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ComEd

April 18, 2000

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

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Annual 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(a)(3)(ii), we are submitting a report of the Emergency Core Cooling System (ECCS) Evaluation Model changes and errors for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2. This report satisfies the annual reporting requirements of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors." The Annual Report is due April 21, 2000.

Attachment 1 provides updated information regarding the Peak Cladding Temperature (PCT) for the limiting Small Break and Large Break Loss of Coolant Accident analysis evaluations for the Byron and Braidwood Stations (i.e., PCT rack-up sheets). Attachment 1 includes all assessments as of March 22, 2000. Attachment 2 contains a detailed description for each change or error reported. Attachment 3 contains a brief description of other Loss of Coolant Accident assessments, which are not included in the rack-up sheets. In all cases outlined in Attachment 3, these assessments resulted in benefits or no penalty to the calculated PCT but we have conservatively chosen not to credit any PCT benefits, i.e., for each change a delta PCT of zero degrees Fahrenheit is assigned.

For all the evaluation model changes and errors contained in this report, we have determined that these changes and errors are not significant as defined in paragraph 10 CFR 50.46(a)(3)(i), and that the Byron and Braidwood Stations continue to comply with the requirements of 10 CFR 50.46. Compliance with 10 CFR 50.46 is maintained with existing PCT penalties. Therefore, no near-term reanalysis is planned for the Byron and Braidwood Stations, and a proposed schedule for providing a reanalysis is not required.

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Should you have any questions concerning this letter, please contact Ms. K.M. Root at (630) 663-7292.

Respectfully,

R.M. Krich
Vice President - Regulatory Services

Attachment 1 – Assessments as of March 22, 2000 (Rack-up Sheets)
Attachment 2 – Assessment Notes
Attachment 3 - Assessment Notes Not Included in Rack-ups

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Braidwood Station
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," Annual Report of the Emergency Core Cooling System Evaluation Model Changes and Errors

Assessments as of March 22, 2000
(Rack-up Sheets)

PLANT NAME: Braidwood Station Unit 1
ECCS EVALUATION MODEL: Small Break Loss of Coolant Accident (SBLOCA)
REPORT REVISION DATE: 3/22/00
CURRENT OPERATING CYCLE: 9

ANALYSIS OF RECORD (AOR)

Evaluation Model: NOTRUMP
Calculation: Westinghouse SEC-LIS-5314-C0, October 1997
Fuel: VANTAGE+ 17 x 17
Heat Flux Hot Channel Factor (FQ) = 2.60
Nuclear Enthalpy Rise Hot Channel Factor (FNΔH) = 1.70
Steam Generator Tube Plugging (SGTP) = 30%

Reference Peak Cladding Temperature (PCT) PCT = 1695.0°F

MARGIN ALLOCATION

A. PRIOR LOSS OF COOLANT ACCIDENT (LOCA) MODEL ASSESSMENTS

None

B. CURRENT LOCA MODEL ASSESSMENTS

None

NET PCT **PCT = 1695.0°F**

PLANT NAME:	Braidwood Station Unit 1
ECCS EVALUATION MODEL:	Large Break Loss of Coolant Accident (LBLOCA)
REPORT REVISION DATE:	3/22/00
CURRENT OPERATING CYCLE:	9

AOR

Evaluation Model: BASH
Calculation: Westinghouse SEC-SAII-4747-C2, May 1996
Fuel: VANTAGE+ 17 x 17
FQ = 2.60 (Fq reduced to 2.5, Note 7)
FNΔH = 1.70
SGTP = 30%

