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United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Byron Station, Unit 1
Facility Operating License No. NPF-37
NRC Docket No. STN 50-454

Subject: Thirty-day Report of the Emergency Core Cooling System Evaluation Model Changes and Errors Required by 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors"

In accordance with 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactor," paragraph (a)(3)(ii), Exelon Generation Company, LLC, is submitting a 30-day report of the emergency core cooling system (ECCS) evaluation model changes and errors for Byron Station, Unit 1. The Byron Station, Unit 1 Cycle 13 core reload analysis resulted in a peak cladding temperature (PCT) penalty assessment of greater than 50°F and is therefore considered a "significant" change as defined by 10 CFR 50.46, paragraph (a)(3)(i); thus requiring a 30-day report. This report is due to the NRC 30 days after the initiation of Byron Station, Unit 1 Cycle 13, i.e., November 12, 2003.

The ECCS evaluation model for large break loss of coolant analysis (LOCA) was assessed a PCT penalty of 80°F for an axial power shape distribution envelope violation. All other evaluation model changes and errors contained in the attached report are not considered "significant" changes. Considering the assessed PCT penalties, Byron Station, Unit 1 continues to comply with the requirements of 10 CFR 50.46 and therefore, no reanalysis or other actions are needed to show compliance with 10 CFR 50.46.

Attachment 1, "Peak Cladding Temperature Rack-Up Sheets," provides updated information regarding the PCT for the limiting small break and large break LOCA analyses evaluations for Byron Station, Unit 1. Attachment 2, "Assessment Notes," contains a detailed description for each change or error reported. Attachment 3, "Assessment Notes Not Included in Peak Cladding Temperature Rack-Up Sheets," contains a brief description of other LOCA assessments not included in the PCT Rack-Up Sheets. All assessments in Attachment 3 resulted in benefits or no penalty to the calculated PCT. Note that we have conservatively chosen not to credit any PCT benefits, i.e., for each beneficial change, a change of PCT of 0°F was assigned.

Please contact Mr. J. A. Bauer at (630) 657-2801 should you have any questions concerning this report.

Respectfully,

Kenneth A. Ainger
Manager, Licensing

Attachment 1: Peak Cladding Temperature Rack-Up Sheets
Attachment 2: Assessment Notes
Attachment 3: Assessment Notes Not Included in Peak Cladding Temperature Rack-Up Sheets

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Byron Station

ATTACHMENT 1

10 CFR 50.46
"Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors"

30-Day Report of the Emergency Core Cooling System Evaluation Model Changes and Errors

Assessments as of October 10, 2003

Peak Cladding Temperature Rack-Up Sheets

PLANT NAME: Byron Station Unit 1
ECCS EVALUATION MODEL: SBLOCA
REPORT REVISION DATE: 02/25/03
CURRENT OPERATING CYCLE: 13

AOR

Evaluation Model: NOTRUMP
Calculation: Westinghouse CN-LIS-00-208, December 2000
Fuel: VANTAGE+ 17 x 17
Limiting Fuel Type: VANTAGE+ 17 x 17
Limiting Single Failure: Loss of one train of ECCS flow
Heat Flux Hot Channel Factor (FQ) = 2.60
Nuclear Enthalpy Rise Hot Channel Factor (FNΔH) = 1.70
Steam Generator Tube Plugging (SGTP) = 5%
Limiting Break: 2" Low Tavg

Notes: Zr-4/ZIRLO Clad Fuel

Reference PCT PCT = 1624.0°F

MARGIN ALLOCATION

A. PRIOR LOCA MODEL ASSESSMENTS

None

B. CURRENT LOCA MODEL ASSESSMENTS

None

NET PCT **PCT = 1624.0°F**

PLANT NAME:	Byron Station Unit 1
ECCS EVALUATION MODEL:	LBLOCA
REPORT REVISION DATE:	06/30/03
CURRENT OPERATING CYCLE:	13

