Dominion Energy Kewaunee, Inc.
5000 Dominion Boulevard, Glen Allen, VA 23060

Dominion

September 24, 2007

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

Serial No. 07-0403
NL&OS/CDS: R8
Docket No. 50-305
License No. DPR-43

DOMINION ENERGY KEWAUNEE, INC.
KEWAUNEE POWER STATION
LICENSE AMENDMENT REQUEST 228
INCORPORATION OF DOMINION NUCLEAR ANALYSIS AND FUEL TOPICAL REPORT DOM-NAF-5 INTO KEWAUNEE TECHNICAL SPECIFICATIONS

Pursuant to 10 CFR 50.90, Dominion Energy Kewaunee, Inc. (DEK) requests an amendment to facility operating license number DPR-43 for Kewaunee Power Station (KPS). The proposed amendment would add a reference to Dominion Topical Report DOM-NAF-5, "Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS)," to the KPS Technical Specification (TS) list of approved analytical methods. The proposed amendment would also:

- Change the TS to accommodate use of methodologies proposed in DOM-NAF-5.
- Delete one approved analytical method in the KPS TS that will no longer be used.
- Delete date and revision numbers from the current TS list of approved analytical methods, consistent with TSTF 363-A (reference 5).

DOM-NAF-5 was submitted for NRC review and approval by letters dated August 16, 2006, December 6, 2006, and April 16, 2007 (references 1, 2, and 3). By letter dated May 4, 2007, DEK also submitted the KPS plant-specific application of the NRC approved Dominion Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," for KPS cores containing Westinghouse 422 V+ fuel assemblies with the VIPRE-D/WRB-1 code correlation (reference 4).

The proposed changes will permit application of the above Dominion nuclear core design and safety analysis methods to KPS, including the methodology to perform core thermal-hydraulic analysis to predict critical heat flux and departure from nucleate boiling ratio for the Westinghouse 422 V+ fuel design. DOM-NAF-5 and the proposed DNBR statistical design limit have been evaluated and approved by the NRC staff (reference 6).

Attachment 1 to this letter contains a description, a safety evaluation, a significant hazards determination, and environmental considerations for the proposed amendment. Attachment 2 contains the marked-up Kewaunee Technical Specification pages. Attachment 3 contains the proposed new Kewaunee Technical Specification pages.

DEK has evaluated the proposed amendment and has concluded that it does not involve a significant hazards consideration as defined in 10 CFR 50.92. The basis for the determination that the proposed changes do not involve a significant hazards consideration is included in attachment 1.

The proposed amendment and supporting evaluation have been reviewed and approved by the KPS Plant Operating Review Committee. A copy of this submittal has been provided to the State of Wisconsin in accordance with 10 CFR 50.91(b).

DEK requests NRC review and approval of this LAR by January 31, 2008. The requested LAR approval date supports application of DOM-NAF-5 to KPS cycle 29, which is scheduled to begin in April 2008. Once approved, the amendment will be implemented within 60 days.

If you have any questions or require additional information, please contact Mr. Craig D. Sly at 804-273-2784.

Very truly yours,

Gerald T. Bischof
Vice President – Nuclear Engineering

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Gerald T. Bischof, who is Vice President, Nuclear Engineering of Dominion Energy Kewaunee, Inc. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 24th day of September, 2007.

My Commission Expires: May 31, 2010.

Vicki L. Hull
Notary Public

VICKI L. HULL
Notary Public
Commonwealth of Virginia
140842
My Commission Expires May 31, 2010

References:

1. Letter from G. T. Bischof (DEK) to NRC, "Request for Approval of Topical Report DOM-NAF-5, 'Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS),'" dated August 16, 2006.
2. Letter from G. T. Bischof (DEK) to NRC, "Attachment A to Topical Report DOM-NAF-5, 'Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS),'" dated December 6, 2006.
3. Letter from G. T. Bischof (DEK) to NRC, "Request for Approval of Topical Report DOM-NAF-5, "Application of Dominion Nuclear Design and Safety Analysis Methods to the Kewaunee Power Station (KPS)," dated April 16, 2007.
4. Letter from G. T. Bischof (DEK) to NRC, "Implementation of the Dominion Statistical DNBR Methodology with VIPRE-D/WRB-1 at Kewaunee Power Station," dated May 4, 2007.
5. Technical Specification Task Force Improved Standard Technical Specifications Change Traveler (TSTF) 363-A, Revision 0, "Revise Topical Report references in ITS 5.6.5, COLR," dated August 4, 2003.
6. Letter from P. D. Milano (NRC) to D. A. Christian (DEK), "Kewaunee Power Station – Safety Evaluation for Topical Report DOM-NAF-5 (TAC No. MD2829)," dated August 30, 2007.

Attachments:

1. Discussion of Change, Safety Evaluation, Significant Hazards Determination and Environmental Considerations
2. Marked-up Technical Specification Pages
3. Affected Technical Specification Pages
4. Marked-up Technical Specification Bases Pages
5. Affected Technical Specification Bases Pages

Commitments made in this letter: None

cc: Regional Administrator
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ATTACHMENT 1

LICENSE AMENDMENT REQUEST 228

INCORPORATION OF DOMINION NUCLEAR ANALYSIS AND FUEL TOPICAL REPORT DOM-NAF-5 INTO KEWAUNEE TECHNICAL SPECIFICATIONS

DISCUSSION OF CHANGE, SAFETY EVALUATION, SIGNIFICANT HAZARDS DETERMINATION AND ENVIRONMENTAL CONSIDERATIONS

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, Dominion Energy Kewaunee, Inc. (DEK) requests an amendment to facility operating license number DPR-43 for Kewaunee Power Station (KPS). The proposed amendment would add a new analytical method, Dominion Topical Report DOM-NAF-5, "Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS)," to the KPS Technical Specification (TS) list of approved analytical methods used to determine core operating limits. In addition, the proposed amendment would:

- Delete one methodology in the current TS list of approved analytical methods that will no longer be used.
- Change TS 3.10.b, "Power Distribution Limits," to reference a more generic nomenclature for height-dependent hot channel factor in place of the current Westinghouse method-specific nomenclature.
- Change TS 2.1, "Safety Limits - Reactor Core," Specification b, which currently specifies the departure from nucleate boiling ratio (DNBR) be maintained above certain specific limits, to require DNBR be maintained greater than the 95/95 DNBR criterion developed with the methodologies in TS 6.9.a.4, "Core Operating Limits Report," (i.e. DOM-NAF-5).
- Delete date and revision numbers from the current TS list of approved analytical methods, consistent with Technical Specification Task Force Traveler (TSTF) 363-A (reference 5).

Incorporating DOM-NAF-5 into the TS list of approved analytical methods in conjunction with the other changes proposed in this amendment will permit the application of approved Dominion nuclear core design and safety analysis methods to KPS. These safety analysis methods include the methodology to perform core thermal-hydraulic analysis to predict critical heat flux and departure from nucleate boiling ratio for the Westinghouse 422 V+ fuel design.

Approval of the proposed amendment is requested prior to January 31, 2008 in order to support application of the methods contained in DOM-NAF-5 to the KPS cycle 29 core.

2.0 DETAILED DESCRIPTION

2.1 Changes to TS 6.9.a.4, "Core Operating Limits Report"

The proposed amendment would change the list of approved analytical methods contained in KPS TS 6.9.a.4.B. KPS TS 6.9.a.4.B states that the analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. TS 6.9.a.4.B also provides a list of NRC approved analytical methods for KPS.

2.1.1 Add DOM-NAF-5 to TS List of Approved Methodologies and Delete Old Methodology

The proposed amendment would add Dominion Topical Report DOM-NAF-5 to the KPS TS 6.9.a.4.B list of approved methodologies. DOM-NAF-5 documents justification for application of Dominion nuclear core design and safety analysis methods to KPS. This topical report describes Dominion core design and safety analysis methods and documents assessments of the applicability of Dominion nuclear core design and safety analysis methods to KPS. The new analytical method would read as follows:

"(16) Topical Report DOM-NAF-5-A, "Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS)."

The proposed amendment would also modify KPS 6.9.a.4.B by deleting an existing analytical method for determining core operating limits. This analytical method will no longer be used for KPS. The description of the deleted analytical method will be replaced by the word "Deleted." The analytical method that would be deleted is as follows:

"(1) Safety Evaluation by the Office of Nuclear Reactor Regulation on "Qualifications of Reactor Physics Methods For Application To Kewaunee" Report, dated August 21, 1979, report date September 29, 1978."

2.1.2 Delete Date and Revision Information from List of Approved Methodologies

Consistent with TSTF 363-A (reference 5), the proposed amendment would delete method revision numbers and dates from the current list of approved methodologies and add a new TS 6.9.a.4.E, which states the following:

"E. The COLR will contain the complete identification of the TS approved analytical methods used to prepare the COLR (i.e. report number, title, revision, date, and any supplements)."

2.2 Change the Nomenclature for Height Dependent Hot Channel Factor

The current KPS TS 3.10.b, "Power Distribution Limits" would be revised to change the nomenclature for the height dependent hot channel factor (F_Q^{EQ} (Z)) to a more generic nomenclature (F_Q^N (Z)) for this core surveillance parameter. The more generic nomenclature (F_Q^N (Z)) would allow application of either Westinghouse (RAOC) or Dominion (RPDC) power distribution control and analysis methods to the KPS core. This proposed change affects TS 3.10.b.3.C, TS 3.10.b.5, TS 3.10.b.6, TS 3.10.b.6.C.i, TS 3.10.b.6.C.ii, TS 3.10.b.7, TS 3.10.b.7.A, TS 3.10.b.7.C, TS 6.9.a.4.A(9) and TS 6.9.a.4.A(10).

