TXU Power

Ref: 10CFR50.90

TXU Power
Comanche Peak Steam
Electric Station
P. O. Box 1002 (E01)
Glen Rose, TX 76043
Tel: 254 897 5209
Fax: 254 897 6652
mike.blevins@txu.com

Mike Blevins
Senior Vice President &
Chief Nuclear Officer

CPSES- 200700605
Log # TXX-07063
File # 00236

April 10, 2007

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NOS. 50-445 AND 50-446
LICENSE AMENDMENT REQUEST (LAR) 07-003
REVISION TO TECHNICAL SPECIFICATION 3.1, "REACTIVITY CONTROL SYSTEMS," 3.2, "POWER DISTRIBUTION LIMITS," 3.3, "INSTRUMENTATION," AND 5.6.5b, "CORE OPERATING LIMITS REPORT (COLR)."

REF: Letter logged TXX-07047, dated Feb, 22, 2007 from Mike Blevins to the NRC.

Dear Sir or Madam:

Pursuant to 10CFR50.90, TXU Generation Company LP (TXU Power) hereby requests an amendment to the CPSES Unit 1 Operating License (NPF-87) and CPSES Unit 2 Operating License (NPF-89) by incorporating the attached change into the CPSES Unit 1 and 2 Technical Specifications. This change request applies to both units.

The proposed change will revise Technical Specifications (TS) 3.1 entitled "Reactivity Control Systems," 3.2 entitled "Power Distribution Limits," 3.3 entitled "Instrumentation," and 5.6.5b entitled "Core Operating Limits Report (COLR)." The requested change proposes to incorporate standard Westinghouse-developed and NRC-approved analytical methods into the lists of methodologies used to establish the core operating limits. These NRC-approved methods include the use of an alternate axial offset control methodology and the use of updated methods for determining core power distribution. Thus, conforming changes to the core power distribution limits and core power distribution measurement methods are also proposed.

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Attachment 1 provides a detailed description of the proposed changes, a technical analysis of the proposed changes, TXU Power's determination that the proposed changes do not involve a significant hazard consideration, a regulatory analysis of the proposed changes and an environmental evaluation. Attachment 2 provides the affected Technical Specification (TS) pages marked-up to reflect the proposed changes. Attachment 3 provides proposed changes to the Technical Specification Bases for information only. These changes will be processed per CPSES site procedures. Attachment 4 provides retyped Technical Specification pages which incorporate the requested changes. Attachment 5 provides retyped Technical Specification Bases pages which incorporate the proposed changes.

Per the referenced letter, TXU Power committed to provide the NRC with a license amendment request to allow use of the Westinghouse NOTRUMP-based small break LOCA methodology to establish future Comanche Peak core operating limits by April 30, 2007 (Commitment Number 27436). TXU Power will continue to use the currently approved COLR methodologies to support the remaining Unit 1, Cycle 13 operation and complete the transition to the proposed methodologies prior to Unit 1, Cycle 14 operation (Fall 2008). However, TXU Power currently plans to use these proposed Core Operating Limit Report (COLR) methodologies to support Unit 2, Cycle 11 operation (Spring of 2008). Therefore in order to comply with the commitment identified above, TXU Power requests approval of the proposed License Amendment by February 15, 2008, to be implemented within 120 days of the issuance of the license amendment.

In accordance with 10CFR50.91(b), TXU Power is providing the State of Texas with a copy of this proposed amendment.

This communication contains no new licensing basis commitments regarding CPSES Units 1 and 2.

Should you have any questions, please contact Mr. J. D. Seawright at (254) 897-0140.

I state under penalty of perjury that the foregoing is true and correct.

Executed on April 10, 2007.

Sincerely,

TXU Generation Company LP

By: TXU Generation Management Company LLC
Its General Partner

Mike Blevins

By: Fred W. Madden
Director, Oversight and Regulatory Affairs

Attachments
1. Description and Assessment
2. Proposed Technical Specifications Changes
3. Proposed Technical Specifications Bases Changes (for information)
4. Retyped Technical Specification Pages
5. Retyped Technical Specification Bases Pages (for information)

c - B. S. Mallett, Region IV
M. C. Thadani, NRR
Resident Inspectors, CPSES

Ms. Alice K. Rogers
Environmental & Consumer Safety Section
Texas Department of State Health Services
1100 West 49th Street
Austin, Texas 78756-3189

ATTACHMENT 1 to TXX-07063

DESCRIPTION AND ASSESSMENT

LICENSEE'S EVALUATION

1.0 DESCRIPTION

By this letter, TXU Power requests an amendment to the Comanche Peak Steam Electric Station (CPSES) Unit 1 Operating License (NPF-87) and CPSES Unit 2 Operating License (NPF-89) by incorporating the attached change into the CPSES Unit 1 and 2 Technical Specifications. Proposed change LAR 07-003 is a request to incorporate standard Westinghouse-developed and NRC-approved analytical methods into the lists of methodologies used to establish the core operating limits. These NRC-approved methods include the use of an alternate axial offset control methodology and the use of updated methods for determining the core power distributions. Thus, conforming changes to the core power distribution limits and core power distribution measurement methods are also proposed.

The proposed changes include the use of different methodologies to evaluate the Emergency Core Cooling System (ECCS). These changes only allow the use of these analytical methods; plant-specific ECCS evaluation models using these methodologies will be submitted separately.

The TS Bases changes are provided for information only.

The proposed change has been reviewed, and it has been determined that no significant hazards consideration exists, as defined in 10 CFR 50.92. In addition, it has been determined that the change qualifies for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9); therefore, no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed change.

2.0 PROPOSED CHANGES

The proposed change would revise the CPSES Units 1 and 2 Technical Specifications as follows:

2.1 Section 5.6.5b Core Operating Limits Report (COLR)

Revise TS 5.6.5b to add the following NRC approved analytical methods:

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

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

WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analysis," April 1999.

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

WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.

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

WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005.

WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

2.2 Section 3.2 Power Distribution Limits

The proposed use of the NRC-approved methodology for controlling the axial power distribution, described in WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification," requires conforming revisions to Technical Specification 3.2.1 and 3.2.3. The proposed Technical Specifications are essentially the same as the NRC-approved Improved Standard Technical Specification (NUREG-1431, Volume 1, Revision 3), Section 3.2.1B, "Heat Flux Hot Channel Factor (F^C_Q (Z) (RAOC-W(Z) Methodology)" and Section 3.2.3B, "Axial Flux Difference (AFD) (Relaxed Axial Offset Control (RAOC) Methodology). One difference is that, in Technical Specification 3.2.1, the Overpower N-16 trip setpoint is used for CPSES rather than the Overpower ΔT trip setpoint identified in the Standard Technical Specification. Other differences in Section 3.2.1 include an editorial modification to the NOTE preceding the Surveillance Requirements and small changes in the Surveillance Frequencies consistent with the current licensing basis.

2.3 BEACON-related Changes (Technical Specifications 3.1.7, 3.2.1, 3.2.2, 3.2.4, and 3.3.1)

The proposed use of the BEACON Core Monitoring methodology requires conforming revisions to Technical Specifications TS 3.1.7, "Rod Position Indication," TS 3.2.1, "Heat Flux Hot Channel Factor," TS 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor," TS 3.2.4, "Quadrant Power Tilt Ratio," and TS 3.3.1, "Reactor Trip System (RTS) Instrumentation." In each of these sections, the proposed revisions would change the phrase "moveable incore detectors" to "core power distribution measurement information," and the phrase "flux map" to "power distribution measurements."

3.0 BACKGROUND

3.1 Section 5.6.5b Core Operating Limits Report (COLR)

The analytical methods used to establish the core operating limits are listed in Technical Specification 5.6.5b. The methods relevant to the proposed change are the non-LOCA transient and accident analytical tools, the core thermal-hydraulic analytical tools, the small break loss of coolant accident (LOCA) analysis method, the statistical best-estimate approach for large break LOCA analysis method, and the BEACON core power distribution monitoring process. All of these methodologies have been previously approved by the NRC for use at Westinghouse nuclear power plants, such as the Comanche Peak units.

<u>Revised Thermal Design Procedure (RTDP) (WCAP-11397-P-A)</u>

With the Revised Thermal Design Procedure (RTDP) methodology (WCAP-11397-P-A), uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and Departure from Nucleate Boiling (DNB) correlation predictions are combined statistically to obtain the overall DNB uncertainty factor which is used to define the design limit Departure from Nucleate Boiling Ratio (DNBR) that satisfies the DNB design criterion. The criterion is that the probability that DNB will not occur in the most limiting fuel rod is at least 95% (at a 95% confidence level) for a Condition I or II event. Since the parameter uncertainties are considered in determining the RTDP

design limit, DNBR values, the plant safety analyses are performed using input at their nominal values. The uncertainties included are:

- the nuclear enthalpy hot-channel factor;
- the enthalpy rise engineering hot-channel factor; and
- the THINC-IV transient codes uncertainties, based on surveillance data, associated with
    - reactor vessel coolant flow,
    - core power,
    - coolant temperature,
    - system pressure and
    - effective core flow fraction.

Per WCAP-14565-P-A, VIPRE-01 will be used in lieu of THINC-IV for CPSES applications.

The NRC's Safety Evaluation report (SER) was reviewed to identify any limitations or conditions on the use of the RTDP at CPSES. The CPSES application meets the guidelines presented in WCAP 11397-P-A. Specifically, Sensitivity factors and their ranges will be included in the Safety Analysis Report or reload submittal. As discussed in the NRC's SER, 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 will be re-evaluated to assure the continued validity of the sensitivity factors and the use of Equation (2-3) of the topical report. If the sensitivity factors are changed as a result of correlation changes or changes in the application or use of the THINC-IV code, then the use of an uncertainty allowance for application of Equation (2-3) will be re-evaluated and the linearity assumption made to obtain Equation (2-27) of the topical report will be validated. (Note, per WCAP-14565-P-A, VIPRE-01 will be used in lieu of THINC-IV for CPSES applications.) Any variances and distributions for input parameters will be justified. Nominal initial condition assumptions apply only to DNBR analyses using RTDP and other analyses will follow the appropriate conservative initial condition assumptions. As described in the NRC's SER, the nominal conditions chosen for use in CPSES analyses bound all permitted methods of plant operation and code uncertainties must be included in the DNBR analyses using RTDP.

<u>Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions (WCAP-8745-P-A)</u>

The Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions (WCAP-8745-P-A) are designed to provide primary protection against DNB and fuel centerline melt through excessive linear heat generation rates (LHGR) during postulated transients. The threshold for fuel centerline melt has been correlated

with a limiting value of the LHGR. The correlation includes the effects of burnup, flow rate, power distribution asymmetry and initial fill gas pressure level, and is based on a NRC approved fuel rod design analysis. The functional forms of the trip setpoints appropriately account for effects such as coolant density and pressure variation, adverse core power distribution, and instrumentation and piping delays.

As described in the NRC's SER appended to WCAP-8745-P-A, the relevant General Design Criteria are specified as UO_2 melting temperature will not be exceeded for 95% of the fuel rods at the 95% confidence level, and at least a 95% probability that the DNB will not occur at the limiting fuel rod at 95% confidence level. TXU Power will meet these criteria by restricting the calculated fuel centerline temperature to less than 4700°F and limiting the minimum DNBR to the correlation limit. The hot leg temperature must be maintained below the saturation temperature to assure the validity of the vessel average inlet/outlet coolant temperature difference. Additionally, the application of the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Function methodology for CPSES will account for changes in system design and operation. The adequacy of the standard power shapes in establishing the core DNB protection system will be evaluated whenever changes are introduced that could potentially affect the core power distribution.

Even though this methodology was developed for the generic Westinghouse overtemperature and overpower ΔT core protection systems, the protection philosophy is applicable to the overtemperature and overpower N-16 (Nitrogen-16) based core protection system used at CPSES. The overtemperature and overpower systems are functionally equivalent; the ΔT-based system uses the hot leg to cold leg temperature difference as an indication of the reactor power, whereas the N-16 based system uses the normalized N-16 gamma activity measured in the hot leg as an indication of the reactor power. Both of the overtemperature systems serve as primary protection functions for the prevention of conditions that could result in DNB, and both of the overpower systems serve as primary protection functions for the prevention of conditions that could result in exceeding the LHGR limits.

