ENCLOSURE 5 TO AEP-NRC-2014-42

Attachment #2 (NP—Attachment) of Westinghouse Letter, LTR-PL-14-22, Westinghouse Responses to NRC, "Donald C. Cook Nuclear Plant Unit 1 – Request for Additional Information on the Application for Amendment to Restore Normal Reactor Coolant System Pressure and Temperature Consistent with Previously Licensed Conditions (TAC No. MF2916)," dated May 28, 2014

RAIs: SCVB RAI-1a&b, SCVB RAI-2a-c, SCVB RAI-3a&b, SCVB RAI-4a-f, SCVB RAI-5a, SCVB RAI-6, SCVB RAI-7, SCVB RAI-8, SCVB RAI-12, SNBP RAI-1, SNBP RAI-2, SNBP RAI-3a&b

NP-Attachment

(Non-Proprietary)

Westinghouse Responses to NRC, "Donald C. Cook Nuclear Plant Unit 1 - Request for Additional Information on the Application for Amendment to Restore Normal Reactor Coolant System Pressure and Temperature Consistent with Previously Licensed Conditions (TAC No. MF2916)" Set #2

SCVB RAI-1a & b; SCVB RAI-2a-c; SCVB RAI-3a & b; SCVB RAI-4a-f; SCVB RAI-5a; SCVB RAI-6; SCVB RAI-7; SCVB RAI-8; SCVB RAI-12; SNBP RAI-1; SNBP RAI-2; SNBP RAI-3a & b.

NP-Attachment

(Non-Proprietary)

Westinghouse Electric Company LLC
1000 Westinghouse Drive
Cranberry Township, PA 16066

SCVB RAI-1

Reference 1, Enclosure 6, Section 5.4.1.5: it is noted that the proposed vessel/core inlet temperature 514.6°F is greater than the Analysis of Record (AOR) vessel/core inlet temperature of 506.6°F and the proposed RCS pressure 2317 psia (includes uncertainty) is greater than the AOR RCS pressure of 2100 psia.

a) Please justify why the most limiting Loss of Coolant Accident (LOCA) short term Mass and Energy (M&E) release and containment response for the proposed minimum vessel/core inlet temperature 514.6°F and pressure of 2317 psia (includes uncertainty) is bounded by the LOCA short term M&E release and containment response in the AOR vessel/core inlet temperature of 506.6°F and RCS pressure of 2100 psia.
    - WEC Response – Short term M&E releases are strongly influenced by two key inputs; RCS temperature and RCS pressure. Short term blowdown releases are linked directly to critical mass flux, which is maximized with increasing pressure and decreasing temperature. The direction of conservatism is high for RCS pressure, and low for RCS temperature (increased fluid density increases short term break flow). The analysis of record short term mass and energy releases were calculated to be bounding for both D.C. Cook Unit 1 and Unit 2, and the vessel/core inlet temperature of 506.6°F and RCS pressure of 2317 psia are thus bounding values.

b) Please explain why subtracting (instead of adding) uncertainty from the realistic value would give a conservative input value of 514.6°F for the vessel/core inlet temperature and would give conservative results for the M&E release.
    - WEC Response – Due to the increased density of the RCS fluid, low RCS temperatures and high RCS pressures are the limiting condition for short term LOCA mass and energy releases.

SCVB RAI-2

Reference 1, Enclosure 6, Section 5.4.2.2:

a) Describe the most limiting LOCA break in the AOR from the containment response standpoint. Explain why the most limiting break in the AOR would also be the most limiting break for the proposed RCS NOP/NOT conditions.
    - WEC Response – The most limiting break in the AOR, which is bounding for D.C. Cook Unit 1 and Unit 2, is the double ended pump suction break with a loss of offsite power and minimum safeguards assumptions (i.e. one train of emergency diesel generator failure to start). This scenario maximizes mass and energy releases, while minimizing the containment active heat removal via containment spray. The analysis of record considered the most limiting conditions for both Unit 1 and Unit 2, and as such was performed at nominal T_{avg} and RCS pressure plus any applicable uncertainties. Analyzing at NOP/NOT conditions will not result in a more limiting break location or limiting single failure.

