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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, D. C. 20555-0001

Donald C. Cook Nuclear Plant Unit 1
REPORT OF LOSS-OF-COOLANT ACCIDENT
EVALUATION MODEL CHANGES

Reference: (1) Letter from M. W. Rencheck (I&M) to U. S. Nuclear Regulatory Commission Document Control Desk, "Donald C. Cook Nuclear Plant Unit 1, Annual Report of Loss-of-Coolant Accident Evaluation Model Changes and Submittal of New Large Break Loss-of-Coolant Accident Analysis of Record for Unit 1," Submittal C0300-07, dated March 31, 2000.

Pursuant to 10 CFR 50.46, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1, is providing an annual report of changes to the loss-of-coolant accident (LOCA) evaluation models that have been used for this unit. The changes to the LOCA analyses are classified as significant in accordance with 10 CFR 50.46(a)(3)(i). Therefore, this submittal also constitutes the 30-day report required by 10 CFR 50.46(a)(3)(ii).

The large break LOCA (LBLOCA) and small break LOCA (SBLOCA) events have been re-analyzed for the upcoming Unit 1 operating cycle (Cycle 17). The new analyses reflect the impact of steam generator replacement and fuel upgrades on the peak cladding temperature (PCT) calculations for these events. The new analyses are based on analytical codes that have been updated to address the impacts of assessments identified in previous 10 CFR 50.46 reports. For the LOCA analysis, additional assessments have also been performed to estimate the impact of a plant design change reducing auxiliary feedwater flow rate and to estimate the burst and blockage/time in life penalty.

When compared to the PCT values reported in the previous Unit 1 10 CFR 50.46 annual report (Reference 1), the new analytical changes, including margin

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allocations, result in a 482°F increase in the calculated SBLOCA PCT and a 124°F decrease in the calculated LBLOCA PCT. Because the calculated PCT values differ by more than 50°F from the previously reported values, these changes are considered significant in accordance with 10 CFR 50.46(a)(3)(i). The PCT values calculated for both the SBLOCA and LBLOCA continue to be less than the 2200°F limit specified by 10 CFR 50.46(b)(1).

Attachment 1 describes the background for the new LBLOCA and SBLOCA analyses-of-record and summarizes the effect of the assessments on these analyses. Attachment 2 provides the PCT margin utilization tables for LBLOCA and SBLOCA showing that the calculated PCTs remain within the limit specified in 10 CFR 50.46.

Westinghouse Electric Company (Westinghouse) is the current supplier of the CNP LOCA analyses conducted pursuant to 10 CFR 50.46. I&M and Westinghouse have ongoing processes to ensure that the as-operated plant values for PCT-sensitive parameters are bounded by the values assumed in the LOCA analyses. These processes include implementing the systematic evaluation methodology discussed in licensing topical report WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology," to confirm the validity of the safety analyses for all plant operating cycles. In a letter dated May 28, 1985, the NRC found this WCAP to be acceptable for reference in licensing documents.

No new commitments are made in this submittal.

Should you have any questions concerning this subject, please contact Mr. Wayne J. Kropp, Director of Regulatory Affairs, at (616) 697-5056.

Sincerely,

M. W. Rencheck
Vice President Nuclear Engineering

Attachments

/dmb

c: J. E. Dyer
MDEQ – DW & RPD, w/o attachments
NRC Resident Inspector
R. Whale, w/o attachments

ATTACHMENT 1 TO C1200-03

BACKGROUND AND ASSESSMENTS FOR
NEW LARGE BREAK LOSS-OF-COOLANT ACCIDENT AND
SMALL BREAK LOSS-OF-COOLANT ACCIDENT ANALYSES OF RECORD

The large break LOCA (LBLOCA) and small break LOCA (SBLOCA) events have been re-analyzed for the upcoming Unit 1 operating cycle (Cycle 17). The new analyses reflect the impact of steam generator replacement and fuel upgrades on the peak cladding temperature (PCT) calculations for these events. The new analyses are based on analytical codes that have been updated to address the impacts of assessments identified in previous 10 CFR 50.46 reports. For the LOCA analysis, additional assessments have also been performed to estimate the impact of a plant design change reducing auxiliary feedwater flow rate and to estimate the burst and blockage/time in life penalty.