AOR

Evaluation Model: WCOBRA/TRAC
Calculation: Westinghouse CN-LIS-00-7, September 2000
Fuel: VANTAGE+ 17 x 17
Limiting Fuel Type: VANTAGE+ 17 x 17
Limiting Single Failure: Loss of one train of ECCS flow
Steam Generator Tube Plugging (SGTP) = 5%
Heat Flux Hot Channel Factor (FQ) = 2.60
Nuclear Enthalpy Rise Hot Channel Factor (FNΔH) = 1.70
Limiting Break Size: Guillotine

Notes: Zr-4/ZIRLO Clad Fuel

Reference PCT	PCT = 2044.0°F

MARGIN ALLOCATION

A. PRIOR LOCA MODEL ASSESSMENTS

Decay Heat Uncertainty Error (Note 1)	ΔPCT = 12.0°F

B. CURRENT LOCA MODEL ASSESSMENTS

Axial Power Shape Distribution Violation (Note 2)	ΔPCT = 80.0°F

NET PCT	**PCT = 2136.0°F**

Attachment 2

10 CFR 50.46,
"Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors,"

Report of the Emergency Core Cooling System Evaluation Model Changes and Errors

Assessment Notes

1. Decay Heat Uncertainty Error in Monte Carlo Calculations

It was determined that an error existed in the calculation of the decay heat uncertainty in the Monte Carlo code used for calculation of the 95th percentile peak cladding temperature (PCT) for Best Estimate (BE) Large Break (LB) Loss of Coolant Accident (LOCA). This issue was determined to be a "Non-Discretionary Change" as defined by Section 4.1.2 of WCAP-13451, "Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting," October 1992.

The increase in PCT for each phase of the BELOCA transient was calculated and the most limiting PCT penalty was calculated to be 12°F, independent of the transient phase, (i.e., blowdown, reflood 1 and reflood 2) and this PCT penalty was applied to the composite results.

2. Axial Power Shape Distribution Envelope Violation (PMID, PBOT)

The LBLOCA analysis is performed based on assuming an axial power shape distribution envelope (i.e., PMID, PBOT), where PMID is the power in the middle one-third of the core; and PBOT is the power in the lower one-third of the core. The envelope is pertinent to the BELOCA analysis and was presented in a letter from R. M. Krich (Commonwealth Edison Company; now Exelon Generation Company, LLC) to the NRC, "Request for a License Amendment for Plant Specific Use of Best Estimate Large Break Loss of Coolant Accident Analysis," dated October 24, 2000. For every reload cycle, Westinghouse verifies that the envelope remains limiting. If there is a violation then a PCT penalty is calculated.

For Byron Station, Unit 1 Cycle 13, there was a violation; a PCT penalty of 80°F was calculated.

Attachment 3

10 CFR 50.46,
"Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors,"

Report of the Emergency Core Cooling System Evaluation Model Changes and Errors

Assessment Notes Not Included in Peak Cladding Temperature Rack-up Sheets

The following is a brief description of other loss of coolant accident (LOCA) assessments that reflect changes to the evaluation models, which are not included in the peak cladding temperature (PCT) rack-up sheets. These assessments, in all cases, resulted in benefits or 0°F penalty to the calculated PCT. However, we have conservatively chosen not to credit these PCT benefits, i.e., for each change a delta PCT of 0°F is assigned. Evaluations of these changes are based upon conservative generic studies for Westinghouse designed nuclear steam supply systems (NSSSs) or engineering judgment. If a re-analysis or an evaluation is obtained from Westinghouse, the impact of these changes will be included and the effect of these changes will be reported as applicable.

Performance Analysis and Design Model (PAD) 4.0 Implementation

The Westinghouse Performance Analysis and Design Model (PAD) is used to generate fuel-related input data for use in LOCA licensing calculations. As documented in the below referenced document, the Safety Evaluation Report for Version 4.0 of the PAD model was issued by the NRC on April 24, 2000. Use of PAD Version 4.0 is considered to represent a "Discretionary Change" and will be implemented on a forward-fit basis, in accordance with Section 4.1.1 of WCAP-13451, "Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting," October 1992.