b) The UFSAR Section 14.3.4.1.3.1.3 input assumption 7 states that the air recirculation fan is effective 132 seconds after the high-1 containment pressure bistable signal is actuated. Please explain the basis for changing this time to 300 seconds.
   - WEC Response - The change in air recirculation fan activation time was driven by the LOCA Peak Clad Temperature (PCT) analysis. The effect of the delay was evaluated for the LOCA containment integrity case, and shown to have a negligible impact on the peak calculated containment pressure.

c) Please explain the basis for the containment spray actuation time of 315 seconds in the proposed evaluation. What is the containment spray actuation time in the AOR? In case the AOR spray actuation time is different from 315 seconds, please justify that the change if the change is less conservative.
   - WEC Response – The containment actuation time in the AOR was 115 seconds. The change in containment spray activation time was driven by the LOCA PCT analysis. The effect of the delay was evaluated for the LOCA containment integrity case, and shown to have a negligible impact on the peak calculated containment pressure.

<u>SCVB RAI-3</u>

Reference 1, Enclosure 6, Section 5.4.2.4 states:
"For the containment integrity analysis, this was completed by evaluating the effects of increased delay times for CTS actuation and containment air recirculation fan actuation on the LOCA containment integrity analysis."

a) Explain the method of evaluation described in the above statement.
   - WEC Response - Sensitivity studies on the effect of the delay in CTS and containment air recirculation fan were completed using the licensed LOTIC1 containment response computer code (WCAP-8354-P-A). These sensitivity studies showed a negligible change in peak containment pressure.

b) If the subject evaluation/analysis methodology is different from the currently used methodology, provide justification.
   - WEC Response - The methodology used for the evaluation was not different from the currently used methodology for ice condenser containment pressure calculations. This methodology is documented in WCAP-8354-P-A.

SCVB RAI-4

Reference 1, Enclosure 6, Section 5.4.2.5;

a) The AOR vessel/core inlet temperature stated in the table is 552.5°F. Please explain why this is different from the AOR vessel/core inlet temperature of 506.6°F stated in Section 5.4.1.5.
   - WEC Response - Section 5.4.1.5 considered short term mass and energy releases and containment response, while Section 5.4.2.5 considered long term mass and energy releases and containment response. The direction of conservatism in the RCS temperature is low for short term M&Es, and high for long term M&Es.

b) Please explain why the RCS pressure of 2100 psia in the AOR and 2250 psia (2317 psia including uncertainty) in the proposed change in NOP/NOT are not included for comparison as key input parameters.
   - WEC Response - The analysis of record for long term mass and energy releases and containment response was bounding for D.C. Cook Unit 1 and Unit 2. Therefore, the AOR considered an RCS pressure of 2317 psia and was effectively completed at NOP/NOT conditions relative to D.C. Cook Unit 1, and thus not included as a parameter for comparison in Section 5.4.2.5.

c) Please justify why the AOR is bounding for both M&E release and containment response even though the proposed RCS pressure of 2317 psia (uncertainty included) is significantly greater than the AOR RCS pressure of 2100 psia.
   - WEC Response - The analysis of record for D.C. Cook Unit 1 and Unit 2 is a single bounding analysis. This analysis was performed considering an RCS pressure of 2317 psia.

d) Aside from the key parameters stated in the table, what are other input parameters for long term M&E analysis that differ in the AOR and the proposed analysis. Please provide these values.
   - WEC Response - An additional parameter not listed in Section 5.4.2.5 is core power. The AOR core power was 3481 MWt, as compared to the NOP/NOT core power of 3327 MWt.

e) Aside from the key parameters stated in the table, what are other input parameters, for long term containment gas temperature response for equipment environmental qualification (EEQ) that differ in the AOR and the proposed analysis. Please provide these values.
   - WEC Response - Westinghouse provides long term containment response results to AEP for EEQ consideration. The long term containment temperature is largely a function of decay heat and containment heat removal systems. The increased delay time in the CTS and containment air recirculation fans were shown to have a negligible effect on the containment pressure/temperature transient. The decay heat resulting from the initial core power in the AOR of 3481 MWt is bounding relative to the Unit1 core power of 3327 MWt.