VIPRE-01 (WCAP-14565-P-A)

VIPRE-01 (WCAP-14565-P-A) is a subchannel thermal-hydraulic computer code that is used to describe the reactor core. Boundary conditions for the coolant entering the core, the power generation rate, and the dimensional and material properties of the nuclear fuel are entered as input to the code. The boundary conditions for the coolant include inlet flow rate, enthalpy and pressure or the pressure, inlet enthalpy and differential pressure from which inlet flow can be derived. The core power generation input includes spatial as well as temporal

variations. Multiple channels can be described and cross flow is calculated based on user supplied input. The intended use of the VIPRE code is for DNB analysis for those Final Safety Analysis Report (FSAR) Chapter 15 transients and accidents for which DNB might be of concern. The CPSES application meets the criteria described in the NRC's SER appended to WCAP-14565-P-A by using conservative reactor core boundary conditions as input into VIPRE for reactor transient analysis.

VIPRE, as described in the NRC's Safety Evaluation appended to WCAP-14565-P-A, has been approved by the NRC staff for use in place of the THINC-IV and FACTRAN computer codes in the Reload Safety Evaluation and RTDP methodologies presented in WCAP-9272-P-A and WCAP-11397-P-A, respectively.

RETRAN-02 (WCAP-14882-P-A)

RETRAN-02 was approved by the NRC to perform non-LOCA safety analyses in WCAP-14882-P-A. RETRAN-02 is a thermal-hydraulic computer code used to evaluate the effect of various reactor conditions on the Reactor Coolant System. RETRAN-02 was developed by Energy, Incorporated, for the Electric Power Research Institute (EPRI) and the modifications made by Westinghouse have been minor and consist of configuration control subroutines to satisfy Westinghouse's quality assurance requirements, an increase in dynamic storage to accommodate the large number of nodes in Westinghouse input models, and a correction in the kinetic energy calculation for opening and closing valves. The NRC concluded the use of RETRAN-02 is acceptable for licensing calculations for those transients and accidents listed in the NRC's SER appended to WCAP-14882-P-A. The NRC's SER also states that licensing applications using RETRAN-02 should include the source of and justification for the input data used in the analysis.

RETRAN-02 as described in WCAP-14882-P-A has been approved by the NRC staff for use in place of the LOFTRAN computer code referenced in the Reload Safety Evaluation and RTDP methodologies, WCAP-9272-P-A and WCAP-11397-P-A, respectively.

The NRC's SER was reviewed to identify any limitations or conditions on the use of the RETRAN-02 at CPSES. RETRAN-02 is a NRC-approved methodology for Westinghouse designed 2, 3, and 4 loop plants of the type that are currently operating, including CPSES. The application at CPSES meets the guidelines presented in WCAP 14882-P-A. Specifically, TXU Power intends to use RETRAN-02 to analyze those transients and accidents listed in the NRC's SER. Input for the RETRAN-02 models will be conservative for the specific application and include the source of and justification for the data used in the analyses consistent with NRC's SER.

Small Break LOCA (WCAP-10054-P-A)

NOTRUMP (WCAP-10054-P-A; WCAP-10054-P-A, Addendum 2 (Revision 1); and WCAP-10079-P-A) is a thermal-hydraulic computer program developed for analysis of FSAR Chapter 15 transient and accident events including the Small Break LOCA (SBLOCA). The code models one-dimensional thermal-hydraulics using control volumes interconnected by flow paths. Reactivity feedback is modeled with point kinetic neutronics. NOTRUMP incorporates special models to calculate responses of the reactor coolant pumps, steam separators, and the core fuel pins. The code also has a node stacking capability for calculating a single mixture elevation to eliminate unrealistic layering of steam and liquid mixture in adjacent vertical control volumes.

The NRC's SER was reviewed to identify any limitations or conditions on the use of the Westinghouse Small Break LOCA methodology at CPSES. As described in the NRC's SER appended to WCAP-10054-P-A, NOTRUMP is a NRC-approved methodology suitable for analyzing the SBLOCA accident for Westinghouse-designed 2, 3, and 4-loop Westinghouse plants, including both CPSES units.

Best Estimate Large Break LOCA (WCAP-16009-P-A)

The NRC-approved best-estimate large break LOCA (BELOCA) analysis includes a reference transient calculation and confirmation of certain conservatisms, the application of the ASTRUM (WCAP-16009-P-A) statistical treatment, the determination of a 95th percentile peak clad temperature (PCT), and oxidation calculations. The proposed ASTRUM Large Break LOCA (LBLOCA) analysis methodology differs from previously approved methodology primarily in the statistical approach. The NRC-approved ASTRUM methodology applies a non-parametric statistical technique directly to a random sample of outputs. The sample outputs are computed by applying Monte Carlo sampling to the inputs of WCOBRA/TRAC calculations. This allows the formulation of a simple singular statement of uncertainty in the form of a tolerance interval for the numerical acceptance criteria of 10 CFR 50.46. The ASTRUM methodology uses a 95/95

tolerance level to demonstrate conformance to 10 CFR 50.46. ASTRUM methodology accounts for the requirement that a spectrum of breaks be considered in the analysis by sampling three distributions: break type, cold-leg break area, and discharge coefficient. In the NRC'S SER appended to WCAP-16009-P-A, the NRC staff found this treatment of LBLOCA break size and type acceptable.

The NRC's SER was reviewed to identify any limitations or conditions on the use of the ASTRUM LBLOCA methodology at CPSES. The ASTRUM LBLOCA methodology is NRC approved for Westinghouse designed 2, 3, and 4-loop plants of the type that are currently operating, including both CPSES units. The application of the ASTRUM LBLOCA methodology at CPSES will entail the determination of the maximum local oxidation and whole core hydrogen generation. TXU Power will follow the conditions and limitations previously identified for WCOBRA/TRAC and will address 10CFR 50.46 parts b.1 through b.4 for LBLOCA as required in the NRC's SER.

BEACON Core Monitoring and Operation Support System (WCAP-12472-P-A)

In the NRC-approved WCAP-12472-P-A, Westinghouse described a method of monitoring the core power distributions using a power distribution monitoring system (PDMS). The Best Estimate Analyzer for Core Operation Nuclear (BEACON) system was developed by Westinghouse to improve the monitoring support for Westinghouse-designed pressurized water reactors (PWRs). It is a core monitoring and support package which uses plant data fed to the plant process computer from the incore thermocouples and excore nuclear instruments in conjunction with an analytical methodology for on-line generation of three-dimensional power distributions. The system provides core monitoring, core measurement reduction, core analysis, and core predictions. In its SER, the NRC concluded that BEACON provides a greatly improved continuous online power distribution measurement and operation prediction information system for Westinghouse reactors.

CPSES proposes to use BEACON to augment the functional capability of the flux mapping system for the purpose of power distribution surveillances. WCAP-12472-P-A discusses an application of BEACON in which selected Technical Specifications and core power distribution limits are changed to take credit for continuous monitoring by plant operators. The CPSES application proposes to use a more conservative application of BEACON where the core power distribution limits remain unchanged. This limited application of BEACON is referred to as the BEACON Technical Specification Monitor (TSM). TXU Power intends to use the BEACON PDMS as the primary method for power distribution measurements

and as the flux mapping system. When the PDMS is inoperable, the existing movable incore detector system can be used.

3.2 Section 3.2 Power Distribution Limits

NUREG-1431 Vol. 1, Rev. 3 specifies Improved Standard Technical Specifications (ISTS) for Westinghouse Plants. The proposed revision to the CPSES Technical Specifications Section 3.2 conforms to changes in the methodologies used to establish the core operating limits as described in WCAP-10216-P-A. Section 3.2.1B specifies Power Distribution Limits for the Heat Flux Hot Channel Factor using the Relaxation of Constant Axial Offset Control (RAOC) methodology described in WCAP-10216-P-A, currently listed in Section 5.6.5b of the CPSES Technical Specifications. The proposed modified Section 3.2.1 will be similar to Section 3.2.1B of NUREG-1431 Vol. 1, Rev. 3. The differences are:

- Required Actions A.3 and B.3 will read "Overpower N-16" instead of "Overpower ΔT"
- The Note prior to the surveillance requirements will retain the current verbiage of "During power escalation following shutdown…" rather than the ISTS verbiage of "During power escalation at the beginning of each cycle…"
- The Frequency for performance of SR 3.2.1.1 will retain the current plant-specific criterion of "Once within 24 hours after achieving equilibrium conditions after exceeding, by ≥ 20% RTP, the THERMAL POWER at which $F^C_Q(X)$ was last verified."
- The description of the factor used to increase the value of $F^W_Q(Z)$ is modified consistent with the current licensing basis and the limited application of the BEACON PDMS.

The overtemperature and overpower systems are functionally equivalent; the ΔT-based system uses the hot leg to cold leg temperature difference as an indication of the reactor power, whereas the N-16-based system uses the normalized N-16 gamma activity measured in the hot leg as an indication of the reactor power. Both of the overtemperature systems serve as primary protection functions for the prevention of conditions that could result in DNB, and both of the overpower systems serve as primary protection functions for the prevention of conditions that could result in exceeding the LHGR limits.

The changes to the ISTS SR 3.2.1.1 Note are consistent with the current CPSES Technical Specifications and eliminate redundancies between the Note and the surveillance frequencies for SR 3.2.1.1 and SR 3.2.1.2 which already include the requirement "Once after each refueling prior to THERMAL POWER exceeding

75% RTP." The proposed change to the ISTS Note also clearly indicates that the surveillances are required following any protracted shutdown, not just a refueling shutdown.

The retention of the current licensing basis completion times for the performance of SR 3.2.1.1 allows for the completion of the surveillance in a reasonable time period but does not allow for plant operation in an uncertain condition for a protracted time period. These completion times are also consistent with Specification 3.0.4 that allow 24 hours for the completion of a surveillance after prerequisite plant conditions are attained.

The description of the factor used to increase the value of $F^W_Q(Z)$ is modified to be consistent with the current licensing basis and the limited application of the BEACON PDMS.

Proposed changes to Section 3.2.3 will conform to Section 3.2.3B of the Improved Standard Technical Specifications without modification.

3.3 BEACON-related Changes (TS 3.1.7, 3.2.1, 3.2.2, 3.2.4, and 3.3.1)

CPSES proposes to use the BEACON system to augment the functional capability of the moveable incore flux mapping system for the purpose of power distribution surveillances. WCAP-12472-P-A discusses an application of BEACON in which the Technical Specifications and core power distribution limits are changed to take credit for continuous monitoring by plant operators. CPSES will use a conservative application of BEACON where the core power distribution limits remain unchanged; referred to as the BEACON Technical Specification monitor (TSM). TXU Power intends to use the BEACON PDMS as the primary method for power distribution measurements and as the flux mapping system, if required, provided that thermal power is greater than 25 percent rated thermal power (RTP). At thermal power levels less than or equal to 25 percent RTP, or when PDMS is inoperable, the existing moveable incore detector system will be used.

The PDMS instrumentation provides the capability to monitor core parameters at more frequent intervals than is currently required by the current Technical Specifications. The PDMS combines inputs from currently installed plant instrumentation and design data for each fuel cycle, and does not modify or eliminate existing plant instrumentation. It provides a means to continuously monitor the power distribution limits including limiting peaking factors and quadrant power tilt ratio. The PDMS instrumentation does not change any of the key safety parameter limits or levels of margin as considered in the reference design basis evaluations. These limits are not revised by this license amendment, and can be determined independently of the operability of the PDMS.

The actual changes to the Technical Specifications involve changing the phrase "moveable incore detectors" to "core power distribution measurement information," and the phrase "flux map" to "power distribution measurements." This approach would allow the use of the PDMS when available, as well as the use of the traditional moveable incore instrumentation system when the PDMS was not available. These changes are consistent with those proposed changes outlined in WCAP-12472-P-A; however, those changes were not based on the format and content of the current Improved Standard Technical Specifications. The changes proposed for CPSES are consistent with those changes recently approved by the NRC for Diablo Canyon in Reference 8.2.

The PDMS itself does not meet any of the 10 CFR 50.36(c)(2)(ii) selection criteria for inclusion into the Technical Specifications. Therefore, the PDMS does not require a Technical Specification controlling its operability. Therefore, the PDMS instrumentation requirement will be controlled administratively.