2.3 Change DNBR Limit from Specified Value to 95/95 Criterion

TS 2.1, "Safety Limits - Reactor Core," Specification b, would be revised to allow for the application of either Westinghouse or Dominion DNBR analysis methods to the KPS core.

TS 2.1.b currently reads as follows:

> *"The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.14 for the HTP DNB correlation and 1.17 for the WRB-1 DNB correlation."*

TS 2.1.b would be revised to the following:

> *"The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ the 95/95 DNBR criterion for the DNB correlations and methodologies specified in Section 6.9."*

The applicable affected and marked-up changes to the Kewaunee Technical Specifications Bases are provided in attachments 4 and 5.

3.0 TECHNICAL EVALUATION

3.1 Add DOM-NAF-5 to TS List of Approved Methodologies

DOM-NAF-5 describes the Dominion nuclear core design and safety analysis methods that will be applied to KPS. The addition of DOM-NAF-5 as a new analytical method would permit the use of Dominion analysis methodologies to perform nuclear core design and safety analyses for KPS. DOM-NAF-5 also encompasses use of Dominion's core thermal-hydraulic analysis methods to predict critical heat flux (CHF) and departure from nucleate boiling ratio (DNBR) for the KPS fuel (Westinghouse 422 V+ fuel design).

DOM-NAF-5 was previously submitted for NRC review and approval by letters dated August 16, 2006, December 6, 2006, and April 16, 2007 (references 1, 2, and 3). In addition, the KPS specific application of the core thermal hydraulic analysis methodology also required NRC approval of the Westinghouse 422 V+ fuel specific DNBR statistical design limit (SDL). By letter dated May 4, 2007 (reference 4), DEK submitted the KPS plant specific application of the NRC approved Dominion Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," for KPS cores containing Westinghouse 422 V+ fuel assemblies with the VIPRE-D/WRB-1 code correlation.

Dominion currently applies its own nuclear core design and safety analysis methods to its nuclear power stations, while the fuel vendor is responsible for fuel design analyses and reload fuel performance assessments. Dominion has performed reload design and safety analyses for approximately 65 reload cores at its Surry and North Anna units using both vendor and Dominion-developed analysis tools. Dominion will apply the nuclear core design and safety analysis methods described in DOM-NAF-5 to KPS in the same manner it applies these methods to the other plants in its nuclear fleet. The KPS fuel vendor will retain responsibility for licensing the fuel design, performing fuel rod design analysis, and for reload fuel performance assessment. The fuel vendor will also perform certain specific safety analyses for KPS (e.g. small break and large break LOCA analyses).

The Dominion nuclear core design methods within the scope of DOM-NAF-5 that will be applied to KPS are as follows:

a) VEP-FRD-42 (current version: VEP-FRD-42, Revision 2.1-A), "Reload Nuclear Design Methodology."
b) VEP-NE-1 (current version: VEP-NE-1, Revision 0.1-A), "Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications."
c) DOM-NAF-1 (current version: DOM-NAF-1, Revision 0.0-P-A), "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations."

The Dominion safety analysis methods within the scope of DOM-NAF-5 that will be applied to KPS are as follows:

d) VEP-FRD-41 (current version: VEP-FRD-41, Revision 0.1-A), "VEPCO Reactor System Transient Analyses Using the RETRAN Computer Code."
e) VEP-NE-2 (current version: VEP-NE-2-A), "Statistical DNBR Evaluation Methodology."
f) DOM-NAF-2 (current version: DOM-NAF-2, Revision 0.0-A), "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code."

Dominion intends to apply the DOM-NAF-5 nuclear core design and safety analysis methods to KPS. Each of the above methods was assessed for applicability to KPS (reference 1). As described in DOM-NAF-5, these methods were determined to be applicable to KPS, and can be employed in design and licensing analyses for KPS.

Kewaunee TS 6.9.a.4.B states that the analytical methods used to determine core operating limits shall be those previously reviewed and approved by NRC. This TS also provides a list of NRC-approved analytical methods for KPS. Therefore, in order for DEK to use the analytical methods described in DOM-NAF-5, this topical report must be added to the list in TS 6.9.a.4.B. By letter dated August 30, 2007, the NRC staff approved the use of DOM-NAF-5 and the proposed DNBR statistical design limit in licensing applications for KPS (reference 6).

3.2 Delete Date and Revision Information from List of Approved Methodologies

Each of the currently listed methods in KPS TS 6.9.a.4.B, "Core Operating Limits Report," contains a complete description of the method, including date and revision number. The proposed amendment would delete the revision numbers and dates and add a new specification TS 6.9.a.4.E that requires the COLR contain a complete identification of each TS approved analytical method used to prepare the COLR.

This proposed change is consistent with TSTF 363-A, Revision 0 (reference 5). This method of referencing the analytical methods would allow KPS to use current methods to support limits in the COLR without having to submit an amendment to the facility operating license each time a method is revised. The COLR would continue to provide a complete identification (including report number, title, revision, date, and any supplements) of the particular approved method used to determine the core limits for the particular cycle in the COLR report. This change would eliminate unnecessary processing of TS submittals to support fuel reload.

3.3 Change the Nomenclature for Height-Dependent Hot Channel Factor and Change DNBR Limit from Specified Value to 95/95 Criterion

KPS TS 3.10.b "Power Distribution Limits" would also be revised to change the nomenclature for the height dependent hot channel factor F_Q^{EQ} (Z) to a more generic nomenclature F_Q^N (Z) for this core surveillance parameter. The more generic nomenclature F_Q^N (Z) would also allow application of either Westinghouse (RAOC) or Dominion (RPDC) power distribution control and analysis methods to the KPS core. This change maintains the distinction inherent in the current TS 3.10.b between equilibrium and transient hot channel factors without encumbering the description with vendor-specific nomenclature.

A change to TS 2.1, "Safety Limits – Reactor Core," is proposed. TS 2.1 currently specifies the departure from nucleate boiling ratio (DNBR) be maintained above certain specific limits (greater than or equal to 1.14 for the HTP DNB correlation and 1.17 for the WRB-1 DNB correlation). These specific limits would be replaced with a citation of more generic functional statement (i.e. the 95/95 DNBR criterion). Specifying that DNBR must be maintained greater than or equal to the 95/95 DNBR criterion for the DNB methodologies specified in TS 6.9 would allow for the application of either Westinghouse or Dominion DNBR analysis methods to the KPS core.

The applicable Westinghouse methodologies have been previously approved for use at KPS. By letter dated August 30, 2007, the NRC staff approved the use of the corresponding Dominion methodologies (reference 6). Therefore, based on the fact that these methods have been approved by the NRC, these changes are considered acceptable.

3.4 Conclusions

Therefore, based on the fact that NRC has approved Topical Report DOM-NAF-5, which includes VEP-NE-2-A, for use at KPS, addition of DOM-NAF-5 to the list of approved analytical methods in TS 6.9.a.4.B and the proposed changes to TS 2.1 and 3.10.b are considered acceptable.

4.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.90, Dominion Energy Kewaunee, Inc. (DEK) is requesting an amendment to facility operating license number DPR-43 for Kewaunee Power Station (KPS). The proposed amendment would add Dominion Topical Report DOM-NAF-5, "Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS)," to the KPS Technical Specification (TS) list of approved analytical methods for determining core operating limits. The amendment would also delete one existing methodology in the current TS list of approved analytical methods that is no longer used and make other changes necessary to allow effective implementation of DOM-NAF-5. In addition, the proposed change would delete date and revision numbers from the current TS list of approved analytical methods, consistent with Technical Specification Task Force Improved Standard Technical Specifications Change Traveler (TSTF) 363-A, "Revise Topical Report References in ITS 5.6.5 (COLR)."

Adding DOM-NAF-5 to the KPS TS will permit the application of approved Dominion nuclear core design and safety analysis methods to KPS including the methodology to perform core thermal-hydraulic analysis to predict critical heat flux and departure from nucleate boiling ratio for the Westinghouse 422 V+ fuel design.

DOM-NAF-5 was submitted for NRC review and approval by letters dated August 16, 2006, December 6, 2006 and April 16, 2007. In addition on May 4, 2007, DEK submitted the KPS plant specific application of the NRC approved Dominion Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," for KPS cores containing Westinghouse 422 V+ fuel assemblies with the VIPRE-D/WRB-1 code correlation. These submittals have been reviewed and approved by the NRC staff and Topical Report DOM-NAF-5 and the proposed statistical design limit has been found acceptable for use in licensing applications for KPS.

4.1 Significant Hazards Consideration

Dominion Energy Kewaunee 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 10 CFR 50.92 (c), "Issuance of amendment," as discussed below:

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

   Response: No

   The analysis methods of DOM-NAF-5 do not make any contribution to the potential accident initiators and thus do not increase the probability of any accident previously evaluated. The use of the approved Dominion analysis methodologies will not increase the probability of an accident because plant systems, structures, and components (SSC) will not be affected or operated in a different manner, and system interfaces will not change.

   Since the applicable safety analysis and nuclear core design acceptance criteria will be satisfied when the Dominion analysis methods are applied to KPS, the use of the approved Dominion analysis methods does not increase the potential consequences of any accident previously evaluated. The use of the approved Dominion methods will not result in a significant impact on normal operating plant releases, and will not increase the predicted radiological consequences of postulated accidents described in the USAR.