SCVB RAI-5

Reference 1, Enclosure 6, Section 5.4.2.6 states:
"The evaluation of the long-term LOCA M&E and peak containment pressure is predicated upon the continued application of the operability assessment supporting NSAL-11-5 (Reference 3), in conjunction with the AOR."

a) Is the AOR based on corrected M&E release, resolving the issues identified in NSAL-11-5 (Reference 2)?
   - WEC Response - The AOR is not based on corrected M&E releases relative to the issues identified in NSAL-11-5.

References Cited in RAI Transmittal

1) Letter from I&M to NRC dated October 8, 2013, "Donald C. Cook Nuclear Plant Unit 1 Docket No. 50-315 License Amendment Request Regarding Restoration of Normal Reactor Coolant System Operating Pressure and Temperature Consistent With Previously Licensed Conditions" (ADAMS Accession No. ML13283A121).
2) NSAL-11-05, "Westinghouse LOCA Mass and Energy Release Calculation Issues," July 26, 2011 (ADAMS Accession No. ML13239A479).

SCVB RAI-4

Reference 1, Enclosure 6, Section 5.4.2.5;

f. Please explain why the NOP/NOT core stored energy of [ ]a,c is less than the AOR core stored energy of [ ]a,c, even though the average proposed RCS temperature of [ ]a,c is greater than the average RCS AOR temperature of [ ]a,c.

Westinghouse Response:

The D. C. Cook Unit 1 Analysis of Record (AOR) core stored energy value, [ ]a,c, was previously calculated using [

]a,c. The Normal Operating Pressure / Normal Operating Temperature (NOP/NOT) core stored energy value for D. C. Cook Unit 1, [ ]a,c, was calculated using [

]a,c. So while a higher Reactor Coolant System (RCS) temperature would increase the overall core stored energy, [ ]a,c.

SCVB RAI-6

"Reference 1, Enclosure 6, Section 5.5.1.4:

Explain the basis for selecting the break sizes 1.4 ft^2, 0.865 ft^2, 0.857 ft^2, 0.834 ft^2, 0.808 ft^2, and 1.0 ft^2 and their corresponding power levels at which the M&E release analysis is performed."

Response

The safety analysis methodology documented in Reference 1 describes the basis for the choice of the break sizes and the initial power levels analyzed for the SLB M&E releases inside containment. Section 2.1 of Reference 1 discusses the break spectrum for the SLB M&E safety analysis. This description includes four plant power levels of 102%, 70%, 30%, and 0% of nominal full load and three different break sizes: a full double-ended rupture (DER), a small DER, and a small split rupture.

The discussion in Section 5.5.1.2 of Enclosure 6 presents the D. C. Cook Unit 1 plant-specific break spectrum based on the approved methodology. The break size of the full DER is defined as the 1.4 ft^2 cross-sectional area of the flow restrictor integral with the discharge nozzle of the faulted steam generator. This is the break size in the forward-flow direction. The area of the steam pipe defines the break size in the reverse-flow direction; this area is modeled as 4.7465 ft^2 for the initial blow down. The 1.0 ft^2 small DER is analyzed only at 0% power initial conditions assuming no entrainment to show that this break size is more limiting than the full DER at 0% power with entrainment (see the response to SCVB RAI-7).

Each of the split rupture break areas has been determined as the largest cross-sectional area that does not produce a steamline isolation signal from the primary protection equipment nor results in water entrainment in the break effluent as discussed in Section 2.1 of Reference 1. These areas were determined for each initial power level based on the D. C. Cook Unit 1 plant-specific values for the secondary-side protection system setpoints.