The justification for not including PDMS instrumentation in the Technical Specifications is outlined below. The purpose of this evaluation is to demonstrate that the structures, systems, or components associated with PDMS instrumentation are not required to be contained in the Technical Specifications. This evaluation is done in accordance with the requirements contained in 10 CFR 50.36(c)(2)(ii).

> A TS Limiting Condition for Operation must be established for each item meeting one or more of the following criteria:
>
> **(A) Installed Instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.**
>
> PDMS instrumentation is not associated with monitoring of any aspect of the reactor coolant pressure boundary.
>
> **(B) A process variable, design feature, or operating restriction that is an initial condition of a design basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.**
>
> The limits for the power distribution parameters $F_Q(Z)$ and $F^N_{\Delta H}$ are operating restrictions, which ensure that all analyzed DBAs remain valid. These limits are included in the Technical Specifications. The PDMS instrumentation, however, provides the capability to monitor these parameters at more frequent intervals than is currently required by the Technical Specifications.

Additionally, these limits can be determined independent of the operability of PDMS. Therefore, the PDMS instrumentation is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(C) A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

PDMS instrumentation provides the capability to monitor core power distribution parameters at more frequent intervals than is currently required by Technical Specifications. PDMS instrumentation does not change any of the key safety parameter limits or levels of margin as considered in the reference design basis evaluations. The PDMS instrumentation has no functions or actuations that mitigate any DBA or transient analysis that either assumes the failure of, or presents the challenge to the integrity of a fission product barrier.

(D) A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

PDMS instrumentation provides the capability to monitor power distribution parameters at more frequent intervals than is currently required by Technical Specifications. PDMS instrumentation is a system utilized to monitor the core power distribution and has no impact on the results or consequences of any DBA or transient analysis. Therefore it has no impact on public health and safety.

The evaluation completed above indicates that PDMS instrumentation does not meet any of the criteria for inclusion in the Technical Specifications. The administrative controls for PDMS operability will reflect the minimum requirements presented in WCAP-12742-P-A except for changes due to CPSES' use of the BEACON TSM, according to vendor instructions.

In summary, the proposed amendment would allow the use of the Westinghouse proprietary 3-D nodal code BEACON for performing power distribution surveillances provided that the PDMS instrumentation is operable.

This amendment would also continue to allow the use of the movable incore detector system for meeting power distribution surveillances and Technical Specifications actions, and for calibration of BEACON.

4.0 TECHNICAL ANALYSIS

The proposed changes to Technical Specifications 5.6.5.b define NRC-approved methods that will be used to establish cycle operating limits. The limits established with the referenced methodologies will ensure that reload design, analysis, and plant operation will remain within the regulatory requirements established for fuel assembly and core designs. The changes to the power distribution limits Technical Specifications (TS 3.2.1 and 3.2.3), and to those Technical Specifications requiring power distribution measurements (TS 3.1.7, 3.2.1, 3.2.2, 3.2.4, and 3.3.1) are proposed to conform to the NRC-approved methodologies used to establish the core operating limits. TXU Power has reviewed the changes and determined that the documents referenced completely address the cycle specific reload design and analysis activities required to determine the core operating limits. All referenced methodologies have been approved by the NRC for the intended application.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

TXU Power has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10CFR50.92, "Issuance of amendment," as discussed below:

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

   Response: No

   No physical plant changes or changes in manner in which the plant will be operated as a result of the methodology changes. The proposed changes do not impact the condition or performance of any plant structure, system or component. The core operating limits are established to support Technical Specifications 3.1, 3.2, 3.3, and 3.4. The core operating limits ensure that fuel design limits are not exceeded during any conditions of normal operation or in the event of any Anticipated Operational Occurrence (AOO). The methods used to establish the core operating limits for each operating cycle are based on methods previously found acceptable by the NRC and listed in Technical Specifications section

5.6.5.b. Application of these NRC-approved methods will continue to ensure that acceptable operating limits are established to protect the fuel cladding integrity during normal operation and AOOs. The requested Technical Specification changes, including those changes proposed to conform with the NRC-approved analysis methodologies, do not involve any plant modifications or operational changes that could affect system reliability, performance, or possibility of operator error. The requested changes do not affect any postulated accident precursors, does not affect any accident mitigation systems, and does not introduce any new accident initiation mechanisms.

As a result, the proposed changes to the CPSES Technical Specifications do not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated since neither accident probabilities nor consequences are being affected by this proposed change.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

There are no physical changes being made to the plant. No new modes of plant operation are being introduced. The parameters assumed in the analyses are within the design limits of the existing plant equipment. All plant systems will perform as designed during the response to a potential accident.

Therefore, the proposed change to the CPSES Technical Specifications does not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any accident previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No

The NRC-approved accident analysis methodologies include restrictions on the choice of inputs, the degree of conservatism inherent in the calculations, and specified event acceptance criteria. Analyses performed in accordance with these methodologies will not result in adverse effects on the regulated margin of safety. Similarly, the use of axial power distribution controls based on the relaxed axial offset control strategy is a

time-proven and NRC-approved method. The method is consistent with the accident analyses assumptions as described in the list of NRC-approved methodologies proposed to be used to establish the core operating limits. Finally, the proposed changes to allow operation with the BEACON power distribution monitoring tool provide additional information to the reactor operators on the state of the reactor core. Again, the use of the BEACON tool and the methodology used to develop the inputs to the tool are consistent with and controlled by the NRC-approved methodologies used to establish the core operating limits. As such, the margin of safety assumed in the plant safety analysis is not adversely affected by the proposed changes.

Based on the above evaluations, TXU Power concludes that the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10CFR50.92(c) and, accordingly, a finding of no significant hazards consideration is justified.

5.2 Applicable Regulatory Requirements/Criteria

The proposed changes will ensure that the fuel design and core operating limits determined for the operating cycles will be developed using NRC-approved methods identified in Technical Specifications 5.6.5.b, which are based on applicable regulatory criteria. In conclusion, (1) there is reasonable assurance that the health and safety of the public will not be endangered by the operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need to be prepared in connection with the proposed amendment.

7.0 PRECEDENTS

The changes to the Technical Specifications 3.1.7, 3.2.1, 3.2.2, 3.2.4, and 3.3.1 involve changing the phrase "moveable incore detectors" to "core power distribution measurement information," and the phrase "flux map" to "power distribution measurements." This approach would allow the use of the power distribution monitoring system (PDMS) when available, as well as the use of the traditional moveable incore instrumentation system when the PDMS was not available. These changes are consistent with those proposed changes outlined in WCAP-12472-P-A and with those changes recently approved by the NRC for Diablo Canyon (see, Reference 8.2).

8.0 REFERENCES

8.1 NUREG-1431 Volume 1, Revision 3.0, "Standard Technical Specifications, Westinghouse Plants," June 2004.

8.2 Diablo Canyon Power Plant, Unit No. 1 (TAC No. MB9640) And Unit No. 2 (TAC No. MB9641) - Issuance Of Amendment Re: Use Of A Power Distribution Monitoring System, March 31, 2004.

ATTACHMENT 2 to TXX-07063

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

Pages	i
	3.1-16
	3.1-17
	3.2-1
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(continued)

Relaxed Axial Offset Control (RAOC)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 The Digital Rod Position Indication (DRPI) System and the Demand Position Indication System shall be OPERABLE

APPLICABILITY: MODES 1 and 2.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each inoperable rod position indicator and each demand position indicator per bank.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DRPI per group inoperable for one or more groups.	A.1 Verify the position of the rods with inoperable position indicators indirectly by using ~~movable incore detectors.~~ core power distribution measurement information	Once per 8 hours
	OR	
	A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. More than one DRPI per group inoperable.	B.1	Place the control rods under manual control.	Immediately
	AND		
	B.2	Monitor and record RCS T_{avg}.	Once per 1 hour
	AND		
	B.3	Verify the position of the rods with inoperable position indicators indirectly by using ~~movable incore detectors.~~ core power distribution measurement information	Once per 8 hours
	AND		
	B.4	Restore inoperable position indicators to OPERABLE status such that a maximum of one DRPI per group is inoperable.	24 hours
C. One or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	C.1	Verify the position of the rods with inoperable position indicators indirectly by using ~~movable incore detectors.~~	4 hours
	OR		
	C.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)

LCO 3.2.1 F_Q (Z), as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. $F_Q^C(Z)$ not within limit.	A.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% $FQ^C(Z)$ exceeds limit.	15 minutes after each $F_Q^C(Z)$ determination
	AND	
	A.2 Reduce Power Range Neutron Flux- High trip setpoints ≥ 1% for each 1% $FQ^C(Z)$ exceeds limit.	72 hours after each $F_Q^C(Z)$ determination
	AND	
	A.3 Reduce Overpower N-16 trip setpoints ≥ 1% for each 1% $FQ^C(Z)$ exceeds limit.	72 hours after each $F_Q^C(Z)$ determination
	AND	
	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

----------NOTE----------
Required Action A.4 shall be completed whenever this Condition is entered.

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. $F_Q^W(Z)$ not within limits.	B.1 Reduce AFD limits ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	4 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 2.	6 hours

----------NOTE----------
Required Action B.4 shall be completed whenever this Condition is entered.

AND

B.2 Reduce Power Range Neutron Flux - High trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced. — 72 hours

AND

B.3 Reduce Overpower N-16 trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced. — 72 hours

AND

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

SURVEILLANCE REQUIREMENTS

----------NOTE----------
During power escalation following shutdown, THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution ~~map~~ measurement is obtained.

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify $F_Q^C(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND Once within 24 hours after achieving equilibrium conditions after exceeding, by ≥ 20% RTP, the THERMAL POWER at which $F_Q^C(Z)$was last verified AND 31 EFPD thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	----------NOTE---------- If $F_Q^C(Z)$ measurements indicate maximum over z $\left[\frac{F_Q^C(Z)}{K(Z)}\right]$ has increased since the previous evaluation of $F_Q^C(Z)$: a. Increase $F_Q^W(Z)$ by ~~the appropriate factor~~ an appropriate factor specified in the COLR and reverify $F_Q^W(Z)$ is within limits; or b. Repeat SR 3.2.1.2 once per 7 EFPD until two successive ~~flux maps~~ power distribution measurements indicate maximum over z $\left[\frac{F_Q^C(Z)}{K(Z)}\right]$ has not increased. ---------- either a. above is met or Verify $F_Q^W(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND (continued)

SURVEILLANCE REQUIREMENTS

----------NOTE----------
During power escalation following shutdown, THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution ~~map~~ measurement is obtained.

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND 31 EFPD thereafter

RAOC

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD) ~~(Constant Axial Offset Control (CAOC)Methodology)~~

(Relaxed Axial Offset Control (RAOC) Methodology)

LCO 3.2.3

~~The AFD:~~

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.

~~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, 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

≥ 50%

RAOC

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~~
~~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~~
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

~~(continued)~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~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~~	~~Not Used.~~	
~~SR 3.2.3.3~~	~~NOTE~~ ~~The initial target flux difference after each refueling may be determined from design predictions.~~ ~~Determine, by measurement, the target flux difference of each OPERABLE excore channel.~~	~~Whenever $F_Q^W(Z)$ is verified per 3.2.1.2.~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	----------NOTES---------- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. ---------- Verify QPTR is within limit by calculation.	7 days
SR 3.2.4.2	----------NOTE---------- Not required to be performed until 12 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP. ---------- Verify QPTR is within limit using the ~~movable incore detectors.~~ core power distribution measurement information	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.2	----------NOTES---------- 1. Adjust NIS and N-16 Power Monitor channel if absolute difference is > 2%. 2. Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP. ---------- Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) and N-16 Power Monitor channel output.	24 hours
SR 3.3.1.3	----------NOTES---------- 1. Adjust NIS channel if absolute difference is ≥ 3%. 2. Not required to be performed until 24 hours after THERMAL POWER is ≥ 50% RTP. ---------- Compare results of the ~~incore detector~~ core power distribution measurements to NIS AFD.	31 effective full power days (EFPD)
SR 3.3.1.4	----------NOTE---------- This Surveillance must be performed on the reactor trip bypass breaker for the local manual shunt trip only prior to placing the bypass breaker in service. ---------- Perform TADOT.	62 days on a STAGGERED TEST BASIS

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.1.6	----------NOTE---------- Not required to be performed until 72 hours after achieving equilibrium conditions with THERMAL POWER ≥ 75 % RTP. ---------- Calibrate excore channels to agree with ~~incore detector~~ core power distribution measurements.	92 EFPD
SR 3.3.1.7	----------NOTES---------- 1. Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3. 2. Source range instrumentation shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. ---------- Perform COT.	184 days

(continued)

5.6 Reporting Requirements (continued)

5.6.5 Core Operating Limits Report (COLR) (continued)

allowing use of 100.6 percent of rated power in safety analysis methodology when the LEFM√ is used for feedwater flow measurement.