The implementation of PAD Version 4.0 with respect to the large break (LB) and small break (SB) LOCA analyses will be handled on a forward-fit basis from this submittal and is assigned a PCT estimated change of 0°F for 10 CFR 50.46 reporting purposes.

Reference: WCAP-15063-P-A, Revision 1, with Errata, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," J. P. Foster and S. Sidener, July 2000.

Improved Code I/O and Diagnostics and General Code Maintenance

Various changes in code input and output format have been made to enhance usability and help preclude errors in analyses. This includes both input changes (e.g., more relevant input variables defined and more common input values used as defaults) and input diagnostics designed to preclude unreasonable values from being used, as well as various changes to code output which have no effect on calculated results. In addition, various blocks of coding were written to eliminate inactive coding, optimize the active coding, and improve commenting, both for enhanced usability and to facilitate code debugging when necessary. These changes were determined to be Discretionary Changes in accordance with Section 4.1.1 of WCAP- 13451.

The nature of these changes leads to an estimated PCT impact of 0°F.

Trapped Nitrogen in Accumulator Lines

To address the potential for gas accumulation in the ECCS piping between the two check valves in the accumulator line (i.e., between valves SI8948 and SI8956), an evaluation was performed by Westinghouse.

This evaluation determined that there was no PCT impact for either the SBLOCA or the LBLOCA analyses.

Removed Upper Internal Assembly Alignment Pins for Byron Units

This assessment addresses the removal of the upper internal alignment pins at the Byron Station. Two pins have been removed from Byron Station Unit 1. Westinghouse performed an evaluation, considering uprated power conditions, to determine the impact of the removal of the fuel alignment pins. The results of the evaluation determined that the impact was insignificant for LBLOCA and therefore there is no PCT penalty. For a SBLOCA, the transient is slow in terms of core flows, providing sufficient time to maintain equilibrium between assembly flow channels having minor differences in hydraulic resistances. Therefore, there is no PCT penalty for the SBLOCA analysis.

Passive Heat Sink Evaluation

The amount of passive heat sinks assumed in the LBLOCA Analysis of Record (AOR) is documented in the Byron Station Updated Final Safety Analysis Report (UFSAR) Table 6.2-55, "Passive Heat Sink Data For Minimum Post LOCA Containment Pressure." For future modifications inside the containment, additional amounts of passive heat sinks were assumed in the AOR. Subsequent to the AOR, several modifications were done inside the containment. Evaluations were performed and it was determined that the additional amount of heat sinks due to these modifications was less than the additional allowance assumed in the AOR; therefore, there is no PCT impact for the LBLOCA analysis. In the SBLOCA analysis, the containment is not modeled; thus, there is no impact on the SBLOCA analysis.

Oxidation Thickness Index Error For Best Estimate WCOBRA/TRAC

A coding error has been identified in the initial outside oxidation thickness array used for fuel rods. The error was an incorrect index for storage of the oxide thickness for each fuel rod. Coding used the rod number index instead of the rod type index. This issue was determined to be a "Non-Discretionary Change" in accordance with Section 4.1.2 of WCAP-13451.

The error was found to have no effect for standard Best Estimate (BE) LOCA analyses that follow the published guidance material for input of this variable. The error also did not affect any test simulations performed to support the licensing of the BE evaluation model. Thus, there was no use of erroneous oxidation thickness and there is no PCT impact for this error. The error will be corrected during the next revision of the BE WCOBRA/TRAC code.

Neutronics Calculation Moderator Density Weighting Factor Error

An error was discovered in WCOBRA/TRAC whereby power used in normalization of moderator density weighting factors was double-accounted for channels with multiple simulated rods. The error biases the average moderator density to be slightly higher, resulting in slightly higher power generation in the hot rod. The error is qualitatively conservative and quantitatively insignificant. This issue was determined to be a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP- 13451.