Reference(s)

1. WCAP-8822 (Proprietary) and WCAP-8860 (Nonproprietary), "Mass and Energy Releases Following a Steam Line Rupture," September 1976; WCAP-8822-S1-P-A (Proprietary) and WCAP-8860-S1-A (Nonproprietary), "Supplement 1 – Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," September 1986.

SCVB RAI-7

"Reference 1, Enclosure 6, Section 5.5.1.5 states:

> *'All of the analyzed breaks conservatively assumed dry saturated steam releases (no entrainment) except the full DER at 0 percent initial power. As a result, the small DER with dry saturated steam release was analyzed at 0 percent power, represented by a 1.0 ft^2 break (smaller than the area of a single integral flow restrictor) from the faulted-loop SG and a 1.0 ft^2 break for the reverse-flow blowdown from the intact-loop SGs.'*

Please explain why entrainment was assumed in the full DER at zero percent initial power, and explain why as a result the small DER with dry saturated steam release was analyzed at zero-percent power."

Response

Calculations have been performed and documented for the full DER at 0 percent power with no entrainment similar to the other full DER SLBs identified in Section 5.5.1.4 of Enclosure 6. However, the containment response for D. C. Cook Unit 1 with the 0 percent power no entrainment M&E releases

produced a peak temperature that exceeded the 324.7°F limit noted in Section 5.5.2.3 of Enclosure 6. The full DER at 0 percent power with entrainment was assumed in order to reduce the energy content of the steam effluent during the initial blowdown. The peak containment temperature for this break is less than 321°F as shown in Table 5.5.2-1 of Enclosure 6. However, to satisfy the requirement that the maximum release rate has been considered in the M&E releases as stipulated in Reference 2, the 1.0 ft^2 small DER has been analyzed with no entrainment to show that that the peak containment temperature does not exceed the limit value as shown in Table 5.5.2-1 of Enclosure 6.

Reference(s)

2. NUREG-0800, Standard Review Plan, Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," Revision 2 – March 2007.

SCVB RAI-8

"Reference 1, Enclosure 6, Table 5.5.1-1;

Explain what is meant by the title of the fourth column 'Rod Motion (sec)'. Describe its method of calculation, and how it affects the M&E release."

Response

Rod Motion is the nomenclature used in the table of the sequence of events to indicate the time of reactor trip. Following a finite delay after the Reactor Trip Signal (third column in Table 5.5.1-1 of Enclosure 6), the control and shutdown rods are released from the grippers and begin to fall into the core. The protection system signals that actuate the reactor trip are the low steam pressure setpoint for the full DERs and the high-1 containment pressure setpoint for the small DER and split ruptures as described in Section 5.5.1.5 of Enclosure 6.

The method of calculation is to input the D. C. Cook Unit 1 plant-specific protection setpoints and delays into the protection model for the SLB safety analysis. As the protection functions are actuated in the transient SLB M&E analysis, reactor trip and rod motion occurs following the applicable delays for the specific protection function. It is not the motion of the rods that is significant to the analysis of the SLB M&E releases, but the occurrence of the reactor trip, which reduces the core power and thus the long-term energy of the reactor coolant system.

SCVB RAI-12) NUREG-0800, Standard Review Plan 6.2.1.5 describes the minimum containment pressure analysis for the ECCS performance capability. Regulatory Guide (RG) 1.157, Section 3.12.1 provides guidance for calculating the containment pressure response used for evaluating cooling effectiveness during the post-blowdown phase of a LOCA. The RG states that the containment pressure should be calculated by including the effects of containment heat sinks and operation of all pressure-reducing equipment assumed to be available. Using the above guidance please describe the impact of the changes in M&E input on the minimum containment pressure analyses for ECCS performance during a LOCA and MSLB accident.

LOCA Response:

A conservatively low containment pressure for the Return to Reactor Coolant System (RCS) Normal Operating Pressure (NOP) / Normal Operating Temperature (NOT) large-break loss-of-coolant accident (LBLOCA) evaluation was calculated in accordance with Sections 11-4-11 and 12-3-4 of the generically approved ASTRUM evaluation methodology (Reference 1). The pressure was calculated using the generically approved LOTIC2 containment pressure model (References 2 and 3) and mass and energy (M&E) release from WCOBRA/TRAC at NOP/NOT conditions.