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

2. Does the proposed amendment create the possibility of a new or different type of accident from any accident previously evaluated?

   Response: No

   The use of Dominion analysis methods and the Dominion statistical design limit (SDL) for fuel departure from nucleate boiling ratio (DNBR) and fuel critical heat flux (CHF) does not impact any of the applicable core design criteria. All pertinent licensing basis limits and acceptance criteria will continue to be met. Demonstrated adherence to these limits and acceptance criteria precludes new challenges to SSCs that might introduce a new type of accident. All design and performance criteria will continue to be met and no new single failure mechanisms will be created. The use of the Dominion methods does not involve any alteration to plant equipment or procedures that might introduce any new or unique operational modes or accident precursors.

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

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

   Response: No

Nuclear core design and safety analysis acceptance criteria will continue to be satisfied with the application of Dominion methods. Meeting the analysis acceptance criteria and limits ensures that the margin of safety is not significantly reduced. Nuclear core design and safety analysis acceptance criteria will continue to be satisfied with the application of Dominion methods. In particular, use of VIPRE-D with the proposed SDL provides at least a 95% probability at a 95% confidence level that DNBR will not occur (the 95/95 DNBR criterion). The required DNBR margin of safety for KPS, which is the margin between the 95/95 DNBR criterion and clad failure, is therefore not reduced.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above, DEK concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

4.2 Applicable Regulatory Requirements/Criteria

10 CFR 50.36, *Technical specifications*, paragraph (c) (5) states that technical specifications will include administrative controls. Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

4.3 Precedent

The portion of this license amendment request that deletes date and revision information of approved methodologies is consistent with Technical Specification Task Force Improved Standard Technical Specifications Change Traveler (TSTF) 363-A, "Revise Topical Report references in ITS 5.6.5, COLR" (reference 5). In addition, the proposed change to TS 2.1, Safety Limits, describes the DNBR safety limit with language which is consistent with the content in Section 2.1 of the North Anna Unit 1 and 2 Technical Specifications.

4.4 Conclusions

In conclusion, based on the considerations discussed above:

1. There is reasonable assurance that the health and safety of the public will not be endangered by 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.

5.0 ENVIRONMENTAL ASSESSMENT

The proposed amendment is confined to (i) changes to surety, insurance, and/or indemnity requirements, or (ii) changes to recordkeeping, reporting or administrative procedures or requirements. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(10). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. Letter from G. T. Bischof (DEK) to NRC, "Request for Approval of Topical Report DOM-NAF-5, 'Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS),'" dated August 16, 2006 (ADAMS Accession No. ML062370351).
2. Letter from G. T. Bischof (DEK) to NRC, "Attachment A to Topical Report DOM-NAF-5, 'Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS),'" dated December 6, 2006 (ADAMS Accession No. ML063410177).
3. Letter from G. T. Bischof (DEK) to NRC, "Request for Approval of Topical Report DOM-NAF-5, "Application of Dominion Nuclear Design and Safety Analysis Methods to the Kewaunee Power Station (KPS)," dated April 16, 2007 (ADAMS Accession No. ML071060392).
4. Letter from G. T. Bischof (DEK) to NRC, "Implementation of the Dominion Statistical DNBR Methodology with VIPRE–D/WRB–1 at Kewaunee Power Station," dated May 4, 2007 (ADAMS Accession No. ML071270581).
5. Technical Specification Task Force Improved Standard Technical Specifications Change Traveler (TSTF) 363-A, Revision 0, "Revise Topical Report references in ITS 5.6.5, COLR," dated August 4, 2003.
6. Letter from P. D. Milano (NRC) to D. A. Christian (DEK), "Kewaunee Power Station – Safety Evaluation for Topical Report DOM-NAF-5 (TAC No. MD2829)," dated August 30, 2007 (ADAMS Accession No. ML072290373).

ATTACHMENT 2

LICENSE AMENDMENT REQUEST 228
INCORPORATION OF DOMINION NUCLEAR ANALYSIS AND FUEL TOPICAL REPORT DOM-NAF-5 INTO KEWAUNEE TECHNICAL SPECIFICATIONS

MARKED-UP TECHNICAL SPECIFICATION PAGES

Pages

TS 2.1-1
TS 3.10-2
TS 3.10-3
TS 6.9-3
TS 6.9-4
TS 6.9-5

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS - REACTOR CORE

APPLICABILITY

Applies to the limiting combination of thermal power, Reactor Coolant System pressure and coolant temperature during the OPERATING and HOT STANDBY MODES.

OBJECTIVE

To maintain the integrity of the fuel cladding.

SPECIFICATION

a. The combination of RATED POWER level, coolant pressure, and coolant temperature shall not exceed the limits specified in the COLR. The SAFETY LIMIT is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

b. The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ ~~1.14 for the HTP DNB correlation and 1.17 for the WRB-1 DNB correlation~~the 95/95 DNBR criterion for the DNB correlations and methodologies specified in Section 6.9.

c. The peak fuel centerline temperature shall be maintained < 5080°F decreasing by 58°F per 10,000 MWD/MTU of burnup.

C. Identify and correct the cause of the out-of-limit condition prior to increasing thermal power above the reduced thermal power limit required by action A and/or B, above. Subsequent power increases may proceed provided that $F_{\Delta H}^{N}$ is demonstrated, through incore flux mapping, to be within its limits prior to exceeding the following thermal power levels:

   i. 50% of RATED POWER,
   ii. 75% of RATED POWER, and
   iii. Within 24 hours of attaining ≥ 95% of RATED POWER

3. If the $F_Q^N(Z)$ equilibrium relationship is not within its limit:

   A. Reduce the thermal power ≥ 1% RATED POWER for each 1% the $F_Q^N(Z)$ equilibrium relationship exceeds its limit within 15 minutes after each determination and similarly reduce the Power Range Neutron Flux-High Trip Setpoints and the Overpower ΔT Trip Setpoints within 72 hours by ≥ 1% for each 1% $F_Q^N(Z)$ equilibrium relationship exceeds its limit.

   B. If the actions of TS 3.10.b.3.A are not completed within the specified time, then reduce thermal power to ≤ 5% of RATED POWER within the next 6 hours.

   C. Verify the $F_Q^N(Z)$ equilibrium relationship and the $F_Q^N(Z)$ $\cancel{F_Q^{EQ}(Z)}$ transient relationship~~s~~ are within limits prior to increasing thermal power above the reduced thermal power limit required by action A, above.

4. Power distribution maps using the movable detection system shall be made to confirm that the hot channel factor limits of TS 3.10.b.1 are satisfied. (Note: time requirements may be extended by 25%)

   A. For $F_Q^N(Z)$ equilibrium relationship, once after each refueling prior to thermal power exceeding 75% of RATED POWER; and once within 12 hours after achieving equilibrium conditions, after exceeding, by ≥ 10% of RATED POWER, the thermal power at which the $F_Q^N(Z)$ equilibrium relationship was last verified; and 31 effective full power days thereafter.

   B. For $F_{\Delta H}^{N}$, following each refueling prior to exceeding 75% RATED POWER and 31 effective full power days thereafter.

5. The measured $\cancel{F_Q^{EQ}(Z)}$ ~~hot channel factors~~ $F_Q^N(Z)$ under equilibrium conditions shall satisfy the $F_Q^N(Z)$ transient relationship for the central axial 80% of the core as specified in the COLR.

6. Power distribution maps using the movable detector system shall be made to confirm the transient relationship of $\cancel{F_Q^{EQ}(Z)}$ $F_Q^N(Z)$ specified in the COLR according to the following schedules with allowances for a 25% grace period:

   A. Once after each refueling prior to exceeding 75% RATED POWER and every 31 effective full power days thereafter.

B. Once within 12 hours of achieving equilibrium conditions after reaching a thermal power level > 10% higher than the power level at which the last power distribution measurement was performed in accordance with TS 3.10.b.6.A.

C. If a power distribution map measurement indicates that the $\underline{F_Q^N(Z)}$ $\cancel{F_Q^{EQ}(Z)}$ transient relationship's margin to the limit, as specified in the COLR, has decreased since the previous evaluation, then either of the following actions shall be taken:

   i. $\cancel{F_Q^{EQ}(Z)}\underline{F_Q^N(Z)}$ transient relationship shall be increased by the penalty factor specified in the COLR for comparison to the transient limit as specified in the COLR and reverified within the transient limit, or

   ii. Repeat the determination of the $\cancel{F_Q^{EQ}(Z)}\underline{F_Q^N(Z)}$ transient relationship once every seven effective full-power days until either i. above is met, or two successive maps indicate that the $\underline{F_Q^N(Z)}\cancel{F_Q^{EQ}(Z)}$ transient relationship's margin to the transient limit has not decreased.

7. If, for a measured $\underline{F_Q^N(Z)}\cancel{F_Q^{EQ}}$, the transient relationship~~s~~ of $\underline{F_Q^N(Z)}\cancel{F_Q^{EQ}(Z)}$ specified in the COLR ~~are~~ <u>is</u> not within limits, then take the following actions:

   A. Reduce the axial flux difference limits ≥ 1% for each 1% the $\underline{F_Q^N(Z)}\cancel{F_Q^{EQ}(Z)}$ transient relationship exceeds its limit within 4 hours after each determination and similarly reduce the Power Range Neutron Flux-High Trip Setpoints and Overpower ΔT Trip Setpoints within 72 hours by ≥ 1% that the maximum allowable power of the axial flux difference limits is reduced.

