WOLF CREEK
NUCLEAR OPERATING CORPORATION

October 18, 2017

Jaime H. McCoy
Vice President Engineering

ET 17-0024

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

References: 1) Letter ET 17-0001, dated January 17, 2017, from J. H. McCoy, WCNOC, to USNRC

2) Electronic mail dated September 21, 2017, from B. K. Singal, USNRC, to W. T. Muilenburg, WCNOC, "Request for Additional Information - License Amendment Request for Transition to Westinghouse Core Design and Safety Analysis Including Adoption of Alternative Source Term Wolf Creek Generating Station (CAC No. MF 9307)"

Subject: Docket No. 50-482: Response to Request for Additional Information Regarding License Amendment Request to Revise Technical Specifications to Transition to Westinghouse Core Design and Safety Analyses Including Adoption of Alternative Source Term

To Whom It May Concern:

Reference 1 provided the Wolf Creek Nuclear Operating Corporation (WCNOC) application to revise the Wolf Creek Generating Station (WCGS) Technical Specifications (TS). The proposed amendment would support transition to the Westinghouse Core Design and Safety Analysis methodologies. In addition, the amendment request included revising the WCGS licensing basis by adopting the Alternative Source Term radiological analysis methodology in accordance with 10 CFR 50.67, "Accident Source Term." Reference 2 provided a request for additional information related to the application. The Attachment provides WCNOC's response to the request for additional information.

The additional information does not expand the scope of the application and does not impact the no significant hazards consideration determination presented in Reference 1.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," a copy of this submittal is being provided to the designated Kansas State official.

A001
NRR

Enclosure I provides the non-proprietary Westinghouse Electric Company LLC Attachment 2 to LTR-LIS-17-353, Revision 0, "Response to Nuclear Regulatory Commission Request for Additional Information Regarding Wolf Creek Generating Station Transition to Westinghouse Methodologies and Safety Analyses." Enclosure II provides the proprietary Westinghouse Electric Company LLC Attachment 3 to LTR-LIS-17-353, Revision 0. As Enclosure II contains information proprietary to Westinghouse Electric Company LLC, it is supported by an affidavit signed by Westinghouse Electric Company LLC, the owner of the information. 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 10 CFR 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 2.390 of the Commission's regulations. This affidavit, along with Westinghouse authorization letter, CAW-17-4663, Revision 0, "Application for Withholding Proprietary Information from Public Disclosure," is contained in Enclosure III.

This letter contains no commitments. If you have any questions concerning this matter, please contact me at (620) 364-4156, or Cynthia R. Hafenstine at (620) 364-4204.

Sincerely,

Jaime H. McCoy

Jaime H. McCoy

JHM/rlt

Attachment: I Response to Request for Additional Information

Enclosures:

I Attachment 2 to LTR-LIS-17-353, Revision 0, "Response to Nuclear Regulatory Commission Request for Additional Information Regarding Wolf Creek Generating Station Transition to Westinghouse Methodologies and Safety Analyses" – Non-Proprietary

II Attachment 3 to LTR-LIS-17-353, Revision 0, "Response to Nuclear Regulatory Commission Request for Additional Information Regarding Wolf Creek Generating Station Transition to Westinghouse Methodologies and Safety Analyses" - Proprietary

III CAW-17-4663, Revision 0, "Application for Withholding Proprietary Information from Public Disclosure"

cc: K. M. Kennedy (NRC), w/a, w/e
B. K. Singal (NRC), w/a, w/e
K. S. Steves (KDHE), w/a, w/e (Enclosure I only)
N. H. Taylor (NRC), w/a, w/e
Senior Resident Inspector (NRC), w/a, w/e

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Jaime H. McCoy, 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 Jaime H McCoy
Jaime H. McCoy
Vice President Engineering

SUBSCRIBED and sworn to before me this 18th day of October, 2017.

STATE OF NOTARY PUBLIC KANSAS
GAYLE SHEPHEARD
My Appointment Expires
July 24, 2019

Gayle Shepheard
Notary Public

Expiration Date 7/24/2019

Response to Request for Additional Information

Reference 1 provided the Wolf Creek Nuclear Operating Corporation (WCNOC) application to revise the Wolf Creek Generating Station (WCGS) Technical Specifications (TS). The proposed change replaces the WCNOC methodology for performing core design, non-loss-of-coolant-accident (non-LOCA) and LOCA safety analyses to the standard Westinghouse methodologies for performing these analyses, and associated TS changes. Reference 1 would also revise WCGS TS's and the Updated Safety Analysis Report Chapter 15 radiological consequence analyses using an updated accident source term consistent with Title 10 of the Code of Federal Regulations (10 CFR), Section 50.67, "Accident source term." Reference 2 provided a request for additional information related to the application. The specific NRC question is provided in italics.

1. *Title 10 of the Code of Federal Regulations (10 CFR) Section 50.46(b)(5) requires that "[a]fter any calculated successful initial operation of the ECCS [emergency core cooling system], 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." Additional guidance for the performance of long term core cooling (LTCC) analyses was provided in the Nuclear Regulatory Commission (NRC) staff letters dated August 1 and November 23, 2005 letters (References 1 and 2, respectively) to Westinghouse Electric Company, LLC (Westinghouse) listing concerns with, and suspending NRC approval for use of, the Westinghouse LTCC methodology documented in CENPD-254-P-A (Reference 3).*

   The licensee submitted analyses performed by Westinghouse, documented in Section 2.7.4.2 of WCAP-17658-NP (Reference 4). The analyses seek to demonstrate that LTCC will not be inhibited by precipitation of boric acid. Because the method, code(s), and assumptions used to calculate core boric acid concentrations following a loss-of-coolant accident (LOCA) are not clearly documented in the licensee's submittal, the NRC staff requests the following additional information regarding the method used to demonstrate adequate post-LOCA long term core cooling in compliance with the 10 CFR 50.46(b)(5).

   a. *Please identify and provide a technical description of the code used to perform the evaluation, which at a minimum should include a basic overview of the modeling, major assumptions, and any pre- or post-processing steps associated with the core boron concentration calculations.*

   Response: A technical description of the computer code used to perform the Wolf Creek Generating Station (WCGS) core boron concentration calculations, SKBOR Version 9.0, is enclosed in Attachment 2 (non-proprietary) and Attachment 3 (proprietary).

   There are no significant pre-processing steps such as void distribution associated with the boron concentration calculations. There are, however, several post-processing steps such as loop differential pressure (a.1), hot leg entrainment criteria (a.2), and core dilution following the transfer to hot leg recirculation (a.3) associated with the calculations.

a.1 Loop Differential Pressure

Presently, the SKBOR computer code does not have the capability to evaluate the loop differential pressure to confirm that there is sufficient margin in the calculated mixing volume to account for loop pressure drop effects. The evaluation is performed as a post-processing step.