The modeling assumptions related to the containment heat sinks and the operation of all pressure-reducing equipment was calculated consistent with References 2 and 3. The modeling of the air recirculation fan delay time, the containment spray initiation delay time, and the containment spray flow were updated as shown in Table 5.1.1-3 of Enclosure 6 to Reference 4. The M&E release from WCOBRA/TRAC captures the effect of the changes to the operating conditions and plant configuration listed in Table 5.1.1-3 of Enclosure 6 to Reference 4. Input assumptions not affected by the change in operating conditions and plant configuration were unchanged from the D.C. Cook Unit 1 Analysis-of-Record (AOR) (Reference 5). The combined effect of these changes was an increase to the predicted minimum containment backpressure.

References:

1. WCAP-16009-P-A, Revision 0, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment Of Uncertainty Method (ASTRUM)," January 2005.
2. WCAP-8354-P-A, Revision 0, Supplement 1, "Long Term Ice Condenser Containment Code-LOTIC Code," April 1976.
3. WCAP-8339, Revision 0, "Westinghouse Emergency Core Cooling System Evaluation Model – Summary," June 1974.
4. Letter from J. P. Gebbie (I&M) to NRC Document Control Desk, "Donald C. Cook Nuclear Plant Unit 1, Docket No. 50-315, License Amendment Request Regarding Restoration of Normal Reactor Coolant System Operating Pressure and Temperature Consistent With Previously Licensed Conditions," October 8, 2013 (Agencywide Documents Access and Management System (ADAMS), Accession No. ML13283A121).
5. Letter from T. A. Beltz (NRC) to M. W. Rencheck (I&M), "Donald C. Cook Nuclear Plant, Unit 1 – Issuance of Amendment to Renewed Facility Operating Licensing Regarding Use of the Westinghouse ASTRUM Large Break Loss-of-Coolant Accident Analysis Methodology (TAC No. MD7556)," October 17 2008 (Agencywide Documents Access and Management System (ADAMS), Accession No. ML082670351).

Request for Additional Information SNPB RAI-1:

Table 2.1-1 of WCAP-17762-NP contains NSSS design parameters which, as described in Section 2.1.2, "are used as the basis for the design transients and for the systems, structures, components, accidents and fuel analyses and evaluations." Each of the eight cases listed in Table 2.1-1 has a different steam outlet pressure listed, ranging from 618 to 851 psia.

The description of the NSSS design transient evaluations in Section 3.1, however, states that, consistent with the measurement uncertainty recapture (MUR) uprate license amendment issued by the NRC on December 20, 2002 (ADAMS Accession No. ML023470126), "the full power steam pressure will continue to be limited to a minimum of 679 psia (administratively limited to 690 psia for conservatism)." These limits are higher than the steam outlet pressures listed in Table 2.1-1 for Cases 1, 2, and 6.

Please clarify how the limits from Section 3.1 listed above interact with the design parameters from Table 2.1-1 for the design transient evaluations.

Response:

The NSSS design parameters in Tables 2.1-1 and 2.1-2 of WCAP-17762-NP provide a range of maximum and minimum design conditions based on the program-specified full power thermal design conditions (e.g., NSSS power, feedwater temperature, desired Tavg range, thermal design flow, RCS pressure, and Steam Generator Tube Plugging (SGTP) range) to be used by various downstream analyses. The NSSS Design Transients in Section 3.1, which were provided as input to the structural design and fatigue/stress analyses for the NSSS components, bound all eight cases in Tables 2.1-1 and 2.1-2. However, to ensure that the steam generator primary to secondary Δp limit is not exceeded during the limiting transients, the minimum full power steam pressure is limited to 679 psia.

In general, any limitations to the NSSS design parameters are provided in the section that documents the basis for the restriction. The limitations are not cross-referenced in the NSSS Design Parameter section such that all analyses start with the same bounding range of conditions.