Reference PCT	PCT =	1968.0°F

MARGIN ALLOCATION

A. PRIOR LOCA MODEL ASSESSMENTS

Translation of Fluid Conditions from SATAN to LOCTA (Note 8)	ΔPCT =	15.0°F
Reactor Coolant System (RCS) Crossover Leg Volume (Note 9)	ΔPCT =	3.0°F
Replacement Steam Generator (RSG) (SGTP 20%, Note 10)	ΔPCT =	21.0°F
Passive Heat Sink Increase (Note 11)	ΔPCT =	16.0°F
Reactor Coolant Fan Cooler (RCFC) Performance (Note 11)	ΔPCT =	1.0°F
LOCBART Fuel Rod Outside Diameter (FOD) Input Error (Note 13)	ΔPCT =	2.0°F
Initial Containment Pressure (Note 11)	ΔPCT =	-5.0°F
LBLOCA Burst Location Change (Note 12)	ΔPCT =	94.0°F
Removal of Translation of Fluid Conditions from SATAN to LOCTA (Note 8)	ΔPCT =	-15.0°F

B. CURRENT LOCA MODEL ASSESSMENTS

Removal of Reactor Coolant System (RCS) Crossover Leg Volume (Notes 7 and 9)	ΔPCT =	-3.0°F
Accumulator Line/Pressurizer Surge Line Data (Note 1), LOCBART Space Grid Single Phase Heat Transfer Error (Note 5), LOCBART Zirc-Water Oxidation Error (Note 6), and Reanalysis of Limiting AOR (Note 7)	ΔPCT =	-32.0°F
Storage of lead shielding blankets in Containment (Note 15)	ΔPCT =	2.0°F
Decrease in RCFC start time (Note 15)	ΔPCT =	2.0°F
Increase in Containment Spray Flow rate (Note 15)	ΔPCT =	1.0°F

NET PCT	**PCT =**	**2070.0°F**

PLANT NAME: Braidwood Station Unit 2
ECCS EVALUATION MODEL: SBLOCA
REPORT REVISION DATE: 3/22/00
CURRENT OPERATING CYCLE: 8

AOR

Evaluation Model: NOTRUMP
Calculation: Westinghouse SEC-LIS-5396-C0, January 1999
Fuel: VANTAGE+ 17 x 17
FQ = 2.70
FNΔH = 1.75
SGTP = 30%

Reference PCT	PCT =	1806.0°F

MARGIN ALLOCATION

A. PRIOR LOCA MODEL ASSESSMENTS

Burst and Blockage/Time in Life (Note 3)	ΔPCT =	19.0°F

B. CURRENT LOCA MODEL ASSESSMENTS

None

NET PCT	**PCT =**	**1825.0°F**

PLANT NAME: Braidwood Station Unit 2
ECCS EVALUATION MODEL: LBLOCA
REPORT REVISION DATE: 3/22/00
CURRENT OPERATING CYCLE: 8

AOR

Evaluation Model: BASH
Calculation: Westinghouse SEC-SAII-4747-C2, May 1996
Fuel: VANTAGE+ 17 x 17
FQ = 2.60 (Fq reduced to 2.5, Note 7)
FNΔH = 1.70
SGTP = 30%

Reference PCT	PCT =	1968.0°F

MARGIN ALLOCATION

A. PRIOR LOCA MODEL ASSESSMENTS

Translation of Fluid Conditions from SATAN to LOCTA (Note 8)	ΔPCT =	15.0°F
RCS Crossover Leg Volume (Note 9)	ΔPCT =	3.0°F
Passive Heat Sink Increase (Note 11)	ΔPCT =	16.0°F
RCFC Performance (Note 11)	ΔPCT =	1.0°F
LOCBART FOD Input Error (Note 13)	ΔPCT =	2.0°F
Initial Containment Pressure (Note 11)	ΔPCT =	-5.0°F
LBLOCA Burst Location Change (Note 12)	ΔPCT =	94.0°F
Removal of Translation of Fluid Conditions from SATAN to LOCTA (Note 8)	ΔPCT =	-15.0°F

B. CURRENT LOCA MODEL ASSESSMENTS

Removal of Reactor Coolant System (RCS) Crossover Leg Volume (Notes 7 and 9)	ΔPCT =	-3.0°F
Accumulator Line/Pressurizer Surge Line Data (Note 1), LOCBART Space Grid Single Phase Heat Transfer Error (Note 5), LOCBART Zirc-Water Oxidation Error (Note 6), and Reanalysis of Limiting AOR (Note 7)	ΔPCT =	-32.0°F
Storage of lead shielding blankets in Containment (Note 15)	ΔPCT =	2.0°F
Decrease in RCFC start time (Note 15)	ΔPCT =	2.0°F
Increase in Containment Spray Flow rate (Note 15)	ΔPCT =	1.0°F

