WOLF CREEK
NUCLEAR OPERATING CORPORATION

John P. Broschak
Vice President Engineering

February 26, 2014
ET 14-0008

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

Reference: 1) WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005

Subject: Docket No. 50-482: Application for Amendment to Revise Specification 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," for Large Break Loss-of-Coolant Accident Analysis Methodology

Gentlemen:

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Wolf Creek Nuclear Operating Corporation (WCNOC) hereby requests an amendment to Renewed Facility Operating License No. NPF-42 to revise the Wolf Creek Generating Station (WCGS) Technical Specifications (TS). Specification 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," is revised to incorporate WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," (Reference 1) to the list of analytical methods used to determine the core operating limits.

Attachments I through V provide the Evaluation, Markup of TSs, Retyped TS pages, Proposed TS Bases changes, and proposed COLR changes, respectively, in support of this amendment request. Attachment IV and V are provided for information only. Final TS Bases changes will be implemented pursuant to TS 5.5.14, "Technical Specification (TS) Bases Control Program," at the time the amendment is implemented.

It has been determined that this amendment application does not involve a significant hazard consideration as determined per 10 CFR 50.92, "Issuance of amendment." Pursuant to 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," Section (b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment.

A002
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This amendment application was reviewed by the Plant Safety Review Committee. In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," Section (b)(1), a copy of this amendment application, with attachments, is being provided to the designated Kansas State official.

WCNOC requests approval of the proposed amendment by February 26, 2015. It is anticipated that the license amendment, as approved, will be effective upon issuance and will be implemented within 90 days from the date of issuance. This implementation period will provide adequate time for the affected station documents to be revised using the appropriate change control mechanisms.

Enclosure I provides the proprietary Westinghouse Electric Company LLC, LTR-LIS-13-557 Revision 0, P-Attachment, "Application of Westinghouse Best-Estimate Large Break LOCA Methodology to the Wolf Creek Generating Station." Enclosure II provides the non-proprietary Westinghouse Electric Company LLC, LTR-LIS-13-557 Revision 0, NP-Attachment, "Application of Westinghouse Best-Estimate Large Break LOCA Methodology to the Wolf Creek Generating Station." As Enclosure I contains information that is proprietary to Westinghouse Electric Company LLC, it is supported by an affidavit signed by Westinghouse Electric Company LLC, the owner of the information. Enclosure III is the Westinghouse Application for Withholding Proprietary Information from Public Disclosure, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information, which is proprietary to Westinghouse, be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

This letter contains no commitments. If you have any questions concerning this matter, please contact me at (620) 364-4085, or Mr. Michael J. Westman at (620) 364-4009.

Sincerely,

John P. Broschak

John P. Broschak

JPB/rlt

Attachment:
- I Evaluation
- II Proposed Technical Specification Changes (Markup)
- III Revised Technical Specification Pages
- IV Proposed TS Bases Changes (for information only)
- V Proposed COLR Changes (for information only)

Enclosure:
- I Westinghouse Electric Company LLC, LTR-LIS-13-557 Revision 0, P-Attachment, "Application of Westinghouse Best-Estimate Large Break LOCA Methodology to the Wolf Creek Generating Station" (Proprietary)
- II Westinghouse Electric Company LLC, LTR-LIS-13-557 Revision 0, NP-Attachment, "Application of Westinghouse Best-Estimate Large Break LOCA Methodology to the Wolf Creek Generating Station" (Non-Proprietary)
- III Westinghouse Electric Company LLC, CAW-14-3879, "Application for Withholding Proprietary Information from Public Disclosure"

cc: T. A. Conley (KDHE), w/a, w/e (only non-proprietary Enclosures)
M. L. Dapas (NRC), w/a, w/e
C. F. Lyon (NRC), w/a, w/e
N. F. O'Keefe (NRC), w/a, w/e
Senior Resident Inspector (NRC), w/a, w/e

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

John P. Broschak, of lawful age, being first duly sworn upon oath says that he is Vice President Engineering of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the contents thereof; that he has executed the same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By John P. Broschak
John P. Broschak
Vice President Engineering

SUBSCRIBED and sworn to before me this 26th day of Feb., 2014.

GAYLE SHEPHEARD
Notary Public - State of Kansas
My Appt. Expires 7/24/2015

Gayle Shepheard
Notary Public

Expiration Date 7/24/2015

EVALUATION

EVALUATION

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Wolf Creek Nuclear Operating Corporation (WCNOC) hereby requests an amendment to Renewed Facility Operating License No. NPF-42 to revise the Wolf Creek Generating Station (WCGS) Technical Specifications (TS). Specification 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," is revised to incorporate WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," (Reference 1) to the list of analytical methods used to determine the core operating limits. The Nuclear Regulatory Commission (NRC) approved WCAP-16009-P in Reference 2.