The approved analytical methods are described in the following documents:

1) WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary)

2) WCAP-10216-P-A, Revision 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary).

3) RXE-90-006-P-A, "Power Distribution Control Analysis and Overtemperature N-16 and Overpower N-16 Trip Setpoint Methodology," June 1994.

4) RXE-88-102-P-A, "TUE-1 Departure from Nucleate Boiling Correlation," July 1992.

5) RXE-88-102-P, Sup. 1, "TUE-1 DNB Correlation - Supplement 1," December 1990.

6) RXE-89-002-A, "VIPRE-01 Core Thermal-Hydraulic Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications," September 1993.

7) RXE-91-001-A, "Transient Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications," October 1993.

8) RXE-91-002-A, "Reactivity Anomaly Events Methodology," October 1993.

9) ERX-2000-002-P, "Revised Large Break Loss of Coolant Accident Analysis Methodology," March 2000.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 Core Operating Limits Report (COLR) (continued)

10) TXX-88306, "Steam Generator Tube Rupture Analysis," March 15, 1988.

11) RXE-91-005-A, "Methodology for Reactor Core Response to Steamline Break Events," February 1994.

12) RXE-94-001-A, "Safety Analysis of Postulated Inadvertent Boron Dilution Event in Modes 3, 4, and 5," February 1994.

13) RXE-95-001-P-A, "Small Break Loss of Coolant Accident Analysis Methodology," September 1996.

14) Caldon, Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power level Using the LEFM√ System," Revision 0, March 1997 and Caldon Engineering Report – 160P, "Supplement to Topical Report ER-80P; Basis for a Power Uprate With the LEFM√tm System," Revision 0, May 2000.

15) ERX-2001-005-P, "ZIRLO™ Cladding and Boron Coating Models for TXU Electric's Loss of Coolant Accident Analysis Methodologies," October 2001.

16) WCAP-10444-P-A, "Reference Core Report VANTAGE 5 Fuel Assembly," September 1985.

17) WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles for Modified LPD Mixing Vane Grids," April 1999.

18) WCAP-13060-P-A, "Westinghouse Fuel Assembly Reconstitution Evaluation Methodology," July, 1993.

19) ERX-04-004-A, "Replacement Steam Generator Supplement To TXU Power's Large and Small Break Loss Of Coolant Accident Analysis Methodologies," Revision 0, March 2007.

INSERT A

20) ERX-04-005-A, "Application of TXU Power's Non-LOCA Transient Analysis Methodologies to a Feed Ring Steam Generator Design," Revision 0, March 2007.

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.

(continued)

INSERT A for Section 5.6

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

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

23) WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

24) WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.

25) WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.

26) WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.

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

28) WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005.

29) WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

ATTACHMENT 3 to TXX-07063

PROPOSED TECHNICAL SPECIFICATIONS BASES CHANGES
(Markup For Information Only)

Pages B 3.1-20
B 3.1-23
B 3.1-39
B 3.1-40
B 3.2-1 thru B 3.2-11
B 3.2-13
B 3.2-16 thru B 3.2-20
Insert A and Insert B
B 3.2-21 thru B 3.2-25
B 3.2-28
B 3.2-29
B 3.2-31
B 3.3-32
B 3.3-45

BASES

APPLICABLE SAFETY ANALYSES (continued)

core power distribution measurement

directly by ~~incore mapping.~~ Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements (i.e., trippability to meet SDM) are separate from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. A rod is considered OPERABLE based on the last satisfactory performance of SR 3.1.4.2 and has met the rod drop time criteria during the last performance of SR 3.1.4.3. Rod control malfunctions that result in the inability to move a rod (e.g., rod urgent failures), which do not impact trippability within the time requirements of SR 3.1.4.3, do not result in rod inoperability.

The requirement to maintain the rod alignment to within plus or minus 12 steps of their group step counter demand position is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2, because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the rods are typically fully inserted and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect

(continued)

BASES

ACTIONS

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6 (continued)

a core power distribution measurement

Verifying that $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, and $F^N_{\Delta H}$ are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain ~~flux maps of the core power distribution using the incore flux mapping system~~ and to calculate $F_Q(Z)$ and $F^N_{\Delta H}$.

Once current conditions have been verified acceptable, time is available to perform evaluations of the affected accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. The accident analyses presented in FSAR Chapter 15 (Ref. 3) that may be adversely affected will be evaluated to ensure that the analyses results remain valid for the duration of continued operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

C.1

When Required Actions of Condition B cannot be completed within their Completion Time, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

D.1.1 and D.1.2

More than one control rod becoming misaligned from its group demand position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. Verification of shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.6, "CONTROL BANK INSERTION LIMITS" ensure SDM is maintained provided the misaligned rod is above the insertion limit. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the required pumps. Boration will continue until the required SDM is restored.

(continued)

BASES

ACTIONS (continued)

or an OPERABLE PDMS

A.1

When one DRPI per group fails, the position of the rod may still be indirectly determined by use of the incore movable detectors. The Required Action may also be satisfied by ensuring at least once per 8 hours that F_Q satisfies LCO 3.2.1, $F_{\Delta H}$ satisfies LCO 3.2.2, and SHUTDOWN MARGIN is within the limits provided in the COLR, provided the nonindicating rods have not been moved. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of C.1 or C.2 below is required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

A.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 2).

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above.

B.1, B.2, B.3 and B.4

When more than one DRPI per group fail, additional actions are necessary to ensure that acceptable power distribution limits are maintained, minimum SDM is maintained, and the potential effects of rod misalignment on associated accident analyses are limited. Placing the Rod Control System in manual assures unplanned rod motion will not occur. Together with the indirect position determination available via movable incore detectors will minimize the potential for rod misalignment.

The immediate Completion Time for placing the Rod Control System in manual reflects the urgency with which unplanned rod motion must be prevented while in this Condition. Monitoring and recording reactor coolant T_{avg} help assure that significant changes in power distribution and SDM are avoided. The once per hour Completion Time is acceptable because only minor fluctuations in RCS temperature are expected at steady state plant operating conditions.

(continued)

BASES

ACTIONS

B.1, B.2, B.3 and B.4 (continued)

or an OPERABLE PDMS

The position of the rods ~~may be~~ determined indirectly by use of the movable incore detectors. The Required Action may also be satisfied by ensuring at least once per 8 hours that F_Q satisfies LCO 3.2.1, $F_{\Delta H}$ satisfies LCO 3.2.2, and SHUTDOWN MARGIN is within the limits provided in the COLR, provided the nonindicating rods have not been moved. Verification of RCCA position once per 8 hours is adequate for allowing continued full power operation for a limited, 24 hour period, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. The 24 hour Completion Time provides sufficient time to troubleshoot and restore the DRPI system to operation while avoiding the plant challenges associated with a shutdown without full rod position indication (Ref. 4).

Based on operating experience, normal power operation does not require excessive rod movement. If one or more rods has been significantly moved, the Required Action of C.1 or C.2 below is required.

C.1 and C.2

These Required Actions clarify that when one or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of A.1 and A.2 or B.3 are still appropriate but must be initiated promptly under Required Action C.1 to begin indirectly verifying that these rods are still properly positioned, relative to their group positions using the movable incore detectors.

If, within 4 hours, the rod positions have not been determined, THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of 4 hours provides an acceptable period of time to verify the rod positions.

D.1.1 and D.1.2

With one demand position indicator per bank inoperable, the rod positions can be determined by the DRPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means (e.g., observation of appropriate DRPI status indications) that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are ≤ 12 steps apart within the allowed Completion Time of once every 8 hours is adequate.

(continued)

RAOC-W(Z)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$) (~~F_Q~~ Methodology)

BASES

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT TILT POWER RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.7, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

measured periodically using the incore detector system or an OPERABLE PDMS.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is ~~not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. The results of the three-dimensional power distribution map are analyzed to derive a measured value for $F_Q(Z)$~~. These measurements are generally taken with the core at or near equilibrium conditions.

Using the measured three dimensional power distributions, it is possible to derive a measured value for FQ(Z).

However, because this value represents an equilibrium condition, it does not include the variations in the value of $F_Q(Z)$ that are present during non-equilibrium situations, such as load following. To account for these possible variations, the steady state value of $F_Q(Z)$ is adjusted by an elevation dependent factor, W(Z), that accounts for calculated ~~transient~~ conditions.

worst case

Core monitoring and control under non-steady state 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's principal effect is to preclude core power distributions that could lead to violation of the following fuel design criterion:

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

(continued)

RAOC-W(Z)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm~~.~~

, and

c. 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.

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 LOCA peak cladding temperature is typically most limiting.

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

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

LCO

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

$$F_Q(Z) \leq \frac{\cancel{F_Q^{RTP}}}{P} K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \leq \frac{\cancel{F_Q^{RTP}}}{0.5} K(Z) \qquad \text{for } P \leq 0.5$$

C F Q

where: $\cancel{F_Q^{RTP}}$ 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 = THERMAL POWER / RTP

The actual values of $\cancel{F_Q^{RTP}}$ and K(Z) are given in the COLR.

Relaxed

For ~~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 $F_Q(Z)$.

(continued)

RAOC-W(Z)

BASES

LCO (continued)

core power distribution measurement

An $F_Q^C(Z)$ evaluation requires obtaining a~~n incore flux map~~ in MODE 1.

From the ~~incore flux map~~ results we obtain the measured value ($F_Q^M(Z)$) of $F_Q(Z)$.

If the PDMS is used, the appropriate measurement uncertainty and manufacturing allowance are automatically calculated and applied to the measured FQ (Ref. 7).

~~The~~ computed heat flux hot channel factor, $F_Q^C(Z)$, is obtained by the equation:

$$F_Q^C(Z) = F_Q^M(Z) \cdot 1.03 \cdot 1.05.$$

If the movable incore detector system is used, the

$F_Q^M(Z)$ is increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainties.

$F_Q^C(Z)$ 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:

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

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

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

C

This LCO requires operation within the bounds assumed in the safety analyses. If $F_Q(Z)$ cannot be maintained within the LCO limits, a reduction of the core power is required.

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 FQ(Z) limits.

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

(continued)

and if FQW(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)

~~If the power distribution measurements are performed at a power level less than 100% RTP, then the $F_Q^C(Z)$ and $F_Q^W(Z)$ values that would result from measurements if the core was at 100% RTP should be inferred from the available information. A comparison of these inferred values with F_Q^{RTP} assures compliance with the LCO at all power levels.~~

APPLICABILITY

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

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^C(Z)$ is $F_Q^M(Z)$ multiplied by factors that account 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.

A.2

A reduction of the Power Range Neutron Flux-High trip setpoints by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit is a conservative action for

(continued)

RAOC-W(Z)

BASES

ACTIONS

A.2 (continued)

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 N-16 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 N-16 trip setpoints initially determined by Required Action A.3 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Overpower N-16 trip setpoint reductions within 72 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Overpower N-16 trip setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum Overpower N-16 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 are consistent with safety analyses assumptions.

(continued)

B.4 Verification that FQW(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 maximum allowable power limit imposed by Required Action B.1 ensures that core conditions during operation at higher power levels and future operation are consistent with safety analysis assumptions.

RAOC-W(Z)

BASES

ACTIONS

~~A.4 (continued)~~

~~Inherent in this action is the ident~~ ~~it~~
~~condition and the correction of th~~ ~~llow~~
~~safe operation at the higher powe~~
~~determined by extrapolating~~ F_Q^C ~~ior to~~
~~increasing power above the extra~~ ~~ration~~
~~of any reduced Reactor Trip Syst~~

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 FQ(Z) is properly evaluated prior to increasing THERMAL POWER.