NET PCT	**PCT =**	**2049.0°F**

PLANT NAME: Byron Station Unit 1
ECCS EVALUATION MODEL: SBLOCA
REPORT REVISION DATE: 3/22/00
CURRENT OPERATING CYCLE: 10

AOR

Evaluation Model: NOTRUMP
Calculation: Westinghouse SEC-LIS-5314-C0, October 1997
Fuel: VANTAGE+ 17 x 17
FQ = 2.60
FNΔH = 1.70
SGTP = 30%

Reference PCT PCT = 1695.0°F

MARGIN ALLOCATION

A. PRIOR LOCA MODEL ASSESSMENTS

None

B. CURRENT LOCA MODEL ASSESSMENTS

None

NET PCT **PCT = 1695.0°F**

PLANT NAME:	Byron Station Unit 1
ECCS EVALUATION MODEL:	LBLOCA
REPORT REVISION DATE:	3/22/00
CURRENT OPERATING CYCLE:	10

AOR

Evaluation Model: BASH
Calculation: Westinghouse SEC-SAII-4747-C2, May 1996
Fuel: VANTAGE+ 17 x 17
FQ = 2.60 (Fq reduced to 2.5, Note 7)
FNΔH = 1.70
SGTP = 30%

Reference PCT	PCT =	1968.0°F

MARGIN ALLOCATION

A. PRIOR LOCA MODEL ASSESSMENTS

Removed Upper Internal Assembly Alignment Pins (Note 2)	ΔPCT =	5.0°F
Assembly Guide Pin Flakes (Note 4)	ΔPCT =	6.0°F
Translation of Fluid Conditions from SATAN to LOCTA (Note 8)	ΔPCT =	15.0°F
RCS Crossover Leg Volume (Note 9)	ΔPCT =	3.0°F
Replacement Steam Generator (RSG) (SGTP 20%, Note 10)	ΔPCT =	21.0°F
Passive Heat Sink Increase (Note 11)	ΔPCT =	16.0°F
RCFC Performance (Note 11)	ΔPCT =	1.0°F
LOCBART FOD Input Error (Note 13)	ΔPCT =	2.0°F
Initial Containment Pressure (Note 11)	ΔPCT =	-5.0°F
LBLOCA Burst Location Change (Note 12)	ΔPCT =	94.0°F
Removal of Translation of Fluid Conditions from SATAN to LOCTA (Note 8)	ΔPCT =	-15.0°F

B. CURRENT LOCA MODEL ASSESSMENTS

Removal of Reactor Coolant System (RCS) Crossover Leg Volume (Notes 7 and 9)	ΔPCT =	-3.0°F
Accumulator Line/Pressurizer Surge Line Data (Note 1), LOCBART Spacer Grid Single Phase Heat Transfer Error (Note 5), LOCBART Zirc-Water Oxidation Error (Note 6), and Reanalysis of Limiting AOR (Note 7)	ΔPCT =	-32.0°F
Storage of lead shielding blankets in Containment (Note 15)	ΔPCT =	2.0°F
Decrease in RCFC start time (Note 15)	ΔPCT =	2.0°F
Increase in Containment Spray Flow rate (Note 15)	ΔPCT =	1.0°F

NET PCT	**PCT =**	**2081.0°F**

PLANT NAME: Byron Station Unit 2
ECCS EVALUATION MODEL: SBLOCA
REPORT REVISION DATE: 3/22/00
CURRENT OPERATING CYCLE: 9

AOR

Evaluation Model: NOTRUMP
Calculation: Westinghouse SEC-LIS-5396-C0, January 1999
Fuel: VANTAGE+ 17 x 17
FQ = 2.70
FNΔH = 1.75
SGTP = 30%