The time-dependent static head of the liquid in the downcomer and the time-dependent collapsed liquid level in the inner vessel are calculated using the time-dependent steaming rate calculated by SKBOR. This establishes the supportable loop pressure drop, $\Delta P_{Supportable}$, the difference between the static head of the liquid in the downcomer, ΔP_{DC}, and the static head of the liquid in the inner vessel, ΔP_{IV}:

$$\Delta P_{Supportable} = \Delta P_{DC} - \Delta P_{IV}$$

The mixing volume is justified if the following condition is met:

$$\Delta P_{Supportable} \geq \Delta P_{Loop}$$

therefore,

$$\Delta P_{Loop} \leq \Delta P_{DC} - \Delta P_{IV}$$

Static Head of the Downcomer Liquid

Assuming a downcomer void fraction, α_{DC}, due to boiling, the downcomer collapsed liquid level (CLL_{DC}) is equal to the bottom of the cold leg elevation referenced to the bottom of the active fuel (ΔZ_{DC}) adjusted to account for the void fraction:

$$CLL_{DC} = \Delta Z_{DC} \times (1 - \alpha_{DC})$$

Static Head of the Inner Vessel Liquid

The inner vessel collapsed liquid level (CLL_{IV}) is calculated, which is then be used with the downcomer collapsed liquid level (CLL_{DC}) to calculate a supportable loop pressure drop:

$$CLL_{IV} = \frac{VCORET - (LPFRAC \times VLP)}{ACC}$$

Where:

$VCORET$ = Time-dependent mixing volume calculated by SKBOR, ft^3

$LPFRAC$ = Fraction of lower plenum credited

VLP = Volume of lower plenum, ft^3

ACC = Core flow area, ft^2

Loop Pressure Drop

The loop pressure drop is calculated using the following form of the Darcy formula by considering the steam flow from the core is equally divided among all loops:

$$\Delta P_{loop} = K_{loop} \times \frac{1}{\rho_g} \times \left(\frac{\dot{m}_{boil}}{N}\right)^2$$

$$= K_{loop} \times \frac{1}{\rho_g} \times \left(\frac{\dot{m}_{boil}}{N}\right)^2 \times \left(\frac{7.48\ gal}{ft^3}\right)^2 \times \left(\frac{60\ s}{min}\right)^2 \times \left(\frac{g}{g_c}\right) \times \left(\frac{ft}{12\ in}\right)^2$$

where:

K_{loop} = hydraulic loss coefficient with locked reactor coolant pump rotor, $\frac{ft}{gpm^2}$

ρ_g = gas density, $\frac{lbm}{ft^3}$

$\dot{m}_{boil}$ = boil-off mass flow rate, $\frac{lbm}{s}$

N = number of loops

g = gravitational acceleration = 32.174 $\frac{ft}{s^2}$

g_c = gravitational constant = 32.174 $\frac{ft\text{-}lbm}{lbf\text{-}s^2}$

Assessment of Loop Pressure Drop Effects

The loop pressure drop that can be supported is:

$$\Delta P_{Supportable} = \Delta P_{DC} - \Delta P_{IV}$$

and the mixing volume used in SKBOR is justified when:

$$\Delta P_{Loop} \leq \Delta P_{DC} - \Delta P_{IV}$$

where:

$$\Delta P_{DC} = CLL_{DC} \times \rho_f \times \left(\frac{g}{g_c}\right) \times \left(\frac{ft}{12\ in}\right)^2$$

$$= \Delta Z_{DC} \times (1 - \alpha_{DC}) \times \rho_f \times \left(\frac{g}{g_c}\right) \times \left(\frac{ft}{12\ in}\right)^2$$

and:

$$\Delta P_{IV} = CLL_{IV} \times \rho_f \times \left(\frac{g}{g_c}\right) \times \left(\frac{ft}{12\ in}\right)^2$$

However, as the boric acid concentration in the core increases so does the density of solution. Therefore, ΔP_{IV} is adjusted as follows:

$$\Delta P_{IV}^{adj} = \Delta P_{IV} \times \left(1 + \Delta\rho_{solute}\left(C_{Core} - C_{Sump}\right)\right)$$

where:

C_{Core} = core region boric acid in weight percent, w/o

C_{Sump} = sump boric acid concentration, w/o

$\Delta\rho_{solute}$ = density change due to solute = $0.0034/_{w/o}$

When accounting for the density change due to solute in the mixing volume, the supportable loop pressure drop, $\Delta P_{Supportable}$, becomes:

$$\Delta P_{Supportable} = \Delta P_{DC} - \Delta P_{IV}^{adj}$$

And the loop pressure drop margin, ΔP_{Loop}^{margin}, is:

$$\Delta P_{Loop}^{margin} = \Delta P_{Supportable} - \Delta P_{Loop}$$

Therefore, the mixing volume is justified when:

$$\Delta P_{Loop}^{margin} \geq 0$$

a.2 Hot Leg Entrainment Criteria

The liquid film entrainment threshold in the hot leg is evaluated by applying both the Wallis-Steen liquid entrainment onset criterion (Reference 1) and the Ishii-Grolmes inception criteria (Reference 2). These entrainment correlations are valid for flow conditions where the liquid phase does not take up a significant volume of the pipe (such as in the hot legs post-LOCA) and viscous effects in the liquid are not dominant, i.e., the liquid phase is in the turbulent regime.

<u>Wallis-Steen Liquid Entrainment Onset Criterion</u>

The liquid entrainment onset correlation (Eq. 12.43 of Reference 1) can be rearranged and expressed as follows to solve for, j_g, the superficial velocity of the gas phase:

$$j_g \geq \pi_2 \left(\frac{\rho_f}{\rho_g}\right)^{1/2} \left(\frac{\sigma}{\mu_g}\right)$$

where:

σ = surface tension of liquid, $\frac{lbf}{ft}$

μ_g = viscosity of gas, $\frac{lbf\text{-}s}{ft^2}$

ρ_f = density of liquid, $\frac{lbm}{ft^3}$

ρ_g = density of gas, $\frac{lbm}{ft^3}$

g_c = gravitational constant = $32.174 \frac{ft\text{-}lbm}{lbf\text{-}s^2}$

π_2 is the dimensionless gas velocity for onset of entrainment. Steen suggested a value of 2.46×10^{-4}; however, a more conservative value of 2.0×10^{-4} is used in the evaluation.

The total gas mass flow rate, $\dot{m}_{gas}$, at the entrainment threshold is calculated:

$$\dot{m}_{gas} = j_g \times (N \times A_{hot\ leg}) \times \rho_g$$

where:

N = number of loops

$A_{hot\ leg}$ = hot leg flow area, ft^2

The decay heat fraction can be related to the core steam mass flow rate as follows, where PZERO is the licensed power including calorimetric uncertainty.

$$\dot{m}_{gas} = \frac{PZERO \times \left(\frac{P}{P_0}\right) \times \left(948 \frac{Btu}{s \cdot MW}\right)}{h_{fg}}$$

$$\frac{P}{P_0} = \frac{\dot{m}_{gas} \times (h_{fg})}{PZERO \times \left(948 \frac{Btu}{s \cdot MW}\right)}$$

with:

h_{fg} = latent heat of vaporization, $\frac{Btu}{lbm}$

<u>Ishii-Grolmes Liquid Entrainment Onset Criterion</u>

The Ishii-Grolmes entrainment inception criterion has three separate regimes based on the liquid film Reynolds number, Re_f, in the channel. Two of the regimes are further subdivided based on the magnitude of the liquid viscosity number, N_μ. Based on the hot leg injection flow rate, the liquid film will not be in the low Reynolds number regime $(Re_f \leq 160)$. Of the two remaining regimes, transition and rough turbulent, the rough turbulent regime requires the lowest gas velocity to entrain droplets from the film as shown in Figures 1 and 3 of the paper (Reference 2). In short, in the rough turbulent regime it is easier to strip droplets off an already unstable interface; whereas, in the laminar and transition regimes, a higher gas velocity is needed to create the instability at the interface. The entrainment onset criterion for the rough turbulent regime can be applied irrespective of the liquid flow direction and is, therefore, applicable for countercurrent flow that occurs in the hot leg during simultaneous hot and cold side injection.