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 A 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 ensure FQ(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 ≥ 1% for each 1% by which $F_Q^W(Z)$ exceeds its limit within the allowed Completion Time of 4 hours, restricts the axial flux distribution such that even if a transient occurred, core peaking factor limits are not exceeded.

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, is a conservative action for protection against the consequence 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 reductions in THERMAL POWER as a result of reducing AFD limits in accordance with Required Action B.1.

C.1

through B.4

If Required Actions A.1 through A.4 or B.1 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 power ascensions following a plant shutdown (leaving Mode 1). The note allows for power ascensions if the surveillances are not current. 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 20% RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because

measurement

B.3 Reduction in the Overpower N-16 setpoints value by ≥ 1% for each 1% by which the maximum allowable power is reduced, is a conservative action for protection against the consequences of sever 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 reductions in THERMAL POWER as a result of reducing AFD limits in accordance with Required Action B.1.

(continued)

RAOC-W(Z)

BASES

SURVEILLANCE REQUIREMENTS (continued)

$F_Q^C(Z)$ and $F_Q^W(Z)$ could not have previously been measured for a reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of $F_Q^C(Z)$ and $F_Q^W(Z)$ are 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 20%, ensures that they are verified ~~within 24 hours from when equilibrium conditions are achieved at RTP (or any other level for extended operation)~~. ~~Equilibrium conditions are achieved when the core is sufficiently stable such that the uncertainty allowances associated with the measurement are valid.~~ 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 20% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 20% higher than that power at which F_Q was last measured.

as soon as RTP (or any other level for extended operation) is achieved.

SR 3.2.1.1

If the PDMS is used, the appropriate measurement uncertainty and manufacturing allowance are automatically calculated and applied to the measured FQ (Ref. 7). If the movable incore detector system is used,

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 $F_Q^C(Z)=F_Q^M(Z)\cdot1.03\cdot1.05$ (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, ~~or at a reduced power level at any other time, and meeting the 100% RTP $F_Q(Z)$~~

(continued)

RAOC-W(Z)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

~~limit~~, provides assurance that the $F_Q^C(Z)$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

If THERMAL POWER has been increased by ≥ 20% RTP since the last determination of $F_Q^C(Z)$, another evaluation of this factor is required 24 hours after achieving equilibrium conditions at this higher power level (to ensure that $F_Q^C(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits).

The Frequency of 31 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).

The nuclear design process includes calculations performed to determine that the core can be operated within the FQ(Z) limits.

SR 3.2.1.2

power distribution measurements

at or near

Because ~~flux maps~~ are taken ~~in~~ equilibrium 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~~ 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, $F_Q^W(Z)$.

core power distribution measurement

above 50% RTP

The limit with which $F_Q^W(Z)$ is compared varies inversely with power 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 typically taken for 30 to 75 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.

(continued)

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. When $F_Q^W(Z)$ is evaluated, an evaluation of the expression below is required to account for any increase to $F_Q^C(Z)$ 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

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

it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by the appropriate factor of ≥ 1.02 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.

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP, ~~or at a reduced power level at any other time, and meeting the 100% RTP $F_Q(Z)$ limit,~~ provides assurance that the $F_Q(Z)$ limit will be met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

$F_Q(Z)$ is verified at power levels ≥ 20% RTP above the THERMAL POWER of its last verification, 24 hours after achieving equilibrium conditions to ensure that $F_Q(Z)$ is within its limit at higher power levels.

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.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50.46, 1974.

2. Regulatory Guide 1.77, Rev. 0, May 1974.

3. 10 CFR 50, Appendix A, GDC 26.

4. RXE-90-006-P-A, "Power Distribution Control Analysis and Overtemperature N-16 and Overpower N-16 Trip Setpoint Methodology," TU Electric, June 1994.

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

6. WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control (and) FQ Surveillance Technical Specification," February 1994.

7. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during normal operation, operational transients, and any transient condition arising from events of moderate frequency analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod. $F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup.

measurement

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution ~~map~~ obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution ~~map~~ are analyzed to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables. Compliance with these LCOs, along with the LCOs governing shutdown and control rod insertion and alignment, maintains the core limits on power distribution on a continuous basis.

or an OPERABLE PDMS

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

or an OPERABLE PDMS

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

$F^N_{\Delta H}$ satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

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

The $F^N_{\Delta H}$ limit is representative of the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a DNB condition.

The limiting value of $F^N_{\Delta H}$ described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.

A power multiplication factor in this equation includes an additional allowance for higher radial peaking factors from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase by a cycle-dependent factor ($PF_{\Delta H}$, as specified

(continued)

BASES

ACTIONS

A.2 (continued)

Required Action A.1.1, or the power level has been reduced to < 50% RTP per Required Action A.1.2.1, ~~an incore flux map~~ (SR 3.2.2.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by either Action A.1.1 or Action A.1.2.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period.

core power distribution measurement

Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the ~~incore flux map,~~ perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.3

Verification that $F^N_{\Delta H}$ is within its specified limits after an out of limit occurrence ensures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is identified, to the extent necessary, and corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP. SR 3.2.2.1 must be satisfied prior to increasing power above the extrapolated allowable power level or restoration of any reduced Reactor Trip System setpoints. When $F^N_{\Delta H}$ is measured at reduced power levels, the allowable power level is determined by evaluating $F^N_{\Delta H}$ for higher power levels.

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action.

B.1

When Required Actions A.1.1 through A.3 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at

(continued)

BASES

ACTIONS

B.1 (continued)

least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

measurement

SR 3.2.2.1 is modified by a Note. The Note applies during power ascensions following a plant shutdown (leaving Mode 1). The note allows for power ascensions if the surveillances are not current. 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. Equilibrium conditions are achieved when the core is sufficiently stable such that the uncertainty allowances associated with the measurement are valid.

or an OPERABLE PDMS

power distribution measurement

The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a ~~flux distribution map.~~ A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by 1.04 to account for measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.

If the PDMS is used, the appropriate measurement uncertainty is automatically calculated and applied to the measured $F^N_{\Delta H}$ (Ref. 4).

If the moveable incore detector system is used, t

After each refueling, $F^N_{\Delta H}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle. Performing this Surveillance in MODE 1 prior to exceeding 75% RTP, or at a reduced power level at any other time, and meeting the 100% RTP $F^N_{\Delta H}$ limit, provides assurance that the $F^N_{\Delta H}$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F^N_{\Delta H}$ limit cannot be exceeded for any significant period of operation.

(continued)

BASES (continued)

REFERENCES

1. Regulatory Guide 1.77, Rev. 0, May 1974.
2. 10 CFR 50, Appendix A, GDC 26.
3. 10 CFR 50.46.
4. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

RAOC

B 3.2 POWER DISTRIBUTION LIMITS

Relaxed

B 3.2.3 AXIAL FLUX DIFFERENCE (AFD) (~~Constant~~ Axial Offset Control (~~CAOC~~) Methodology)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the 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.

Insert A attached

~~The operating scheme used to control the axial power distribution, 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. 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., ≥ 180 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.~~

~~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 AFD is 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 if the AFDs for two or more OPERABLE excore channels are outside 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.~~

(continued)

RAOC

BASES (continued)

APPLICABLE SAFEY ANALYSES

~~The Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) and QPTR LCOs limit the radial component of the peaking factors. The AFD is a measure of axial power distribution skewing to 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.~~

Insert B attached

~~The CAOC methodology (Refs. 1, 2, and 3) entails:~~

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

~~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 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. Compliance with this limit ensures that acceptable levels of 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 complete loss of forced RCS flow accident. The most significant Condition II events are uncontrolled bank withdrawal and boration or dilution accidents. Condition II accidents are used to confirm the adequacy of Overpower N-16 and Overtemperature N-16 trip setpoints.

The limits on the AFD satisfy Criterion 2 of the 10CFR50.36(c)(2)(ii).

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 responding to

(continued)

Insert A

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 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 s 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 to or more OPERABLE excore channels is outside its specified limits.

Although the RAOC defines limits that must be met to satisfy safety analyses, typically an operating scheme, Constant Axial Offset Control (CAOC), is used to control axial power distribution in day to day operation (Ref. 2). CAOC requires that the AFD be controlled within a narrow tolerance band around a burnup dependent target to minimize the variation of axial peaking factors and axial xenon distribution during unit maneuvers.

The CAOC operating space is typically smaller and lies within the RAOC operating space. Control within the CAOC operating space constrains the variation of axial xenon distributions and axial power distributions. RAOC calculations assume a wide range of xenon distributions and then confirm that the resulting power distributions satisfy the requirements of the accident analyses.

Insert B

The AFD is a measure of the 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 concentration.

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. 5) 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 loss of coolant accident 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.

BASES

LCO (continued)

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 (Ref. 4). 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.

The AFD limits are provided in the COLR. The AFD limits for RAOC do not dpend on the target flux difference. However, the target flux difference may be used to minimize changes in the axial power distribution.

~~This LCO is modified by a Note that states the conditions necessary for declaring the AFD outside of the target band. 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.~~

~~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.~~

~~Parts B and C of this LCO are affected by Notes that 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. This penalty time is accumulated at the rate of 1 minute for each 1 minute of operating time within the power range of Part B of this LCO (i.e., THERMAL POWER ≥ 50% RTP). The cumulative penalty time is the sum of penalty times from Parts B and C of this LCO.~~

~~For THERMAL POWER levels > 15% RTP and < 50% RTP (i.e., Part C of~~

(continued)

BASES

LCO (continued)

~~this LCO), deviations of the AFD outside of the target band are less significant. The accumulation of 1/2 minute penalty deviation time per 1 minute of actual time outside the target band 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 distributions are prevented by limiting the accumulated penalty deviation time.~~

Violating the LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its limits.

APPLICABILITY

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

greater than or equal to

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

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.

~~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.~~

~~For surveillance of the power range channels performed according to SR 3.3.1.6, deviation outside the target band is permitted for 16 hours and no penalty deviation time is accumulated. Some deviation in the AFD may be required for the performance of the NIS calibration with the incore detector system. This calibration is typically performed every 92 days.~~

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

ACTIONS

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.

A.1

~~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.~~

(continued)

BASES

ACTIONS (continued)

~~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 but within the acceptable operation limits provided in the COLR provided in the COLR is allowed for up to 1 hour. 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 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.~~

~~(continued)~~

BASES

ACTIONS

~~D.1 (continued)~~

~~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.~~

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

its specified limits

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.

requirements

~~SR 3.2.3.2~~

~~Not Used.~~

~~SR 3.2.3.3~~

~~Measurement of the target flux difference is accomplished by taking a flux map when the core is at equilibrium xenon conditions, preferably at high power levels with the control banks nearly withdrawn. This flux map provides the equilibrium xenon axial power distribution from which the target value can be determined. The target flux difference varies slowly with core burnup.~~

~~The target AFD must be determined in conjunction with the measurement of~~ $F_Q^W(Z)$~~; therefore, the frequency for the performance of this surveillance is the same as that required for the performance of the~~ $F_Q^W(Z)$ ~~surveillance per SR 3.2.1.2.~~

~~A Note modifies this SR to allow the predicted beginning of cycle AFD from the Startup and Operations Report to be used to determine the initial target flux difference after each refueling. This note allows operation until the power level for extended operations has been achieved and an equilibrium power distribution can be obtained.~~

REFERENCES

1. RXE-90-006-P-A,"Power Distribution Control Analysis and Overtemperature N-16 and Overpower N-16 Trip Setpoint Methodology," TU Electric, June, 1994.

(continued)

BASES

REFERENCES (continued)

2. WCAP-8385 (W proprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.

3. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC), Attachment: "Operation and Safety Analysis Aspects of an Improved Load Follow Package," January 31, 1980.

4. FSAR, Chapter 7.

5. WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control (and) FQ Surveillance Technical Specification," February 1994.

BASES

ACTIONS

A.1 (continued)

allowable THERMAL POWER level. Decreases in QPTR would allow raising the maximum allowable THERMAL POWER level and increasing THERMAL POWER up to this revised limit.