Reference PCT	PCT =	1806.0°F

MARGIN ALLOCATION

A. PRIOR LOCA MODEL ASSESSMENTS

Burst and Blockage/Time in Life (Note 3)	ΔPCT =	19.0°F

B. CURRENT LOCA MODEL ASSESSMENTS

None

NET PCT	**PCT =**	**1825.0°F**

PLANT NAME: Byron Station Unit 2
ECCS EVALUATION MODEL: LBLOCA
REPORT REVISION DATE: 3/22/00
CURRENT OPERATING CYCLE: 9

AOR

Evaluation Model: BASH
Calculation: Westinghouse SEC-SAII-4747-C2, May 1996
Fuel: VANTAGE+ 17 x 17
FQ = 2.60 (Fq reduced to 2.5, Note 7)
FNΔH = 1.70
SGTP = 30%

Reference PCT — PCT = 1968.0°F

MARGIN ALLOCATION

A. PRIOR LOCA MODEL ASSESSMENTS

Removed Upper Internal Assembly Alignment Pins (Note 2)	ΔPCT =	28.0°F
Translation of Fluid Conditions from SATAN to LOCTA (Note 8)	ΔPCT =	15.0°F
RCS Crossover Leg Volume (Note 9)	ΔPCT =	3.0°F
Passive Heat Sink Increase (Note 11)	ΔPCT =	16.0°F
RCFC Performance (Note 11)	ΔPCT =	1.0°F
LOCBART FOD Input Error (Note 13)	ΔPCT =	2.0°F
Initial Containment Pressure (Note 11)	ΔPCT =	-5.0°F
LBLOCA Burst Location Change (Note 12)	ΔPCT =	94.0°F
Removal of Translation of Fluid Conditions from SATAN to LOCTA (Note 8)	ΔPCT =	-15.0°F

B. CURRENT LOCA MODEL ASSESSMENTS

Removal of Reactor Coolant System (RCS) Crossover Leg Volume (Notes 7 and 9)	ΔPCT =	-3.0°F
Accumulator Line/Pressurizer Surge Line Data (Note 1), LOCBART Space Grid Single Phase Heat Transfer Error (Note 5), LOCBART Zirc-Water Oxidation Error (Note 6), and Reanalysis of Limiting AOR (Note 7)	ΔPCT =	-32.0°F
Storage of lead shielding blankets in Containment (Note 15)	ΔPCT =	2.0°F
Decrease in RCFC start time (Note 15)	ΔPCT =	2.0°F
Increase in Containment Spray Flow rate (Note 15)	ΔPCT =	1.0°F

NET PCT — **PCT = 2077.0°F**

Attachment 2

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

Assessment Notes

1. Accumulator Line/Pressurizer Surge Line Data

Westinghouse identified an issue where the accumulator line piping schedule installed at a plant was different than the design value. This discovery led to a review of various geometric data related to the accumulator lines and pressurizer surge lines. Revised data was compared to the Loss of Coolant Accident (LOCA) analysis values to determine the effect on existing analysis results.

For the Small Break Loss of Coolant Accident (SBLOCA), the estimated effect of this issue on PCT is 0°F, based on the following general characteristics of limiting small break transients. Only a small fraction of the available accumulator capacity is generally required to replenish vessel inventory to a level sufficient to terminate the cladding temperature excursion, and small variations in the rate of accumulator injection would be expected to have a minimal effect on PCT results. Furthermore, the pressurizer empties well before any core uncovery occurs, so variations in the rate of pressurizer discharge would also be expected to have a minimal effect on PCT.

For the Large Break Loss of Coolant Accident (LBLOCA), the effect of this issue on PCT was determined on a plant-specific basis (see Note 7).

