation", Function 8.a, Pressurizer Pressure-Low: Increase Pressurizer Pressure-Low Allowable Value.

Future reload designs for Prairie Island Units 1 and 2 will incorporate OTΔT and OPΔT parameters developed according to the NRC approved Westinghouse methodology described in Reference 7. Based on that methodology, the Pressurizer Pressure-Low reactor trip safety limit (in conjunction with the Pressurizer Pressure-High reactor trip safety limit) provides limitations on the range of protection required by the OTΔT and OPΔT reactor trips. The proposed change to the Allowable Value for the Pressurizer Pressure-Low reactor trip has been determined in accordance with the PINGP setpoint methodology (Reference 8) using the safety limits from this methodology (Reference 7), and is expected to provide acceptable limitations for the OTΔT and OPΔT reactor trip functions. The specific parameter values for the OTΔT and OPΔT reactor trips will be determined during the cycle specific reload evaluation process.

This TS change is in the conservative direction for this reactor trip function and the TS with this change will continue to protect the health and safety of the public.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

GROUP 1 – Implementation of relaxed axial offset control

A. **TS 3.2.1, Heat Flux Hot Channel Factor - $F_Q(Z)$ and Bases:** Modification of Required Actions and Completion Time if F^W_Q (Z) is not within its limit and update Bases.

B. **TS 3.2.3, Axial Flux Difference (AFD) and Bases:** Modification of Limiting Conditions for Operation, Actions and Surveillance Requirements and revision of the Bases.

This license amendment request proposes to revise the Technical Specifications to implement the relaxed axial offset control methodology to address the heat flux hot channel factor and axial flux difference limits.

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

   This license amendment request proposes to revise the Technical Specifications to implement the relaxed axial offset control methodology to address the heat flux hot channel factor and axial flux difference limits. The revised Technical Specifications and parameter changes associated with relaxed axial offset control assure that the limiting safety analysis inputs (such as, heat flux hot channel factor and axial flux difference limits) are not exceeded. The bounding power distribution transient factor values, W(Z), and the axial flux difference limits that are documented in the Core Operating Limits Report will be determined by NRC approved analytical methods and will be validated as part of the cycle specific reload evaluation process.

   Heat flux hot channel factors and axial flux difference limits are not assumed accident initiators. Therefore, the relaxed axial offset control related Technical Specification changes do not involve a significant increase in the probability of an accident.

Likewise, operation of the plant within the proposed Technical Specification controls and limits assures that safety analysis assumptions are met, thus, if an accident were to occur, the consequences would continue to be bounded by the accident analyses. Therefore, the relaxed axial offset control related technical specification changes do not involve a significant increase in the consequences of an accident.

The relaxed axial offset control related technical specification changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. <u>The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.</u>

   This proposed change does not involve a physical alteration of the plant; that is, no new or different type of equipment will be installed. This proposed change does not introduce any new mode of plant operation or change the methods governing normal plant operation. No new failure mode has been created and no new equipment performance burdens are imposed. Therefore the possibility of a new or different kind of accident from those previously analyzed has not been created.

3. <u>The proposed amendment will not involve a significant reduction in the margin of safety.</u>

   This license amendment request proposes to revise the Technical Specifications to implement the relaxed axial offset control methodology to address the heat flux hot channel factor and axial flux difference limits. The supporting Technical Specification limits are defined by NRC approved analytical methods which are performed to conservatively bound the operating conditions defined by the Technical Specifications and to demonstrate meeting the regulatory acceptance limits. The heat flux hot channel factor licensed safety margins are maintained. The heat flux hot channel factor conforms to plant design bases and limits actual plant operation within analyzed and licensed boundaries. The relaxed axial offset control methodology has been demonstrated to ensure that core heat flux hot channel factors will remain below accident analysis limits. The margin of safety provided by the analyses in accordance with the acceptance limits is maintained and not reduced. Thus, the implementation of relaxed axial offset control at Prairie Island does not involve a significant reduction in a margin of safety.

GROUP 2 - Relocation of Technical Specifications Safety Limits Figure and Overtemperature Delta-T and Overpower Delta-T parameter values to the Core Operating Limits Report, and miscellaneous administrative changes

A. **TS 2.1.1, "Reactor Core SLs" and Bases:** Relocate the safety limits Figure to the Core Operating Limits Report, update TS 2.1.1 and Bases.

B. **TS 3.3.1, Table 3.3.1-1 (Pages 2, 7 and 8), "Reactor Trip System Instrumentation", Overpower Delta-T Trip Function, and Overtemperature Delta-T and Overpower Delta-T parameter values:** Delete SR 3.3.1.3, SR 3.3.1.6, and remove f(ΔI) from Overpower Delta-T Trip Function, relocate overtemperature delta-T and overpower delta-T parameter values and revise the Bases.

C. **TS 5.6.5, CORE OPERATING LIMITS REPORT (COLR):** Additions to document Technical Specifications with limits in the Core Operating Limits Report and the analytical methods used to determine the values for relocated safety limits and overtemperature delta-T and overpower delta-T parameters and miscellaneous administrative changes.

This license amendment request proposes to relocate the safety limits and overtemperature delta-T and overpower delta-T parameter values to the Core Operating Limits Report. Relocation of these limits and parameter values to the Core Operating Limits Report allows them to be changed under licensee controls. This license amendment also proposes to include, in the Technical Specifications administrative controls section, the appropriate references to the NRC approved methodologies which will be used to determine the safety limits and overtemperature delta-T and overpower delta-T parameter values. These changes are acceptable because the values used to operate the Prairie Island plant will be determined using NRC approved methods and these changes are consistent with the guidance of the industry standard Technical Specifications, NUREG-1431, Revision 2, "Standard Technical Specifications Westinghouse Plants". This license amendment request also proposes to delete references to an NRC Safety Evaluation and make some editorial corrections in the Technical Specifications administrative controls section. These changes are acceptable since they are administrative and do not affect plant operation.

1. <u>The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

   This license amendment request proposes to relocate the safety limits and overtemperature delta-T and overpower delta-T parameter values to the Core Operating Limits Report and to include, in the Technical Specifications administrative controls section, the appropriate references to the NRC approved methodologies which support determination of these limits and parameter values. The safety limits and overtemperature delta-T and overpower delta-T parameter values that are documented in the Core Operating Limits Report will be determined by NRC approved analytical methods and will be validated as part of the cycle specific reload evaluation process.

Safety limits are not assumed accident initiators. Thus relocation of the safety limits does not involve a significant increase in the probability of an accident. Overtemperature delta-T and overpower delta-T parameter values are inputs to the reactor trip system which is provided to mitigate the consequences of an accident. The reactor trip system is not an accident initiator and therefore, changes to input values do not increase the probability of an accident.

Safety limits define bounding values within which plant operation will not initiate an accident condition. Safety limits relocated to the Core Operating Limits Report and determined by use of NRC approved methodologies will continue to determine the safe limits of plant operation, thus this change does not involve a significant increase in the consequences of an accident. The reactor trip system, with inputs from the overtemperature delta-T and overpower delta-T trip functions, mitigates the consequences of accidents. The overtemperature delta-T and overpower delta-T trip parameter values are determined to assure that the design limit departure from nucleate boiling ratio is met and fuel integrity is maintained. Overtemperature delta-T and overpower delta-T trip parameters relocated to the Core Operating Limits Report and values determined by use of NRC approved methodologies will continue to determine the inputs for these trip functions which mitigate the design basis accident consequences, thus this change does not involve a significant increase in the consequences of an accident.

Addition of references to NRC approved methodologies in the Technical Specifications administrative controls section is an administrative change which does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed miscellaneous administrative changes in the Technical Specifications administrative controls section do not affect plant operation and therefore do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

The malfunction of safety related equipment, assumed to be operable in the accident analyses, would not be impacted as a result of the proposed technical specification changes. No new failure mode has been created and no new equipment performance burdens are imposed. Therefore the possibility of a new or different kind of accident from those previously analyzed has not been created. The proposed administrative changes do not create the possibility of a new or different kind of accident from those previously analyzed.

3. <u>The proposed amendment will not involve a significant reduction in the margin of safety.</u>

   This license amendment request proposes to relocate the safety limits and overtemperature delta-T and overpower delta-T parameter values to the Core Operating Limits Report and to include, in the Technical Specifications administrative controls section, the appropriate references to the NRC approved methodologies which support determination of these limits and parameter values. This proposed change also allows these relocated limits and parameter values to be changed under licensee controls. Safety limits in the Core Operating Limits Report will be determined by use of NRC approved methodologies and will continue to determine the safe limits of plant operation. Overtemperature delta-T and overpower delta-T trip parameter values in the Core Operating Limits Report will be determined by use of NRC approved methodologies and will continue to determine the inputs for these trip functions which mitigate design basis accidents. The Safety Limits licensed safety margins are maintained. The Safety Limits conform to plant design bases and limit actual plant operation within analyzed and licensed boundaries. The methodology described in WCAP-8745, along with the low pressurizer pressure allowable value, ensures that the overtemperature delta-T and overpower delta-T trips will protect against fuel centerline melting and departure from nucleate boiling during Condition II events. Thus, these changes do not involve a significant reduction in the margin of safety.

   This license amendment request proposes to delete references to an NRC Safety Evaluation and make some editorial corrections in the Technical Specifications administrative controls section. These changes are administrative and thus do not involve a significant reduction in the margin of safety.

GROUP 3 – Revision of Pressurizer Pressure-Low reactor trip Allowable Value

TS 3.3.1, Table 3.3.1-1 (Page 2), "Reactor Trip System Instrumentation", Function 8.a, Pressurizer Pressure-Low: Increase Pressurizer Pressure-Low Allowable Value.

This license amendment request proposes to increase the Allowable Value defined in Table 3.3.1-1 for the Pressurizer Pressure-Low reactor trip.

1. <u>The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

   This license amendment request proposes to increase the Allowable Value defined in Table 3.3.1-1 for the Pressurizer Pressure-Low reactor trip. Pressurizer Pressure-Low reactor trip is an input to the reactor trip system which is provided to mitigate the consequences of an accident. The reactor trip system is not an accident initiator and therefore, changes to the Pressurizer Pressure-Low Allowable Value do not involve an increase in the probability of an accident.