A.2

After completion of Required Action A.1, the QPTR may still exceed its limits. Any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ and $F_Q(Z)$ within the Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 ensures that these primary indicators of power distribution are within their respective limits. Equilibrium conditions are achieved when the core is sufficiently stable at the intended operating conditions to support ~~flux mapping.~~ A Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a ~~flux map.~~ If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta H}$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

core power distribution measurements

A.4

Although $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction

(continued)

BASES

ACTIONS

A.4 (continued)

accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This evaluation is required to ensure that, before increasing THERMAL POWER to above the limits of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

If the QPTR remains above the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized to restore QPTR to within limit prior to increasing THERMAL POWER to above the limit of Required Action A.1. Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.00. This is done to detect any subsequent significant changes in QPTR.

Required Action A.5 is modified by two Notes. Note 1 states that the excore detectors are not normalized to restore to restore QPTR to within limits until after the evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). Note 2 states that if Required Action A.5 is performed, then Required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limit, which restores compliance with ~~LCO 3.2.4~~. Thus, Note 2 prevents exiting the Actions prior to completing ~~flux mapping~~ to verify peaking factors per Required Action A.6. These notes are intended to prevent any ambiguity about the required sequence of actions.

core power distribution measurements

A.6

Once the excore detectors are normalized to restore QPTR to within limit (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits within 24 hours of achieving equilibrium conditions at RTP. As an added precaution, if the core power does not reach equilibrium conditions at

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is not required until 12 hours after the inputs from one or more Power Range Neutron Flux channels are inoperable and the THERMAL POWER is > 75% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

or an OPERABLE PDMS

When using the moveable incore detector system, t

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors may be used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8.

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full core flux map, to generate an incore QPTR. Therefore, incore monitoring of QPTR can be used to confirm that QPTR is within limits.

an OPERABLE PDMS

core power distribution measurement

With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.

REFERENCES

1. 10 CFR 50.46.

2. Regulatory Guide 1.77, Rev 0, May 1974.

3. 10 CFR 50, Appendix A, GDC 26.

BASES

ACTIONS

D.1.1, D.1.2, and D.2 (continued)

or an OPERABLE PDMS

Required Action D.1.1 has been modified by a Note which only requires SR 3.2.4.2 to be performed if the Power Range Neutron Flux input QPTR becomes inoperable. Failure of a component in the Power Range Neutron Flux Channel which renders the High Flux Trip Function inoperable may not affect the capability to monitor QPTR. As such, determining QPTR using the movable incore detectors once per 12 hours may not be necessary.

The NIS power range detectors provide input to the CRD System and, therefore, have a two-out-of-four trip logic. A known inoperable channel must be placed in the tripped condition. This results in a partial trip condition requiring only one-out-of-three logic for actuation. The 72 hours allowed to place the inoperable channel in the tripped condition is justified in WCAP-14333-P-A (Ref. 11).

As an alternative to the above Actions, the plant must be placed in a MODE where this Function is no longer required OPERABLE. Seventy-eight (78) hours are allowed to place the plant in MODE 3. The 78-hour Completion Time includes 72 hours for channel corrective maintenance, and an additional 6 hours for the MODE reduction as required by Required Action D.2. This is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. If Required Actions cannot be completed within their allowed Completion Times, LCO 3.0.3 must be entered.

The Required Actions are modified by a Note that allows placing one channel in bypass for 12 hours while performing routine surveillance testing, and setpoint adjustments when a setpoint reduction is required by other Technical Specifications. The 12 hour time limit is justified in Reference 11.

E.1 and E.2

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux-Low;
- Overtemperature N-16;
- Overpower N-16;
- Power Range Neutron Flux-High Positive Rate;
- Pressurizer Pressure-High; and
- SG Water Level-Low Low.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 92 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. The Frequency of every 92 days on a STAGGERED TEST BASIS is justified in Reference 12.

SR 3.3.1.6

core power distribution measurement

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~~ core power distribution measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the f(Δq) input to the overtemperature N-16 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 72 hours is allowed for performing the first surveillance after reaching equilibrium conditions at a THERMAL POWER ≥ 75% RTP. The SR is deferred until a scheduled testing plateau above 75% is attained during the post-outage power ascension. During a typical post-refueling power ascension, it is usually necessary to control the axial flux difference at lower power levels through control rod insertion. Due to rod shadowing effects on the base flux map and, to a lesser degree, the dependency of the axially-dependent radial leakage on the power level, a multi-point calibration performed well below 75% RTP may result in excessive incore-excore axial flux difference deviations at full power. After equilibrium conditions are achieved at the specified power plateau, a base flux map must be taken, required AFD swings initiated, and the required data collected. The data is typically analyzed and the appropriate excore calibrations are completed within 48 hours after achieving equilibrium conditions. An additional time allowance of 24 hours is provided during which the effects of equipment failures may be remedied and any required re-testing may be performed.

The allowance of 72 hours after equilibrium conditions are attained at the testing plateau provides sufficient time to allow power ascensions and

(continued)

ATTACHMENT 4 to TXX-07063

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 The Digital Rod Position Indication (DRPI) System and the Demand Position Indication System shall be OPERABLE

APPLICABILITY: MODES 1 and 2.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each inoperable rod position indicator and each demand position indicator per bank.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DRPI per group inoperable for one or more groups.	A.1 Verify the position of the rods with inoperable position indicators indirectly by using core power distribution measurement information.	Once per 8 hours
	OR	
	A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. More than one DRPI per group inoperable.	B.1	Place the control rods under manual control.	Immediately
	AND		
	B.2	Monitor and record RCS T_{avg}.	Once per 1 hour
	AND		
	B.3	Verify the position of the rods with inoperable position indicators indirectly by using core power distribution measurement information.	Once per 8 hours
	AND		
	B.4	Restore inoperable position indicators to OPERABLE status such that a maximum of one DRPI per group is inoperable.	24 hours
C. One or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	C.1	Verify the position of the rods with inoperable position indicators indirectly by using core power distribution measurement information.	4 hours
	OR		
	C.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor (FQ(Z)) (FQ Methodology)

LCO 3.2.1 FQ (Z), as approximated by $FQ^C(Z)$ and $FQ^W(Z)$, shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
----------NOTE---------- Required Action A.4 shall be completed whenever this Condition is entered. ----------			
A. $FQ^C(Z)$ not within limit.	A.1	Reduce THERMAL POWER ≥ 1% RTP for each 1% $FQ^C(Z)$ exceeds limit.	15 minutes after each $FQ^C(Z)$ determination
	AND		
	A.2	Reduce Power Range Neutron Flux C High trip setpoints ≥ 1% for each 1% $FQ^C(Z)$ exceeds limit.	72 hours after each $FQ^C(Z)$ determination
	AND		
	A.3	Reduce Overpower N-16 trip setpoints ≥ 1% for each 1% $FQ^C(Z)$ exceeds limit.	72 hours after each $FQ^C(Z)$ determination
	AND		
	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

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
----------NOTE---------- Required Action B.4 shall be completed whenever this Condition is entered. ----------			
B. $F_Q^W(Z)$ not within limits.	B.1	Reduce AFD limits ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	4 hours
	AND		
	B.2	Reduce Power Range Neutron Flux – High trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced.	72 hours
	AND		
	B.3	Reduce Overpower N-16 trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced.	72 hours
	AND		
	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

SURVEILLANCE REQUIREMENTS

----------NOTE----------
During power escalation following shutdown, THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution measurement is obtained.

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify $F_Q^C(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND Once within 24 hours after achieving equilibrium conditions after exceeding, by ≥ 20% RTP, the THERMAL POWER at which $F_Q^C(Z)$was last verified AND 31 EFPD thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	----------NOTE---------- If $F_Q^C(Z)$ measurements indicate maximum over z $\left[\frac{F_Q^C(Z)}{K(Z)}\right]$ has increased since the previous evaluation of $F_Q^C(Z)$: a. Increase $F_Q^W(Z)$ by an appropriate factor specified in the COLR and reverify $F_Q^W(Z)$ is within limits; or b. Repeat SR 3.2.1.2 once per 7 EFPD until either a. above is met or two successive power distribution measurements indicate maximum over z $\left[\frac{F_Q^C(Z)}{K(Z)}\right]$ has not increased. ----------	
	Verify $F_Q^W(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND (continued)

SURVEILLANCE REQUIREMENTS

----------NOTE----------
During power escalation following shutdown, THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution measurement is obtained.

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND 31 EFPD thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control (RAOC)Methodology)

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 $\geq$ 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 is within limits for each OPERABLE excore channel.	7 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	----------NOTES---------- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. ---------- Verify QPTR is within limit by calculation.	7 days
SR 3.2.4.2	----------NOTE---------- Not required to be performed until 12 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP. ---------- Verify QPTR is within limit using the core power distribution measurement information.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.2	----------NOTES---------- 1. Adjust NIS and N-16 Power Monitor channel if absolute difference is > 2%. 2. Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP. ---------- Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) and N-16 Power Monitor channel output.	24 hours
SR 3.3.1.3	----------NOTES---------- 1. Adjust NIS channel if absolute difference is ≥ 3%. 2. Not required to be performed until 24 hours after THERMAL POWER is ≥ 50% RTP. ---------- Compare results of the core power distribution measurements to NIS AFD.	31 effective full power days (EFPD)
SR 3.3.1.4	----------NOTE---------- This Surveillance must be performed on the reactor trip bypass breaker for the local manual shunt trip only prior to placing the bypass breaker in service. ---------- Perform TADOT.	62 days on a STAGGERED TEST BASIS

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.1.6	----------NOTE---------- Not required to be performed until 72 hours after achieving equilibrium conditions with THERMAL POWER ≥ 75% RTP. ---------- Calibrate excore channels to agree with core power distribution measurements.	92 EFPD
SR 3.3.1.7	----------NOTES---------- 1. Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3. 2. Source range instrumentation shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. ---------- Perform COT.	184 days

(continued)

5.6 Reporting Requirements (continued)

5.6.5 Core Operating Limits Report (COLR) (continued)

allowing use of 100.6 percent of rated power in safety analysis methodology when the LEFM√ is used for feedwater flow measurement.

The approved analytical methods are described in the following documents:

1) WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary)

2) WCAP-10216-P-A, Revision 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary).

3) RXE-90-006-P-A, "Power Distribution Control Analysis and Overtemperature N-16 and Overpower N-16 Trip Setpoint Methodology," June 1994.

4) RXE-88-102-P-A, "TUE-1 Departure from Nucleate Boiling Correlation," July 1992.

5) RXE-88-102-P, Sup. 1, "TUE-1 DNB Correlation - Supplement 1," December 1990.

6) RXE-89-002-A, "VIPRE-01 Core Thermal-Hydraulic Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications," September 1993.

7) RXE-91-001-A, "Transient Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications," October 1993.

8) RXE-91-002-A, "Reactivity Anomaly Events Methodology," October 1993.

9) ERX-2000-002-P, "Revised Large Break Loss of Coolant Accident Analysis Methodology," March 2000.

10) TXX-88306, "Steam Generator Tube Rupture Analysis," March 15, 1988.

11) RXE-91-005-A, "Methodology for Reactor Core Response to Steamline Break Events," February 1994.

12) RXE-94-001-A, "Safety Analysis of Postulated Inadvertent Boron Dilution Event in Modes 3, 4, and 5," February 1994.

13) RXE-95-001-P-A, "Small Break Loss of Coolant Accident Analysis Methodology," September 1996.

14) Caldon, Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power level Using the LEFM√ System," Revision 0, March 1997 and Caldon Engineering Report – 160P, "Supplement to Topical Report ER-80P; Basis for a Power Uprate With the LEFM√™ System," Revision 0, May 2000.

15) ERX-2001-005-P, "ZIRLO™ Cladding and Boron Coating Models for TXU Electric's Loss of Coolant Accident Analysis Methodologies," October 2001.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 Core Operating Limits Report (COLR) (continued)

16) WCAP-10444-P-A, "Reference Core Report VANTAGE 5 Fuel Assembly," September 1985.

17) WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles for Modified LPD Mixing Vane Grids," April 1999.

18) WCAP-13060-P-A, "Westinghouse Fuel Assembly Reconstitution Evaluation Methodology," July, 1993.

19) ERX-04-004-A; "Replacement Steam Generator Supplement To TXU Power's Large and Small Break Loss Of Coolant Accident Analysis Methodologies" Revision 0, March 2007.