The liquid entrainment onset correlation per Reference 2 can be expressed as follows:

$$j_g \geq N_\mu^{0.8}\left(\frac{\rho_f}{\rho_g}\right)^{0.5}\left(\frac{\sigma}{\mu_f}\right) \qquad \text{for } N_\mu < \frac{1}{15}$$

N_μ is the liquid viscosity number:

$$N_\mu = \frac{\mu_f}{\left[\rho_f \frac{\sigma}{g_c}\left(\sqrt{\frac{\sigma g_c}{g\Delta\rho}}\right)\right]^{1/2}} \qquad \text{with } \Delta\rho = \rho_f - \rho_g$$

The total gas mass flow rate, $\dot{m}_{gas}$, at the entrainment threshold is calculated:

$$\dot{m}_{gas} = j_g \times \left(N \times A_{hot\,leg}\right) \times \rho_g$$

where:

N = number of loops

$A_{hot\,leg}$ = hot leg flow area, ft^2

The decay heat fraction can be related to the core steam mass flow rate as follows, where PZERO is the licensed power including calorimetric uncertainty.

$$\dot{m}_{gas} = \frac{PZERO \times \left(\frac{P}{P_0}\right) \times \left(948 \frac{Btu}{s \cdot MW}\right)}{h_{fg}}$$

$$\frac{P}{P_0} = \frac{\dot{m}_{gas} \times \left(h_{fg}\right)}{PZERO \times \left(948 \frac{Btu}{s \cdot MW}\right)}$$

Evaluation of Hot Leg Entrainment Criteria

With the decay heat fraction $\left(\frac{P}{P_0}\right)$ known, the corresponding time following shutdown to drop below the hot leg entrainment threshold for both the Wallis-Steen and Ishii-Grolmes correlations is determined from tabulated values based on Appendix K decay heat. The latest time to drop below the entrainment threshold is then compared to the earliest time to initiate the transfer to hot leg recirculation to ensure that the ECCS recirculation flow to the hot legs will be effective.

a.3 Core Dilution Following the Transfer to Hot Leg Recirculation

A simplistic post-processing step is performed to demonstrate dilution of the boron concentration in the mixing volume following the initiation of hot leg recirculation. This is achieved using the calculation results from the SKBOR computer code to define the volume and boron concentration of the mixing volume when hot leg recirculation begins. The post-processing calculation simply adds dilute solution from the sump at a rate corresponding to the hot leg recirculation dilution flow (ECCS flow in excess of boil-off) while continuing to account for the boron left behind due to decay heat boil-off as calculated by SKBOR. The mixing volume in the post-processing dilution calculation is assumed to continuously expand as dilution flow (ECCS flow in excess of boil-off) is added thus resulting in a slower rate of dilution than would occur if mixing volume were conserved.

b. *Please provide a description of the void model used in the method, including a discussion of the conditions under which it was validated and whether the void calculation is axially dependent.*

Response: The void fraction model used in the method is the Cunningham-Yeh model presented in References 3, 4, and 5. The implementation of this model in the boron concentration calculation is described in the technical description of the SKBOR computer code in the response to RAI 1, Item a. The void model used in the SKBOR code allows the user to specify the number of axial cells to be used in the void calculation. The user guidance recommends using a sufficient number of cells to ensure that the cell length is ≤ 3 inches, e.g., 48 cells for a typical 12 foot core. The power distribution used in the calculation is assumed to be uniform both axially and radially.

The void correlation has been validated against experimental measurements over a wide pressure range in numerous publications and conference proceedings. For instance, in Reference 5 a pressure range of 0.14 MPa to 15 MPa (20 psia to 2175 psia) was considered. More recently, the validation was extended to low-pressure, low-flow conditions expected during the recirculation phase following a LOCA. This validation which considered a pressure range of 102 kPa to 340 kPa (15 psia to 50 psia) was published in Reference 6. Other correlations were also reviewed in Reference 6 and authors noted that there is no significant improvement in mean error or standard deviation in complex correlations in comparison to the more simplistic Cunningham-Yeh model.

c. *Please provide additional detail on the assumptions regarding reactor coolant system pressure and their impacts on the void calculations. The response should also include a discussion on how the loop pressure drop calculation requested in the NRC staff letters August 1, 2005 and November 23, 2005 (References 1 and 2) has been incorporated into the methodology.*

Response: The void calculations as implemented in the SKBOR computer code are performed assuming saturated conditions at the defined system pressure, i.e., no ECCS subcooling is modeled. Accordingly, the void fraction decreases as the system pressure increases. This effect is taken into account in the large and small break loop pressure drop assessments.

The boron concentration calculation performed by SKBOR does not directly incorporate the loop pressure drop calculation requested in the NRC staff letters. The effect of the loop pressure drop is therefore evaluated in a post-processing step as described in RAI Response a.1, Loop Pressure Differential. The evaluation is performed to demonstrate the static head of liquid in the reactor vessel downcomer can support the two-mixture level used in the mixing volume calculation while accounting for the additional backpressure due to the loop pressure drop. This evaluation is done for both the large break and small break boron concentration calculations using the void fraction and steaming rate at the corresponding pressure, i.e., 14.7 psia for the large break calculation and 120 psia for the small break calculation.

d. *Please provide additional detail on the regions of the vessel and core assumed to form the mixing volume, including assumptions regarding voiding in these regions.*

Response: The mixing volume consists of the core region, the upper plenum region from the top of the core to the bottom of the hot leg, and the lower plenum of the reactor vessel. Although the configuration of the reactor internals is an upflow barrel-baffle design which has direct communication between the core and barrel-baffle region, the volume within the barrel-baffle region was not credited in the boron concentration calculations for WCGS.

The time-dependent core region void fraction is calculated at each timestep in the boron concentration calculation using the void fraction model described in in the technical description of the SKBOR computer code in the response to RAI 1, Item a.

The core exit void fraction is applied to the upper plenum region from the top of the core to the bottom of the hot leg. This approach is conservative since the increase in flow area in the upper plenum will reduce the vapor superficial velocity; thus, void fraction in this region relative to the core exit. The result of this modeling approach is a smaller mixing volume than can be credited in the boron concentration calculation.

The volume of the lower plenum is reduced by 50 percent to account for the boric acid concentration gradient that will exist in this region.

e. *Please provide a description of the method used to verify that sufficient decay heat removal will be provided after the initial switch to cold leg recirculation and the subsequent switch to hot leg injection.*

Response: The method used to verify sufficient decay heat removal during cold leg recirculation was communicated in Nuclear Safety Advisory Letter (NSAL) 95-001 (Reference 7). In this NSAL, Westinghouse recommends that the minimum ECCS flow during cold leg recirculation should be at least 1.2 times the decay heat boil-off at the time cold leg recirculation is initiated. For both large and small break scenarios, the WCGS ECCS performance exceeds the recommended minimum cold leg recirculation flow after the switch to cold leg recirculation.

The method used to verify sufficient decay heat removal during hot leg recirculation was communicated in NSAL-92-010 (Reference 8) and NSAL 95-001 (Reference 7). For units such as Wolf Creek that simultaneously provide ECCS recirculation flow to both the hot legs and cold legs following the switch to hot leg recirculation, Westinghouse recommends that the minimum ECCS recirculation flow should be at least 1.3 times the decay heat boil-off to the hot legs and at least 1.2 times the decay heat boil-off to the cold legs at the time hot leg recirculation is initiated. For both large and small break scenarios, the Wolf Creek ECCS performance exceeds the recommended minimum simultaneous hot leg and cold leg recirculation flow after the switch to hot leg recirculation.