The Pressurizer Pressure-Low Allowable Value is being increased which is a conservative change. The increase in the Pressurizer Pressure-Low reactor trip Allowable Value will assure that the overtemperature delta-T and overpower delta-T reactor trip functions, with values determined in accordance with NRC approved methodologies, provide protection against fuel centerline melting and departure from nucleate boiling for overpower and overtemperature events. Therefore, this change does not involve an increase in the consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   This proposed change does not involve a physical alteration of the plant; that is, no new or different type of equipment will be installed. This proposed change does not introduce any new mode of plant operation or change the methods governing normal plant operation. No new failure mode has been created and no new equipment performance burdens are imposed. Therefore the possibility of a new or different kind of accident from those previously analyzed has not been created.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   This license amendment request proposes to increase the Allowable Value defined in Table 3.3.1-1 for the Pressurizer Pressure-Low reactor trip. The Allowable Value is determined in accordance with an NRC accepted setpoint methodology with input from NRC approved analytical methods. These determinations are performed to conservatively bound the operating conditions defined by the Technical Specifications and to demonstrate meeting the regulatory acceptance limits.

   Performance of analyses and evaluations for the cycle specific reload evaluation process will confirm that the operating envelope defined by the Technical Specifications continues to be bounded by the analytical basis and in no case exceeds the acceptance limits. The proposed Pressurizer Pressure-Low Allowable Value along with the overtemperature delta-T and overpower delta-T trips will protect against fuel centerline melting and departure from nucleate boiling during Condition II events. The proposed Allowable Value conforms to plant design bases and limits actual plant operation within analyzed and licensed boundaries. The margin of safety provided by the proposed Pressurizer Pressure-Low Allowable Value is maintained and not reduced. Thus, the increase in the Pressurizer Pressure-Low reactor trip Allowable Value does not involve a significant reduction in a margin of safety.

CONCLUSION

Considering the above evaluations and pursuant to 10CFR50.91, the Nuclear Management Company has determined that operation of the Prairie Island Nuclear Generating Plant in accordance with the proposed license amendment request does not involve a significant hazards consideration as defined by Nuclear Regulatory Commission regulations in 10CFR50.92.

ENVIRONMENTAL ASSESSMENT

The Nuclear Management Company has evaluated the proposed change and determined that:

1. The changes do not involve a significant hazards consideration,

2. The changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and

3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22(c)(9). Therefore, pursuant to 10CFR51.22(b), an environmental assessment of the proposed changes is not required.

REFERENCES

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

2. WCAP 14483-A, "Generic Methodology for Expanded Core Operating Limits Report," January 1999.

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

4. NRC Generic Letter 88-16, "Guidance for Technical Specification Changes for Cycle-Specific Parameter Limits," October 1988.

5. NRC SER Letter, "Acceptance for Referencing of Licensing Topical Report WCAP 14483-A, "Generic Methodology for Expanded Core Operating Limits Report," (TAC No. M94338), January 1999.

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

7. WCAP 8745-P-A, "Design Basis for the Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Functions," September 1986.

8. Engineering Manual Section 3.3.4.1, "Engineering Design Standard for Instrument Setpoint/Uncertainty Calculations", accepted by NRC SER dated July 26, 2002, Section G.2.1 (TAC NOS. MB0695 and MB0696).

EXHIBIT B

PRAIRIE ISLAND NUCLEAR GENERATING STATION

License Amendment Request dated March 25, 2003

Marked Up Pages
(shaded material to be added, strikethrough material to be removed)

Technical Specification Pages

2.0-1	3.3.1-18
2.0-2	3.3.1-23
3.2.1-2	3.3.1-24
3.2.1-3	5.0-35
3.2.3-1	5.0-36
3.2.3-2	5.0-37
3.2.3-3	5.0-38
3.2.3-4	5.0-39
Insert A 3.2.3-1	5.0-40

Bases Pages

B 2.1.1-2	B 3.2.1-13
B 2.1.1-3	B 3.2.1-14
B 2.1.1-4	B 3.2.3-1
B 2.1.1-5	B 3.2.3-2
B 2.1.1-6	B 3.2.3-3
B 3.2.1-2	B 3.2.3-4
B 3.2.1-3	B 3.2.3-5
B 3.2.1-4	B 3.2.3-6
B 3.2.1-5	B 3.2.3-7
B 3.2.1-6	B 3.2.3-8
B 3.2.1-7	B 3.2.3-9
B 3.2.1-8	B 3.2.3-10
B 3.2.1-9	B 3.2.3-11
B 3.2.1-10	B 3.3.1-20
B 3.2.1-11	B 3.3.1-56
B 3.2.1-12	B 3.3.1-58

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 **limits** ~~SLs~~ specified in **the COLR** ~~Figure 2.1.1-1~~ **and the following SLs shall not be exceeded:**

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.17 for WRB-1 DNB correlation for OFA fuel.

2.1.1.2 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 from Technical Specifications

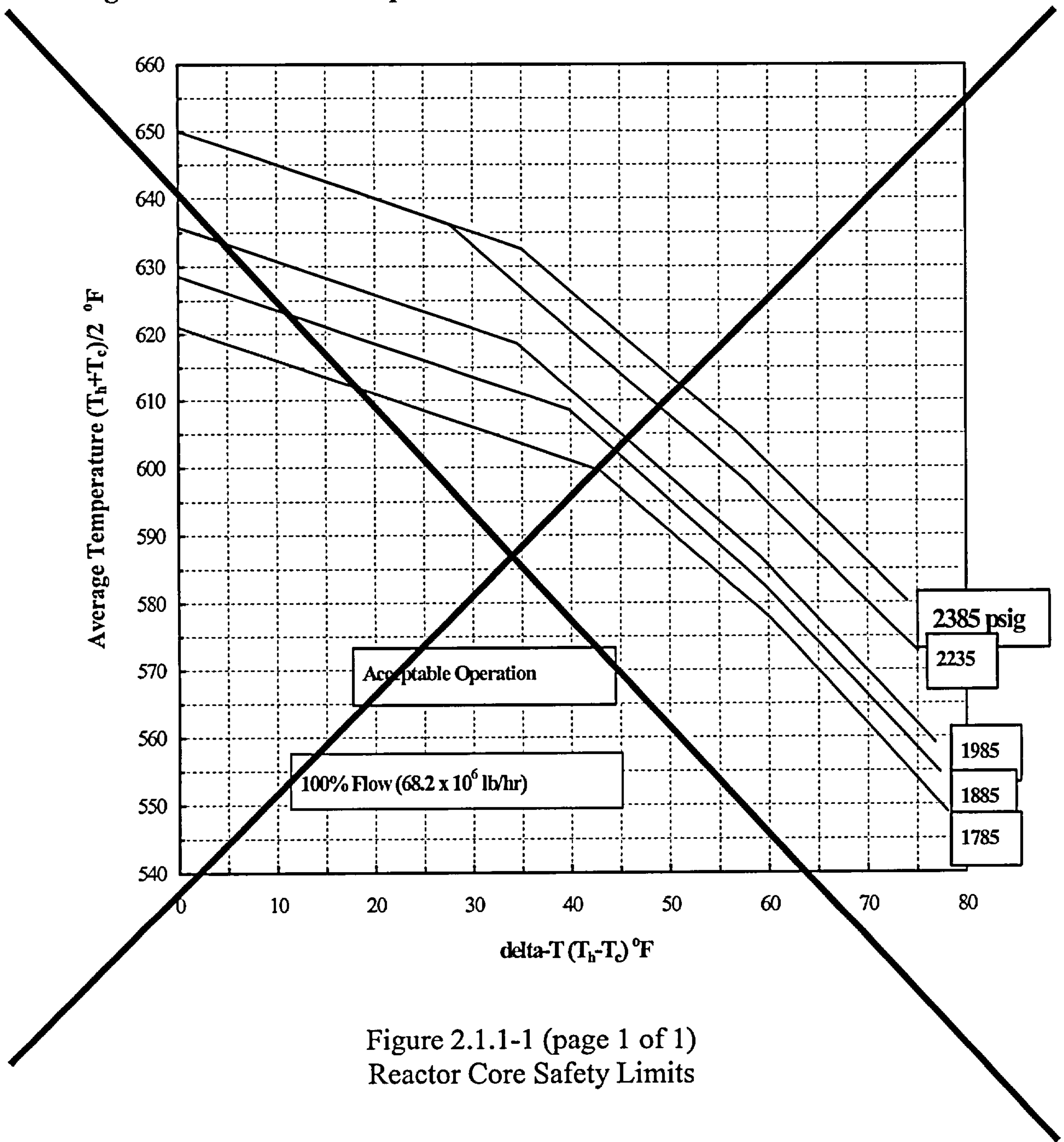

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.4 Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
B. -----------NOTE----------- Required Action B.4 shall be completed whenever this Condition is entered. ----------------------------- $F^W_Q(Z)$ not within limits.	B.1 Reduce AFD limits~~THERMAL POWER~~ ≥ 1% ~~RTP~~ for each 1% $F^W_Q(Z)$ exceeds limit. AND B.2 Reduce Power Range Neutron Flux-High trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limit is reduced~~$F^W_Q(Z)$ exceeds limit~~. AND B.3 Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limit is reduced~~$F^W_Q(Z)$ exceeds limit~~. AND	4 hours after each $F^W_Q(Z)$ determination 72 hours after each $F^W_Q(Z)$ determination 72 hours after each $F^W_Q(Z)$ determination

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.4 Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above **the maximum allowable power of the AFD limits** ~~the limit of Required Action B.1~~
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 2.	6 hours

~~3.2 POWER DISTRIBUTION LIMITS~~

~~3.2.3 AXIAL FLUX DIFFERENCE (AFD)~~

Replace CAOC Specification with RAOC Specification
INSERT A

~~LCO 3.2.3 The AFD:~~

~~a. Shall be maintained within the target band about the target flux difference. The target band is specified in the COLR.~~

~~b. May deviate outside the target band with THERMAL POWER < 90% RTP but ≥ 50% RTP, provided AFD is within the acceptable operation limits and cumulative penalty deviation time is ≤ 1 hour during the previous 24 hours. The acceptable operation limits are specified in the COLR.~~

~~c. May deviate outside the target band with THERMAL POWER < 50% RTP.~~

~~NOTES~~

~~1. The AFD shall be considered outside the target band when two or more OPERABLE excore channels indicate AFD to be outside the target band.~~

~~2. With THERMAL POWER ≥ 50% RTP, penalty deviation time shall be accumulated on the basis of a 1 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.~~

~~3. With THERMAL POWER < 50% RTP and >15% RTP, penalty deviation time shall be accumulated on the basis of a 0.5 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.~~

~~4. A total of 16 hours of operation may be accumulated with AFD outside the target band without penalty deviation time during surveillance of power range channels in accordance with SR 3.3.1.6, provided AFD is maintained within acceptable operation limits.~~

~~APPLICABILITY: MODE 1 with THERMAL POWER > 15% RTP.~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. THERMAL POWER ≥ 90% RTP.~~ ~~AND~~ ~~AFD not within the target band.~~	~~A.1 Restore AFD to within target band.~~	~~15 minutes~~
~~B. Required Action and associated Completion Time of Condition A not met.~~	~~B.1 Reduce THERMAL POWER to < 90% RTP.~~	~~15 minutes~~

~~ACTIONS (continued)~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~C. ------------NOTE------------ Required Action C.1 must be completed whenever Condition C is entered.~~ ~~THERMAL POWER < 90% and ≥ 50% RTP with cumulative penalty deviation time > 1 hour during the previous 24 hours.~~ ~~OR~~ ~~THERMAL POWER < 90% and ≥ 50% RTP with AFD not within the acceptable operation limits.~~	~~C.1 Reduce THERMAL POWER to < 50% RTP.~~	~~30 minutes~~
~~D. Required Action and associated Completion Time for Condition C not met.~~	~~D.1 Reduce THERMAL POWER to < 15% RTP.~~	~~9 hours~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~		~~FREQUENCY~~
~~SR 3.2.3.1~~	~~Verify AFD is within limits for each OPERABLE excore channel.~~	~~7 days~~
~~SR 3.2.3.2~~	~~NOTE~~ ~~The initial target flux difference after each refueling may be determined from design predictions.~~ ~~Determine, by measurement, and update target flux difference.~~	~~Once within 31 EFPD after each refueling~~ ~~AND~~ ~~31 EFPD thereafter~~

3.2 POWER DISTRIBUTION LIMITS

INSERT A

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

LCO 3.2.3 The AFD in % flux difference units shall be maintained within the limits specified in the COLR.