20) ERX-04-005-A; "Application of TXU Power's Non-LOCA Transient Analysis Methodologies to a Feed Ring Steam Generator Design" Revision 0, March 2007.

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

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

23) WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

24) WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.

25) WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.

26) WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.

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

28) WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005.

29) WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

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.

(continued)

ATTACHMENT 5 to TXX-07063

RETYPED TECHNICAL SPECIFICATION BASES PAGES

Pages B 3.1-20
B 3.1-23
B 3.1-39
B 3.1-40
B 3.2-1 thru B 3.2-29
B 3.3-32
B 3.3-45

BASES

APPLICABLE SAFETY ANALYSES (continued)

directly by core power distribution measurement. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements (i.e., trippability to meet SDM) are separate from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. A rod is considered OPERABLE based on the last satisfactory performance of SR 3.1.4.2 and has met the rod drop time criteria during the last performance of SR 3.1.4.3. Rod control malfunctions that result in the inability to move a rod (e.g., rod urgent failures), which do not impact trippability within the time requirements of SR 3.1.4.3, do not result in rod inoperability.

The requirement to maintain the rod alignment to within plus or minus 12 steps of their group step counter demand position is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2, because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the rods are typically fully inserted and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect

(continued)

BASES

ACTIONS

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6 (continued)

Verifying that $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, and $F^N_{\Delta H}$ are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain a core power distribution measurement and to calculate $F_Q(Z)$ and $F^N_{\Delta H}$. Once current conditions have been verified acceptable, time is available to perform evaluations of the affected accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. The accident analyses presented in FSAR Chapter 15 (Ref. 3) that may be adversely affected will be evaluated to ensure that the analyses results remain valid for the duration of continued operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

C.1

When Required Actions of Condition B cannot be completed within their Completion Time, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

D.1.1 and D.1.2

More than one control rod becoming misaligned from its group demand position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. Verification of shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.6, "CONTROL BANK INSERTION LIMITS" ensure SDM is maintained provided the misaligned rod is above the insertion limit. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the required pumps. Boration will continue until the required SDM is restored.

(continued)

BASES

ACTIONS (continued)

A.1

When one DRPI per group fails, the position of the rod may still be indirectly determined by use of the incore movable detectors or an OPERABLE PDMS. The Required Action may also be satisfied by ensuring at least once per 8 hours that F_Q satisfies LCO 3.2.1, $F_{\Delta H}$ satisfies LCO 3.2.2, and SHUTDOWN MARGIN is within the limits provided in the COLR, provided the nonindicating rods have not been moved. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of C.1 or C.2 below is required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

A.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 2).

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above.

B.1, B.2, B.3 and B.4

When more than one DRPI per group fail, additional actions are necessary to ensure that acceptable power distribution limits are maintained, minimum SDM is maintained, and the potential effects of rod misalignment on associated accident analyses are limited. Placing the Rod Control System in manual assures unplanned rod motion will not occur. Together with the indirect position determination available via movable incore detectors will minimize the potential for rod misalignment.

The immediate Completion Time for placing the Rod Control System in manual reflects the urgency with which unplanned rod motion must be prevented while in this Condition. Monitoring and recording reactor coolant T_{avg} help assure that significant changes in power distribution and SDM are avoided. The once per hour Completion Time is acceptable because only minor fluctuations in RCS temperature are expected at steady state plant operating conditions.

(continued)

BASES

ACTIONS

B.1, B.2, B.3 and B.4 (continued)

The position of the rods may be determined indirectly by use of the movable incore detectors or an OPERABLE PDMS. The Required Action may also be satisfied by ensuring at least once per 8 hours that F_Q satisfies LCO 3.2.1, $F_{\Delta H}$ satisfies LCO 3.2.2, and SHUTDOWN MARGIN is within the limits provided in the COLR, provided the nonindicating rods have not been moved. Verification of RCCA position once per 8 hours is adequate for allowing continued full power operation for a limited, 24 hour period, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. The 24 hour Completion Time provides sufficient time to troubleshoot and restore the DRPI system to operation while avoiding the plant challenges associated with a shutdown without full rod position indication (Ref. 4).

Based on operating experience, normal power operation does not require excessive rod movement. If one or more rods has been significantly moved, the Required Action of C.1 or C.2 below is required.

C.1 and C.2

These Required Actions clarify that when one or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of A.1 and A.2 or B.3 are still appropriate but must be initiated promptly under Required Action C.1 to begin indirectly verifying that these rods are still properly positioned, relative to their group positions using the movable incore detectors.

If, within 4 hours, the rod positions have not been determined, THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of 4 hours provides an acceptable period of time to verify the rod positions.

D.1.1 and D.1.2

With one demand position indicator per bank inoperable, the rod positions can be determined by the DRPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means (e.g., observation of appropriate DRPI status indications) that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are ≤ 12 steps apart within the allowed Completion Time of once every 8 hours is adequate.

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$) (RAOC-W(Z) Methodology)

BASES

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT TILT POWER RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.7, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is measured periodically using the incore detector system or an OPERABLE PDMS. These measurements are generally taken with the core at or near equilibrium conditions.

Using the measured three dimensional power distributions, it is possible to derive a measured value of $F_Q(Z)$. However, because this value represents an equilibrium condition, it does not include the variations in the value of $F_Q(Z)$ that are present during non-equilibrium situations, such as load following. To account for these possible variations, the steady state value of $F_Q(Z)$ is adjusted by an elevation dependent factor, W(Z), that accounts for calculated worst case transient conditions.

Core monitoring and control under non-steady state 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's principal effect is to preclude core power distributions that could lead to violation of the following fuel design criterion:

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm, and

c. 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.

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 LOCA peak cladding temperature is typically most limiting.

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

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

LCO

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

$$F_Q(Z) \leq \frac{F_Q^C}{P} K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \leq \frac{F_Q^C}{0.5} K(Z) \qquad \text{for } P \leq 0.5$$

where:

F_Q^C 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 = THERMAL POWER/RTP

The actual values of F_Q^C and K(Z) are given in the COLR.

(continued)

BASES

LCO (continued)

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 $F_Q(Z)$.

An $F_Q^C(Z)$ evaluation requires obtaining core power distribution measurement in MODE 1. From the core power distribution measurement results we obtain the measured value ($F_Q^M(Z)$) of $F_Q(Z)$.

If the PDMS is used, the appropriate measurement uncertainty and manufacturing allowance are automatically calculated and applied to the measured F_Q (Ref. 7).

If the movable incore detector system is used, the computed heat flux hot channel factor, $F_Q^C(Z)$, is obtained by the equation:

$$F_Q^C(Z) = F_Q^M(Z) \bullet 1.03 \bullet 1.05.$$

$F_Q^M(Z)$ is increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainties.

$F_Q^C(Z)$ 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:

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

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

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

(continued)

BASES

LCO (continued)

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$. If $F_Q^C(Z)$ cannot be maintained within the LCO limits, a 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 produce unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

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

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^C(Z)$ is $F_Q^M(Z)$ multiplied by factors that account 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.

(continued)

BASES

ACTIONS (continued)

A.2

A reduction of the Power Range Neutron Flux-High 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 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 N-16 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 N-16 trip setpoints initially determined by Required Action A.3 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Overpower N-16 trip setpoint reductions within 72 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Overpower N-16 trip setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum Overpower N-16 trip setpoints.

(continued)

BASES

ACTIONS (continued)

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 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 by ≥ 1% for each 1% by which $F_Q^W(Z)$ exceeds its limit within the allowed Completion Time of 4 hours, restricts the axial flux distribution such that even if a transient occurred, core peaking factor limits are not exceeded.

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, is a conservative action for protection against the consequence 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 reductions in THERMAL POWER as a result of reducing AFD limits in accordance with Required Action B.1.

(continued)

BASES

ACTIONS (continued)

B.3

Reduction in the Overpower N-16 setpoints value 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 reductions 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 maximum allowable power limit imposed by Required Action B.1 ensures that core conditions during operation at higher power levels and future operation are consistent with safety analysis 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 A 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 ensure $F_Q(Z)$ 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 power ascensions following a plant shutdown (leaving Mode 1). The note allows for power ascensions if the surveillances are not current. It states that THERMAL POWER may be increased until an equilibrium power level has

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

been achieved at which a power distribution measurement 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 20% RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because $F_Q^C(Z)$ and $F_Q^W(Z)$ could not have previously been measured for a reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of $F_Q^C(Z)$ and $F_Q^W(Z)$ are 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 20%, 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 20% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 20% higher than that power at which F_Q 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 $F_Q^C(Z)$. If the PDMS is used, the appropriate measurement uncertainty and manufacturing allowance are automatically calculated and applied to the measured F_Q (Ref. 7). If the movable incore detector system is used, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F_Q^C(Z)=F_Q^M(Z)\bullet 1.03\bullet 1.05$ (Ref. 4). $F_Q^C(Z)$ is then compared to its specified limits.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

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 provides assurance that the $F_Q^C(Z)$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

If THERMAL POWER has been increased by ≥ 20% RTP since the last determination of $F_Q^C(Z)$, another evaluation of this factor is required 24 hours after achieving equilibrium conditions at this higher power level (to ensure that $F_Q^C(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits).

The Frequency of 31 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 power distribution measurements are taken at or near equilibrium conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the core power distribution measurement data. These variations are, however, conservatively calculated 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

map data are typically taken for 30 to 75 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.

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. When $F_Q^W(Z)$ is evaluated, an evaluation of the expression below is required to account for any increase to $F_Q^C(Z)$ 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

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

it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by the appropriate factor of ≥ 1.02 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.

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP provides assurance that the $F_Q(Z)$ limit will be met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

$F_Q(Z)$ is verified at power levels ≥ 20% RTP above the THERMAL POWER of its last verification, 24 hours after achieving equilibrium conditions to ensure that $F_Q(Z)$ is within its limit at higher power levels.

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

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.

REFERENCES

1. 10 CFR 50.46, 1974.
2. Regulatory Guide 1.77, Rev. 0, May 1974.
3. 10 CFR 50, Appendix A, GDC 26.
4. RXE-90-006-P-A, "Power Distribution Control Analysis and Overtemperature N-16 and Overpower N-16 Trip Setpoint Methodology," TU Electric, June 1994.
5. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.
6. WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control (and) F_Q Surveillance Technical Specification," February 1994.
7. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during normal operation, operational transients, and any transient condition arising from events of moderate frequency analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod. $F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution measurement obtained with the movable incore detector system or an OPERABLE PDMS. Specifically, the results of the three dimensional power distribution measurement are analyzed to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables. Compliance with these LCOs, along with the LCOs governing shutdown and control rod insertion and alignment, maintains the core limits on power distribution on a continuous basis.

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

Operation outside the LCO limits may produce unacceptable consequences

(continued)

BASES

BACKGROUND (continued)

if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

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

a. For ANS Condition II events, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;

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

c. During an ejected rod accident, the average fuel pellet enthalpy at the hot spot must not exceed 280 cal/gm (Ref. 1); and

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

The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion applicable to a specific DNBR correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB condition.

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

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

$F^N_{\Delta H}$ satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

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

The $F^N_{\Delta H}$ limit is representative of the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a DNB condition.

The limiting value of $F^N_{\Delta H}$ described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.

A power multiplication factor in this equation includes an additional allowance for higher radial peaking factors from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase by a cycle-dependent factor ($PF_{\Delta H}$, as specified

(continued)

BASES

LCO (continued)

in the COLR) for a 1% RTP reduction in THERMAL POWER.

If the power distribution measurements are performed at a power level less than 100% RTP, then the $F^N_{\Delta H}$ values that would result from measurements if the core was at 100% RTP should be inferred from the available information. A comparison of these inferred values with $F_{\Delta H}^{RTP}$ assures compliance with the LCO at all power levels.

APPLICABILITY

The $F^N_{\Delta H}$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power.

ACTIONS

A.1.1

With $F^N_{\Delta H}$ exceeding its limit, the unit is allowed 4 hours to restore $F^N_{\Delta H}$ to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power dependent limit. When the $F^N_{\Delta H}$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F^N_{\Delta H}$ value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time. The restoration of the peaking factor to within its limits by power reduction or control rod movement does not restore compliance with the LCO. Thus, this condition can not be exited until a valid surveillance demonstrates compliance with the LCO.