When compared to the PCT values reported in the previous Unit 1 10 CFR 50.46 annual report (Reference 1), the new analytical changes, including margin allocations, result in a 482°F increase in the calculated SBLOCA PCT and a 124°F decrease in the calculated LBLOCA PCT. Because the calculated PCT values differ by more than 50°F from the previously reported values, these changes are considered significant in accordance with 10 CFR 50.46(a)(3)(i). The PCT values calculated for both the SBLOCA and LBLOCA continue to be less than the 2200°F limit specified by 10 CFR 50.46(b)(1).

The basis for the new LOCA analyses-of-record and a summary of the effect of the assessments on these analyses are provided below.

CHANGES TO CALCULATED PEAK CLADDING TEMPERATURE FOR SBLOCA

Changes to the calculated PCT for a SBLOCA resulted from a new analysis, an assessment to account for an AFW flow rate reduction, and an assessment to account for burst and blockage/time in life penalty.

New Small Break LOCA Analysis

The new SBLOCA analysis includes changes to reflect replacement steam generator modeling, transition to an enhanced fuel design, and the incorporation of previous computer code changes to account for model assessments.

Replacement Steam Generator Modeling

Indiana Michigan Power Company (I&M) has recently completed replacement of the original Unit 1 Westinghouse Model 51 steam generators with Babcock & Wilcox Model 51R steam generators. In order to minimize the impact on the LOCA analyses-of-record, the nuclear steam supply system (NSSS) design parameters were maintained as close as possible to the design parameters that formed the basis of the current Unit 1 LOCA analyses-of-record.

Major input assumptions used for the previous SBLOCA analysis that were maintained for the new analysis-of-record include a core power level of 3250 megawatts thermal (MWt), a reactor vessel average temperature (T_{avg}) range of 553°F to 576.3°F, a reactor coolant system (RCS) pressure of 2100 pounds per square inch absolute (psia), and a 30 percent average steam generator tube plugging level. The SBLOCA re-analysis also includes a number of physical parameters that differ from the original steam generator parameters. For example, the replacement steam generators have a greater number of tubes, lower resistance to flow, and a slightly increased primary side water inventory.

Transition to Enhanced Fuel Design

The new SBLOCA analysis-of-record incorporates the effects of a change in the fuel assembly design for Cycle 17. The fuel design to be used in Cycle 17 incorporates ZIRLO cladding and intermediate flow mixer grids (IFMs). This fuel design is referred to as VANTAGE+, or by the Westinghouse marketing label of Performance+. By Reference 2, the NRC approved a change to the Technical Specifications (T/S) to allow the use of ZIRLO-clad fuel assemblies. The Cycle 17 core will include both optimized fuel assembly (OFA) fuel assemblies and VANTAGE+ fuel.

Incorporation of Previous Model Assessments

The new Unit 1 SBLOCA analysis also includes computer code changes to reconcile previously reported Unit 1 assessments and to address new changes/errors that had been identified and assessed since the previous Unit 1 annual 10 CFR 50.46 report (Reference 2). The new assessments against the SBLOCA analysis-of-record were the SPIKE correlation revision associated with the burst and blockage/time in life penalty and the NOTRUMP – mixture level tracking/region depletion logic errors. The computer code versions used for the Unit 1 re-analysis include the code updates that address these errors. Therefore, the effect of these impacts is included in the PCT value calculated by the new Unit 1 SBLOCA analysis-of-record and the individual PCT impacts for these assessments are not specifically identified on the SBLOCA margin utilization sheet in Attachment 2. These errors are discussed below.

SBLOCA Burst and Blockage/Time in Life (SPIKE Correlation Revision)

The SPIKE computer program and the associated methodology are used as an evaluation tool to estimate fuel rod burst PCT penalties for SBLOCA analyses. The burst and blockage/time in life assessment is a function of the base PCT plus margin allocations, and as such may either increase or decrease as the margin allocations change. Burst and blockage/time in life effects do not adversely impact SBLOCA analyses with PCTs less than 1700°F.

The SPIKE code has been revised to reflect more recent data that was generated with the current evaluation model and methodology. Using the current SBLOCA evaluation model and considering a series of plant types at varying beginning-of-life non-burst PCTs, a new database of burst data points was developed. The SPIKE computer program and the associated methodology were updated and validated to reflect the new database information. Plant-specific evaluations have been performed for the affected SBLOCA analyses to assess the effect of the revision.