-----------------------------NOTE-----------------------------
The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.3.1 Verify AFD within limits for each OPERABLE excore channel.	7 days

Table 3.3.1-1 (page 2 of 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Source Range Neutron Flux	$2^{(d)}$	2	H, I	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 1.0E6 cps
	$3^{(a)}$, $4^{(a)}$, $5^{(a)}$	2	I, J	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 1.0E6 cps
6. Overtemperature ΔT	1, 2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3 3.1.7 SR 3.3.1.12 SR 3.3.1.16	Refer to Note 1 (Page 3.3.1-23)
7. Overpower ΔT	1, 2	4	E	SR 3.3.1.1 ~~SR 3 3.1.3~~ ~~SR 3 3 1 6~~ SR 3.3.1.7 SR 3.3.1.12 SR 3.3.1.16	Refer to Note 2 (Page 3 3.1-24)
8 Pressurizer Pressure					
a. Low	$1^{(e)}$	4	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ $\boxed{1845}$ ~~1760~~ psig
b. High	1, 2	3	E	SR 3.3.1.1 SR 3 3.1.7 SR 3.3.1.10	≤ 2400 psig

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks
(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

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

<u>Note 1: Overtemperature ΔT</u>

The Overtemperature ΔT Function Allowable Value is defined by the following Trip Setpoint.

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

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, = * ~~567.3~~°F.

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

$K_1 \le$ ~~1.11~~*
$K_2 =$ ~~0.009~~*/°F
$K_3 =$ ~~0.000566~~*/psig
$\tau_1 =$ ~~30~~ * sec
$\tau_2 =$ ~~4~~ *sec

f*(ΔI) = ~~−0.0150~~*{~~12~~ * + (q_t - q_b)}   when q_t - q_b ≤ ~~−12~~*% RTP
~~0~~*% of RTP   when ~~−12~~*% RTP < q_t - q_b ≤ ~~9~~*% RTP
~~0.0250~~*{(q_t - q_b) - ~~9~~*}   when q_t - q_b > ~~9~~*% RTP

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

*** as specified in the COLR**

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

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value is defined by the following Trip Setpoint.

$$\Delta T \leq \Delta T_0 \left\{ K_4 - K_5 \frac{\tau_3 s T}{1+\tau_3 s} - K_6 (T - T') \cancel{- f(\Delta I)} \right\}$$

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, = ~~567.3~~*°F.

$K_4 \leq$ ~~1.10~~*

K_5 = ~~0.0275~~*/°F for increasing T_{avg}
= ~~0~~*/°F for decreasing T_{avg}

K_6 = ~~0.002~~*/°F when T > T′
= ~~0~~*/°F when T ≤ T′

τ_3 = ~~10~~ * sec

~~f(ΔI) = As defined in Note 1~~

* as specified in the COLR

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

TS 2.1.1, "Reactor Core SLs";
LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)";
LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor $(F^N_{\Delta H})$";
LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)";
LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation" Overtemperature ΔT and Overpower ΔT Parameter Values for Table 3.3.1-1;
LCO 3.4.1, "RCS Pressure, Temperature, and Flow - Departure from Nucleate Boiling (DNB) Limits"; and
LCO 3.9.1, "Boron Concentration".

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

1. NSPNAD-8101-PA, "Qualification of Reactor Physics Methods for Application to PI Units" (latest approved version);

2. NSPNAD-8102-PA, "Prairie Island Nuclear Power Plant Reload Safety Evaluation Methods for Application to PI Units" (latest approved version);

3. NSPNAD-97002-PA, "Northern States Power Company's "Steam Line Break Methodology", (latest approved version);

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

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

6. WCAP-10924-P-A, "Westinghouse Large Break LOCA Best-Estimate Methodology", December, 1988;

7. WCAP-10924-P-A, Volume 1, Addendum 4, "Westinghouse Large Break LOCA Best Estimate Methodology", August, 1990;

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

8. XN-NF-77-57 (A), XN-NF-77-57, Supplement 1 (A), "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II", May, 1981;

9. WCAP-13677, "10 CFR 50.46 Evaluation Model Report: W-COBRA/TRAC 2-Loop Upper Plenum Injection Model Update to Support ZIRLO$_{TM}$ Cladding Options", April 1993 (approved by NRC SE dated November 26, 1993);

10. NSPNAD-93003-A, "Transient Power Distribution Methodology", (latest approved version);

11. NAD-PI-003, "Prairie Island Nuclear Power Plant Required Shutdown Margin During Physics Tests," ~~(approved by NRC SE dated July 30, 2002)~~; ~~and~~

12. NAD-PI-004, "Prairie Island Nuclear Power Plant $F_Q^W(Z)$ Penalty With Increasing $\left[F_Q^C(Z)/K(Z)\right]$ Trend," ~~approved by NRC SE dated July 30, 2002).~~;

13. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control/ F_Q Surveillance Technical Specification";

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

15. WCAP-11397-P-A, "Revised Thermal Design Procedure"; and

16. WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report".

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

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

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

a. RCS pressure and temperature limits for heat-up, cooldown, low temperature operation, criticality, and hydrostatic testing, OPPS arming, PORV lift settings and Safety Injection Pump Disable Temperature as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.10, "Pressurizer Safety Valves";
LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) – Reactor Coolant System Cold Leg Temperature (RCSCLT) > Safety Injection (SI) Pump Disable Temperature";
LCO 3.4.13, "Low Temperature Overpressure Protection (LTOP) – Reactor Coolant System Cold Leg Temperature (RCSCLT) ≤ Safety Injection (SI) Pump Disable Temperature"; and
LCO 3.5.3, "ECCS - Shutdown".

5.6 Reporting Requirements

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

b. The analytical methods used to determine the RCS pressure and temperature limits and Cold Overpressure Mitigation System setpoints shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (includes any exemption granted by NRC to ASME Code Case N-514).

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto. Changes to the curves, setpoints, or parameters in the PTLR resulting from new or additional analysis of beltline material properties shall be submitted to the NRC prior to issuance of an updated PTLR.

5.6.7 Steam Generator Tube Inspection Report

1. Following each in-service inspection of steam generator tubes, if there are any tubes requiring plugging or sleeving, the number of tubes plugged or sleeved in each steam generator shall be reported to the Commission within 15 days.

2. The results of steam generator tube in-service inspections shall be included with the summary reports of ASME Code Section XI inspections submitted within 90 days of the end of each refueling outage. Results of steam generator tube in-service inspections not associated with a refueling outage shall be submitted within 90 days of the completion of the inspection. These reports shall include: (1) number and extent of tubes inspected, (2) location and percent of wall-thickness penetration for each indication of an imperfection, and (3) identification of tubes plugged or sleeved.

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report (continued)

3. Results of steam generator tube inspections which fall into Category C-3 require notification to the Commission prior to resumption of plant operation, and reporting as a special report to the Commission within 30 days. This special report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

4. The results of inspections performed under Specification 5.5.8.b for all tubes that have defects below the F* or EF* distance, and were not plugged, shall be reported to the Commission within 15 days following the inspection. The report shall include:

   a. Identification of F* and EF* tubes, and

   b. Location and extent of degradation.

5. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC staff prior to returning the steam generators to service should any of the following conditions arise:

   a. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steamline break) for the next operating cycle.

   b. If circumferential crack-like indications are detected at the tube support plate intersections.

   c. If indications are identified that extend beyond the confines of the tube support plate.

   d. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report (continued)

e. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1E-02, notify the NRC and provide an assessment of the safety significance of the occurrence.

5.6.8 EM Report

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

BASES (continued)

BACKGROUND (continued)

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 steam generator safety valves 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 at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

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

The Reactor Trip System allowable values specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation", in combination with other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, flow, ΔI, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude DNB related flow instabilities.

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

~~a. High pressurizer pressure trip;~~

~~b. Low pressurizer pressure trip;~~

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

~~c. Overtemperature ΔT trip;~~

~~d. Overpower ΔT trip;~~

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

~~f. Steam generator safety valves.~~

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

The SLs represent a design requirement for establishing the RPS allowable values 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 (as indicated in the USAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The figure ~~curves~~ provided in the COLR shows ~~Figure 2.1. 1-1 show~~ the loci of points 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.