At the beginning of the transient calculation, the difference in weighted density is less than 1% for all plant types. This difference is similar to the density difference between (2250 psia, 586°F) and (2250 psia, 588.8°F) thermodynamic state points. The difference in average moderator density affects the reactivity. The difference in reactivity at the

beginning of the transient is negligible. As the transient progresses, with voiding of the core, the strong negative moderator reactivity feedback dominates. Therefore, it was estimated that the error has 0°F PCT impact on plant calculations. The error will be corrected during the next revision of the BE WCOBRA/TRAC code.

Inclusion of Required NOTRUMP Version 38.0 Input Variables in SPADES

Following the release of NOTRUMP Version 38.0 code, which introduced several new input variables to the evaluation model, it became necessary to update the SPADES code to reflect these new input variables. These input variables are required to activate the revised model features incorporated into the NOTRUMP Version 38.0 code. This change was determined to be a Discretionary Change in accordance with Section 4.1.1 of WCAP-13451.

This change simply introduces the new input parameters required by the release of NOTRUMP Version 38.0 to SPADES. The revised NOTRUMP model PCT effects have previously been assessed and this change to SPADES does not introduce an additional PCT impact.

Use of NOTRUMP Subcooled Steam Table Routines in SPADES

A review of SPADES calculation methodology determined that subcooled fluid node properties were being calculated based on steam tables that were inconsistent with those of NOTRUMP. As a result, slight differences in fluid node conditions could be seen between SPADES and NOTRUMP. The SPADES code has been modified to utilize the NOTRUMP subcooled steam table properties. This reduces perturbations incurred during the steady-state simulation period with NOTRUMP resulting from differences in subcooled steam table properties. This revision was determined to be a Discretionary Change in accordance with Section 4.1.1 of WCAP-13451.

The nature of this change leads to an estimated PCT impact of 0°F.

Accumulator Line Friction Factor in the NOTRUMP Evaluation Model

The current input for the NOTRUMP evaluation model uses a dimensionless value of 0.013 for line loss friction factor in the accumulator injection lines. This is based on fully developed, turbulent flow in the general pipe size range for accumulator injection lines applicable to Westinghouse designed NSSSs. However, in the small break LOCA analysis during accumulator injection, the flow seldom obtains velocities high enough to support the fully developed, turbulent flow value. Taking this into account yields a friction factor on the order of 0.016. This revision was determined to be a Discretionary Change in accordance with Section 4.1.1 of WCAP-13451.

The nature of this change leads to an estimated PCT impact of 0°F.

Large Break LOCA Vessel Geometry Input Errors

Several minor geometric errors associated with metal heat slabs in the vessel portion of the WCOBRA/TRAC model of the BELOCA analyses for Byron Station, Unit 1 were identified. Several metal heat slabs in both the lower plenum and upper plenum were

identified to have either overestimated the metal mass or overestimated the heat slab time constant. The upper plenum metal mass errors are generally inconsequential since this zone becomes steam shortly after the onset of the transient. The lower plenum errors are dominated by an overestimation of the radial keys, which is judged conservative since additional metal heat release will occur during reflood. However, the extent of conservatism is minimal since Byron Station analyses are not late reflood limited.

Engineering judgment was applied to assess the errors taking into account their relative magnitude and level of importance. The net impact of these vessel geometric errors is judged to be 0 °F.

Broken Cold Leg Modeling Deviations

The broken pipe modeling described in Section 22-6-1 of WCAP-12945-P-A, "Code Qualification Document for Large Break BELOCA," 1998, states that the piping between the reactor coolant pump (RCP) and the vessel is to be divided into seven cells of the same length. The break location is assumed such that the RCP-side pipe has three cells and the vessel-side pipe has four cells. For Byron Station, the cold leg nozzle cell used the actual nozzle length making the cell length different from the other six cells. A code change was made to ensure that the long pipe logic is applied for guillotine breaks. This coding change was determined to be a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

The nature of this change leads to an estimated PCT impact of 0°F.