NOTRUMP - Mixture Level Tracking/Region Depletion Logic Errors

The NOTRUMP digital computer code is used in SBLOCA analyses to calculate the transient depressurization of the reactor coolant system (RCS), as well as to describe the mass and enthalpy of the flow through the break. The NOTRUMP code models the RCS as individual volumes, or nodes, that are interconnected by flow paths. Transient behavior of the RCS is determined by applying the governing conservation equations of mass, energy, and momentum of the fluid flowing across the boundaries between nodes.

Several closely related errors have been discovered in how NOTRUMP models the transition across a node boundary between stacked nodes with different fluid levels. The first of these errors occurs when the code attempts to model the mixture level crossing a node boundary in a stack of fluid nodes. In such a case, the fluid mixture being modeled can occasionally have difficulty crossing the node boundary. This condition is referred to as "level hang." When a mixture level hang occurs at a node boundary, the flow for the given time step is reset and becomes inconsistent with the matrix solution of the momentum equation over a long period of time. This results in the generation of local mass/energy errors. In addition, it was discovered that the code did not properly update metal node temperatures as a result of the calculational approach that is used when a fluid node empties or fills. This calculational approach is referred to as the "nodal region depletion logic."

Assessment for Auxiliary Feedwater Flow Rate Reduction

To provide additional margin for the steam generator tube rupture (SGTR) event overfill analysis, I&M has recently implemented a modification to throttle the turbine-driven AFW pump discharge valves. Throttling these valves reduces the turbine-driven AFW flow rate from 560 gpm to approximately 220 gpm, thereby reducing the heat removal capabilities of the secondary side and resulting in a calculated PCT increase. This change does not impact the LBLOCA analysis because AFW is not modeled in the LBLOCA analysis. As indicated in the PCT margin utilization table in Attachment 2, the direct effect of the reduced turbine-driven AFW flow rate is a 109°F PCT penalty for the SBLOCA analysis.

Assessment for Burst and Blockage/Time in Life Penalty

An additional assessment has been made to address the burst and blockage/time in life penalty because the PCT for this case is greater than 1700°F. This assessment determined that a total burst and blockage/time in life penalty of 111°F is applicable to the SBLOCA analysis.

CHANGES TO CALCULATED PEAK CLADDING TEMPERATURE FOR LBLOCA

New Large Break LOCA Analysis

The new LBLOCA analysis includes changes to reflect replacement steam generator modeling, transition to an enhanced fuel design, and the incorporation of previous model assessments.

Replacement Steam Generator Modeling

The LBLOCA has been re-analyzed to reflect the design parameters of the replacement steam generators. Although most key input parameters used for the previous LBLOCA analysis-of-record have also been used for the new LBLOCA analysis, one notable exception is the assumed reduction in the steam generator tube plugging (SGTP) level from 25 percent to 15 percent. The reduced SGTP level was assumed in the LBLOCA analysis to obtain additional margin.

Transition to Enhanced Fuel Design

The previous LBLOCA analysis-of-record submitted in Reference 1 reflected the transition from Westinghouse OFA fuel to the more advanced ZIRLO-clad VANTAGE+ fuel with IFMs. The 31°F transition core PCT penalty that was added to the analysis-of-record assessment in Reference 1 will be applicable until a full core with VANTAGE+ fuel is obtained.

Incorporation of Previous Model Assessments

The new Unit 1 LBLOCA analysis also includes computer code changes to reconcile previously reported Unit 1 assessments and to address an error that had been identified and assessed since the previous Unit 1 annual 10 CFR 50.46 report (Reference 2). The new assessment against the LBLOCA analysis-of-record was for an error associated with the vapor film flow regime heat transfer coefficient determination of the LOCBART computer code.

The LOCBART computer code is used for fuel rod heat-up calculations in the LBLOCA analysis. The Berenson model for film boiling is used in the LOCBART computer code to compute the cladding-to-fluid heat transfer coefficient for conduction in the vapor film flow regime, which occurs near the quench front. Heat transfer in this flow regime is assumed to consist of a conduction component and a radiation component. An error was discovered in the LOCBART computer code whereby the multiplier on this correlation was programmed incorrectly, resulting in a minor underestimation of the cladding-to-fluid heat transfer coefficient.

The LOCBART computer code version used for the Unit 1 re-analysis includes the code update that addresses this error. Therefore, the effect of this impact is included in the PCT value calculated by the new Unit 1 LBLOCA analysis-of-record.