   B. If the actions of TS 3.10.b.7.A are not completed within the specified time, then reduce thermal power to ≤ 5% of rated power within the next 6 hours.

   C. Verify the $F_Q^N(Z)$ equilibrium relationship and the $\underline{F_Q^N(Z)}\cancel{F_Q^{EQ}(Z)}$ transient relationship~~s~~ are within limits prior to increasing thermal power above the reduced thermal power limit required by action A, above.

8. Axial Flux Difference

   NOTE: The axial flux difference shall be considered outside limits when two or more operable excore channels indicate that axial flux difference is outside limits.

   A. During power operation with thermal power ≥ 50 percent of RATED POWER, the axial flux difference shall be maintained within the limits specified in the COLR.

      i. If the axial flux difference is not within limits, reduce thermal power to less than 50% RATED POWER within 30 minutes.

3. Deleted.

4. Core Operating Limits Report (COLR)

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

(1)	TS 2.1	Reactor Core Safety Limit
(2)	TS 2.3.a.3.A	Overtemperature ΔT Setpoint
(3)	TS 2.3.a.3.B	Overpower ΔT Setpoint
(4)	TS 3.1.f.3	Moderator Temperature Coefficient (MTC)
(5)	TS 3.8.a.5	Refueling Boron Concentration
(6)	TS 3.10.a	Shutdown Margin
(7)	TS 3.10.b.1.A	$F_Q^N(Z)$ Limits
(8)	TS 3.10.b.1.B	$F_{\Delta H}^N$ Limits
(9)	TS 3.10.b.5	$\underline{F_Q^N(Z)}\cancel{F_Q^{EQ}(Z)}$ Limits
(10)	TS 3.10.b.6.C.i	$\underline{F_Q^N(Z)}\cancel{F_Q^{EQ}(Z)}$ penalty
(11)	TS 3.10.b.8	Axial Flux Difference Target Band
(12)	TS 3.10.b.8.A	Axial Flux Difference Envelope
(13)	TS 3.10.d.1	Shutdown Bank Insertion Limits
(14)	TS 3.10.d.2	Control Bank Insertion Limits
(15)	TS 3.10.k	Core Average Temperature
(16)	TS 3.10.l	Reactor Coolant System Pressure
(17)	TS 3.10.m.1	Reactor Coolant Flow

   B. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102% of the original rated power is specified in a previously approved method, 100.6% of uprated power may be used only when the main feedwater flow measurement (used as the input for reactor thermal output) is provided by the Crossflow ultrasonic flow measurement system (Crossflow system) as described in report (15) listed below. When main feedwater flow measurements from the Crossflow System are unavailable, a power measurement uncertainty consistent with the instrumentation used shall be applied.

   Future revisions of approved analytical methods listed in this Technical Specification that currently reference the original Appendix K uncertainty of 102% of the original rated power should include the condition given above allowing use of 100.6% of uprated power in the safety analysis methodology when the Crossflow system is used for main feedwater flow measurement.

The approved analytical methods are described in the following documents.

(1) ~~Safety Evaluation by the Office of Nuclear Reactor Regulation on "Qualifications of Reactor Physics Methods For Application To Kewaunee" Report, dated August 21, 1979, report date September 29, 1978~~Deleted

(2) Kewaunee Nuclear Power Plant – Review For Kewaunee Reload Safety Evaluation Methods Topical Report WPSRSEM-NP~~, Revision 3 (TAC No. MB0306) dated September 10, 2001~~.

(3) S.M. Bajorek, et al., WCAP-12945-P-A (Proprietary), Westinghouse Code Qualification Document for Best-Estimate Loss-of –Coolant Accident Analysis, Volume I~~, Rev. 2,~~ and Volume II-V~~, Rev.1, and WCAP-14747 (Non-Proprietary) March 1998~~.

(4) N. Lee et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A~~ (Proprietary) and WCAP-10081-NP-A (Non-Proprietary), dated August 1985~~.

(5) C.M. Thompson, et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," WCAP-10054-P-A, Addendum 2~~, Revision 1 (Proprietary) and WCAP-10081-NP (Non-Proprietary), dated July 1997~~.

(6) XN-NF-82-06 (P)(A) Revision 1 and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup, Exxon Nuclear Company~~, dated October 1986~~.

(7) ANF-88-133 (P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation~~, dated December 1991~~.

(8) EMF-92-116 (P)(A)~~ Revision 0~~, "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation~~, dated February 1999~~.

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

(10) WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology~~,~~."~~ July 1985 (W Proprietary).~~

(11) WCAP-8745-P-A, Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT trip functions~~, September 1986~~.

(12) S.I. Dederer, et al., WCAP-14449-P-A, Application of Best-Estimate Large-Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection~~, Rev. 1 (Proprietary and WCAP-14450-NP-A, Rev. 1 (Non-Proprietary), October 1999~~.

(13) WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report~~,~~." ~~April 1995 (Westinghouse Proprietary).~~

(14) WCAP-11397-P-A, "Revised Thermal Design Procedure~~,~~." ~~April 1989.~~

(15) CENP-397-P-A, "Improved Flow Measurement Accuracy Using Cross Flow Ultrasonic Flow Measurement Technology~~,"~~." ~~Rev. 1, May 2000.~~

(16) Topical Report DOM-NAF-5-A, "Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS)."

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.

E. The COLR will contain the complete identification of the TS approved analytical methods used to prepare the COLR (i.e. report number, title, revision, date, and any supplements).

ATTACHMENT 3

LICENSE AMENDMENT REQUEST 228
INCORPORATION OF DOMINION NUCLEAR ANALYSIS AND FUEL TOPICAL REPORT DOM-NAF-5 INTO KEWAUNEE TECHNICAL SPECIFICATIONS

AFFECTED TECHNICAL SPECIFICATION PAGEs

Pages

TS 2.1-1
TS 3.10-2
TS 3.10-3
TS 6.9-3
TS 6.9-4
TS 6.9-5

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS - REACTOR CORE

APPLICABILITY

Applies to the limiting combination of thermal power, Reactor Coolant System pressure and coolant temperature during the OPERATING and HOT STANDBY MODES.

OBJECTIVE

To maintain the integrity of the fuel cladding.

SPECIFICATION

a. The combination of RATED POWER level, coolant pressure, and coolant temperature shall not exceed the limits specified in the COLR. The SAFETY LIMIT is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

b. The departure from nucleate boiling ratio (DNBR) shall be maintained $\geq$ the 95/95 DNBR criterion for the DNB correlations and methodologies specified in Section 6.9.

c. The peak fuel centerline temperature shall be maintained < 5080°F decreasing by 58°F per 10,000 MWD/MTU of burnup.

C. Identify and correct the cause of the out-of-limit condition prior to increasing thermal power above the reduced thermal power limit required by action A and/or B, above. Subsequent power increases may proceed provided that $F_{\Delta H}^{N}$ is demonstrated, through incore flux mapping, to be within its limits prior to exceeding the following thermal power levels:

   i. 50% of RATED POWER,
   ii. 75% of RATED POWER, and
   iii. Within 24 hours of attaining ≥ 95% of RATED POWER

3. If the $F_Q^N(Z)$ equilibrium relationship is not within its limit:

   A. Reduce the thermal power ≥ 1% RATED POWER for each 1% the $F_Q^N(Z)$ equilibrium relationship exceeds its limit within 15 minutes after each determination and similarly reduce the Power Range Neutron Flux-High Trip Setpoints and the Overpower ΔT Trip Setpoints within 72 hours by ≥ 1% for each 1% $F_Q^N(Z)$ equilibrium relationship exceeds its limit.

   B. If the actions of TS 3.10.b.3.A are not completed within the specified time, then reduce thermal power to ≤ 5% of RATED POWER within the next 6 hours.

   C. Verify the $F_Q^N(Z)$ equilibrium relationship and the $\underline{F_Q^N(Z)}$ transient relationship are within limits prior to increasing thermal power above the reduced thermal power limit required by action A, above.

4. Power distribution maps using the movable detection system shall be made to confirm that the hot channel factor limits of TS 3.10.b.1 are satisfied. (Note: time requirements may be extended by 25%)

   A. For $F_Q^N(Z)$ equilibrium relationship, once after each refueling prior to thermal power exceeding 75% of RATED POWER; and once within 12 hours after achieving equilibrium conditions, after exceeding, by ≥ 10% of RATED POWER, the thermal power at which the $F_Q^N(Z)$ equilibrium relationship was last verified; and 31 effective full power days thereafter.

   B. For $F_{\Delta H}^{N}$, following each refueling prior to exceeding 75% RATED POWER and 31 effective full power days thereafter.

5. The measured $\underline{F_Q^N(Z)}$ under equilibrium conditions shall satisfy the $\underline{F_Q^N(Z)}$ transient relationship for the central axial 80% of the core as specified in the COLR.

6. Power distribution maps using the movable detector system shall be made to confirm the transient relationship of $\underline{F_Q^N(Z)}$ specified in the COLR according to the following schedules with allowances for a 25% grace period:

   A. Once after each refueling prior to exceeding 75% RATED POWER and every 31 effective full power days thereafter.

   B. Once within 12 hours of achieving equilibrium conditions after reaching a thermal power level > 10% higher than the power level at which the last power distribution measurement was performed in accordance with TS 3.10.b.6.A.