2.0 DETAILED DESCRIPTION

Specification 5.6.5a. requires core operating limits to be established and documented in the COLR prior to each reload cycle, or prior to any remaining portion of a reload cycle. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, as listed in Specification 5.6.5b. The current methodology used for development of core operating limits related to the large break loss-of-coolant accident (LOCA) is listed in Specification 5.6.5b.7, which states:

7. WCAP-10266-P-A, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code."

The proposed change replaces the reference to WCAP-10266-P-A with a reference to WCAP-16009-P-A, as follows:

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

Listing only the NRC-approved methodology by topical report number and title is consistent with Amendment No. 144 (Reference 3). Amendment No. 144 adopted TSTF-363, "Revise Topical Report References in ITS 5.6.5, COLR," and the NRC concluded in the safety evaluation that the proposed change to only list the NRC approved methodology by topical report number and title is acceptable. Additionally, in a letter from the NRC to the TSTF (Reference 4) the NRC indicated that the NRC staff does not intend to backfit licensees that have these travelers (TSTF-363, TSTF-408 or TSTF-419) already in their TSs.

Attachment V provides proposed changes to the COLR that includes listing the NRC-approved methodology including the specific revision of the topical report.

3.0 TECHNICAL EVALUATION

Westinghouse obtained generic NRC approval of its original topical report describing best-estimate large break LOCA methodology in 1996. NRC approval of the methodology is documented in the NRC safety evaluation report appended to the topical report (Reference 5).

Westinghouse subsequently underwent a program to revise the statistical approach used to develop the peak cladding temperature (PCT) and oxidation results at the 95th percentile. This method is still based on the Code Qualification Document (CQD) methodology (Reference 5) and follows the steps in the Code Scaling Applicability and Uncertainty (CSAU) methodology. However, the uncertainty analysis (Element 3 in CSAU) is replaced by a technique based on order statistics. The ASTRUM methodology replaces the response surface technique with a statistical sampling method where the uncertainty parameters are simultaneously sampled for each case. The approved ASTRUM evaluation model is documented in WCAP-16009-P-A (Reference 1).

A best-estimate large break LOCA analysis was completed for WCGS. The application of the Westinghouse ASTRUM best-estimate LOCA evaluation model for the large break LOCA analyses, along with consideration of the effects of fuel thermal conductivity degradation (TCD), is summarized in Enclosure I. The results of the WCGS ASTRUM analysis are summarized in Table 1.

Table 1
WCGS Best-Estimate Large Break LOCA Results

10 CFR 50.46 Requirement	Value	Criteria
95/95 Peak Cladding Temperature (^{o}F)	1900	< 2200
95/95 Maximum Local Oxidation (%)	4.21	< 17
95/95 Core-Wide Oxidation (%)	0.1352	< 1

WCNOC previously submitted a license amendment request (Reference 6) requesting approval of changes to the WCGS TSs based on WCAP-16009-P-A. The license amendment request was withdrawn on August 23, 2012 (Reference 7) after discussion with the NRC identified the need to re-perform the best-estimate large break LOCA analysis to account for the effects of fuel TCD. Enclosure I includes a description of the approach used for the modeling of TCD in the WCGS best-estimate large break LOCA analysis.

4.0 REGULATORY EVALUATION

This section addresses the standards of 10 CFR 50.92, “Issuance of amendment,” as well as the applicable regulatory requirements and acceptance criteria.

The proposed amendment revises the Wolf Creek Generating Station (WCGS) Technical Specifications to incorporate WCAP-16009-P-A, “Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM),” (Reference 1) into Specification 5.6.5, “CORE OPERATING LIMITS REPORT (COLR).” The Nuclear Regulatory Commission (NRC) approved WCAP-16009-P in Reference 2.

4.1 Significant Hazards Consideration

Wolf Creek Nuclear Operating Corporation (WCNOC) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," Part 50.92(c), 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 proposed change revises Specification 5.6.5 to incorporate a new large break LOCA analysis methodology. Specifically, the proposed change adds WCAP-16009-P-A to Specification 5.6.5b. as a method used for establishing core operating limits.

   Accident analyses are not accident initiators; therefore, the proposed change does not involve a significant increase in the probability of an accident. The analyses using ASTRUM demonstrated that the acceptance criteria in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," were met. Large break LOCA analyses performed consistent with the methodology in Nuclear Regulatory Commission (NRC) approved WCAP-16009-P-A, including applicable assumptions, limitations and conditions, demonstrate that 10 CFR 50.46 acceptance criteria are met; thus, this change does not involve a significant increase in the consequences of an accident. No physical changes to the plant are associated with the proposed change.