RAI 1 References:

1. Wallis, G.B., "One-dimensional Two-phase Flow," McGraw-Hill Book Company, 1969.

2. Ishii, M., and Grolmes, M.A., "Inception Criteria for Droplet Entrainment in Two-Phase Concurrent Film Flow," AIChE Journal, Vol. 21, No. 2, pp. 308-318, March 1975.

3. Cunningham, J.P., and Yeh, H.C., "Experiments and Void Correlation for PWR Small-Break LOCA Conditions," Trans. ANS 17, p. 369-370, 1973.

4. Hochreiter, L.E., and Yeh, H.C., "Mass Effluence During FLECHT Forced Reflood Experiments," Nuclear Engineering and Design, 60, p. 413-429, 1980.

5. Yeh, H.C., "Modification of Void Fraction Calculation," Proceedings of the Fourth International Topical Meeting on Nuclear Thermal-Hydraulics, Operations and Safety, Volume 1, Taipei, Taiwan, June 1988.

6. Frepoli, C., and Ohkawa, K., "Void Fraction Predictions in Rod Bundles at Low-Pressure Low-Flow Conditions Based on Cunningham-Yeh Model," Proceedings of ICONE12, 12th International Conference on Nuclear Engineering, INCONE12-49437, April 2004.

7. Westinghouse Nuclear Safety Advisory Letter, NSAL 95-001, "Minimum Cold Leg Recirculation Flow," January 1995.

8. Westinghouse Nuclear Safety Advisory Letter, NSAL-92-010, "Hot Leg Switchover Methodology," January 1993.

2. *Footnote 11 of Regulatory Guide (RG) 1.183 (Reference 5) contains acceptable gap release fractions for LOCA and non-LOCA transients in Section 3.2, Tables 2 and 3 (for pressurized water reactors). The gap fractions reported in this table assume a linear heat generation rate (LHGR) of 6.2 kilowatts per foot (kW/ft) for peak burnups beyond 54 gigawatt-days per metric ton of uranium (GWd/MTU). However, the footnote allows licensees to perform fission gas release calculations for the NRC staff to review on a case-by-case basis, provided that the calculations use a power history that bounds the limiting plant-specific power history.*

   The licensee's fuel handling accident analysis methodology (reported in Reference 6), stated that the assumed gap release fractions were based on values reported in NUREG/CR-5009 (Reference 7) for high burnup fuel. As discussed in a response to an NRC request for additional information (RAI), the licensee considers these gap fractions to be applicable up to a peak LHGR of 20.5 kW/ft for burnups exceeding 50 GWd/MTU (Reference 8). In Reference 9, the licensee justified the use of NUREG/CR-5009 gap fractions by comparing them to calculations performed using the ANSI/ANS-5.4-2011 standard for fission gas release, as documented in Reference 10.

   However, Appendix A to Reference 10 indicates that the calculations the licensee compared the NUREG/CR-5009 gap fractions to, were performed using a maximum LHGR of 12.2 kW/ft that decreases with burnup to 7.0 kW/ft. The NRC staff is concerned that an assumed

LHGR of 20.5 kW/ft for burnups up to 62 GWd/MTU is unrealistically high and that the gap fractions provided in NUREG/CR-5009 will not be applicable at such high power densities. Please provide further justification for the gap fractions assumed in the fuel handling accident analysis. The justification should provide an analysis using NRC-approved methodologies and a power history that bounds limiting plant-specific power histories at Wolf Creek, per RG 1.183 Footnote 11. One example of such an analysis that was recently accepted by the NRC is documented in Reference 11.

Response: The fuel handling accident dose analysis assumption that 100% of the fuel exceeds the Regulatory Guide (RG) 1.183, Table 3, Footnote 11 gap fraction applicability limits (peak burnup up to 62,000 MWD/MTU and a maximum of 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU) is a bounding assumption that provides fuel management flexibility. The selection of gap fractions based on RG 1.25 as modified by NUREG/CR-5009 provides a conservative estimate of the dose consequences. The reference cycle for the Wolf Creek methods transition (Cycle 19) had no exceedances of the Footnote 11 applicability limits. Cycle 19 is representative of typical Wolf Creek core designs, and thus, it is expected that, for typical Wolf Creek core designs, there would be no exceedances of the Footnote 11 applicability limits. However, there is the possibility that an atypical core design could result in a limited number of rods exceeding the Footnote 11 applicability limits. For example, Wolf Creek Cycle 23 is predicted to have a limited number of rods (much less than 10% of any assembly) that exceed the Footnote 11 applicability limits. While exceeding the Footnote 11 applicability limits, the linear heat rates and burnups are within the PNNL-1221, Rev. 1, Table 2.9 gap fraction applicability limits (12.2 kw/ft up to 35GWD/MTU, decreasing to 7.0 kw/ft at 65 GWD/MTU). Thus, Composite gap fractions consisting of RG 1.183, Table 3 gap fractions for the 90% of the fuel meeting the Footnote 11 applicability limits and PNNL-1221, Rev. 1 Table 2.9 gap fractions for the remaining 10% are calculated and presented in Table 1. These composite gap fractions, which are based upon plant-specific power history and NRC approved methodologies, are bounded by the analyzed gap fractions.

Table 1: Fuel Handling Accident Gap Fractions

Nuclide	RG 1.183, Table 3	PNNL-1221, Rev. 1, Table 2.9	Composite	Analyzed
Kr-85	0.10	0.38	0.128	0.3
I-131	0.08	0.08	0.08	0.12
I-132	0.05	0.09	0.054	0.1
Other Nobles	0.05	0.08	0.053	0.1
Other Halogens	0.05	0.05	0.05	0.1

References:

1. Letter ET 17-0001, dated January 17, 2017, from J. H. McCoy, WCNOC, to USNRC

2. Electronic mail from B. K. Singal, USNRC, to W. T. Muilenburg, WCNOC, "Request for Additional Information - License Amendment Request for Transition to Westinghouse Core Design and Safety Analysis Including Adoption of Alternative Source Term Wolf Creek Generating Station (CAC No. MF 9307)," September 21, 2017. ADAMS Accession No. ML17265A014.

Attachment 2 to LTR-LIS-17-353, Revision 0, "Response to Nuclear Regulatory Commission Request for Additional Information Regarding Wolf Creek Generating Station Transition to Westinghouse Methodologies and Safety Analyses" – Non-Proprietary
(19 pages)

Attachment 2

SKBOR: A Computer Code for Calculating the Accumulation of Boric Acid in the Reactor Vessel [Non-Proprietary]

TABLE OF CONTENTS

1.0 Introduction

This report describes Version 9.0 of the SKBOR computer code used in the Wolf Creek Generating Station (WCGS) boric acid precipitation control (BAPC) analysis. The SKBOR computer code is used to determine the time at which emergency core cooling system (ECCS) recirculation should be realigned to the reactor coolant system (RCS) hot legs to prevent boron precipitation in the long term post-LOCA.

Section 2.0 identifies the limits of applicability regarding the calculations performed by the code. Section 3.0 describes the SKBOR problem formulation and initial conditions and presents the governing equations that are solved by the code. Section 4.0 describes the input data used by the program. Section 5.0 describes the contents of the graphics output binary file that is generated by SKBOR. Finally, Section 6.0 summarizes the releases of SKBOR on UNIX® and Linux® platforms.