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

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

BASES (continued)

SAFETY LIMITS (continued)

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

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

~~The SL curves in Figure 2.1.1-1 define the regions of acceptable operation with respect to average temperatures, power (measured in ΔT), and pressurizer pressure. Each of the curves in the Figure has three slopes. For the 2235 and 2385 psig curves, at lower power (lower ΔT) the vessel exit design temperature, 650°F, is limiting.~~

~~SAFETY LIMITS (continued)~~

~~For the lower pressure curves, at lower ΔT, vessel exit temperature T_{sat} is limiting, to ensure the ΔT measurement remains valid. At all pressures after the first knee, at higher ΔT, the minimum DNBR derived from the critical heat flux correlation is limiting. The change in slope near full power ΔT is due to more restrictive $F_{\Delta H}$ consideration in the DNBR limit at high power.~~

~~The curves are based on enthalpy hot channel factor limits provided in the CORE OPERATING LIMITS REPORT (COLR).~~
~~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.~~

BASES (continued)

~~The SL is higher than the setpoint calculated when the AXIAL FLUX DIFFERENCE (AFD) is within the limits of the f(ΔI) 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 (Ref. 3).~~

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 steam generator safety valves and 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. Allowable values for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs. 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. AEC "General Design Criteria for Nuclear Power Plant Construction Permits", Criterion 6, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

2. USAR, Section 14.3.

~~3. WCAP-13123, December 1991.~~

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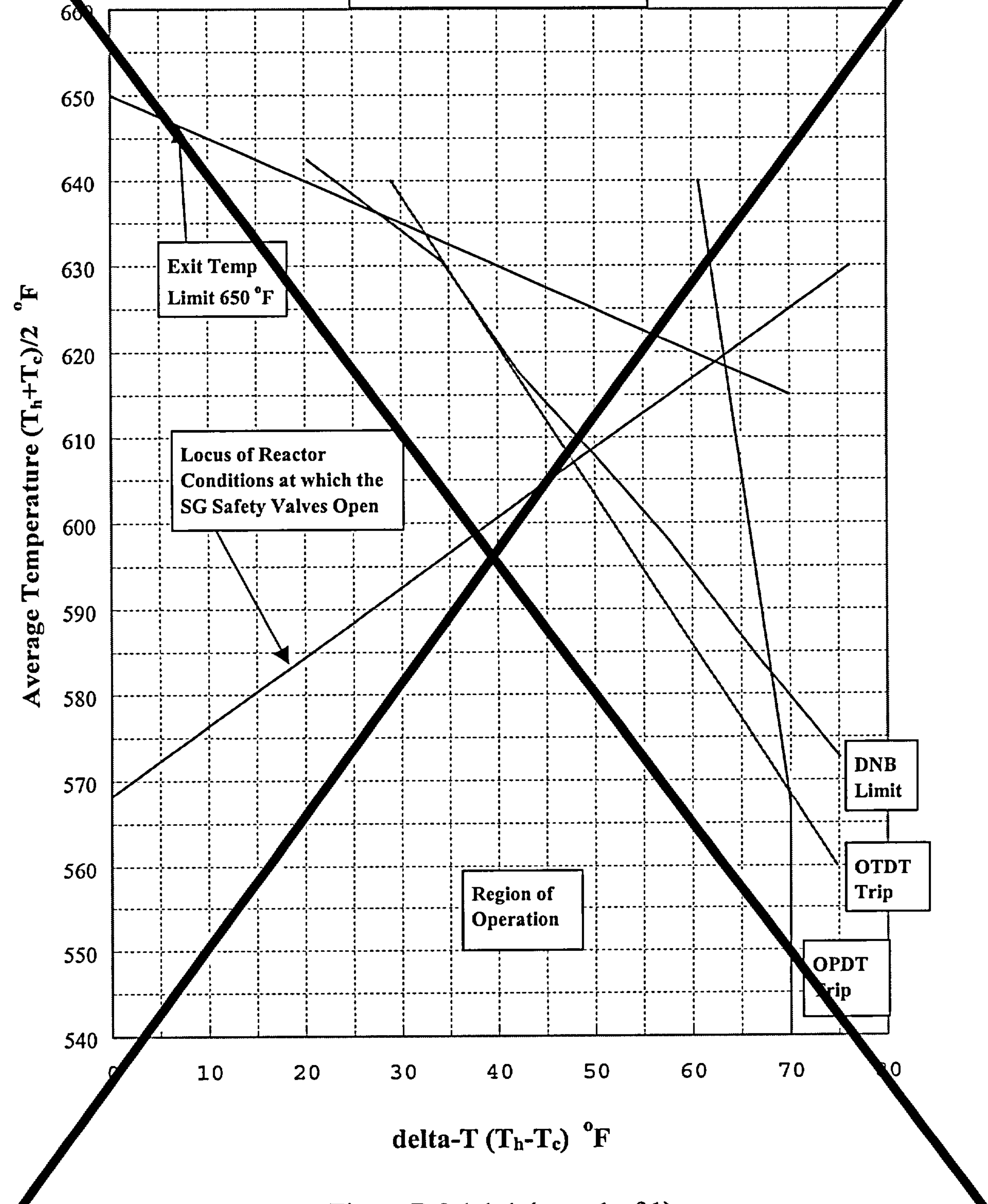

Figure B 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits vs. Boundary of Protection

BASES

BACKGROUND (continued)

To account for these possible variations, the equilibrium value of $F_Q(Z)$ is adjusted as $F_Q^W(Z)$ by an elevation dependent factor that accounts for the calculated worst case transient conditions. Core monitoring and control under non-equilibrium conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

APPLICABLE SAFETY ANALYSES

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

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

b. During transient conditions arising from events of moderate frequency (Condition II events), there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition (Ref. 1);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 200 ~~280~~ cal/gm (Ref. 1); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 2).

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

The Large Break LOCA (LBLOCA) analysis is the analysis that determines the LCO limit for $F_Q(Z)$. The $F_Q(Z)$ assumed in the Safety Analysis for other postulated accidents is either equal to or greater than that assumed in the LBLOCA analysis. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The Heat Flux Hot Channel Factor, $F_Q(Z)$, shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{CFQ}{P} \; K(Z) \quad \text{for } P > 0.5$$

$$F_Q(Z) \leq \frac{CFQ}{0.5} \; K(Z) \quad \text{for } P \leq 0.5$$

where: CFQ is the $F_Q(Z)$ limit at RTP provided in the COLR,

K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR, ~~and is based on the Small Break LOCA analysis,~~ and

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

For Relaxed ~~Constant~~ Axial Offset Control operation, $F_Q(Z)$ is approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$. Thus, both $F_Q^C(Z)$ and $F_Q^W(Z)$ must meet the preceding limits on $F_Q(Z)$.

BASES

LCO (continued)

An $F_Q^C(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results a measured value ($F_Q^M(Z)$) of $F_Q(Z)$ is obtained. Then,

$$F_Q^C(Z) = F_Q^M(Z)*(1.0815)$$

where 1.0815 is a factor that accounts for fuel manufacturing tolerances (1.03) multiplied by a factor associated with the flux map measurement uncertainty (1.05) (Ref. 3).

$F_Q^C(Z)$ is an excellent approximation for $F_Q(Z)$ when the reactor is at the steady state power at which the incore flux map was taken.

The expression for $F_Q^W(Z)$ is:

$$F_Q^W(Z) = F_Q^C(Z)\ W\cancel{V}(Z)$$

where W~~V~~(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. W~~V~~(Z) is included in the COLR. The $F_Q^W(Z)$ is calculated at equilibrium conditions.

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

This LCO precludes core power distributions that could violate the assumptions in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q^C(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required if $F_Q^W(Z)$ cannot be maintained within the LCO limits, reduction of the AFD limits is required. Note that sufficient reduction of the AFD limits will also result in a reduction of the core power.

BASES

LCO (continued)	Violating the LCO limits for $F_Q(Z)$ may result in unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.
APPLICABILITY	The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.
ACTIONS	<u>A.1</u> Reducing THERMAL POWER by $\geq$ 1% RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^C(Z)$ is $F_Q^M(Z)$ multiplied by factors accounting for manufacturing tolerances and measurement uncertainties. $F_Q^M(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time. The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require power reductions within 15 minutes of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable power level. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit.

BASES

ACTIONS (continued)

A.2

A reduction of the Power Range Neutron Flux-High trip setpoints by $\geq$ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Power Range Neutron Flux-High trip setpoints initially determined by Required Action A.2 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Power Range Neutron Flux-High trip setpoint reductions within 72 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux-High trip setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable Power Range Neutron Flux-High trip setpoints.

A.3

Reduction in the Overpower ΔT trip setpoints by $\geq$ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Overpower ΔT trip setpoints initially determined by Required Action A.3 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Overpower ΔT setpoint reductions within 72 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Overpower ΔT trip setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable Overpower ΔT trip setpoints.

BASES

ACTIONS

A.4

Verification that $F_Q^C(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, ensures that core conditions during operation at higher power levels, and future operations, are consistent with safety analyses assumptions.

Condition A is modified by a Note that requires Required Action A.4 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action A.1, even when Condition A is exited prior to performing Required Action A.4. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

B.1

If it is found that the maximum calculated value of $F_Q(Z)$ that can occur during normal maneuvers, $F_Q^W(Z)$, exceeds its specified limits, there exists a potential for $F_Q^C(Z)$ to become excessively high if a normal operational transient occurs. Reducing the AFD limits ~~THERMAL POWER~~ by $\geq$ 1% ~~RTP~~ for each 1% by which $F_Q^W(Z)$ exceeds its limit within the allowed Completion Time of 4 hours, maintains an acceptable absolute power density such that even if a transient occurred, core peaking factors are not exceeded (Ref. 4).

The percent that $F_Q(Z)$ exceeds its transient limit is calculated based on the following expression:

$$\left\{\left(\begin{matrix}\text{maximum}\\ \text{over } z\end{matrix}\left[\frac{F_Q^C(Z) * W(z)}{\frac{CFQ}{P} * K(z)}\right]\right) - 1\right\} * 100 \text{ for } P > 0.5$$

BASES

ACTIONS

B.1 (continued)

$$\left\{\left(\underset{\text{over z}}{\text{maximum}}\left[\frac{F_Q^C(Z) * W(z)}{\frac{CFQ}{0.5} * K(z)}\right]\right) - 1\right\} * 100 \text{ for } P \leq 0.5$$

The implicit assumption is that if W(Z) values were recalculated (consistent with the reduced AFD limits), then $F_Q^C(Z)$ times the recalculated W(Z) values would meet the $F_Q(Z)$ limit. Note that complying with this action (of reducing AFD limits) may also result in a power reduction. Hence the need for B.2, B.3 and B.4.

B.2

A reduction of the Power Range Neutron Flux-High trip setpoints by ≥ 1% for each 1% by which the maximum allowable power is reduced ~~$F_Q^W(Z)$ exceeds its limit~~, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER as a result of reducing AFD limits in accordance with Required Action B.1.

B.3

Reduction in the Overpower ΔT trip setpoints by ≥ 1% for each 1% by which the maximum allowable power is reduced ~~$F_Q^W(Z)$ exceeds its limit~~, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER as a result of reducing AFD limits in accordance with Required

BASES

Action B.1.

ACTIONS (continued)

B.4

Verification that $F_Q^W(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the limit imposed by Required Action B.1, ensures that core conditions during operation at higher power levels, and future operation, are consistent with safety analyses assumptions.