Condition A is modified by a Note that requires that Required Actions A.2 and A.3 must be completed whenever Condition A is entered. Thus, even if this Required Action is completed within the 4 hour time period, Required

(continued)

BASES

ACTIONS

A.1.1 (continued)

Action A.2 requires another measurement and calculation of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2.1.

Required Action A.3 requires that another determination of $F^N_{\Delta H}$ must be done prior to exceeding 50% RTP, prior to exceeding 75% RTP, and within 24 hours after reaching or exceeding 95% RTP; however, THERMAL POWER does not have to be reduced to comply with these requirements. In addition, Required Action A.2 is performed if power ascension is delayed past 24 hours.

A.1.2.1 and A.1.2.2

If the value of $F^N_{\Delta H}$ is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action A.1.2.1 and reduce the Power Range Neutron Flux-High to ≤ 55% RTP in accordance with Required Action A.1.2.2. Reducing power to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 4 hours for Required Action A.1.2.1 is consistent with those allowed for in Required Action A.1.1 and provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The Completion Times of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive.

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints; however, for extended operations at the reduced power level, the reduced trip setpoints are required to protect against events involving positive reactivity excursions. This is a sensitive operation that may inadvertently trip the Reactor Protection System.

A.2

Once actions have been taken to restore $F^N_{\Delta H}$ to within its limits per

(continued)

BASES

ACTIONS

A.2 (continued)

Required Action A.1.1, or the power level has been reduced to < 50% RTP per Required Action A.1.2.1, core power distribution measurement (SR 3.2.2.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by either Action A.1.1 or Action A.1.2.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period.

Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the core power distribution measurement, perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.3

Verification that $F^N_{\Delta H}$ is within its specified limits after an out of limit occurrence ensures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is identified, to the extent necessary, and corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP. SR 3.2.2.1 must be satisfied prior to increasing power above the extrapolated allowable power level or restoration of any reduced Reactor Trip System setpoints. When $F^N_{\Delta H}$ is measured at reduced power levels, the allowable power level is determined by evaluating $F^N_{\Delta H}$ for higher power levels.

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action.

B.1

When Required Actions A.1.1 through A.3 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at

(continued)

BASES

ACTIONS

B.1 (continued)

least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

SR 3.2.2.1 is modified by a Note. The Note applies during power ascensions following a plant shutdown (leaving Mode 1). The note allows for power ascensions if the surveillances are not current. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution measurement can be obtained. Equilibrium conditions are achieved when the core is sufficiently stable such that the uncertainty allowances associated with the measurement are valid.

The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system or an OPERABLE PDMS to obtain a power distribution measurement. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. If the PDMS is used, the appropriate measurement uncertainty is automatically calculated and applied to the measured $F^N_{\Delta H}$ (Ref. 4). If the moveable incore detector system is used, the measured value of $F^N_{\Delta H}$ must be multiplied by 1.04 to account for measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.

After each refueling, $F^N_{\Delta H}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle. Performing this Surveillance in MODE 1 prior to exceeding 75% RTP, or at a reduced power level at any other time, and meeting the 100% RTP $F^N_{\Delta H}$ limit, provides assurance that the $F^N_{\Delta H}$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F^N_{\Delta H}$ limit cannot be exceeded for any significant period of operation.

(continued)

BASES (continued)

REFERENCES

1. Regulatory Guide 1.77, Rev. 0, May 1974.
2. 10 CFR 50, Appendix A, GDC 26.
3. 10 CFR 50.46.
4. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control (RAOC) Methodology)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the 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.

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 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 AFDs 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 to or more OPERABLE excore channels is outside its specified limits.

Although the RAOC defines limits that must be met to satisfy safety analyses, typically an operating scheme, Constant Axial Offset Control (CAOC), is used to control axial power distribution in day to day operation (Ref. 2). CAOC requires that the AFD be controlled within a narrow tolerance band around a burnup dependent target to minimize the variation of axial peaking factors and axial xenon distribution during unit maneuvers.

The CAOC operating space is typically smaller and lies within the RAOC operating space. Control within the CAOC operating space constrains the variation of axial xenon distributions and axial power distributions. RAOC calculations assume a wide range of xenon distributions and then confirm that the resulting power distributions satisfy the requirements of the accident analyses.

APPLICABLE SAFETY ANALYSES

The AFD is a measure of the 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,

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

axial xenon distribution, and, to a lesser extent, reactor coolant temperature and boron concentration.

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. 5) 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 loss of coolant accident 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 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 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. Compliance with this limit ensures that acceptable levels of 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 complete loss of forced RCS flow accident. The most significant Condition II events are uncontrolled bank withdrawal and boration or dilution accidents. Condition II accidents are used to confirm the adequacy of Overpower N-16 and Overtemperature N-16 trip setpoints.

The limits on the AFD satisfy Criterion 2 of the 10CFR50.36(c)(2)(ii).

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 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 (Ref. 4). 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.

(continued)

BASES

LCO (continued)

The AFD limits are provided in the COLR. The AFD limits for RAOC do not depend on the target flux difference. However, the target flux difference may be used to minimize changes in the axial power distribution.

Violating the LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its 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 the core parameters of primary importance in safety analyses (Ref. 1).

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.

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 channels is within its specified limits. The Surveillance Frequency of 7 days is adequate because the AFD is controlled by the operator and monitored by the process computer. Furthermore, any deviation of the AFD from requirements that is not alarmed should be readily noticed.

REFERENCES

1. RXE-90-006-P-A,"Power Distribution Control Analysis and Overtemperature N-16 and Overpower N-16 Trip Setpoint Methodology," TU Electric, June, 1994.

2. WCAP-8385 (W proprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.

(continued)

BASES

REFERENCES (continued)

3. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC), Attachment: "Operation and Safety Analysis Aspects of an Improved Load Follow Package," January 31, 1980.

4. FSAR, Chapter 7.

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.7, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

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, the peak cladding temperature must not exceed 2200°F (Ref. 1);

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

c. During an ejected rod accident, the average fuel pellet enthalpy at the hot spot must not exceed 280 cal/gm (Ref. 2); and

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

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure that $F^N_{\Delta H}$ and $F_Q(Z)$ remain below their limiting

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

values by preventing an undetected change in the gross radial power distribution.

In MODE 1, the $F^N_{\Delta H}$ and $F_Q(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

The QPTR satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

The QPTR limit of 1.02, above which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and ($F^N_{\Delta H}$) is possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1 ≤ 50% RTP and 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 the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition. The maximum allowable THERMAL POWER level initially determined by Required Action A.1 may be affected by subsequent determinations of

(continued)

BASES

ACTIONS

A.1 (continued)

QPTR. Increases in QPTR would require a THERMAL POWER reduction within 2 hours of QPTR determination, if necessary to comply with the decreased maximum allowable THERMAL POWER level. Decreases in QPTR would allow raising the maximum allowable THERMAL POWER level and increasing THERMAL POWER up to this revised limit.

A.2

After completion of Required Action A.1, the QPTR may still exceed its limits. Any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ and $F_Q(Z)$ within the Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 ensures that these primary indicators of power distribution are within their respective limits. Equilibrium conditions are achieved when the core is sufficiently stable at the intended operating conditions to support core power distribution measurements. A Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a core power distribution measurement. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta H}$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the

(continued)

BASES

ACTIONS

A.4 (continued)

safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This evaluation is required to ensure that, before increasing THERMAL POWER to above the limits of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

If the QPTR remains above the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized to restore QPTR to within limit prior to increasing THERMAL POWER to above the limit of Required Action A.1. Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.00. This is done to detect any subsequent significant changes in QPTR.

Required Action A.5 is modified by two Notes. Note 1 states that the excore detectors are not normalized to restore to restore QPTR to within limits until after the evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). Note 2 states that if Required Action A.5 is performed, then Required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limit, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing core power distribution measurements to verify peaking factors per Required Action A.6. These notes are intended to prevent any ambiguity about the required sequence of actions.

A.6

Once the excore detectors are normalized to restore QPTR to within limit (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits within 24 hours of achieving equilibrium conditions at RTP. As an

(continued)

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ACTIONS

A.6 (continued)

added precaution, if the core power does not reach equilibrium conditions at RTP within 24 hours but is increased slowly, then the peaking factor surveillances must be performed within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1. This Completion Time is intended to allow adequate time to increase THERMAL POWER to above the limits of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.6 is modified by a Note that states that the peaking factor surveillances must be completed when the excore detectors have been normalized to restore QPTR to within limit (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are normalized to restore QPTR to within limit.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to < 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is ≤ 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. The Frequency of 7 days takes into account other information and alarms available to the operator in the control room.

For those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is not required until 12 hours after the inputs from one or more Power Range Neutron Flux channels are inoperable and the THERMAL POWER is > 75% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors or an OPERABLE PDMS may be used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. When using the moveable incore detector system, the incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8.

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full core flux map, to generate an incore QPTR. Therefore, incore monitoring of QPTR can be used to confirm that QPTR is within limits.

With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous core power distribution measurement using an OPERABLE PDMS the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.

REFERENCES

1. 10 CFR 50.46.

2. Regulatory Guide 1.77, Rev 0, May 1974.

3. 10 CFR 50, Appendix A, GDC 26.

BASES

ACTIONS

D.1.1, D.1.2, and D.2 (continued)

Required Action D.1.1 has been modified by a Note which only requires SR 3.2.4.2 to be performed if the Power Range Neutron Flux input QPTR becomes inoperable. Failure of a component in the Power Range Neutron Flux Channel which renders the High Flux Trip Function inoperable may not affect the capability to monitor QPTR. As such, determining QPTR using the movable incore detectors or an OPERABLE PDMS once per 12 hours may not be necessary.

The NIS power range detectors provide input to the CRD System and, therefore, have a two-out-of-four trip logic. A known inoperable channel must be placed in the tripped condition. This results in a partial trip condition requiring only one-out-of-three logic for actuation. The 72 hours allowed to place the inoperable channel in the tripped condition is justified in WCAP-14333-P-A (Ref. 11).

As an alternative to the above Actions, the plant must be placed in a MODE where this Function is no longer required OPERABLE. Seventy-eight (78) hours are allowed to place the plant in MODE 3. The 78-hour Completion Time includes 72 hours for channel corrective maintenance, and an additional 6 hours for the MODE reduction as required by Required Action D.2. This is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. If Required Actions cannot be completed within their allowed Completion Times, LCO 3.0.3 must be entered.

The Required Actions are modified by a Note that allows placing one channel in bypass for 12 hours while performing routine surveillance testing, and setpoint adjustments when a setpoint reduction is required by other Technical Specifications. The 12 hour time limit is justified in Reference 11.

E.1 and E.2

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux-Low;
- Overtemperature N-16;
- Overpower N-16;
- Power Range Neutron Flux-High Positive Rate;
- Pressurizer Pressure-High; and
- SG Water Level-Low Low.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 92 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. The Frequency of every 92 days on a STAGGERED TEST BASIS is justified in Reference 12.

SR 3.3.1.6

SR 3.3.1.6 is a calibration of the excore channels to the core power distribution measurement. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the core power distribution measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the f(Δq) input to the overtemperature N-16 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 72 hours is allowed for performing the first surveillance after reaching equilibrium conditions at a THERMAL POWER ≥ 75% RTP. The SR is deferred until a scheduled testing plateau above 75% is attained during the post-outage power ascension. During a typical post-refueling power ascension, it is usually necessary to control the axial flux difference at lower power levels through control rod insertion. Due to rod shadowing effects on the base flux map and, to a lesser degree, the dependency of the axially-dependent radial leakage on the power level, a multi-point calibration performed well below 75% RTP may result in excessive incore-excore axial flux difference deviations at full power. After equilibrium conditions are achieved at the specified power plateau, a base flux map must be taken, required AFD swings initiated, and the required data collected. The data is typically analyzed and the appropriate excore calibrations are completed within 48 hours after achieving equilibrium conditions. An additional time allowance of 24 hours is provided during which the effects of equipment failures may be remedied and any required re-testing may be performed.

The allowance of 72 hours after equilibrium conditions are attained at the testing plateau provides sufficient time to allow power ascensions and

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