2.0 Limits of Applicability

The void correlation used in SKBOR has only been validated for pool boiling scenarios in rod bundle geometries such as for Westinghouse 3-/4-loop nuclear steam supply system (NSSS) such as WCGS and Combustion Engineering NSSS during the cold leg recirculation phase. Presently, different methods are used to calculate the void fraction for Westinghouse 2-loop NSSS with upper plenum injection.

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3.0 Problem Formulation, Initial Conditions, and Governing Equations

A typical SKBOR calculation considers two volumes: one representing the effective vessel mixing volume (referred to as the CORE) and one representing the remaining system inventory (referred to as the SUMP). In some cases, a third volume representing steam/water mixing in the reactor vessel upper plenum (referred to as the UP) is also considered. All mass storage is assumed to occur in either the CORE or SUMP, with any mass entering the UP in a given timestep assumed to return to either the CORE or SUMP at the end of the timestep. Figures 3-1 and 3-2 illustrate the mass and boron calculations in SKBOR.

3.1 Initial Conditions

The initial conditions assumed in SKBOR can be summarized as follows:

- The calculation is initiated at the user-specified start time $TSTART\ (s)$ and a system pressure of $PCORE\ (psia)$ that is assumed to remain constant throughout the calculation.
- The CORE is initially assumed to be full of borated liquid, with a concentration (WTF_{CORE}, weight fraction) equal to the initial system-average boron concentration or a user-specified boron concentration with a density (ρ_{CORE}, lbm/ft^3) given by the following equation:

  $$\rho_{CORE} = \rho_f \times (1 + 0.1629 \times WTF_{CORE})$$

  where $\rho_f\ (lbm/ft^3)$ represents the saturated liquid density of water at a system pressure of $PCORE\ (psia)$.

  Denoting the effective vessel mixing volume as $V_{CORE}\ (ft^3)$, the initial CORE mass, $M_{T,CORE}\ (lbm)$, is:

  $$M_{T,CORE} = V_{CORE} \times \rho_{CORE}$$

  and the CORE boron, $M_{B,CORE}\ (lbm)$, and water, $M_{W,CORE}\ (lbm)$, masses are:

  $$M_{B,CORE} = M_{T,CORE} \times WTF_{CORE}$$

  $$M_{W,CORE} = M_{T,CORE} \times (1 - WTF_{CORE})$$

- The SUMP is initially assumed to contain borated liquid with a concentration (WTF_{SUMP}, weight fraction) equal to the initial system-average boron concentration. Denoting the total system mass as $M_{T,TOT}\ (lbm)$, the initial SUMP mass, $M_{T,SUMP}\ (lbm)$, is:

  $$M_{T,SUMP} = M_{T,TOT} - M_{T,CORE}$$

  Then, the SUMP boron, $M_{B,SUMP}\ (lbm)$, and water, $M_{W,SUMP}\ (lbm)$, masses are:

  $$M_{B,SUMP} = M_{T,SUMP} \times WTF_{SUMP}$$

  $$M_{W,SUMP} = M_{T,SUMP} \times (1 - WTF_{SUMP})$$

3.2 Governing Equations

The mass and boron calculations in standard SKBOR application are illustrated in Figures 3-1 and 3-2, respectively, and can be summarized as follows for a given timestep, $\Delta t\ (s)$ (note that SI and $CONDENSATION$ terms only apply when $WSI \neq 0$, see Section 4.7):

- The core fluid density, ρ_{CORE}, is calculated at the beginning of the timestep.
- The decay heat mass boil-off over the timestep Δt is computed as:

$$M_{T,BOIL} = \frac{Q_{core} \times (P/P_0) \times \Delta t}{h_{fg} + (h_f - h_{LP})}$$

where:

$M_{T,BOIL}$	=	decay heat mass boil-off over timestep (lbm)
Q_{core}	=	initial core power level (BTU/s)
P/P_0	=	normalized core power fraction at beginning of timestep
h_{fg}	=	enthalpy of formation (BTU/lbm)
$(h_f - h_{LP})$	=	lower plenum subcooling (BTU/lbm)

- The CORE and SUMP boron, water, and total masses at the end of the timestep are computed from the values at the beginning of the timestep using the following information:
  - The mass exiting the CORE over the timestep Δt (i.e., $M_{T,BOIL}$) is assumed to consist of unborated vapor at enthalpy h_g.
  - Boric acid is added from the sump at density ρ_f and concentration WTF_{SUMP}, as required to keep V_{CORE} full.
  - The vapor exiting the core is assumed to condense fully in containment and return to the sump as unborated liquid.
  - The total system mass remains constant, with all mass storage assumed to occur in either the CORE or SUMP.
- Denoting boron mass as M_B and total (i.e., boron + water) mass as M_T, the CORE and SUMP boron concentrations are computed using the following general equations:

$$C_B\ (weight\ fraction) = M_B/M_T$$

- The hot leg switchover time is defined as the time at which the core boron concentration reaches the assumed boric acid solubility limit (weight percent, w/o).

3.3 Void Fraction Model

The void fraction model used in SKBOR is the Yeh correlation (References 1 through 3). The model is summarized in this section.

3.3.1 Nomenclature

j_g, j_f, j	Superficial velocity, $\left(\frac{ft}{s}\right)$, of the gas, liquid, and mixture
Q_{core}	Core power, $\left(\frac{Btu}{s}\right)$
q''	Heat flux, $\left(\frac{Btu}{s\text{-}ft^2}\right)$
P_H	Fuel rod heated perimeter, (ft)
P/P_0	Decay heat power fraction
h_f, h_g	Saturation enthalpies for the liquid and gas phases, $\left(\frac{Btu}{lbm}\right)$
h_{fg}	Heat of vaporization, $(h_g - h_f)$, $\left(\frac{Btu}{lbm}\right)$
G	Mass flux, $\left(\frac{lbm}{s\text{-}ft^2}\right)$
$\dot{m}$	Mass flow rate, $\left(\frac{lbm}{s}\right)$
A_{core}	Core flow area, (ft^2)
$A_{\substack{fuel \\ surface}}$	Surface area of the fuel, (ft^2)
z	Core axial elevation, relative to bottom of active fuel, (ft)
z_{core}	Active fuel length, (ft)
t	Time, (s)
g	Gravitational acceleration, $\left(32.174\frac{ft}{s^2}\right)$
g_c	Gravitational constant, $\left(32.174\frac{ft\text{-}lbm}{lbf\text{-}s^2}\right)$
V_{bcr}	Critical bubble velocity, $\left(\frac{ft}{s}\right)$
b	Regime-specific exponent from Table 3.3-1
C	Regime-specific coefficient Table 3.3-1
α	Void fraction
ρ	Density, $\left(\frac{lbm}{ft^3}\right)$
σ	Surface tension, $\left(\frac{lbf}{ft}\right)$

3.3.2 Conservation of Energy

For the case of pool boiling with no liquid subcooling, steam superheating, or liquid carry-over, the energy balance is:

[ ]^{a,c}

For uniformly distributed core power,

[ ]^{a,c}

3.3.3 Conservation of Mass

Over a sufficiently short time interval, the system can be considered to be in a quasi-steady state such that the net change in the mass inventory is approximately zero. Therefore, for the purpose of calculating the void fraction, conservation of mass for the active (heated) core region control volume gives;

[ ]^{a,c} (for small Δt)

Since the flow area through the core A_{core}) does not vary axially over the heated length:

[ ]^{a,c} (for small Δt)

The mass flux at a given elevation is:

[ ]^{a,c} (for small Δt)

and

[ ]^{a,c}

therefore;

[ ]^{a,c}

where:

[ ]^{a,c}

The following expressions are obtained for $j_f(z)$ and $j_g(z)$:

[ ]^{a,c}

[ ]^{a,c}

The time-dependent heat flux is calculated as follows:

[ ]^{a,c}

where:

$$Q_{core}(t) = Q_{core}(t = 0) \times {}^{P}/_{P_0}(t)$$

and

${}^{P}/_{P_0}(t)$ is the decay heat power fraction at time, t, following the shutdown of the reactor core.