Condition B is modified by a Note that requires Required Action B.4 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action B.1, even when Condition B is exited prior to performing Required Action B.4. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

C.1

If Required Actions A.1 through A.4 or B.1 through B.4 are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 and SR 3.2.1.2 are modified by a Note. The Note applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until an equilibrium power

SURVEILLANCE REQUIREMENTS (continued)

level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F_Q^C(Z)$ and $F_Q^W(Z)$ are within their specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because $F_Q^C(Z)$ could not have previously been measured in this reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of $F_Q^C(Z)$ before exceeding 75% RTP. This ensures that some determination of $F_Q(Z)$ is made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of $F_Q^C(Z)$ and $F_Q^W(Z)$ following a power increase of more than 10%, ensures that they are verified as soon as RTP (or any other level for extended operation) is achieved. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of $F_Q^C(Z)$ and $F_Q^W(Z)$. The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which $F_Q(Z)$ was last measured.

SR 3.2.1.1

Verification that $F_Q^C(Z)$ is within its specified limits involves increasing $F_Q^M(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q^C(Z)$. Specifically, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F_Q^C(Z) = F_Q^M(Z)*(1.0815)$ (Ref. 4 ~~3~~). $F_Q^C(Z)$ is then compared to its specified limits. The limit with which $F_Q^C(Z)$ is

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

compared varies inversely with power above 50% RTP and directly with a function called K(Z) provided in the COLR.

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q^C(Z)$ limit is met during the power ascension following a refueling, including when RTP is achieved, because peaking factors generally decrease as power level is increased.

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^C(Z)$, another evaluation of this factor is required 12 hours after achieving equilibrium conditions at this higher power level (to ensure that $F_Q^C(Z)$ values are being reduced sufficiently with the power increase to stay within the LCO limits).

The Frequency of 31 effective full power days (EFPD) is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with the Technical Specifications (TS).

SR 3.2.1.2

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated during the nuclear design process by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called W~~V~~(Z). Multiplying the measured total peaking factor, $F_Q^C(Z)$, by W ~~V~~(Z)

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

gives the maximum $F_Q(Z)$ calculated to occur in normal operation, $F_Q^W(Z)$.

The limit with which $F_Q^W(Z)$ is compared varies inversely with power above 50% RTP and directly with the function K(Z) provided in the COLR.

The W~~V~~(Z) curve is provided in the COLR for discrete core elevations. Flux map data are taken for 61 core elevations. $F_Q^W(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height:

a. Lower core region, from 0 to 15~~10~~% inclusive; and

b. Upper core region, from 85~~90~~ to 100% inclusive.

The top and bottom 15~~10~~% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. If $F_Q^W(Z)$ is evaluated, an evaluation of the expression below is required to account for any increase to $F_Q^M(Z)$ that may occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation.

If the two most recent $F_Q(Z)$ evaluations show an increase in the expression

maximum over z $\left[\dfrac{F_Q^C(Z)}{K(Z)}\right]$

it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

by an appropriate factor specified in the COLR, or to evaluate $F_Q(Z)$ more frequently, each 7 EFPD. These alternative requirements prevent $F_Q(Z)$ from exceeding its limit for any significant period of time without detection.

During the power ascension following a refueling outage, startup physics testing program controls ensure that the $F_Q(Z)$ will not exceed the values assumed in the safety analysis. These controls include flux mapping, ramp rate restrictions, and restrictions on RCCA motion. They provide the necessary controls to precondition the fuel and ensure that the reactor power may be safely increased to equilibrium conditions at or near RTP, at which time $F_Q^W(Z)$ and AFD target band are determined. Performing the Surveillance within 12 hours after achieving equilibrium conditions after each refueling after THERMAL POWER exceeds 75% RTP, ensures that the $F_Q(Z)$ limit is met when the unit is released for normal operations.

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^W(Z)$, another evaluation of this factor is reuquired 12 hours after achieving equilibrium condition at this higher power level (to ensure that $F_Q^W(Z)$ values are being reduced sufficiently with the power increase to stay within the LCO limits).

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(Z)$ evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

BASES (continued)

REFERENCE

1. USAR, Section 14.

2. AEC "General Design Criteria for Nuclear Power Plant Construction Permits", Criterion 29, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

3. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL FLUX DIFFERENCE (AFD)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

~~The operating scheme used to control the axial power distribution, Constant Axial Offset Control (CAOC), involves maintaining the AFD within a tolerance band around a burnup dependent target, known as the target flux difference, to minimize the variation of the axial peaking factor and axial xenon distribution during unit maneuvers.~~

~~The target flux difference is determined at equilibrium xenon conditions in conjunction with verifying $F_Q^W(Z)$ in accordance with SR 3.2.1.2. The control banks must be positioned within the core in accordance with their insertion limits and Control Bank D should be inserted near its normal position (i.e., ≥ 190 steps withdrawn) for steady state operation at high power levels. The power level should be as near RTP as practical. The value of the target flux difference obtained under these conditions divided by the Fraction of RTP is the target flux difference at RTP for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RTP value by the appropriate fractional THERMAL POWER level.~~

Relaxed Axial Offset Control (RAOC) is a calculational procedure that defines the allowed operational space of the AFD versus THERMAL POWER. The AFD limits are selected by considering a

BASES

BACKGROUND (continued)

range of axial xenon distributions that may occur as a result of large variations of the AFD.

Subsequently, power peaking factors and power distributions are examined to ensure that the loss of coolant accident (LOCA), loss of flow accident, and anticipated transient limits are met. Violation of the AFD limits invalidate the conclusions of the accident and transient analyses with regard to fuel cladding integrity.

The AFD is monitored on an automatic basis using the unit process computer, which has an AFD monitor alarm. ~~The AFD is logged manually or monitored on an automatic basis using the unit process computer that has an AFD monitor alarm. The frequency of monitoring the AFD by the unit computer is once per minute providing an essentially continuous accumulation of penalty deviation time that allows the operator to accurately assess the status of the penalty deviation time.~~ The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFDs for two or more OPERABLE excore channels **is** ~~are~~ outside **its specified limits.** ~~the target band and the THERMAL POWER is ≥ 90% RTP. During operation at THERMAL POWER levels < 90% RTP but > 15% RTP, the computer sends an alarm message when the cumulative penalty deviation time is > 1 hour in the previous 24 hours.~~

BASES

BACKGROUND (continued)

~~Periodic updating of the target flux difference value is necessary to follow the change of the flux difference at steady state conditions with burnup.~~

~~The Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) and QPTR LCOs limit the radial component of the peaking factors.~~

APPLICABLE SAFETY ANALYSES

The AFD is a measure of axial power distribution skewing to either the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution and, to a lesser extent, reactor coolant temperature and boron concentrations.

The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

The RAOC methodology (Ref. 2) establishes a xenon distribution library with tentatively wide AFD limits. One dimensional axial power distribution calculations are then performed to demonstrate that normal operation power shapes are acceptable for the LOCA and loss of flow accident, and for initial conditions of anticipated transients. The tentative limits are adjusted as necessary to meet the safety analysis requirements.

~~The CAOC and Transient Power Distribution methodologies (Refs. 1 and 2) entail:~~

~~a. Establishing an envelope of allowed power shapes and power densities;~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~b. Devising an operating strategy for the cycle that maximizes unit flexibility (maneuvering) and minimizes axial power shape changes;~~

~~c. Demonstrating that this strategy does not result in core conditions that violate the envelope of permissible core power characteristics; and~~

~~d. Demonstrating that this power distribution control scheme can be effectively supervised with excore detectors.~~

~~The limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes.~~

~~The Transient Power Distribution methodology (Ref. 2) determines a function, (V(Z)), that when applied to equilibrium $F_Q^c(Z)$ values will bound $F_Q^c(Z)$ values that could be measured at non-equilibrium conditions. This remains valid provided that the AFD is maintained within the target flux band around a target flux difference that was determined in conjunction with determining the equilibrium $F_Q^w(Z)$.~~

The limits on the AFD also limit the range of power distributions that are assumed as initial conditions in analyzing Condition II, III, and IV events. This ensures that fuel cladding integrity is maintained for these postulated accidents. The most important Condition IV event is the loss of coolant accident. The most significant Condition III event is the loss of RCS flow accident. The most significant Condition II events are uncontrolled bank withdrawal at power and Rod Cluster Control Assembly (RCCA) misalignment.

The limits on the AFD satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator~~,~~ through ~~either~~ the manual operation of the control banks~~,~~ or automatic motion of control banks. **The automatic motion of the control banks is in response** ~~responding~~ to temperature deviations resulting from ~~either~~ manual operation of the Chemical and Volume Control System to change boron concentration~~,~~ or from power level changes.

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors. Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detector in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labaeled as Δ flux or % ΔI.

~~The required target band varies with axial burnup distribution, which in turn varies with the core average accumulated burnup. The target band defined in the COLR may provide one target band for the entire cycle or more than one band, each to be followed for a specific range of cycle burnup and target flux difference.~~

~~With THERMAL POWER ≥ 90% RTP, the AFD must be kept within the target band. With the AFD outside the target band with THERMAL POWER ≥ 90% RTP, the assumptions of the accident analyses may be violated.~~

Violating the LCO on the AFD could produce unacceptable consequences if a Condition II, III, or IV event occurs while the AFD is outside its **specified** limits.

~~The LCO is modified by four Notes. Note 1 states the conditions necessary for declaring the AFD outside of the target band.~~

BASES

LCO (continued)

~~Notes 2 and 3 describe how the cumulative penalty deviation time is calculated. It is intended that the unit is operated with the AFD within the target band about the target flux difference. However, during rapid THERMAL POWER reductions, control bank motion may cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation does not affect the xenon distribution sufficiently to change the envelope of peaking factors that may be reached on a subsequent return to RTP with the AFD within the target band, provided the time duration of the deviation is limited. Accordingly, while THERMAL POWER is ≥ 50% RTP and < 90% RTP (i.e., Part b of this LCO), a 1 hour cumulative penalty deviation time limit, cumulative during the preceding 24 hours, is allowed during which the unit may be operated outside of the target band but within the acceptable operation limits provided in the COLR (Note 2). This penalty time is accumulated at the rate of 1 minute for each 1 minute of operating time when THERMAL POWER ≥ 50% RTP. The cumulative penalty time is the sum of penalty times from LCO Notes 2 and 3.~~

~~For THERMAL POWER levels > 15% RTP and < 50% RTP (i.e., Part c of this LCO), deviations of the AFD outside of the target band are less significant. Note 3 allows the accumulation of 1/2 minute penalty deviation time per 1 minute of actual time outside the target band and reflects this reduced significance. With THERMAL POWER < 15% RTP, AFD is not a significant parameter in the assumptions used in the safety analysis and, therefore, requires no limits. Because the xenon distribution produced at THERMAL POWER levels less than RTP does affect the power distribution as power is increased, unanalyzed xenon and power distribution is prevented by limiting the accumulated penalty deviation time.~~

~~For surveillance of the power range channels performed according to SR 3.3.1.6, Note 4 allows deviation outside the target band for 16 hours and no penalty deviation time accumulated. Some~~

BASES

LCO (continued)

~~deviation in the AFD is required for doing the NIS calibration with the incore detector system. This calibration is performed every 92 days.~~

APPLICABILITY

The AFD requirements are applicable in MODE 1 greater than or equal to 50% RTP when the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis.