C. If a power distribution map measurement indicates that the $\underline{F_Q^N(Z)}$ transient relationship's margin to the limit, as specified in the COLR, has decreased since the previous evaluation, then either of the following actions shall be taken:

i. $\underline{F_Q^N(Z)}$ transient relationship shall be increased by the penalty factor specified in the COLR for comparison to the transient limit as specified in the COLR and reverified within the transient limit, or

ii. Repeat the determination of the $\underline{F_Q^N(Z)}$ transient relationship once every seven effective full-power days until either i. above is met, or two successive maps indicate that the $\underline{F_Q^N(Z)}$ transient relationship's margin to the transient limit has not decreased.

7. If, for a measured $\underline{F_Q^N(Z)}$, the transient relationship of $\underline{F_Q^N(Z)}$ specified in the COLR is not within limits, then take the following actions:

A. Reduce the axial flux difference limits ≥ 1% for each 1% the $\underline{F_Q^N(Z)}$ transient relationship exceeds its limit within 4 hours after each determination and similarly reduce the Power Range Neutron Flux-High Trip Setpoints and Overpower ΔT Trip Setpoints within 72 hours by ≥ 1% that the maximum allowable power of the axial flux difference limits is reduced.

B. If the actions of TS 3.10.b.7.A are not completed within the specified time, then reduce thermal power to ≤ 5% of rated power within the next 6 hours.

C. Verify the $F_Q^N(Z)$ equilibrium relationship and the $\underline{F_Q^N(Z)}$ transient relationship are within limits prior to increasing thermal power above the reduced thermal power limit required by action A, above.

8. Axial Flux Difference

NOTE: The axial flux difference shall be considered outside limits when two or more operable excore channels indicate that axial flux difference is outside limits.

A. During power operation with thermal power ≥ 50 percent of RATED POWER, the axial flux difference shall be maintained within the limits specified in the COLR.

i. If the axial flux difference is not within limits, reduce thermal power to less than 50% RATED POWER within 30 minutes.

3. Deleted.

4. Core Operating Limits Report (COLR)

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

(1)	TS 2.1	Reactor Core Safety Limit
(2)	TS 2.3.a.3.A	Overtemperature ΔT Setpoint
(3)	TS 2.3.a.3.B	Overpower ΔT Setpoint
(4)	TS 3.1.f.3	Moderator Temperature Coefficient (MTC)
(5)	TS 3.8.a.5	Refueling Boron Concentration
(6)	TS 3.10.a	Shutdown Margin
(7)	TS 3.10.b.1.A	$F_Q^N(Z)$ Limits
(8)	TS 3.10.b.1.B	$F_{\Delta H}^N$ Limits
(9)	TS 3.10.b.5	$\underline{F_Q^N(Z)}$ Limits
(10)	TS 3.10.b.6.C.i	$\underline{F_Q^N(Z)}$ penalty
(11)	TS 3.10.b.8	Axial Flux Difference Target Band
(12)	TS 3.10.b.8.A	Axial Flux Difference Envelope
(13)	TS 3.10.d.1	Shutdown Bank Insertion Limits
(14)	TS 3.10.d.2	Control Bank Insertion Limits
(15)	TS 3.10.k	Core Average Temperature
(16)	TS 3.10.l	Reactor Coolant System Pressure
(17)	TS 3.10.m.1	Reactor Coolant Flow

   B. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102% of the original rated power is specified in a previously approved method, 100.6% of uprated power may be used only when the main feedwater flow measurement (used as the input for reactor thermal output) is provided by the Crossflow ultrasonic flow measurement system (Crossflow system) as described in report (15) listed below. When main feedwater flow measurements from the Crossflow System are unavailable, a power measurement uncertainty consistent with the instrumentation used shall be applied.

   Future revisions of approved analytical methods listed in this Technical Specification that currently reference the original Appendix K uncertainty of 102% of the original rated power should include the condition given above allowing use of 100.6% of uprated power in the safety analysis methodology when the Crossflow system is used for main feedwater flow measurement.

The approved analytical methods are described in the following documents.

(1) Deleted

(2) Kewaunee Nuclear Power Plant – Review For Kewaunee Reload Safety Evaluation Methods Topical Report WPSRSEM-NP.

(3) S.M. Bajorek, et al., WCAP-12945-P-A (Proprietary), Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Accident Analysis, Volume I and Volume II-V.

(4) N. Lee et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A.

(5) C.M. Thompson, et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," WCAP-10054-P-A, Addendum 2.

(6) XN-NF-82-06 (P)(A) Revision 1 and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup, Exxon Nuclear Company.

(7) ANF-88-133 (P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation.

(8) EMF-92-116 (P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation.

(9) WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification."

(10) WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology."

(11) WCAP-8745-P-A, Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT trip functions.

(12) S.I. Dederer, et al., WCAP-14449-P-A, Application of Best-Estimate Large-Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection.

(13) WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report."

(14) WCAP-11397-P-A, "Revised Thermal Design Procedure."

(15) CENP-397-P-A, "Improved Flow Measurement Accuracy Using Cross Flow Ultrasonic Flow Measurement Technology."

(16) Topical Report DOM-NAF-5-A, "Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS)."

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.

E. The COLR will contain the complete identification of the TS approved analytical methods used to prepare the COLR (i.e. report number, title, revision, date, and any supplements).

ATTACHMENT 4

LICENSE AMENDMENT REQUEST 228
INCORPORATION OF DOMINION NUCLEAR ANALYSIS AND FUEL TOPICAL REPORT DOM-NAF-5 INTO KEWAUNEE TECHNICAL SPECIFICATIONS

MARKED-UP TECHNICAL SPECIFICATION BASES PAGES

Pages:

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

BASIS - Safety Limits-Reactor Core (TS 2.1)

The reactor core safety limits shall not be exceeded during steady state operation, normal operational transients, and anticipated operational occurrences. This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNBR criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

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

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

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all OPERATING conditions. This is accomplished by operating the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters of RATED POWER, reactor coolant temperature and pressure have been related to DNB through a DNB correlation. The DNB correlation has been developed to predict the DNB heat flux and the location of the DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and Condition I and II transients is ~~limited to the DNBR limit. This minimum DNBR~~less than the 95/95 DNBR criterion. The 95/95 DNBR criterion corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all OPERATING conditions.

The SAFETY LIMIT curves as provided in the Core Operating Report Limits Report show the loci of points of thermal power, reactor coolant system average temperature, and reactor coolant system pressure for which the minimum DNBR is not less than the safety analysis limit, that fuel centerline temperature remains below melting, that the average enthalpy at the exit of the core is less than or equal to the enthalpy of saturated liquid, or that the core exit quality is within limits defined by the DNBR correlation. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNBR ratio reaches the DNBR limit and thus, this limit is conservative with respect to maintaining clad integrity. The area where clad integrity is ensured is below the safety limit curves.

The curves are based on the nuclear hot channel factor limits of as specified in the COLR.

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are given in TS 3.10.d. Slightly higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits as specified in the COLR ensure that the increase in peaking factor is more than offset by the decrease in power level.

The Reactor Control and PROTECTION SYSTEM is designed to prevent any anticipated combination of transient conditions that would result in a DNBR less than the ~~DNBR limit~~95/95 DNBR criterion.

~~Two departure from nucleate boiling ratio (DNBR) correlations are used in the generation and validation of the safety limit curves: the WRB-1 DNBR correlation and the high thermal performance (HTP) DNBR correlation. The WRB-1 correlation applies to the Westinghouse 422 V+ fuel. The HTP correlation applies to FRA-ANP fuel with HTP spacers. The DNBR correlations have been qualified and approved for application to Kewaunee. The DNB correlation limits are 1.14 for the HTP DNBR correlation, and 1.17 for the WRB-1 DNBR correlation.~~The applicable departure from nucleate boiling ratio (DNBR) correlations and methodologies are used in the generation and validation of safety limit curves. The DNBR correlations, methodologies, and 95/95 DNBR criterion have been qualified and approved for application to Kewaunee. The approved DNBR correlations and methodologies are documented in Section 6.9.

BASIS - Reactor Coolant System (TS 3.1.a)

Reactor Coolant Pumps (TS 3.1.a.1)

When the boron concentration of the Reactor Coolant System is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the primary system volume in approximately one-half hour.

Part one of the specification requires that both reactor coolant pumps be OPERATING when the reactor is in power operation to provide core cooling. Planned power operation with one loop out-of-service is not allowed in the present design because the system does not meet the single failure (locked rotor) criteria requirement for this MODE of operation. The flow provided in each case in part one will keep Departure from Nucleate Boiling Ratio (DNBR) well above ~~1.30~~the DNBR limit. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. One pump operation is not permitted except for tests. Upon loss of one pump below 10% full power, the core power shall be reduced to a level below the maximum power determined for zero power testing. Natural circulation can remove decay heat up to 10% power. Above 10% power, an automatic reactor trip will occur if flow from either pump is lost.[1]

The RCS will be protected against exceeding the design basis of the Low Temperature Overpressure Protection (LTOP) System by restricting the starting of a Reactor Coolant Pump (RXCP) to when the secondary water temperature of each SG is < 100°F above each RCS cold leg temperature. The restriction on starting a reactor coolant pump (RXCP) when one or more RCS cold leg temperatures is ≤ 200°F is provided to prevent a RCS pressure transient, caused by an energy addition from the secondary system, which could exceed the design basis of the LTOP System.

Decay Heat Removal Capabilities (TS 3.1.a.2)

When the average reactor coolant temperature is ≤ 350°F a combination of the available heat sinks is sufficient to remove the decay heat and provide the necessary redundancy to meet the single failure criterion.