Then, the core average void fraction is calculated as follows,

$$\alpha_{\substack{core \\ avg}} = \frac{\int_0^{z_{core}} \alpha(z)dz}{\int_0^{z_{core}} dz}$$ [ ]a,c

and

$$\alpha_{\substack{core \\ exit}} = \alpha(z_{core})$$

3.3.4 Yeh Correlation Equations

The full form of the Yeh void fraction correlation (References 1 through 3) is:

$$\alpha(z) = C\left(\frac{\rho_g}{\rho_f}\right)^{0.239}\left(\frac{j_g(z)}{V_{bcr}}\right)^{b}\left[\frac{j_g(z)}{j_g(z)+j_f(z)}\right]^{0.6}$$

where:

$$V_{bcr} = 1.53\left[\frac{\sigma(\rho_f-\rho_g)gg_c}{\rho_f^2}\right]^{1/4}$$

The variables b and C vary by regime and are defined as shown in Table 3-1. The superficial velocities, j_f and j_g, are calculated as derived earlier.

Table 3-1 Constants and Exponents for the Yeh Void Fraction Model

Regime		C	b
Small bubble	$\frac{j_g}{V_{bcr}} \leq 1.0$	0.925	0.67
Large bubble	$1.0 < \frac{j_g}{V_{bcr}} < 4.31$	0.925	0.47
	$\frac{j_g}{V_{bcr}} \geq 4.31$	1.035	0.393