For AFD limits developed using RAOC methodology, the value of the AFD does not affect the limiting accident consequences with THERMAL POWER < 50% RTP and for lower operating power MODES.

~~AFD requirements are applicable in MODE 1 above 15% RTP. Above 50% RTP, the combination of THERMAL POWER and core peaking factors are the core parameters of primary importance in safety analyses (Ref. 3).~~

~~Between 15% RTP and 90% RTP, this LCO is applicable to ensure that the distributions of xenon are consistent with safety analysis assumptions.~~

~~At or below 15% RTP and for lower operating MODES, the stored energy in the fuel and the energy being transferred to the reactor coolant are low. The value of the AFD in these conditions does not affect the consequences of the design basis events.~~

~~Low signal levels in the excore channels may preclude obtaining valid AFD signals below 15% RTP.~~

ACTIONS

A.1

As an alternative to restoring the AFD to within its specified limits.

BASES

ACTIONS (continued)

A.1

Required Action A.1 requires a THERMAL POWER reduction to < 50% RTP. This places the core in a condition for which the value of the AFD is not important in the applicable safety analyses. A Completion Time of 30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

~~With the AFD outside the target band and THERMAL POWER ≥ 90% RTP, the assumptions used in the accident analyses may be violated with respect to the maximum heat generation. Therefore, a Completion Time of 15 minutes is allowed to restore the AFD to within the target band because xenon distributions change little in this relatively short time.~~

~~B.1~~

~~If the AFD cannot be restored within the target band, then reducing THERMAL POWER to < 90% RTP places the core in a condition that has been analyzed and found to be acceptable, provided that the AFD is within the acceptable operation limits provided in the COLR.~~

~~The allowed Completion Time of 15 minutes provides an acceptable time to reduce power to < 90% RTP without allowing the plant to remain in an unanalyzed condition for an extended period of time.~~

~~C.1~~

~~With THERMAL POWER < 90% RTP but ≥ 50% RTP, operation with the AFD outside the target band is allowed for up to 1 hour if the AFD is within the acceptable operation limits provided in the COLR. With the AFD within these limits, the resulting axial power distribution is acceptable as an initial condition for accident analyses assuming the then existing xenon distributions. The 1 hour cumulative penalty deviation time restricts the extent of xenon redistribution. Without this limitation, unanalyzed xenon axial distributions may result from a different pattern of xenon buildup~~

BASES

ACTIONS

~~C.1 (continued)~~

~~and decay. The reduction to a power level < 50% RTP puts the reactor at a THERMAL POWER level at which the AFD is not a significant accident analysis parameter.~~

~~If the indicated AFD is outside the target band and outside the acceptable operation limits provided in the COLR, the peaking factors assumed in accident analysis may be exceeded with the existing xenon condition. (Any AFD within the target band is acceptable regardless of its relationship to the acceptable operation limits.) The Completion Time of 30 minutes allows for a prompt, yet orderly, reduction in power.~~

~~Condition C is modified by a Note that requires that Required Action C.1 must be completed whenever this Condition is entered.~~

~~D.1~~

~~If Required Action C.1 is not completed within its required Completion Time of 30 minutes, the axial xenon distribution starts to become significantly skewed with the THERMAL POWER ≥ 50% RTP. In this situation, the assumption that a cumulative penalty deviation time of 1 hour or less during the previous 24 hours while the AFD is outside its target band is acceptable at < 50% RTP, is no longer valid.~~

~~Reducing the power level to <15% RTP within the Completion Time of 9 hours and complying with LCO penalty deviation time requirements for subsequent increases in THERMAL POWER ensure that acceptable xenon conditions are restored.~~

~~This Required Action must also be implemented either if the cumulative penalty deviation time is > 1 hour during the previous 24 hours, or the AFD is not within the target band and not within the acceptable operation limits.~~

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits. The Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.

~~This Surveillance verifies that the AFD as indicated by the NIS excore channels is within the target band. The Surveillance Frequency of 7 days is adequate because the AFD is controlled by the operator and monitored by the process computer. Furthermore, any deviations of the AFD from the target band that is not alarmed should be readily noticed.~~

~~The AFD should be monitored and logged more frequently in periods of operation for which the power level or control bank positions are changing to allow corrective measures when the AFD is more likely to move outside the target band.~~

~~SR 3.2.3.2~~

~~This Surveillance requires that the target flux difference be determined and updated at a Frequency of 31 effective full power days (EFPD) to account for small changes that may occur in the target flux differences in that period due to burnup.~~

~~The target flux difference is determined by averaging the indicated AFD from all OPERABLE excore channels.~~

~~To ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during non-equilibrium state conditions, the Transient Power Distribution methodology, i.e. V(Z), (Ref. 2) requires SR 3.2.1.2 to be performed in conjunction with this SR.~~

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.2.3.2 (continued)~~

~~Following a refueling outage, SR 3.2.1.2, and thus SR 3.2.3.2, are not required to be performed until equilibrium conditions are achieved. Since it may be desirable to provide the operators with some guidance for AFD control during the power ascension, a target flux difference may be posted based on design predictions.~~

~~A Note modifies this SR to allow the predicted end of cycle AFD from the cycle nuclear design to be used to determine the initial target flux difference after each refueling.~~

REFERENCES

1. **WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.** ~~XN-NF-77-57, supplement 1(A), "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II", May, 1981.~~

2. **WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control/ F_Q Surveillance Technical Specification," February 1994.**
~~Transient Power Distribution, NSPNAD-93003-A.~~

3. **USAR, Chapter 7.** ~~WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.~~

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

7. Overpower ΔT

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions. This trip Function also provides a backup to the Power Range Neutron Flux-High Setpoint trip. The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the ΔT of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature – the trip setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;
- rate of change of reactor coolant average temperature – including dynamic compensation for the delays between the core and the temperature measurement system; and
- ~~axial power distribution – f(ΔI), the trip setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the trip setpoint is reduced in accordance with Note 2 of Table 3.3.1-1.~~

The Overpower ΔT trip Function is calculated for each channel as per Note 2 of Table 3.3.1-1. A trip occurs if Overpower ΔT is indicated in two channels. Since the temperature signals are used for other control functions, the actuation logic must be

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued)

The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate the change in the absolute difference between NIS and heat balance calculated powers rarely exceeds 2% in any 24 hour period.

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output every 31 Effective Full Power Days (EFPD). If the absolute difference is ≥ 2%, the NIS channel is still OPERABLE, but must be readjusted.

If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the overtemperature ~~and overpower~~ ΔT Function~~s~~.

Two Notes modify SR 3.3.1.3. Note 1 indicates that the excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is ≥ 2%.

Note 2 clarifies that the Surveillance is required only if reactor power is ≥ 15% RTP and that 72 hours is allowed for performing the first Surveillance after reaching 15% RTP.

The Frequency of every 31 EFPD is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Also, the slow changes in neutron flux during the fuel cycle can be detected during this interval.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.6

SR 3.3.1.6 is a calibration of the excore channels to the incore channels. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the overtemperature ~~and overpower~~ ΔT Functions.

A Note modifies SR 3.3.1.6. The Note states that this Surveillance is required only if reactor power is > 75% RTP and that 24 hours is allowed for performing the first surveillance after reaching 75% RTP.

The Frequency of 92 EFPD is adequate. It is based on industry operating experience, considering instrument reliability and operating history data for instrument drift.

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT every 92 days. A COT is performed on each required channel to ensure the entire channel will perform the intended Function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL OPERATIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specification tests at least once per refueling interval with applicable extensions. Setpoints must be within the Allowable Values specified in Table 3.3.1-1.

EXHIBIT C

PRAIRIE ISLAND NUCLEAR GENERATING STATION

License Amendment Request dated March 25, 2003

Revised Pages

Technical Specification Pages

2.0-1	5.0-35
3.2.1-2	5.0-36
3.2.1-3	5.0-37
3.2.3-1	5.0-38
3.3.1-18	5.0-39
3.3.1-23	5.0-40
3.3.1-24	

Bases Pages

B 2.1.1-2	B 3.2.1-11
B 2.1.1-3	B 3.2.1-12
B 2.1.1-4	B 3.2.1-13
B 2.1.1-5	B 3.2.1-14
B 3.2.1-2	B 3.2.1-15
B 3.2.1-3	B 3.2.1-16
B 3.2.1-4	B 3.2.3-1
B 3.2.1-5	B 3.2.3-2
B 3.2.1-6	B 3.2.3-3
B 3.2.1-7	B 3.2.3-4
B 3.2.1-8	B 3.3.1-20
B 3.2.1-9	B 3.3.1-56
B 3.2.1-10	B 3.3.1-58

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 limits specified in the COLR and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained $\geq$ 1.17 for WRB-1 DNB correlation for OFA fuel.

2.1.1.2 The peak fuel centerline temperature shall be maintained $\leq$ 4700°F.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained $\leq$ 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.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.4 Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
B. -----------NOTE----------- Required Action B.4 shall be completed whenever this Condition is entered. ------------------------------ $F_Q^W(Z)$ not within limits.	B.1 Reduce AFD limits ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit. AND B.2 Reduce Power Range Neutron Flux-High trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limit is reduced. AND B.3 Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limit is reduced. AND	4 hours after each $F_Q^W(Z)$ determination 72 hours after each $F_Q^W(Z)$ determination 72 hours after each $F_Q^W(Z)$ determination

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.4 Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the maximum allowable power of the AFD limits
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 2.	6 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

LCO 3.2.3 The AFD in % flux difference units shall be maintained within the limits specified in the COLR.

-------------------------------------NOTE-----------------------------------
The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD within limits for each OPERABLE excore channel.	7 days

Table 3.3.1-1 (page 2 of 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Source Range Neutron Flux	2(d)	2	H, I	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 1.0E6 cps
	3(a), 4(a), 5(a)	2	I, J	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 1.0E6 cps
6 Overtemperature ΔT	1, 2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.12 SR 3.3.1.16	Refer to Note 1 (Page 3.3.1-23)
7. Overpower ΔT	1, 2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.12 SR 3.3.1.16	Refer to Note 2 (Page 3.3.1-24)
8. Pressurizer Pressure					
a Low	1(e)	4	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ 1845 psig
b. High	1, 2	3	E	SR 3.3.1.1 SR 3.3 1.7 SR 3.3 1.10	≤ 2400 psig

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.
(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

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

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value is defined by the following Trip Setpoint.