Conclusion

Title 10 of the Code of Federal Regulations, Section 50.46(a)(3)(ii), requires that the 30-day report for significant changes/errors must include a proposed schedule for providing a re-analysis to show compliance with 10 CFR 50.46. As shown in Attachment 2, the values for PCT calculated in the new SBLOCA and LBLOCA analyses-of-record are below the 2200°F limit specified in 10 CFR 50.46(b)(1). Therefore, this submittal satisfies the requirements of 10 CFR 50.46(a)(3)(ii) for annual reporting and reporting of significant errors and changes within 30 days, and no additional actions are required.

References

1. Letter from M. W. Rencheck (I&M) to U. S. Nuclear Regulatory Commission Document Control Desk, "Donald C. Cook Nuclear Plant Unit 1, Annual Report of Loss-of-Coolant Accident Evaluation Model Changes and Submittal of New Large Break Loss-of-Coolant Accident Analyses-of-Record for Unit 1," Submittal C0300-07, dated March 31, 2000.
2. Letter from J. F. Stang (NRC) to R. P. Powers (I&M), "Issuance of Amendments - Donald C. Cook Nuclear Plant, Units 1 and 2, Re: Fuel Rod ZIRLO Cladding and Integral Fuel Burnable Absorber Requirements (TAC Nos. MA7041 and MA 7042)," dated January 6, 2000.

ATTACHMENT 2 TO C1200-03

PEAK CLADDING TEMPERATURE MARGIN UTILIZATION TABLES FOR LARGE AND SMALL BREAK LOSS-OF-COOLANT ACCIDENT ANALYSES-OF-RECORD

DONALD C. COOK NUCLEAR PLANT, UNIT 1

TABLE 1

D. C. COOK UNIT 1

LARGE BREAK LOCA

Evaluation Model: BASH

$F_Q = 2.15$ $F_{\Delta H} = 1.55$ SGTP = 15% Break Size: $C_d = 0.4$

Operational Parameters: RHR System Cross-Tie Valve Closed, 3250 MWt Reactor Power

Fuel: Vantage+ (ZIRLO clad, IFM grids)

LICENSING BASIS

	Analysis-of-Record - December 2000	PCT = 2038°F

MARGIN ALLOCATIONS (Delta PCT)

A.	PREVIOUS ECCS MODEL ASSESSMENTS [1]	0°F
B.	2000 10 CFR 50.46 MODEL ASSESSMENTS	0°F
C.	OTHER	
	1. Transition Core Penalty	ΔPCT= + 31°F
D.	LICENSING BASIS PCT + MARGIN ALLOCATIONS	PCT= 2069°F

Footnote:

1. The December 2000 Analysis-of-Record used updated versions of the analytical codes to include the effects of replacement steam generators, enhanced fuel design, and impacts from previous emergency core cooling system (ECCS) model assessments.

TABLE 2

D. C. COOK UNIT 1

SMALL BREAK LOCA

Evaluation Model: NOTRUMP

$F_Q = 2.32$ $F_{\Delta H} = 1.55$ SGTP = 30% 3" cold leg break

Operational Parameters: SI System Cross-Tie Valve Closed, 3250 MWt Reactor Power

Fuel: Vantage+ (ZIRLO clad, IFM grids)

LICENSING BASIS			
	Analysis-of-Record - December 2000	PCT =	1720°F
MARGIN ALLOCATIONS (Delta PCT)			
A.	PREVIOUS ECCS MODEL ASSESSMENTS [1]		0°F
B.	2000 10 CFR 50.46 MODEL ASSESSMENTS		0°F
C.	OTHER		
	1. Asymmetric HHSI delivery [2]	ΔPCT =	+ 50°F
	2. Reduction in Turbine Driven Auxiliary Feedwater Flow	ΔPCT =	+109°F
	3. Burst and Blockage / Time in Life	ΔPCT =	+111°F
D.	LICENSING BASIS PCT + MARGIN ALLOCATIONS	PCT =	1990°F

Footnotes:

1. The December 2000 Analysis-of-Record used updated versions of the analytical codes to include the effects of replacement steam generators, enhanced fuel design, and impacts from previous emergency core cooling system (ECCS) model assessments.

2. See Letter from M. W. Rencheck, I&M, to U. S. Nuclear Regulatory Commission, "Donald C. Cook Nuclear Plant Units 1 and 2, Errors in Loss-Of-Coolant-Accident Evaluation Models," submittal C1299-04, dated December 9, 1999.