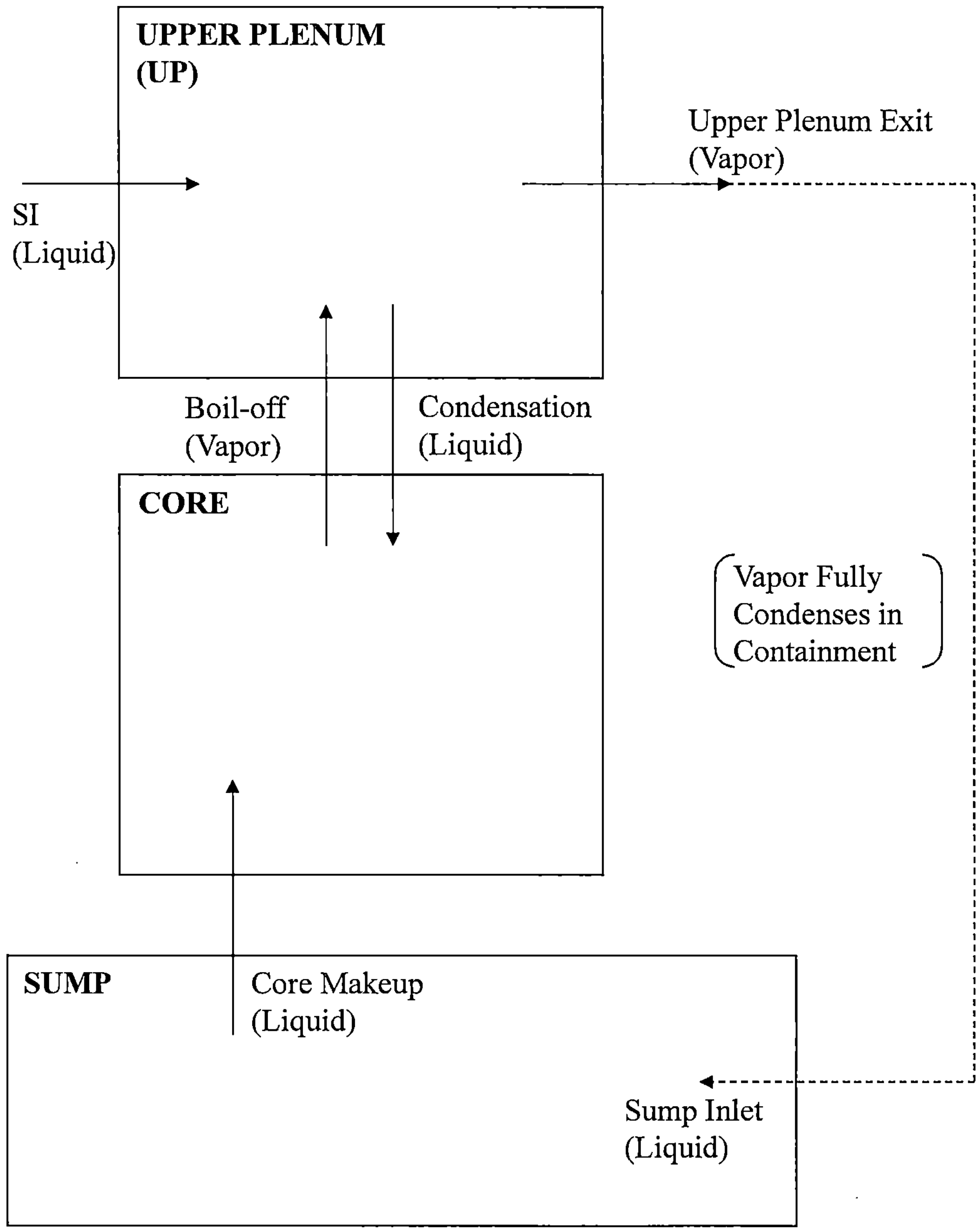

Figure 3-1: Mass Calculations in SKBOR

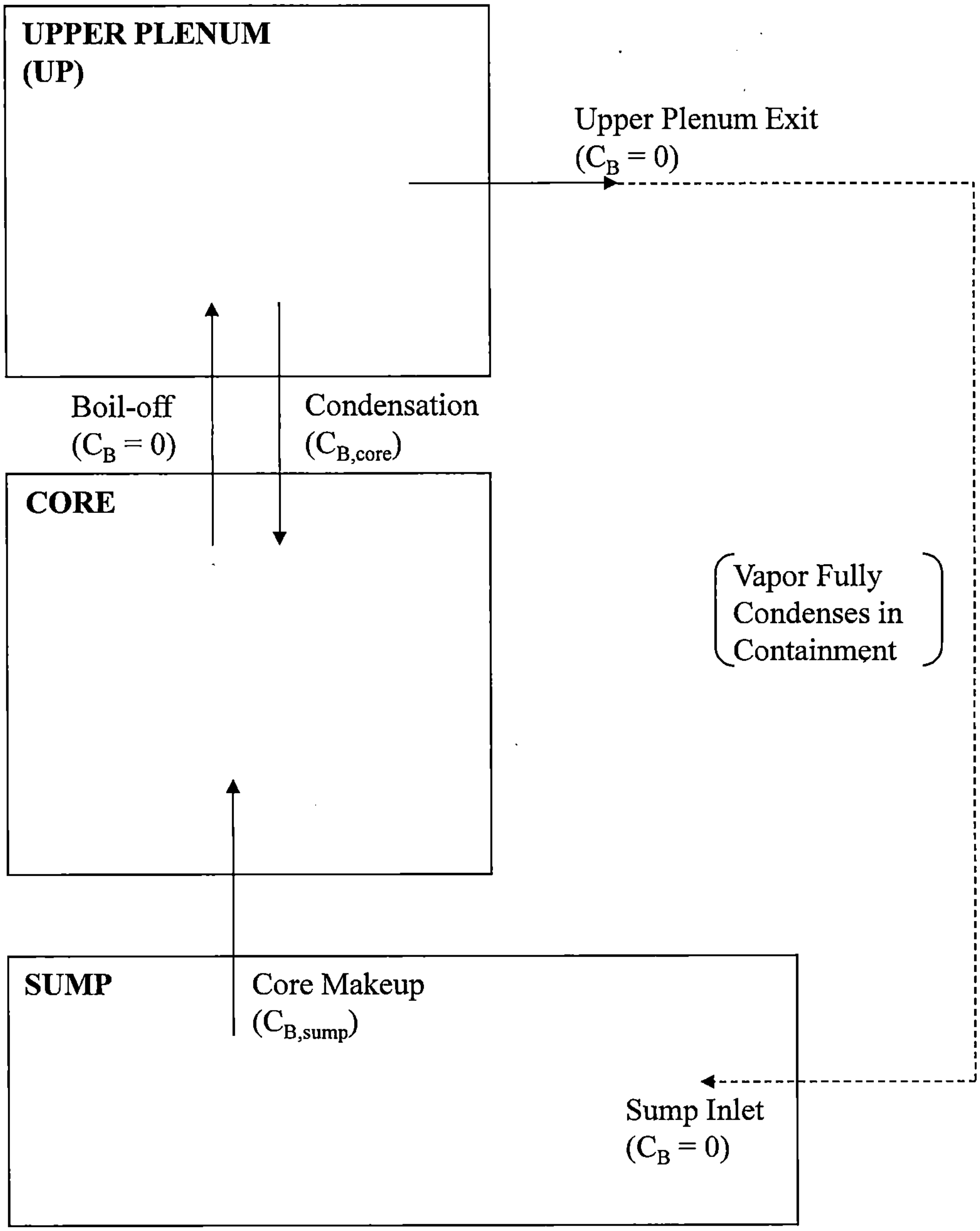

Figure 3-2: Boron Calculations in SKBOR

4.0 Input Description

This section describes the input variables that are available in SKBOR. Default values are provided where available, and recommended input values are suggested where possible. (Note that an entry of {N/D} in the Default column means that the variable has no default value.)

4.1 Timing and Problem Control

The following inputs are used in the various timing and problem control functions in SKBOR:

Name	Default	Units	Description
TSTART	{100}	s	Transient start time. Default value is standard for hot leg switchover time calculations.
TEND	{100,000}	s	Transient end time. Default value is generally sufficient for hot leg switchover time calculations.
DT	{1}	s	Timestep size. Default value is recommended for hot leg switchover time calculations.
DTPLOT	{100}	s	Interval at which plot points will be written to the binary graphics output file.
DTPRINT	{1,000}	s	Interval at which timestep printouts will be written to the ASCII output file.
TITLE	{N/D}		Title for program output.
VOID	{0}	No units	This is a flag to instruct the code as to whether or not the internally coded void fraction calculations are to be used. The values are: 0 No void fraction calculations are to be performed by the code. Note that voiding can still be accounted for by using the VCORET array. 1 Void fraction calculations are to be performed by the code. This option requires additional inputs LCORE, NCELLS, NFA, NRODA, CDO, LPFRAC, LPVOL7, LPVOL8, COREVOL9, UPVOL10, and UPVOL11. This option overrides user-specified inputs for HLP, HSI, VCORE, NVCORET, TVCORET, VCORET, and WSI.
WTPLIMIT	{23.53}	w/o	This variable allows the analyst to specify an alternate boric acid solubility limit. The default value is 23.53 w/o.

4.2 Component Boron Concentrations

In SKBOR, the following inputs are used to define the initial component boron concentrations.

Name	Default	Units	Description
CACC	{0}	ppm	Cold leg accumulator initial boron concentration.
CBIT	{0}	ppm	Boron injection tank (BIT) initial boron concentration.
CICE	{0}	ppm	Ice condenser initial boron concentration.
CPIPE	{0}	ppm	ECCS/BIT piping initial boron concentration.
CRCS	{0}	ppm	Reactor coolant system initial boron concentration.
CRWST	{0}	ppm	Refueling water storage tank initial boron concentration.

4.3 Component Fluid Masses

In SKBOR, the following inputs are used to define the initial component fluid masses.

Name	Default	Units	Description
MTACC	{0}	lbm	Cold leg accumulator initial fluid mass.
MTBIT	{0}	lbm	Boron injection tank initial fluid mass.
MTICE	{0}	lbm	Ice condenser initial fluid mass.
MTPIPE	{0}	lbm	ECCS/BIT piping initial fluid mass.
MTRCS	{0}	lbm	Reactor coolant system initial fluid mass.
MTRWST	{0}	lbm	Refueling water storage tank initial fluid mass.

4.4 Effective Vessel Mixing Volume

The effective vessel mixing volume in SKBOR can be specified several ways: (1) a time dependent volume calculated by SKBOR that accounts for voiding, (2) a user input time dependent volume (VCORET), or (3) a user input constant value (VCORE). The following inputs are used by SKBOR to calculate the effective mixing volume.

Option 1 – Internally Calculated by SKBOR

Name	Default	Units	Description
LCORE	{0.0}	ft	This variable is defined as the length of the active fuel region.
NCELLS	{0}	count	This variable is defined as the number of cells to be used in the core voiding calculation.
NFA	{0}	count	This variable is the number of fuel assemblies.
NRODA	{0}	count	This variable is defined as the number of active fuel rods per fuel assembly.
CDO	{0.0}	in	This variable is defined as the fuel rod cladding outside diameter.
LPFRAC	{0.0}	N/D	This variable is defined as the fraction of the lower plenum to be included in the liquid mixing volume.
LPVOL7	{0.0}	ft^3	This variable is defined as the volume of the reactor vessel lower head region.
LPVOL8	{0.0}	ft^3	This variable is defined as the volume of the reactor vessel lower plenum (core support) region.
COREVOL9	{0.0}	ft^3	This variable is defined as the total core region volume.
UPVOL10	{0.0}	ft^3	This variable is defined as the volume of the core outlet region.
UPVOL11	{0.0}	ft^3	This variable is defined as the volume of the reactor vessel upper plenum.

Option 2 – Variable Mixing Volume vs. Time

Name	Default	Units	Description
NVCORET	{0}	count	Number of points in table of effective vessel mixing volume vs. time (maximum 100).
TVCORET	{100*0.0}	s	Time values in table of effective vessel mixing volume vs. time. Must be in ascending order and bound the problem duration TSTART to TEND.
VCORET	{100*0.0}	ft^3	Mixing volume values in table of effective vessel mixing volume vs. time. Must be greater than 0 for I = 1 to NVCORET.

Option 3 – Constant Mixing Volume vs. Time

Name	Default	Units	Description
VCORE	{0.0}	ft^3	Effective vessel mixing volume.

4.5 Core Power and Decay Heat Model Options

In SKBOR, the initial core power level is denoted as PZERO, and the core power uncertainty factor is denoted as DPZERO. IDKMOD and AMARG are used to select the decay heat model and multipliers, respectively. Alternate decay heat models can be incorporated using NDK, TDK, and DK.