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

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, = *°F.

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

$K_1 \le$ *
K_2 = */°F
K_3 = */psig
τ_1 = * sec
τ_2 = * sec

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

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

* As specified in the COLR.

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

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value is defined by the following Trip Setpoint.

$$\Delta T \le \Delta T_0 \left\{ K_4 - K_5 \frac{\tau_3 s T}{1 + \tau_3 s} - K_6 (T - T') \right\}$$

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, = *°F.

$K_4 \le$ *

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

K_6 = */°F when $T > T'$
= */°F when $T \le T'$

τ_3 = * sec

* As specified in the COLR.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

TS 2.1.1, "Reactor Core SLs";
LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)";
LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor $\left(F_{\Delta H}^{N}\right)$";
LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)";
LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation" Overtemperature ΔT and Overpower ΔT Parameter Values for Table 3.3.1-1;
LCO 3.4.1, "RCS Pressure, Temperature, and Flow - Departure from Nucleate Boiling (DNB) Limits"; and
LCO 3.9.1, "Boron Concentration".

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

1. NSPNAD-8101-PA, "Qualification of Reactor Physics Methods for Application to PI Units" (latest approved version);

2. NSPNAD-8102-PA, "Prairie Island Nuclear Power Plant Reload Safety Evaluation Methods for Application to PI Units" (latest approved version);

3. NSPNAD-97002-PA, "Northern States Power Company's "Steam Line Break Methodology", (latest approved version);

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

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

6. WCAP-10924-P-A, "Westinghouse Large Break LOCA Best-Estimate Methodology", December, 1988;

7. WCAP-10924-P-A, Volume 1, Addendum 4, "Westinghouse Large Break LOCA Best Estimate Methodology", August, 1990;

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

8. XN-NF-77-57 (A), XN-NF-77-57, Supplement 1 (A), "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II", May, 1981;

9. WCAP-13677, "10 CFR 50.46 Evaluation Model Report: W-COBRA/TRAC 2-Loop Upper Plenum Injection Model Update to Support $ZIRLO_{TM}$ Cladding Options", April 1993 (approved by NRC SE dated November 26, 1993);

10. NSPNAD-93003-A, "Transient Power Distribution Methodology", (latest approved version);

11. NAD-PI-003, "Prairie Island Nuclear Power Plant Required Shutdown Margin During Physics Tests";

12. NAD-PI-004, "Prairie Island Nuclear Power Plant $F_Q^W(Z)$ Penalty With Increasing $\left[F_Q^C(Z) / K(Z)\right]$ Trend";

13. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control/ F_Q Surveillance Technical Specification";

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

15. WCAP-11397-P-A, "Revised Thermal Design Procedure"; and

16. WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report".

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

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

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

a. RCS pressure and temperature limits for heat-up, cooldown, low temperature operation, criticality, and hydrostatic testing, OPPS arming, PORV lift settings and Safety Injection Pump Disable Temperature as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.10, "Pressurizer Safety Valves";
LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) – Reactor Coolant System Cold Leg Temperature (RCSCLT) > Safety Injection (SI) Pump Disable Temperature";
LCO 3.4.13, "Low Temperature Overpressure Protection (LTOP) – Reactor Coolant System Cold Leg Temperature (RCSCLT) ≤ Safety Injection (SI) Pump Disable Temperature"; and
LCO 3.5.3, "ECCS - Shutdown".

5.6 Reporting Requirements

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

b. The analytical methods used to determine the RCS pressure and temperature limits and Cold Overpressure Mitigation System setpoints shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (includes any exemption granted by NRC to ASME Code Case N-514).

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto. Changes to the curves, setpoints, or parameters in the PTLR resulting from new or additional analysis of beltline material properties shall be submitted to the NRC prior to issuance of an updated PTLR.

5.6.7 Steam Generator Tube Inspection Report

1. Following each in-service inspection of steam generator tubes, if there are any tubes requiring plugging or sleeving, the number of tubes plugged or sleeved in each steam generator shall be reported to the Commission within 15 days.

2. The results of steam generator tube in-service inspections shall be included with the summary reports of ASME Code Section XI inspections submitted within 90 days of the end of each refueling outage. Results of steam generator tube in-service inspections not associated with a refueling outage shall be submitted within 90 days of the completion of the inspection. These reports shall include: (1) number and extent of tubes inspected, (2) location and percent of wall-thickness penetration for each indication of an imperfection, and (3) identification of tubes plugged or sleeved.

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report (continued)

3. Results of steam generator tube inspections which fall into Category C-3 require notification to the Commission prior to resumption of plant operation, and reporting as a special report to the Commission within 30 days. This special report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

4. The results of inspections performed under Specification 5.5.8.b for all tubes that have defects below the F* or EF* distance, and were not plugged, shall be reported to the Commission within 15 days following the inspection. The report shall include:

   a. Identification of F* and EF* tubes, and

   b. Location and extent of degradation.

5. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC staff prior to returning the steam generators to service should any of the following conditions arise:

   a. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steamline break) for the next operating cycle.

   b. If circumferential crack-like indications are detected at the tube support plate intersections.

   c. If indications are identified that extend beyond the confines of the tube support plate.

   d. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report (continued)

e. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1E-02, notify the NRC and provide an assessment of the safety significance of the occurrence.

5.6.8 EM Report

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

BASES

BACKGROUND (continued)

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 steam generator safety valves 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 at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

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

The Reactor Trip System allowable values specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation", in combination with other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, flow, ΔI, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude DNB related flow instabilities.

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SLs represent a design requirement for establishing the RPS allowable values 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 (as indicated in the USAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The figure provided in the COLR shows the loci of points 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.

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

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

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is,

BASES

SAFETY LIMITS (continued)	it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.
APPLICABILITY	SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves and 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. Allowable values for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.
SAFETY LIMIT VIOLATIONS	The following SL violation responses are applicable to the reactor core SLs. If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable. The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

BASES (continued)

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits", Criterion 6, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

2. USAR, Section 14.3

BASES

BACKGROUND (continued)	To account for these possible variations, the equilibrium value of $F_Q(Z)$ is adjusted as $F_Q^W(Z)$ by an elevation dependent factor that accounts for the calculated worst case transient conditions. Core monitoring and control under non-equilibrium conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

APPLICABLE SAFETY ANALYSES

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

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

b. During transient conditions arising from events of moderate frequency (Condition II events), there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition (Ref. 1);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 200 cal/gm (Ref. 1); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 2).

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

BASES

APPLICABLE SAFETY ANALYSES (continued)	The Large Break LOCA (LBLOCA) analysis is the analysis that determines the LCO limit for $F_Q(Z)$. The $F_Q(Z)$ assumed in the Safety Analysis for other postulated accidents is either equal to or greater than that assumed in the LBLOCA analysis. Therefore, this LCO provides conservative limits for other postulated accidents. $F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The Heat Flux Hot Channel Factor, $F_Q(Z)$, shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{CFQ}{P} \ K(Z) \quad \text{for } P > 0.5$$

$$F_Q(Z) \leq \frac{CFQ}{0.5} \ K(Z) \quad \text{for } P \leq 0.5$$

where: CFQ is the $F_Q(Z)$ limit at RTP provided in the COLR,

K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR, and

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

For Relaxed Axial Offset Control operation, $F_Q(Z)$ is approximated by $F_Q^c(Z)$ and $F_Q^w(Z)$. Thus both $F_Q^c(Z)$ and $F_Q^w(Z)$ must meet the preceding limits on $F_Q(Z)$.

BASES

LCO (continued)

An $F_Q^C(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results a measured value ($F_Q^M(Z)$) of $F_Q(Z)$ is obtained. Then,

$$F_Q^C(Z) = F_Q^M(Z)*(1.0815)$$

where 1.0815 is a factor that accounts for fuel manufacturing tolerances (1.03) multiplied by a factor associated with the flux map measurement uncertainty (1.05) (Ref. 3).

$F_Q^C(Z)$ is an excellent approximation for $F_Q(Z)$ when the reactor is at the steady state power at which the incore flux map was taken.

The expression for $F_Q^W(Z)$ is:

$$F_Q^W(Z) = F_Q^C(Z)\ W(Z)$$

where W(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR. The $F_Q^W(Z)$ is calculated at equilibrium conditions.

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

BASES

LCO (continued)

This LCO precludes core power distributions that could violate the assumptions in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q^C(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required and if $F_Q^W(Z)$ cannot be maintained within the LCO limits, reduction of the AFD limits is required. Note that sufficient reduction of the AFD limits will also result in a reduction of the core power.

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

APPLICABILITY

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

ACTIONS

A.1

Reducing THERMAL POWER by $\geq$ 1% RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^C(Z)$ is $F_Q^M(Z)$ multiplied by factors accounting for manufacturing tolerances and measurement uncertainties. $F_Q^M(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time. The maximum allowable power level initially determined by Required Action A.1 may be

BASES

ACTIONS

A.1 (continued)

affected by subsequent determinations of $F^c_Q(Z)$ and would require power reductions within 15 minutes of the $F^c_Q(Z)$ determination, if necessary to comply with the decreased maximum allowable power level. Decreases in $F^c_Q(Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit.

A.2

A reduction of the Power Range Neutron Flux-High trip setpoints by $\geq$ 1% for each 1% by which $F^c_Q(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Power Range Neutron Flux-High trip setpoints initially determined by Required Action A.2 may be affected by subsequent determinations of $F^c_Q(Z)$ and would require Power Range Neutron Flux-High trip setpoint reductions within 72 hours of the $F^c_Q(Z)$ determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux-High trip setpoints. Decreases in $F^c_Q(Z)$ would allow increasing the maximum allowable Power Range Neutron Flux-High trip setpoints.

BASES

ACTIONS (continued)

A.3

Reduction in the Overpower ΔT trip setpoints by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Overpower ΔT trip setpoints initially determined by Required Action A.3 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Overpower ΔT setpoint reductions within 72 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Overpower ΔT trip setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable Overpower ΔT trip setpoints.

A.4

Verification that $F_Q^C(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, ensures that core conditions during operation at higher power levels, and future operations, are consistent with safety analyses assumptions.

BASES

ACTIONS

A.4 (continued)

Condition A is modified by a Note that requires Required Action A.4 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action A.1, even when Condition A is exited prior to performing Required Action A.4. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

B.1

If it is found that the maximum calculated value of $F_Q(Z)$ that can occur during normal maneuvers, $F_Q^W(Z)$, exceeds its specified limits, there exists a potential for $F_Q^C(Z)$ to become excessively high if a normal operational transient occurs. Reducing the AFD limits by $\geq$ 1% for each 1% by which $F_Q^W(Z)$ exceeds its limit within the allowed Completion Time of 4 hours, maintains an acceptable absolute power density such that even if a transient occurred, core peaking factors are not exceeded (Ref. 4).