SNPB RAI-2

Section 5.1.1 of WCAP-17762-NP describes the best-estimate large break loss-of-coolant accident (BE LBLOCA) analysis for the DC Cook Unit 1 return to NOP/NOT program. This analysis estimates the impact of fuel thermal conductivity degradation (TCD) by using a method found acceptable by NRC staff in a letter dated March 7, 2013 (ADAMS Accession No. ML13077A137). In this method, margin to the 10 CFR 50.46(b)(1) peak cladding temperature limit of 2200°F is recaptured by adjusting input parameters, including accounting for peaking factor burndown in the fuel.

Table 1 of the previous TCD evaluation for DC Cook Unit 1, from March 19, 2012 (ADAMS Accession No. ML12088A104), included burnup-dependent limits for both heat flux hot channel factor, F_Q, and the enthalpy rise hot channel factor, $F_{\Delta H}$. In WCAP-17762-NP, Table 5.1.1-1 presents the F_Q and $F_{\Delta H}$ burndown as a function of rod burnup used in the LBLOCA analysis. The AOR (available in ADAMS at Accession No. ML080090268) accounts for neither TCD nor peaking factor burndown, and provides single limits for F_Q and $F_{\Delta H}$.

If peaking factor burndown is required to recapture margin to the PCT limit of 2200°F for the return to NOP/NOT analysis, how will this be addressed in the CNP Unit 1 Core Operating Limits Report and/or technical specifications?

Westinghouse Input

The following peaking factor changes were included in the Normal Operating Pressure (NOP)/Normal Operating Temperature (NOT) project for D. C. Cook Unit 1:

- reduction in transient F_Q (heat flux hot channel factor), including uncertainties, from 2.15 to 2.09
- reduction in steady-state F_Q, without uncertainties, from 1.70 to 1.65
- reduction in $F_{\Delta H}$ (enthalpy rise hot channel factor), including uncertainties, from 1.545 to 1.530
- corresponding reduction in hot assembly average power, including uncertainties

Of the fuel peaking factor design values, the transient F_Q and $F_{\Delta H}$ parameters have specific limits specified in the Core Operating Limits Report (COLR). Additionally, the Engineering procedure controlling the process for Reactor Core Design is being updated to include the peaking factor burndown values to ensure compliance with the revised peaking factor limits in future core designs. Verification of the peaking factor limits is performed as part of the normal reload design process for each core reload. Steady state F_Q limits are verified by comparing predicted steady-state peaking factors at full power conditions against the steady-state burnup dependent F_Q peaking factor limits. Transient F_Q and $F_{\Delta H}$ limits are verified by comparing the predicted power distributions during normal operation and the operational transients against the applicable burnup-dependent limits. Transient power distributions are generated based on the methodology described in Reference 1. Predicted power distribution used in reload analyses are based on core models developed using the NRC-approved ANC code described in Reference 2. Each reload cycle, these limits are analytically confirmed and verified. If the analytical verification produces unacceptable results, then the core is either redesigned or a Large Break Loss of Coolant Accident (LBLOCA) analysis re-assessment for

thermal conductivity degradation (TCD) is performed with revised peaking factor input. The acceptability of analysis results is based on confirming that the reactor core is operating as designed.

The plant-specific burnup-dependent peaking factor limits used in the LBLOCA TCD evaluations were established with consideration of current operating cycles. The reduced peaking factors at high burnup conditions (or burndown limits) constitute a core design constraint and will be confirmed during the reload design process. The peaking factor burndown limits defined to support the LBLOCA peak cladding temperature (PCT) evaluations reflect the physical phenomenon of lower rod power as the fuel rod becomes less reactive, and are within the practical application of the core design. The plant-specific burndown limits were developed based upon review of a neutronic model's peaking factor behavior versus fuel rod burnup for actual cycle-designs for each plant. The burndown limit lines for the higher burnup fuel were defined such that there was slightly more margin (or "white space") to the predictive peaking factor data, as compared to that defined for the low burnup fuel (where no burndown credit is taken). By preserving more core design margin in the higher burnup region, the limit lines are more conservative for the purposes of a LBLOCA PCT evaluation. From a Technical Specification Surveillance perspective, this results in the lower burnup fuel to be more limiting, implicitly enveloping the higher burnup fuel. Therefore, no further Technical Specification Surveillance actions or COLR changes are required; the burndown credits supporting the LBLOCA PCT evaluations can be confirmed analytically as part of the normal core design process.