When the average reactor coolant temperature is ≤ 200°F, the plant is in a COLD SHUTDOWN condition and there is a negligible amount of sensible heat energy stored in the Reactor Coolant System. Should one residual heat removal train become inoperable under these conditions, the remaining train is capable of removing all of the decay heat being generated.

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[1] USAR Section 7.2.2

The requirement that the reactor is not to be made critical when the moderator coefficient is greater than the value specified in the COLR has been imposed to prevent any unexpected power excursion during normal operation as a result of either an increase in moderator temperature or a decrease in coolant pressure. The moderator temperature coefficient limits are required to maintain plant operation within the assumptions contained in the USAR analyses. Having an initial moderator temperature coefficient no greater than the value specified in the COLR provides reasonable assurance that the moderator temperature coefficient will be negative at 60% rated thermal power. The moderator temperature coefficient requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special OPERATING precautions will be taken. In addition, the strong negative Doppler coefficient[20] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction in moderator density.

Suitable physics measurements of moderator coefficients of reactivity will be made as part of the startup testing program to verify analytical predictions.

Analysis has shown that maintaining the moderator temperature coefficient at criticality less than or equal to the value specified in the COLR will ensure that a negative coefficient will exist at 60% power. Current safety analysis supports OPERATING up to 60% power with a moderator temperature coefficient less than or equal to the value specified in the COLR. At power levels greater than 60%, a negative moderator temperature coefficient must exist.

The calculated hot full power (HFP) moderator temperature coefficient will be more negative than the value specified in the COLR for at least 95% of a cycle's time at HFP to ensure the limitations associated with and anticipated transient without scram (ATWS) event are not exceeded. NRC approved methods~~[27][28]~~ will be used to determine the lowest expected HFP moderator temperature coefficient for the 5% of HFP cycle time with the highest boron concentration. The cycle time at HFP is the maximum number of days that the cycle could be at HFP based on the design calculation of cycle length. The cycle time at HFP can also be expressed in terms of burnup by converting the maximum number of days at full power to an equivalent burnup. If this HFP moderator temperature coefficient is more negative than the value specified in the COLR, then the ATWS design limit will be met for 95% of the cycle's time at HFP. If this HFP moderator temperature coefficient design limit is still not met after excluding the 5% of the cycle burnup with the highest boron concentration, then the core loading must be revised.

~~The results of this design limit consideration will be reported in the Reload Safety Evaluation Report.~~

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[20] USAR Section 3.2.1

~~[27] "NRC Safety Evaluation Report for Qualification of Reactor Physics, Methods for Application to Kewaunee," dated October 22, 1979.~~

~~[28] "NRC Safety Evaluation Report for the Reload Safety Evaluation Methods for Application to Kewaunee," dated April 11, 1988.~~

$F_Q^N(Z)$, Height Dependent Nuclear Flux Hot Channel Factor

$F_Q^N(Z)$, Height Dependent Nuclear Flux Hot Channel Factor, is defined as the maximum local linear power density in the core at core elevation Z divided by the core average linear power density, assuming nominal fuel rod dimensions.

An upper bound envelope for $F_Q^N(Z)$ as specified in the COLR has been determined from extensive analyses considering all OPERATING maneuvers consistent with the Technical Specifications on power distribution control as given in TS 3.10. The results of the loss-of-coolant accident analyses based on this upper bound envelope indicate the peak clad temperatures, with a high probability, remain less than the 2200 $^\circ$F limit.

The $F_Q^N(Z)$ limits as specified in the COLR are derived from the LOCA analyses. ~~The LOCA analyses are performed for Westinghouse 422 V+ fuel, FRA-ANP heavy fuel and for FRA-ANP standard fuel.~~

When a $F_Q^N(Z)$ measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

$F_Q^N(Z)$ is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests).

$\cancel{F_Q^{EQ}(Z)}$ ~~is t~~The measured $F_Q^N(Z)$ is obtained at equilibrium conditions during the target flux determination. $\cancel{F_Q^{EQ}(Z)}$The measured $F_Q^N(Z)$ must satisfy the equilibrium and transient relationships that ~~is~~ are in the COLR. The $F_Q^N(Z)$ equilibrium relationship is the inequality relationship between $F_Q^N(Z)$ and its limit. The $F_Q^N(Z)$ equilibrium relationship does not include the transient condition multiplier.

Because the value of $F_Q^N(Z)$ represents an equilibrium condition, it does not include the variations of $F_Q^N(Z)$ which are present during non-equilibrium situations such as load following or power ascension. To account for these possible variations, the equilibrium value of $F_Q^N(Z)$ is adjusted by an elevation dependent factor~~, W(z),~~ that accounts for the calculated worst case transient conditions. Core power distribution is controlled under non-equilibrium conditions by operating the core within the core operating limits on axial flux distribution, quadrant power tilt, and control rod insertion.

$F_Q^N(Z)$ transient is the measured $F_Q^N(Z)$ obtained at equilibrium conditions multiplied by the elevation dependent factor that accounts for the worst case transient conditions. The $F_Q^N(Z)$ transient relationship is the inequality relationship between the $F_Q^N(Z)$ transient and its limit. The $F_Q^N(Z)$ transient relationship includes the transient condition multiplier.

If a power distribution measurement indicates that the $F_Q^N(Z)\cancel{F_Q^{EQ}(Z)}$ transient relationship's margin to the limit has decreased since the previous evaluation then TS 3.10.b.6.C provides two options of either increasing the $F_Q^N(Z)\cancel{F_Q^{EQ}(Z)}$ transient relationship by the appropriate penalty factor or increasing the power distribution surveillance to once every 7 EFPD until two successive flux maps indicate that the $F_Q^N(Z)\cancel{F_Q^{EQ}(Z)}$ transient relationship's margin to the limit has not decreased. IF $F_Q^N(Z)\cancel{F_Q^{EQ}(Z)}$ with the penalty factor applied is greater than the limit, then TS 3.10.b.6 is not satisfied and TS 3.10.b.7 should be applied to maintain the normal surveillance interval. Based on TS 3.10.b.7.A, the axial flux distribution (AFD) limits are reduced by 1% for each 1% that the $F_Q^N(Z)\cancel{F_Q^{EQ}(Z)}$ transient relationship exceeds its limit within the allowed time of 4 hours.

The contingency actions of TS 3.10.b.6 and TS 3.10.b.7 are to ensure that $F_Q^N(Z)$ does not exceed its limit for any significant period of time without detection. Satisfying limits on $F_Q^N(Z)$ ensures that the safety analyses remain bounding and valid.

$F_{\Delta H}^N$ Nuclear Enthalpy Rise Hot Channel Factor

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the maximum integral of linear power along a fuel rod to the core average integral fuel rod power.

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in DNBR calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus, the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

The $F_{\Delta H}^N$ limit is determined from safety analyses of the limiting DNBR transient events. ~~The safety analyses are performed for FRA-ANP heavy fuel, FRA-ANP standard fuel, and Westinghouse 422 V+ fuel.~~ In these analyses, the important operational parameters are selected to minimize DNBR. The results of the safety analyses must demonstrate that minimum DNBR is greater than the DNBR limit for a fuel rod operating at the $F_{\Delta H}^N$ limit.

The use of $F_{\Delta H}^N$ in TS 3.10.b.5.C is to monitor "upburn" which is defined as an increase in $F_{\Delta H}^N$ with exposure. Since this is not to be confused with observed changes in peak power resulting from such phenomena as xenon redistribution, control rod movement, power level changes, or changes in the number of instrumented thimbles recorded, an allowance of 2% is used to account for such changes.

Rod Bow Effects

~~No penalty for rod bow effects needs to be included in TS 3.10.b.1 for FRA-ANP fuel.~~[(1)] Penalty for rod bow effects is applied based on approved methodology.

Surveillance

Measurements of the hot channel factors are required as part of startup physics tests, at least each full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identifies operational anomalies which would otherwise affect these bases.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met. These conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than an indicated 12 steps from the bank demand position where reactor power is ≥85%, or an indicated 24 steps when reactor power is < 85%.

2. Control rod banks are sequenced with overlapping banks as specified in the COLR.

3. The control bank insertion limits as specified in the COLR are not violated, except as allowed by TS 3.10.d.2.

4. The axial power distribution, expressed in terms of axial flux difference, is maintained within the limits.

The limits on axial flux difference (AFD) assure that the axial power distribution is maintained such that the FQ(Z) upper bound envelope of FQLIMIT times the normalized axial peaking factor [K(Z)] is not exceeded during either normal operation or in the event of xenon redistribution following power changes. This ensures that the power distributions assumed in the large and small break LOCA analyses will bound those that occur during plant operation.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD monitor program. The computer determines the AFD for each of the operable excore channels and provides a computer alarm if the AFD for at least 2 of 4 or 2 of 3 operable excore channels are outside the AFD limits and reactor power is greater than 50 percent or RATED POWER.

For Condition II events the core is protected from overpower and a minimum DNBR less than the DNBR limit by an automatic Protection System. Compliance with the specification is assumed as a precondition for Condition II transients; however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

~~[(1)] N. E. Hoppe, "Mechanical Design Report Supplement for Kewaunee High Burnup (49 GWD/MTU) Fuel Assemblies," XN-NF-84-28(P), Exxon Nuclear Company, July 1984.~~

Inoperable Rod Position Indicator Channels (TS 3.10.f)

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the Bank Demand Position Indication System (commonly called group step counters) and the Individual Rod Position Indication (IRPI) System.