The percent that $F_Q(Z)$ exceeds its transient limit is calculated based on the following expression:

$$\left\{ \left(\begin{matrix} \text{maximum} \\ \text{over z} \end{matrix} \left[\frac{F_Q^C(Z) * W(z)}{\frac{CFQ}{P} * K(z)} \right] \right) - 1 \right\} * 100 \text{ for } P > 0.5$$

BASES

ACTIONS

B.1 (continued)

$$\left\{\left(\begin{matrix}\text{maximum}\\ \text{over z}\end{matrix}\left[\frac{F_Q^C(Z)*W(z)}{\frac{CFQ}{0.5}*K(z)}\right]\right)-1\right\}*100 \text{ for } P \leq 0.5$$

The implicit assumption is that if W(Z) values were recalculated (consistent with the reduced AFD limits), then $F_Q^C(Z)$ times the recalculated W(Z) values would meet the $F_Q(Z)$ limit. Note that complying with this action (of reducing AFD limits) may also result in a power reduction. Hence the need for B.2, B.3 and B.4.

B.2

A reduction of the Power Range Neutron Flux-High trip setpoints by $\geq$ 1% for each 1% by which the maximum allowable power is reduced, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER as a result of reducing AFD limits in accordance with Required Action B.1.

BASES

ACTIONS (continued)

B.3

Reduction in the Overpower ΔT trip setpoints by ≥ 1% for each 1% by which the maximum allowable power is reduced, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER as a result of reducing AFD limits in accordance with Required Action B.1.

B.4

Verification that $F_Q^W(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the limit imposed by Required Action B.1, ensures that core conditions during operation at higher power levels, and future operation, are consistent with safety analyses assumptions.

Condition B is modified by a Note that requires Required Action B.4 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action B.1, even when Condition B is exited prior to performing Required Action B.4. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

BASES

ACTIONS (continued)

C.1

If Required Actions A.1 through A.4 or B.1 through B.4 are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 and SR 3.2.1.2 are modified by a Note. The Note applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F_Q^C(Z)$ and $F_Q^W(Z)$ are within their specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because $F_Q^C(Z)$ could not have previously been measured in this reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of $F_Q^C(Z)$ before exceeding 75% RTP. This ensures that some determination of $F_Q(Z)$ is made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of $F_Q^C(Z)$ and $F_Q^W(Z)$ following a power increase of more than 10%, ensures that they are verified as soon as RTP (or any other level for extended operation) is achieved.

BASES

SURVEILLANCE REQUIREMENTS (continued)

In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of $F_Q^C(Z)$ and $F_Q^W(Z)$. The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which $F_Q(Z)$ was last measured.

SR 3.2.1.1

Verification that $F_Q^C(Z)$ is within its specified limits involves increasing $F_Q^M(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q^C(Z)$. Specifically, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F_Q^C(Z) = F_Q^M(Z)*(1.0815)$ (Ref. 4). $F_Q^C(Z)$ is then compared to its specified limits. The limit with which $F_Q^C(Z)$ is compared varies inversely with power above 50% RTP and directly with a function called K(Z) provided in the COLR.

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q^C(Z)$ limit is met during the power ascension following a refueling, including when RTP is achieved, because peaking factors generally decrease as power level is increased.

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^C(Z)$, another evaluation of this factor is required 12 hours after achieving equilibrium conditions at this higher power level (to ensure that $F_Q^C(Z)$ values are being reduced sufficiently with the power increase to stay within the LCO limits).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

The Frequency of 31 effective full power days (EFPD) is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with the Technical Specifications (TS).

SR 3.2.1.2

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated during the nuclear design process by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called W(Z). Multiplying the measured total peaking factor, $F_Q^C(Z)$, by W(Z) gives the maximum $F_Q(Z)$ calculated to occur in normal operation, $F_Q^W(Z)$.

The limit with which $F_Q^W(Z)$ is compared varies inversely with power above 50% RTP and directly with the function K(Z) provided in the COLR.

The W(Z) curve is provided in the COLR for discrete core elevations. Flux map data are taken for 61 core elevations. $F_Q^W(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height:

a. Lower core region, from 0 to 15% inclusive; and

b. Upper core region, from 85 to 100% inclusive.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. If $F_Q^W(Z)$ is evaluated, an evaluation of the expression below is required to account for any increase to $F_Q^M(Z)$ that may occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation.

If the two most recent $F_Q(Z)$ evaluations show an increase in the expression

maximum over z $\left[\frac{F_Q^C(Z)}{K(Z)}\right]$

it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by an appropriate factor specified in the COLR, or to evaluate $F_Q(Z)$ more frequently, each 7 EFPD. These alternative requirements prevent $F_Q(Z)$ from exceeding its limit for any significant period of time without detection.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

During the power ascension following a refueling outage, startup physics testing program controls ensure that the $F_Q(Z)$ will not exceed the values assumed in the safety analysis. These controls include flux mapping, ramp rate restrictions, and restrictions on RCCA motion. They provide the necessary controls to precondition the fuel and ensure that the reactor power may be safely increased to equilibrium conditions at or near RTP, at which time $F_Q^W(Z)$ and AFD target band are determined. Performing the Surveillance within 12 hours after achieving equilibrium conditions after each refueling after THERMAL POWER exceeds 75% RTP, ensures that the $F_Q(Z)$ limit is met when the unit is released for normal operations.

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^W(Z)$, another evaluation of this factor is required 12 hours after achieving equilibrium condition at this higher power level (to ensure that $F_Q^W(Z)$ values are being reduced sufficiently with the power increase to stay within the LCO limits).

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(Z)$ evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

BASES (continued)

REFERENCES

1. USAR, Section 14.

2. AEC "General Design Criteria for Nuclear Power Plant Construction Permits", Criterion 29, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

3. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL FLUX DIFFERENCE (AFD)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

Relaxed Axial Offset Control (RAOC) is a calculational procedure that defines the allowed operational space of the AFD versus THERMAL POWER. The AFD limits are selected by considering a range of axial xenon distributions that may occur as a result of large variations of the AFD. Subsequently, power peaking factors and power distributions are examined to ensure that the loss of coolant accident (LOCA), loss of flow accident, and anticipated transient limits are met. Violation of the AFD limits invalidate the conclusions of the accident and transient analyses with regard to fuel cladding integrity.

The AFD is monitored on an automatic basis using the unit process computer, which has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPEARABLE excore channels is outside its specified limits.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The AFD is a measure of axial power distribution skewing to either the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution and, to a lesser extent, reactor coolant temperature and boron concentrations.

The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

The RAOC methodology (Ref. 2) establishes a xenon distribution library with tentatively wide AFD limits. One dimensional axial power distribution calculations are then performed to demonstrate that normal operation power shapes are acceptable for the LOCA and loss of flow accident, and for initial conditions of anticipated transients. The tentative limits are adjusted as necessary to meet the safety analysis requirements.

The limits on the AFD also limit the range of power distributions that are assumed as initial conditions in analyzing Condition II, III, and IV events. This ensures that fuel cladding integrity is maintained for these postulated accidents. The most important Condition IV event is the loss of coolant accident. The most significant Condition III event is the loss of RCS flow accident. The most significant Condition II events are uncontrolled bank withdrawal at power and Rod Cluster Control Assembly (RCCA) misalignment.

The limits on the AFD satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator through the manual operation of the control banks or automatic motion of control banks. The automatic motion of the control banks is in response to temperature deviations resulting from manual operation of the Chemical and Volume Control System to change boron concentration or from power level changes.

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors. Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detector in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as Δ flux or % ΔI.

Violating this LCO on the AFD could produce unacceptable consequences if a Condition II, III, and IV event occurs while the AFD is outside its specified limits.

APPLICABILITY

The AFD requirements are applicable in MODE 1 greater than or equal to 50% RTP when the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis.

For AFD limits developed using RAOC methodology, the value of the AFD does not affect the limiting accident consequences with THERMAL POWER < 50% RTP and for lower operating power MODES.

BASES (continued)

ACTIONS

A.1

As an alternative to restoring the AFD to within its specified limits, Required Action A.1 requires a THERMAL POWER reduction to < 50% RTP. This places the core in a condition for which the value of the AFD is not important in the applicable safety analyses. A Completion Time of 30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits. The Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.

REFERENCES

1. WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.
2. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control/ F_Q Surveillance Technical Specification," February 1994.
3. USAR, Chapter 7.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

7. Overpower ΔT

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions. This trip Function also provides a backup to the Power Range Neutron Flux-High Setpoint trip. The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the ΔT of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature – the trip setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;

- rate of change of reactor coolant average temperature – including dynamic compensation for the delays between the core and the temperature measurement system; and

The Overpower ΔT trip Function is calculated for each channel as per Note 2 of Table 3.3.1-1. A trip occurs if Overpower ΔT is indicated in two channels. Since the temperature signals are used for other control functions, the actuation logic must be

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued)

The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate the change in the absolute difference between NIS and heat balance calculated powers rarely exceeds 2% in any 24 hour period.

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output every 31 Effective Full Power Days (EFPD). If the absolute difference is $\geq$ 2%, the NIS channel is still OPERABLE, but must be readjusted.

If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the overtemperature ΔT Function.

Two Notes modify SR 3.3.1.3. Note 1 indicates that the excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is $\geq$ 2%.

Note 2 clarifies that the Surveillance is required only if reactor power is $\geq$ 15% RTP and that 72 hours is allowed for performing the first Surveillance after reaching 15% RTP.

The Frequency of every 31 EFPD is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Also, the slow changes in neutron flux during the fuel cycle can be detected during this interval.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.6

SR 3.3.1.6 is a calibration of the excore channels to the incore channels. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the overtemperature ΔT Function.

A Note modifies SR 3.3.1.6. The Note states that this Surveillance is required only if reactor power is > 75% RTP and that 24 hours is allowed for performing the first surveillance after reaching 75% RTP.

The Frequency of 92 EFPD is adequate. It is based on industry operating experience, considering instrument reliability and operating history data for instrument drift.

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT every 92 days. A COT is performed on each required channel to ensure the entire channel will perform the intended Function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL OPERATIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specification tests at least once per refueling interval with applicable extensions. Setpoints must be within the Allowable Values specified in Table 3.3.1-1.

EXHIBIT D

PRAIRIE ISLAND NUCLEAR GENERATING STATION

License Amendment Request dated March 25, 2003

Instrument Uncertainties for Plant Operating Parameter Inputs to the Westinghouse Revised Thermal Design Procedure