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

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

   Response: No

   The proposed change revises Specification 5.6.5 to incorporate a new large break LOCA analysis methodology. Specifically, the proposed change adds WCAP-16009-P-A to Specification 5.6.5b. as a method used for establishing core operating limits. There are no physical changes being made to the plant as a result of using the Westinghouse ASTRUM analysis methodology in WCAP-16009-P-A for performance of the large break LOCA analyses. Large break LOCA analyses performed consistent with the methodology in NRC approved WCAP-16009-P-A, including applicable assumptions, limitations and conditions; demonstrate that 10 CFR 50.46 acceptance criteria are met. No new modes of plant operation are being introduced. The configuration, operation, and accident response of the structures or components are unchanged by use of the new analysis methodology. Analyses of transient events have confirmed that no transient event results in a new sequence of events that could lead to a new accident scenario. The parameters assumed in the analyses are within the design limits of existing plant equipment.

   In addition, employing the Westinghouse ASTRUM large break LOCA analysis methodology does not create any new failure modes that could lead to a different kind of accident. The design of systems remains unchanged and no new equipment or systems have been installed which could potentially introduce new failure modes or accident sequences. No changes have been made to instrumentation actuation setpoints. Adding the reference to WCAP-16009-P-A in Specification 5.6.5b. is an administrative change that does not create the possibility of a new or different kind of accident.

Therefore, the proposed change does not create the possibility of 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

   The proposed change revises Specification 5.6.5 to incorporate a new large break LOCA analysis methodology. Specifically, the proposed change adds WCAP-16009-P-A to Specification 5.6.5b. as a method used for establishing core operating limits. The analyses using ASTRUM demonstrated that the applicable acceptance criteria in 10 CFR 50.46 are met. Margins of safety for large break LOCAs include quantitative limits for fuel performance established in 10 CFR 50.46. These acceptance criteria are not being changed by this proposed new methodology. Large break LOCA analyses performed consistent with the methodology in NRC approved WCAP-16009-P-A, including applicable assumptions, limitations and conditions, demonstrate that 10 CFR 50.46 acceptance criteria are met; thus, this change does not involve a significant reduction in a margin of safety.

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

4.2 Applicable Regulatory Requirements/Criteria

Section 50.46 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.46), "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," requires, in part, that Emergency Core Cooling System (ECCS) cooling performance be calculated in accordance with an acceptable evaluation model and be calculated for a number of postulated LOCAs of different sizes, locations, and other properties. It also requires that "... uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded."

Section 50.46(b) of 10 CFR 50 also states detailed acceptance criteria for LOCA evaluations. These are as follows:

(1) The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

(2) The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

4.3 Conclusion

Since the issuance of 10 CFR 50, Appendix K, the NRC and the nuclear industry have developed improved thermal-hydraulic computer codes and modes that more accurately and realistically perform accident analysis calculations. Westinghouse has developed the ASTRUM methodology for performing best-estimate large break LOCA analysis as documented in WCAP-16009-P-A. The NRC has approved WCAP-16009-P-A for application to Westinghouse four loop plants. WCGS is a Westinghouse four-loop plant.

A large break LOCA analysis has been performed for WCGS using the ASTRUM methodology. The results demonstrate that the acceptance criteria of 10 CFR 50.46 are met. The proposed change incorporates the best-estimate large break LOCA analysis using ASTRUM into the WCGS licensing basis and revises Specification 5.6.5b. to add WCAP-16009-P-A to the list of NRC-approved methods for establishing core operating limits.

Based on the considerations discussed above, 1) there is a 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 CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amount of effluent that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environment impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)," Revision 0, January 2005.

2. Letter from H. N. Berkow (USNRC) to J. A. Gresham (Westinghouse Electric Company), "Final Safety Evaluation for WCAP-16009-P, Revision 0, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)" (TAC NO. MB9483)," November 5, 2004.

3. License Amendment No. 144, "Wolf Creek Generating Station – Issuance of Amendment RE: Relocation of Cycle Specific Parameters to the Core Operating Limits Report (TAC NO. MB1638)," March 28, 2002. ADAMS Accession No. ML020180190.

4. NRC letter from J. R. Jolicoeur to TSTF, "Implementation Of Travelers TSTF-363, Revision 0, "Revise Topical Report References in ITS 5.6.5, COLR [CORE OPERATING LIMITS REPORT]," TSTF-408, Revision 1, "Relocation of LTOP [LOW TEMPERATURE OVERPRESSURE PROTECTION] Enable Temperature and PORV [POWER-OPERATED RELIEF VALVE] Lift Setting to the PTLR [PRESSURE-TEMPERATURE LIMITS REPORT]," AND TSTF-419, Revision 0, "Revise PTLR Definition and References in ISTS [IMPROVED STANDARD TECHNICAL SPECIFICATION] 5.6.6, RCS [REACTOR COOLANT SYSTEM] PTLR," August 4, 2011. ADAMS Accession No. ML110660285.