2. Removed Upper Internal Assembly Alignment Pins

This penalty addresses the removal of upper internal alignment pins at the Byron Station. Two pins have been removed from Byron Station Unit 1 and six pins have been removed from Byron Station Unit 2. Removal of the alignment pins resulted in a LBLOCA PCT penalty of +5.0°F for Byron Station Unit 1. Byron Station Unit 2 previously accounted for the cut pins by penalizing Heat Flux Hot Channel Factor (FQ). Starting with Byron Station Unit 2 Cycle 6 a LBLOCA PCT penalty of +28.0°F was assessed instead of the FQ penalty. This will establish consistent treatment of the cut alignment pins for both units at Byron Station.

3. Burst and Blockage/Time in Life

Typically the SBLOCA analysis was performed using Beginning of Life (BOL) fuel performance data (i.e., PAD) and evaluated at other burnups using the SPIKE computer code. Presently this is explicitly modeled using a "time in life study." The burst and blockage model does not have any effect on the PCT if the PCT is less than 1700°F.

For Byron Station Unit 2 and Braidwood Station Unit 2 analysis the burst and blockage penalty is 19 °F and this is based on direct burnup studies. For Byron Station Unit 1 and Braidwood Station Unit 1 analysis there is no burst and blockage penalty because the PCT is less than 1700°F (see Note 14).

4. Assembly Guide Pin Flakes

Bending of fuel assembly alignment pins to angles greater than 5 degrees may result in the generation of pin flakes or fragments. The flakes could potentially lodge themselves in an assembly and locally reduce assembly flow. The flakes could increase blockage of the hot rod subchannel during the reflood period and increase the PCT. This penalty of 6 °F is only applicable to Byron Station Unit 1.

5. LOCBART Spacer Grid Single-Phase Heat Transfer Error

As discussed in WCAP-10484-P-A, "Spacer Grid Heat Transfer Effects During Reflood," March 1991, the Yao-Hochreiter-Leech correlation is used in the LOCBART computer code to calculate the single-phase heat transfer enhancement for axial elevations located downstream of spacer grids. The Safety Evaluation Report to WCAP-10484-P-A requires that a length-averaged value be used to specify the heat transfer coefficient for a given fluid cell, since use of a local value corresponding to the forward edge or the rear edge of the cell could be non-conservative. It was determined that the length-averaging in LOCBART was not being done correctly in all cases.

The effect of this error on existing results was determined on a plant-specific basis (see Note 7).

6. LOCBART Zirc-Water Oxidation Error

Westinghouse identified a logic error in the LOCBART computer code that caused the Baker-Just metal-water reaction calculations to be performed three times per timestep. Correcting the error was found to reduce the total cladding oxidation while increasing the heat deposition in the cladding.

The effect of this error on existing results was determined on a plant-specific basis (see Note 7).

7. Limiting LBLOCA Analysis of Record (AOR) Case Reanalyzed

The limiting LBLOCA AOR case was reanalyzed. The reanalyzed case incorporated Note 1, Accumulator Line/Pressurizer Surge Line Data, Note 5, LOCBART Spacer Grid Single-Phase Heat Transfer Error, Note 6, LOCBART Zirc-Water Oxidation Error, and Note 9, RCS Crossover Leg Volume Error. The reanalyzed case reduced the total peaking factor (Fq) value assumed in the AOR from 2.6 to 2.5.

Incorporation of the above changes resulted in a reduction in the PCT by 32°F. Additionally, the 3°F PCT penalty due to the RCS crossover leg volume error was removed since the error was correctly modeled in the reanalyzed case.

8. Translation of Fluid Conditions from SATAN to LOCTA

An error was discovered in the coding related to the translation of fluid conditions between the SATAN blowdown hydraulics computer code and the LOCTA computer code used for subchannel analysis of the fuel rods. In performing axial interpolations to translate the SATAN fluid conditions onto the mesh nodalization used by the LOCTA computer code, the length of the lower core channel fluid connection to the lower plenum node was incorrectly calculated. Calculations with the corrected model resulted in an increase of 15°F in the PCT for the LBLOCA. This penalty applies to both the Byron and Braidwood Station LBLOCA analyses.