The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± ⅝ inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The IRPI System provides an indirect indication of actual control rod position, but at a lower precision than the step counters. The rod position indicator channel is sufficiently accurate to detect a rod ± 12 steps away from its demand position. If the rod position indicator channel is not OPERABLE, special surveillance of core power tilt indications, using established procedures and relying on movable incore detectors, will be used to verify power distribution symmetry.

A note indicating individual control rod position indications may not be within limits for up to and including one hour following substantial control rod movement modifies this LCO. This allows up to one hour of thermal soak time to allow the control rod drive shaft to reach thermal equilibrium and thus present a consistent position indication. Substantial rod movement is considered to be 10 or more steps in one direction in less than or equal to one hour.

3.10.f.1
When one IRPI channel per group fails, the position of the rod 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 $\underline{F_Q^N(Z)}$$\cancel{F_Q}$ satisfies TS 3.10.b.1.A ~~($F_Q^N(Z)$)~~, TS 3.10.b.5 ~~F_Q^{EQ}~~), $F_{\Delta H}^N$ satisfies TS 3.10.b.1.B, and SHUTDOWN MARGIN satisfies TS 3.10.a, provided the non-indicating 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 (≥ 24 steps), the required action of TS 3.10.f.3 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 reduction of reactor thermal power to ≤ 50% RATED POWER puts the core into a condition where COLR limits are sufficiently relaxed such that rod position will not cause the core to violate COLR limits[1]. The allowed completion time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RATED POWER from full power conditions without challenging plant systems and allowing for rod position determination by movable incore detectors.

3.10.f.2
When more than one IRPI 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. This together with the indirect position determination available via movable incore detectors will minimize the

[1] USAR Chapter 14

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 Tavg helps 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. 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 $\underline{F_Q{}^N(Z)}$~~FQ~~ satisfies TS 3.10.b.1.A ~~($F_Q{}^N(Z)$)~~, TS 3.10.b.5 ($\underline{F_Q{}^N(Z)}$~~$F_Q{}^{EQ}$~~), $F_{\Delta H}{}^N$ satisfies TS 3.10.b.1.B, and SHUTDOWN MARGIN satisfies TS 3.10.a, provided the non-indicating rods have not been moved. Verification of control rod 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 IRPI system to operation while avoiding the plant challenges associated with the shutdown without full rod position indication.

3.10.f.3

Based on operating experience, normal power operation does not require excessive rod movement. If one or more rods has been significantly moved. 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 one or more inoperable individual rod position indicators, as applicable, are still appropriate but must be initiated under TS 3.10.f.3 to begin verifying that these rods are still properly positioned, relative to their group positions. If, within 4 hours, the rod positions have not been determined, thermal power must be reduced to ≤ 50% RATED POWER within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RATED POWER, 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.

3.10.f.4

With one demand position indicator per bank inoperable, the IRPI System can determine the rod positions. Since normal power operation does not require excessive movement of rods, verification by administrative means (logging IRPI position and verifying within rod alignment limitations) that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are ≤ 12 steps apart when operating at > 85% RATED POWER or ≤ 24 steps apart when operating at ≤ 85% RATED POWER within the allowed Completion Time of once every 8 hours is adequate. A reduction of reactor thermal power to ≤ 50% RATED POWER puts the core into a condition where COLR limits are sufficiently relaxed such that rod position will not cause the core to violate COLR limits. The allowed completion time of 8 hours provides an acceptable period of time to verify the rod positions or reduce power to ≤ 50% RATED POWER.

Inoperable Rod Limitations (TS 3.10.g)

One inoperable control rod is acceptable provided the potential consequences of accidents are not worse than the cases analyzed in the safety analysis report. A 30-day period is provided for the reanalysis of all accidents sensitive to the changed initial condition.

Rod Drop Time (TS 3.10.h)

The required drop time to dashpot entry is consistent with safety analysis.

ATTACHMENT 5

LICENSE AMENDMENT REQUEST 228
INCORPORATION OF DOMINION NUCLEAR ANALYSIS AND FUEL TOPICAL REPORT DOM-NAF-5 INTO KEWAUNEE TECHNICAL SPECIFICATIONS

AFFECTED TECHNICAL SPECIFICATION BASES PAGES

Pages:

TS B2.1-1
TS B2.1-2
TS B3.1-1
TS B3.1-14
TS B3.10-2
TS B3.10-3
TS B3.10-4
TS B3.10-7
TS B3.10-8

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

BASIS - Safety Limits-Reactor Core (TS 2.1)

The reactor core safety limits shall not be exceeded during steady state operation, normal operational transients, and anticipated operational occurrences. This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNBR criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

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

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

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all OPERATING conditions. This is accomplished by operating the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters of RATED POWER, reactor coolant temperature and pressure have been related to DNB through a DNB correlation. The DNB correlation has been developed to predict the DNB heat flux and the location of the DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and Condition I and II transients is less than the 95/95 DNBR criterion. The 95/95 DNBR crtierion corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all OPERATING conditions.

The SAFETY LIMIT curves as provided in the Core Operating Report Limits Report show the loci of points of thermal power, reactor coolant system average temperature, and reactor coolant system pressure for which the minimum DNBR is not less than the safety analysis limit, that fuel centerline temperature remains below melting, that the average enthalpy at the exit of the core is less than or equal to the enthalpy of saturated liquid, or that the core exit quality is within limits defined by the DNBR correlation. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNBR ratio reaches the DNBR limit and thus, this limit is conservative with respect to maintaining clad integrity. The area where clad integrity is ensured is below the safety limit curves.

The curves are based on the nuclear hot channel factor limits of as specified in the COLR.

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are given in TS 3.10.d. Slightly higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits as specified in the COLR ensure that the increase in peaking factor is more than offset by the decrease in power level.

The Reactor Control and PROTECTION SYSTEM is designed to prevent any anticipated combination of transient conditions that would result in a DNBR less than the <u>95/95 DNBR criterion</u>.

<u>The applicable departure from nucleate boiling ratio (DNBR) correlations and methodologies are used in the generation and validation of safety limit curves. The DNBR correlations, methodologies, and 95/95 DNBR criterion have been qualified and approved for application to Kewaunee. The approved DNBR correlations and methodologies are documented in Section 6.9.</u>

BASIS - Reactor Coolant System (TS 3.1.a)

Reactor Coolant Pumps (TS 3.1.a.1)

When the boron concentration of the Reactor Coolant System is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the primary system volume in approximately one-half hour.

Part one of the specification requires that both reactor coolant pumps be OPERATING when the reactor is in power operation to provide core cooling. Planned power operation with one loop out-of-service is not allowed in the present design because the system does not meet the single failure (locked rotor) criteria requirement for this MODE of operation. The flow provided in each case in part one will keep Departure from Nucleate Boiling Ratio (DNBR) well above the DNBR limit. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. One pump operation is not permitted except for tests. Upon loss of one pump below 10% full power, the core power shall be reduced to a level below the maximum power determined for zero power testing. Natural circulation can remove decay heat up to 10% power. Above 10% power, an automatic reactor trip will occur if flow from either pump is lost.[1]

The RCS will be protected against exceeding the design basis of the Low Temperature Overpressure Protection (LTOP) System by restricting the starting of a Reactor Coolant Pump (RXCP) to when the secondary water temperature of each SG is < 100°F above each RCS cold leg temperature. The restriction on starting a reactor coolant pump (RXCP) when one or more RCS cold leg temperatures is ≤ 200°F is provided to prevent a RCS pressure transient, caused by an energy addition from the secondary system, which could exceed the design basis of the LTOP System.

Decay Heat Removal Capabilities (TS 3.1.a.2)

When the average reactor coolant temperature is ≤ 350°F a combination of the available heat sinks is sufficient to remove the decay heat and provide the necessary redundancy to meet the single failure criterion.

When the average reactor coolant temperature is ≤ 200°F, the plant is in a COLD SHUTDOWN condition and there is a negligible amount of sensible heat energy stored in the Reactor Coolant System. Should one residual heat removal train become inoperable under these conditions, the remaining train is capable of removing all of the decay heat being generated.

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[1] USAR Section 7.2.2

The requirement that the reactor is not to be made critical when the moderator coefficient is greater than the value specified in the COLR has been imposed to prevent any unexpected power excursion during normal operation as a result of either an increase in moderator temperature or a decrease in coolant pressure. The moderator temperature coefficient limits are required to maintain plant operation within the assumptions contained in the USAR analyses. Having an initial moderator temperature coefficient no greater than the value specified in the COLR provides reasonable assurance that the moderator temperature coefficient will be negative at 60% rated thermal power. The moderator temperature coefficient requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special OPERATING precautions will be taken. In addition, the strong negative Doppler coefficient[20] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction in moderator density.

Suitable physics measurements of moderator coefficients of reactivity will be made as part of the startup testing program to verify analytical predictions.

Analysis has shown that maintaining the moderator temperature coefficient at criticality less than or equal to the value specified in the COLR will ensure that a negative coefficient will exist at 60% power. Current safety analysis supports OPERATING up to 60% power with a moderator temperature coefficient less than or equal to the value specified in the COLR. At power levels greater than 60%, a negative moderator temperature coefficient must exist.

The calculated hot full power (HFP) moderator temperature coefficient will be more negative than the value specified in the COLR for at least 95% of a cycle's time at HFP to ensure the limitations associated with and anticipated transient without scram (ATWS) event are not exceeded. NRC approved methods will be used to determine the lowest expected HFP moderator temperature coefficient for the 5% of HFP cycle time with the highest boron concentration. The cycle time at HFP is the maximum number of days that the cycle could be at HFP based on the design calculation of cycle length. The cycle time at HFP can also be expressed in terms of burnup by converting the maximum number of days at full power to an equivalent burnup. If this HFP moderator temperature coefficient is more negative than the value specified in the COLR, then the ATWS design limit will be met for 95% of the cycle's time at HFP. If this HFP moderator temperature coefficient design limit is still not met after excluding the 5% of the cycle burnup with the highest boron concentration, then the core loading must be revised.