Name	Default	Units	Description
DPZERO	{1.02}	No units	Core power uncertainty factor.
PZERO	{0}	MWt	Core power level without uncertainty.
IDKMOD	{1}	No units	Decay heat model option: 0 User input table 1 1971 ANS infinite without residual fissions 2 1971 ANS finite without residual fissions The default value corresponds to the model prescribed in Section I.A.4 of 10 CFR 50, Appendix K.
AMARG(I)	{3*1.2, 1.0}	No units	Array of decay heat multipliers: 1 Multiplier on fission product decay, 0-1,000 s 2 Multiplier on fission product decay, 1,000-10,000,000 s 3 Multiplier on fission product decay, 10,000,000 s and beyond 4 Multiplier used with U-239 and Np-239 terms The default values correspond to the model prescribed in Section I.A.4 of 10 CFR 50, Appendix K.
NDK	{0}	count	Number of points in table of normalized core power fraction vs. time (maximum 100), used with the IDKMOD = 0 option.
TDK	{100*0.0}	s	Time values in table of normalized core power fraction vs. time, used with the IDKMOD = 0 option. Must be in ascending order and bound the problem duration TSTART to TEND.
DK	{100*0.0}	No units	Power values in table of normalized core power fraction vs. time, used with the IDKMOD = 0 option. Must be greater than 0 and less than or equal to 1 for I = 1 to NDK.

4.6 Lower Plenum Subcooling Model

In SKBOR, credit for lower plenum subcooling can be modeled to reduce the effective core boil-off rate and extend the hot leg switchover time. This model is not active when the internally coded void fraction model is used.

Name	Default	Units	Description
HLP	{-1.0}	Btu/lbm	Lower plenum enthalpy (e.g., HLP = 148.016 Btu/lbm for P = 14.7 psia and T = 180 °F). If no credit for lower plenum subcooling is taken, then the default value should be used.

4.7 Upper Plenum Condensation Model

In SKBOR, credit for upper plenum condensation can be used to reduce the effective core boil-off rate and extend the hot leg switchover time. This model is not active when the internally coded void fraction model is used.

Name	Default	Units	Description
WSI	{0}	lbm/s	Mass flow rate of pumped injection entering the upper plenum that interacts with steam. The default value is typically used for hot leg switchover calculations and cycling calculations during cold leg injection; refer to Section 8.0 for guidance on performing cycling calculations during hot leg injection.
HSI	{-1.0}	Btu/lbm	Enthalpy of pumped injection entering the upper plenum that interacts with steam (e.g. HSI = 148 Btu/lbm for P = 14.7 psia and T = 180 °F). If no credit for upper plenum subcooling is taken, then the default value should be used.

4.8 Core Pressure

PCORE allows the user to specify a constant core pressure between 14.7 and 200 psia (inclusive).

Name	Default	Units	Description
PCORE	{14.7}	psia	Core pressure, assumed to remain constant with time. Must be between 14.7 and 200 psia (inclusive).

5.0 Output Description

SKBOR generates both an ASCII output file and a binary graphics output file. The ASCII output is self-explanatory and does not warrant further discussion here. Section 5.1 describes the contents of the binary graphics output file.

5.1 Graphics Output Binary File

The following variables are written to the graphics output binary file:

Variable	Units	Description
T	s	Transient time.
PFRAC		Normalized core power fraction.
RHOCORE	lbm/ft^3	Core fluid density.
WBOIL	lbm/s	Average core exit mass flow rate over timestep DT.
MBCORE	lbm	Core boron mass.
MWCORE	lbm	Core water mass.
MTCORE	lbm	Core total (i.e., boron + water) mass.
MBSUMP	lbm	Sump boron mass.
MWSUMP	lbm	Sump water mass.
MTSUMP	lbm	Sump total (i.e., boron + water) mass.
WTFCORE	w/f	Core boron concentration, in weight fraction.
WTPCORE	w/o	Core boron concentration, in weight percent
PPMCORE	ppm	Core boron concentration, in ppm
WTFSUMP	w/f	Sump boron concentration, in weight fraction.
WTPSUMP	w/o	Sump boron concentration, in weight percent
PPMSUMP	ppm	Sump boron concentration, in ppm
DILSUMP	ppm	Sump dilution, in ppm.
PCORE	psia	Core pressure
HEATF	$Btu/s\text{-}ft^2$	Heat Flux
JFIN	ft/s	Superficial Velocity
JGOUT	ft/s	Superficial Velocity
COREVOID	no units	Void Fraction
UPVOID	no units	Void Fraction
VCORE	ft^3	Core mixing volume

6.0 Summary of SKBOR Releases

Table 6-1 summarizes the releases of SKBOR on UNIX and Linux platforms.

Table 6-1: SKBOR Releases

Version	Release Date
3.1	5/94
4.0	4/99
5.0	7/00
6.0	6/01
7.0	11/01
8.0	11/05
9.0	5/07

7.0 References

1. Cunningham, J.P., and Yeh, H.C., "Experiments and Void Correlation for PWR Small-Break LOCA Conditions," Trans. ANS 17, p. 369-370, 1973.
2. Hochreiter, L.E., and Yeh, H.C., "Mass Effluence During FLECHT Forced Reflood Experiments," Nuclear Engineering and Design, 60, p. 413-429, 1980.
3. Yeh, H.C., "Modification of Void Fraction Calculation," Proceedings of the Fourth International Topical Meeting on Nuclear Thermal-Hydraulics, Operations and Safety, Volume 1, Taipei, Taiwan, June 1988.

CAW-17-4663, Revision 0, "Application for Withholding Proprietary Information from Public Disclosure"
(8 pages)

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CAW-17-4663

October 17, 2017

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-LIS-17-353, Attachment 3, "SKBOR: A Computer Code for Calculating the Accumulation of Boric Acid in the Reactor Vessel"

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC ("Westinghouse"), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-17-4663 signed by the owner of the proprietary information, Westinghouse. The Affidavit, which accompanies this letter, 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 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Wolf Creek Nuclear Generating Station.

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-17-4663, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

James A. Gresham, Manager
Regulatory Compliance

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, James A. Gresham, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse") and declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

Executed on: 10/17/17

James A. Gresham, Manager
Regulatory Compliance

(1) I am Manager, Regulatory Compliance, Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-LIS-17-353, Attachment 3, "SKBOR: A Computer Code for Calculating the Accumulation of Boric Acid in the Reactor Vessel (Proprietary) for submittal to the Commission, being transmitted by Wolf Creek Nuclear Generation Station letter. The proprietary information as submitted by Westinghouse is that associated with Westinghouse Alternate Source Term analysis and Methodology Transition, and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to support Wolf Creek for the Alternate Source Term analysis and Methodology Transition.

(b) Further, this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of Alternate Source Term analysis or Methodology Transition.

(ii) Westinghouse can sell support and defense of industry guidelines and acceptance criteria for plant-specific applications.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and non-proprietary versions of a document, furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

COPYRIGHT NOTICE

Wolf Creek Nuclear Generation Station

Letter for Transmittal to the NRC

The following paragraphs should be included in your letter to the NRC Document Control Desk:

Enclosed are:

1. "SKBOR: A Computer Code for Calculating the Accumulation of Boric Acid in the Reactor Vessel" (Proprietary)

2. "SKBOR: A Computer Code for Calculating the Accumulation of Boric Acid in the Reactor Vessel" (Non-Proprietary)

Also enclosed are the Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-17-4663, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice.

As Item 1 contains information proprietary to Westinghouse Electric Company LLC ("Westinghouse"), it is supported by an Affidavit signed by Westinghouse, the owner of the information. The Affidavit sets forth the basis on which the information may be withheld from public disclosure by the Nuclear Regulatory Commission ("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.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-17-4663 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.