1-D Minimum Film Boiling Temperature Model Selection Error

Section 6-3-6 of WCAP-12945-P-A indicates that the minimum film boiling temperature calculation for 1-D components is calculated as the maximum of the homogeneous nucleation temperature and that predicted by the Iloeje correlation. The comparison of these two correlations is made if a program flag (ITMIN) is set greater than zero; otherwise, the homogeneous nucleation temperature is used. It was found that ITMIN was not initialized, resulting in the Iloeje correlation not being considered. This error has the potential to affect the heat transfer calculations in the steam generator tubes of the STGEN subroutine. The coding was corrected to be consistent with the description in Section 6-3-6. This coding change was determined to be a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

The nature of this change leads to an estimated PCT impact of 0°F.

1-D Condensation Ramp Error

Section 5-3-5 of WCAP-12945-P-A indicates that condensation in specified one-dimensional components is suppressed if the pressure drops significantly below the containment pressure, using Equation 5-95a. This ramp was erroneously applied to the interfacial heat transfer for superheated liquid, affecting the evaporation process as well as the condensation due to subcooled liquid. The coding has been corrected so that it is applied to condensation conditions only. This coding change was determined to be a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

The nature of this change leads to an estimated PCT impact of 0°F.

Cladding Axial Thermal Expansion Error

The cladding axial thermal expansion enters into the calculation of the fuel rod internal pressure via the time-dependent gas plenum volume (reference Equation 7-46 of WCAP-12945-P-A). Equation 7-39 shows how the cladding axial thermal expansion over the length of the rod is calculated. Table 7-1 of WCAP-12945-P-A shows that the cladding axial thermal expansion is based on a linear interpolation scheme over a temperature range of 1073-1273°K. The CALL statement for the interpolation subroutine had a typographical error in one of the arguments, such that the axial thermal expansion was evaluated incorrectly. The error was corrected. This coding change was determined to be a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

The nature of this change leads to an estimated PCT impact of 0°F.

Error in Time After Shutdown for Neutron Capture Term

Equation 8-45 of WCAP-12945-P-A shows the neutron capture correction factor specified by the American National Standards Institute / American Nuclear Society (ANSI/ANS) 5.1-1979 standard, "Decay Heat Power in Light Water Reactors." The time after shutdown term, t, was incorrectly programmed to use the total calculation time, including the steady state calculation. The coding has been corrected so that it is defined as the time after initiation of the break. This coding change was determined to be a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

The nature of this change leads to an estimated PCT impact of 0°F.

SBLOCTA Code Time Step Selection Logic

The SBLOCTA code was updated to resolve some inconsistencies in the time step selection logic, pertaining to the use of the fluid vs. fuel rod time step. This represents a closely related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451.

The nature of this change leads to an estimated PCT impact of 0°F.

SBLOCTA ZIRLO™ Cladding Specific Heat Model

For consistency with the change made to LOCBART code, the ZIRLO™ cladding specific heat model in the SBLOCTA code has been revised to reflect data collected at the Thermophysical Properties Research Laboratory. This represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

The nature of this change leads to an estimated PCT impact of 0°F.

Simplified Isothermal Solution for SBLOCTA Subroutine Rate

As discussed in Reference 1, LOCBART was revised in 1999 to correct a logic error that caused the Baker-Just metal-water reaction calculations to be performed three times per time step. During the review of the corresponding code logic, it was determined that the complicated solution technique described in Section 3.3.2 of Reference 2 could be replaced with a simplified isothermal solution, with only a minimal effect on results. This change was made in LOCBART per Reference 3 and has also been implemented in

SBLOCTA which uses similar logic. This represents a Discretionary Change that will be implemented on a forward-fit basis in accordance with Section 4.1.1 of WCAP-13451. The nature of this change leads to an estimated PCT impact of 0°F.

References:

1. Westinghouse Letter NSBU-NRC-00-5970, "1999 Annual Notification of Changes to the Westinghouse Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10 CFR 50.46 (a)(3)(ii)", May 12, 2000

2. WCAP-8301, "LOCTA-IV Program: Loss-of-Coolant Transient Analysis", June 1974

3. Westinghouse Letter LTR-NRC-01-6, "U. S. Nuclear Regulatory Commission, 10 CFR 50.46 Annual Notification and Reporting for 2000", March 13, 2001