Reactivity and power distribution measurements are performed periodically during the cycle as required by Technical Specifications (TS) 3.1.2 (Core Reactivity), TS 3.2.1 (Heat Flux Hot Channel Factor ($F_Q(Z)$)), and TS 3.2.2 (Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)) to verify that core reactivity and peaking factors are within their respective design limits. Measured power distributions and core reactivity are also compared against predicted power distributions and core reactivity. These comparisons, when coupled with startup physics testing results following refueling, are used to verify the core is operating as designed. This confirmation provides verification that the LBLOCA accident analysis input is within the specified limits. If the core is determined not to be operating as designed, an evaluation would be performed to assess analysis margins, understand the reasons for the deviation, and make appropriate adjustments on a case-by-case basis to plant operations or setpoints to ensure operation remains within LBLOCA analysis limits.

References:

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

2. WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code," September 1986.

SNPB RAI-3

Sections 6.2.3 and 6.2.4 of WCAP-17762-NP discuss the evaluation of fuel performance at the Cook Unit 1 return to NOP/NOT conditions. Section 6.2.3 provides the acceptance criteria for the fuel, and Section 6.2.4 provides the results of the evaluations with respect to these criteria.

a. *It is stated in Section 6.2.4 that "[n]o explicit PAD calculations were used to evaluate the fuel rod design criteria at NOP/NOT conditions." Discussion in several of the evaluations notes that the effects of returning to NOP/NOT at Cook Unit 1 will be offset by available margin. Was a quantitative margin assessment performed?*

b. *It is also noted in Section 6.2.4 that "[t]he PAD code with USNRC-approved models... for in-reactor behavior is used to calculate the fuel rod performance over its irradiation history." The NRC-approved current version of the PAD code, PAD 4.0, does not include approved method for modeling TCD.*

   In evaluating the fuel acceptance criteria for the return to NOP/NOT conditions, was consideration given to the effects of TCD on fuel performance? If a quantitative margin assessment was performed per SNPB-RAI-3.a, did this margin assessment include the effects of TCD?

Westinghouse Response:

While no explicit, quantitative calculations were performed to evaluate the fuel rod design criteria, a qualitative assessment was done as part of the Engineering Report (Reference 2) to compare the estimated impact of NOP/NOT operation with the adequate available margin for recent cycles of D. C. Cook Unit 1. It was determined that the available margin for D. C. Cook Unit 1 is sufficient to offset both the estimated impacts of transition to NOP/NOT operation, as well as the maximum impacts of thermal conductivity degradation (TCD), which have been calculated generically. Additionally, all fuel rod design criteria are evaluated on a cycle-specific basis based on models developed by Nuclear Design, as described in Section 6.1 of Reference 1 Enclosure 6, so they are ensured to remain met each cycle.

References

1. Nuclear Regulatory Commission Agencywide Documents Access Management System (ADAMS) Accession No. ML13283A122, "WCAP-17762-NP, Revision 1, 'D. C. Cook Unit 1 Return to Reactor Coolant System Normal Operating Pressure/Normal Operating Temperature Program – Licensing Report,' and Enclosures 7 through 9."
2. Westinghouse Topical Report WCAP-17761-P, "D. C. Cook Unit 1 Return to Reactor Coolant System Normal Operating Pressure/Normal Operating Temperature Program – Engineering Report."

Supplemental Information

Per Mr. Don Hafer's request, the following non-proprietary Figure 5.1.1-5 from the D.C. Cook Unit 1 NOP/NOT engineering report (WCAP-17761-P [2]) is to be provided with AEP's response to EEEB RAI-1.