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[20] USAR Section 3.2.1

$F_Q^N(Z)$, Height Dependent Nuclear Flux Hot Channel Factor

$F_Q^N(Z)$, Height Dependent Nuclear Flux Hot Channel Factor, is defined as the maximum local linear power density in the core at core elevation Z divided by the core average linear power density, assuming nominal fuel rod dimensions.

An upper bound envelope for $F_Q^N(Z)$ as specified in the COLR has been determined from extensive analyses considering all OPERATING maneuvers consistent with the Technical Specifications on power distribution control as given in TS 3.10. The results of the loss-of-coolant accident analyses based on this upper bound envelope indicate the peak clad temperatures, with a high probability, remain less than the 2200 °F limit.

The $F_Q^N(Z)$ limits as specified in the COLR are derived from the LOCA analyses.

When a $F_Q^N(Z)$ measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

$F_Q^N(Z)$ is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests).

The measured $F_Q^N(Z)$ is obtained at equilibrium conditions during the target flux determination. The measured $F_Q^N(Z)$ must satisfy the equilibrium and transient relationships that are in the COLR. The $F_Q^N(Z)$ equilibrium relationship is the inequality relationship between $F_Q^N(Z)$ and its limit. The $F_Q^N(Z)$ equilibrium relationship does not include the transient condition multiplier.

Because the value of $F_Q^N(Z)$ represents an equilibrium condition, it does not include the variations of $F_Q^N(Z)$ which are present during non-equilibrium situations such as load following or power ascension. To account for these possible variations, the equilibrium value of $F_Q^N(Z)$ is adjusted by an elevation dependent factor that accounts for the calculated worst case transient conditions. Core power distribution is controlled under non-equilibrium conditions by operating the core within the core operating limits on axial flux distribution, quadrant power tilt, and control rod insertion.

The $F_Q^N(Z)$ transient is the measured $F_Q^N(Z)$ obtained at equilibrium conditions multiplied by the elevation dependent factor that accounts for the worst case transient conditions. The $F_Q^N(Z)$ transient relationship is the inequality relationship between the $F_Q^N(Z)$ transient and its limit. The $F_Q^N(Z)$ transient relationship includes the transient condition multiplier.

If a power distribution measurement indicates that the $F_Q^N(Z)$ transient relationship's margin to the limit has decreased since the previous evaluation then TS 3.10.b.6.C provides two options of either increasing the $F_Q^N(Z)$ transient relationship by the appropriate penalty factor or increasing the power distribution surveillance to once every 7 EFPD until two successive flux maps indicate that the $F_Q^N(Z)$ transient relationship's margin to the limit has not decreased. IF $F_Q^N(Z)$ with the penalty factor applied is greater than the limit, then TS 3.10.b.6 is not satisfied and TS 3.10.b.7 should be applied to maintain the normal surveillance interval. Based on TS 3.10.b.7.A, the axial flux distribution (AFD) limits are reduced by 1% for each 1% that the $F_Q^N(Z)$ transient relationship exceeds its limit within the allowed time of 4 hours.

The contingency actions of TS 3.10.b.6 and TS 3.10.b.7 are to ensure that $F_Q^N(Z)$ does not exceed its limit for any significant period of time without detection. Satisfying limits on $F_Q^N(Z)$ ensures that the safety analyses remain bounding and valid.

$F_{\Delta H}^{N}$ Nuclear Enthalpy Rise Hot Channel Factor

$F_{\Delta H}^{N}$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the maximum integral of linear power along a fuel rod to the core average integral fuel rod power.

It should be noted that $F_{\Delta H}^{N}$ is based on an integral and is used as such in DNBR calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus, the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^{N}$.

The $F_{\Delta H}^{N}$ limit is determined from safety analyses of the limiting DNBR transient events. In these analyses, the important operational parameters are selected to minimize DNBR. The results of the safety analyses must demonstrate that minimum DNBR is greater than the DNBR limit for a fuel rod operating at the $F_{\Delta H}^{N}$ limit.

The use of $F_{\Delta H}^{N}$ in TS 3.10.b.5.C is to monitor "upburn" which is defined as an increase in $F_{\Delta H}^{N}$ with exposure. Since this is not to be confused with observed changes in peak power resulting from such phenomena as xenon redistribution, control rod movement, power level changes, or changes in the number of instrumented thimbles recorded, an allowance of 2% is used to account for such changes.

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Penalty for rod bow effects is applied based on approved methodology.

Surveillance

Measurements of the hot channel factors are required as part of startup physics tests, at least each full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identifies operational anomalies which would otherwise affect these bases.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met. These conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than an indicated 12 steps from the bank demand position where reactor power is ≥85%, or an indicated 24 steps when reactor power is < 85%.

2. Control rod banks are sequenced with overlapping banks as specified in the COLR.

3. The control bank insertion limits as specified in the COLR are not violated, except as allowed by TS 3.10.d.2.

4. The axial power distribution, expressed in terms of axial flux difference, is maintained within the limits.

The limits on axial flux difference (AFD) assure that the axial power distribution is maintained such that the FQ(Z) upper bound envelope of FQLIMIT times the normalized axial peaking factor [K(Z)] is not exceeded during either normal operation or in the event of xenon redistribution following power changes. This ensures that the power distributions assumed in the large and small break LOCA analyses will bound those that occur during plant operation.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD monitor program. The computer determines the AFD for each of the operable excore channels and provides a computer alarm if the AFD for at least 2 of 4 or 2 of 3 operable excore channels are outside the AFD limits and reactor power is greater than 50 percent or RATED POWER.

For Condition II events the core is protected from overpower and a minimum DNBR less than the DNBR limit by an automatic Protection System. Compliance with the specification is assumed as a precondition for Condition II transients; however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

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The IRPI System provides an indirect indication of actual control rod position, but at a lower precision than the step counters. The rod position indicator channel is sufficiently accurate to detect a rod ± 12 steps away from its demand position. If the rod position indicator channel is not OPERABLE, special surveillance of core power tilt indications, using established procedures and relying on movable incore detectors, will be used to verify power distribution symmetry.

A note indicating individual control rod position indications may not be within limits for up to and including one hour following substantial control rod movement modifies this LCO. This allows up to one hour of thermal soak time to allow the control rod drive shaft to reach thermal equilibrium and thus present a consistent position indication. Substantial rod movement is considered to be 10 or more steps in one direction in less than or equal to one hour.

3.10.f.1

When one IRPI channel per group fails, the position of the rod 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 $F_Q^N(Z)$ satisfies TS 3.10.b.1.A, TS 3.10.b.5, $F_{\Delta H}^N$ satisfies TS 3.10.b.1.B, and SHUTDOWN MARGIN satisfies TS 3.10.a, provided the non-indicating 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 (≥ 24 steps), the required action of TS 3.10.f.3 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 reduction of reactor thermal power to ≤ 50% RATED POWER puts the core into a condition where COLR limits are sufficiently relaxed such that rod position will not cause the core to violate COLR limits[1]. The allowed completion time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RATED POWER from full power conditions without challenging plant systems and allowing for rod position determination by movable incore detectors.

3.10.f.2

When more than one IRPI 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. This together with the indirect position determination available via movable incore detectors will minimize the

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[1] USAR Chapter 14

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 Tavg helps 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. 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 $F_Q^N(Z)$ satisfies TS 3.10.b.1.A, TS 3.10.b.5, $F_{\Delta H}^N$ satisfies TS 3.10.b.1.B, and SHUTDOWN MARGIN satisfies TS 3.10.a, provided the non-indicating rods have not been moved. Verification of control rod 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 IRPI system to operation while avoiding the plant challenges associated with the shutdown without full rod position indication.

3.10.f.3

Based on operating experience, normal power operation does not require excessive rod movement. If one or more rods has been significantly moved. 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 one or more inoperable individual rod position indicators, as applicable, are still appropriate but must be initiated under TS 3.10.f.3 to begin verifying that these rods are still properly positioned, relative to their group positions. If, within 4 hours, the rod positions have not been determined, thermal power must be reduced to ≤ 50% RATED POWER within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RATED POWER, 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.

3.10.f.4

With one demand position indicator per bank inoperable, the IRPI System can determine the rod positions. Since normal power operation does not require excessive movement of rods, verification by administrative means (logging IRPI position and verifying within rod alignment limitations) that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are ≤ 12 steps apart when operating at > 85% RATED POWER or ≤ 24 steps apart when operating at ≤ 85% RATED POWER within the allowed Completion Time of once every 8 hours is adequate. A reduction of reactor thermal power to ≤ 50% RATED POWER puts the core into a condition where COLR limits are sufficiently relaxed such that rod position will not cause the core to violate COLR limits. The allowed completion time of 8 hours provides an acceptable period of time to verify the rod positions or reduce power to ≤ 50% RATED POWER.

Inoperable Rod Limitations (TS 3.10.g)

One inoperable control rod is acceptable provided the potential consequences of accidents are not worse than the cases analyzed in the safety analysis report. A 30-day period is provided for the reanalysis of all accidents sensitive to the changed initial condition.

Rod Drop Time (TS 3.10.h)

The required drop time to dashpot entry is consistent with safety analysis.