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

6. WCNOC Letter ET 10-0025, "Application to Revise Technical Specification 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," for Large Break Loss-of-Coolant Accident Analysis Methodology," November 4, 2010. ADAMS Accession No. ML103200209.

7. WCNOC Letter ET 12-0020, "Withdrawal of License Amendment Request to Revise Technical Specification 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," for Large Break Loss-of-Coolant Accident Analysis Methodology," August 23, 2012. ADAMS Accession No. ML12248A261.

Proposed Technical Specification Changes (Markup)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification."

5. WCNOC Topical Report NSAG-007, "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station."

6. NRC Safety Evaluation Report dated March 30, 1993, for the "Revision to Technical Specification for Cycle 7."

7. ~~WCAP-10266-P-A, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code."~~

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

9. WCAP 10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code."

10. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report."

11. WCAP-8745-P-A, "Design Bases for the Thermal Power ΔT and Thermal Overtemperature ΔT Trip Functions."

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.

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

(continued)

Revised Technical Specification Pages

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification."

5. WCNOC Topical Report NSAG-007, "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station."

6. NRC Safety Evaluation Report dated March 30, 1993, for the "Revision to Technical Specification for Cycle 7."

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

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

9. WCAP 10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code."

10. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report."

11. WCAP-8745-P-A, "Design Bases for the Thermal Power ΔT and Thermal Overtemperature ΔT Trip Functions."

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

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

(continued)

Proposed TS Bases Changes (for information only)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated primarily by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and ECCS pumps play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the centrifugal charging pumps become solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 2) will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

Since the accumulators empty themselves by the beginning stages of the reflood phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

~~For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used~~. The contained water volume is the same as the available deliverable volume for the accumulators. For large breaks, an increase in water volume can be either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core reflooding portion of the transient. The analysis ~~makes a conservative assumption with respect to ignoring or taking credit for~~ line water volume from the accumulator to the check valve. ~~To allow for instrument inaccuracy, an accumulator volume ranging between 6122 gallons and 6594 gallons is specified.~~

For the small break LOCA analysis, a nominal contained accumulator water volume is used, while the large break LOCA analysis samples the accumulator water volume over the specified range of 6119 gallons to 6590 gallons to allow for instrument inaccuracy.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The minimum boron concentration limit is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump boron concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

The ~~large and~~ small break LOCA ~~analyses are~~ analysis is performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity. The large break LOCA analysis samples the accumulator pressure over the range of 568.1 psig to 681.9 psig.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 1 and 3).

The accumulators satisfy Criterion 2 and Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above 1000 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases,

Proposed COLR Changes (for information only)

WOLF CREEK
NUCLEAR OPERATING CORPORATION

Wolf Creek Generating Station
Cycle ~~20~~
Core Operating Limits Report
Revision 0

7. WCAP-10266-P-A, Revision 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," March 1987.

   NRC letter dated November 13, 1986, "Acceptance for Referencing of Licensing Topical Report WCAP-10266 "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code.""

   WCAP-10266-P-A, Addendum 1, Revision 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code Addendum 1: Power Shape Sensitivity Studies," December 1987.

   NRC letter dated September 15, 1987, "Acceptance for Referencing of Addendum 1 to WCAP-10266, BASH Power Shape Sensitivity Studies."

   WCAP-10266-P-A, Addendum 2, Revision 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code Addendum 2: BASH Methodology Improvements and Reliability Enhancements," May 1988

   NRC letter dated January 20, 1988, "Acceptance for Referencing Topical Report Addendum 2 to WCAP-10266, Revision 2, "BASH Methodology Improvements and Reliability Enhancements."

INSERT A

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

   NRC Safety Evaluation Report dated May 17, 1988, "Acceptance for Referencing of Westinghouse Topical Report WCAP-11596 - Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores."

9. WCAP 10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code," September 1988.

   NRC letter dated June 23, 1986, "Acceptance for Referencing of Topical Report WCAP 10965-P and WCAP 10966-NP."

10. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995.

    NRC Safety Evaluation Reports dated July 1, 1991, "Acceptance for Referencing of Topical Report WCAP-12610, 'VANTAGE+ Fuel Assembly Reference Core Report' (TAC NO. 77258)."

    NRC Safety Evaluation Report dated September 15, 1994, "Acceptance for Referencing of Topical Report WCAP-12610, Appendix B, Addendum 1, 'Extended Burnup Fuel Design Methodology and ZIRLO Fuel Performance Models' (TAC NO. M86416)."

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

    NRC Safety Evaluation Report dated April 17, 1986, "Acceptance for Referencing of Licensing Topical Report WCAP-8745(P)/8746(NP), 'Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions.'"

INSERT A

WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)," Revision 0, January 2005.

NRC letter dated November 5, 2004, "Final Safety Evaluation for WCAP-16009-P, Revision 0, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)" (TAC NO. MB9483)."