Sensitivity studies to rebaseline the Analysis of Record (AOR) using updated computer code versions/methodology and minor input changes, described in Note 12, resulted in a PCT of 2062°F, a 94°F increase relative to AOR. These results incorporate the SATAN/LOCTA error included in the letter from R. M. Krich (ComEd) to U.S. NRC, "Annual 10 CFR 50.46 Report," dated April 23, 1998. Therefore, this 15°F penalty is removed in this report.

9. Reactor Coolant System (RCS) Crossover Leg Volume Error

IMP is an electronic database containing a variety of plant geometry data whose primary purpose is to interactively support the LOCA Evaluation Model input preprocessors. Secondary purposes are to provide a convenient repository and ready reference for Nuclear Steam Supply System (NSSS) geometry information for a variety of functional groups that generally utilize the database simply for limited hand data extractions. Westinghouse discovered an error in a recent edition of the Byron/Braidwood IMP Database associated with the crossover leg volume. The IMP Database has been subsequently updated.

The impact of the error is a 3°F increase in PCT in the LBLOCA (see Note 7). The SBLOCA AOR bounds the increased crossover leg volume configuration and there is no adverse impact to SBLOCA PCT. This item applies to the Byron and Braidwood Stations.

10. Replacement Steam Generator (RSG)

Westinghouse performed an evaluation to demonstrate the applicability of the current LBLOCA AOR to the Babcock & Wilcox International (BWI) RSGs. The evaluation consisted of evaluating the differences between the BWI and Westinghouse designed steam generators and the impact on the PCT. The evaluation resulted in a 21°F PCT penalty for the BWI RSGs. The RSG supports a steam generator plugging level of up to 20%. This is applicable to Byron Station Unit 1 and Braidwood Station Unit 1.

11. Passive Heat Sink /Reactor Containment Fan Cooler (RCFC) Performance Data/Initial Containment Pressure Assumption

In the LBLOCA analysis it is conservative to assume data which minimizes the containment pressure during the transient, such as the assumption of the total amount of passive heat sinks, the performance of RCFCs, the assumption of initial containment pressure, containment spray flows, etc.

a) Passive Heat Sink

The amount of passive heat sinks assumed in the AOR LOCA analysis is documented in the Updated Final Safety Analysis Report (UFSAR) Table 6.2-55. Subsequent to the AOR, numerous modifications were done inside the containment. To perform the modifications materials such as steel were installed inside the containment. Increased steel results in a decrease in the containment pressure during the LOCA transient and is potentially non-conservative.

Anytime modifications such as replacing the steam generators, addition of Safety Injection (SI) Tank Access Galleries, etc. are done inside the containment, evaluations are performed to determine the impact of these modifications on the LOCA analysis. Westinghouse representatives performed an evaluation to determine the impact on the PCT due to the addition of various passive heat sinks. The Westinghouse evaluation addressed the addition of increased steel due to RSGs, SI Tank Access Gallery steel, and an assumed miscellaneous containment metal mass corresponding to 20,000 sq. ft. at 0.2083 ft. thickness for future modifications. This evaluation determined a PCT penalty of 16°F for these additional passive heat sinks.

b) RCFC Performance Data

The RCFC Performance data assumed in the AOR was determined to be incorrect. Revised corrected data was provided to Westinghouse to evaluate the impact on the LOCA analysis. The Westinghouse evaluation determined a PCT penalty of 1°F for the corrected data.

c) Initial Containment Pressure Assumption

The AOR assumed a conservative initial containment pressure of -1.0 psig. To partially offset the penalty due to the above items (a) and (b), the assumption of initial containment pressure was revised. The revised assumption of initial containment pressure is -0.5 psig. This revised assumption for the initial containment pressure resulted in a PCT benefit of 5 °F.

12. PCT Assessment for the LBLOCA for the Byron and Braidwood Stations Due to Burst Location Change

The LBLOCA AOR for the Byron and Braidwood Stations is presented in Chapter 15 of the UFSAR, Revision 7, and has a limiting case PCT of 1968°F. Westinghouse performed a series of sensitivity studies to rebaseline the AOR. One of the sensitivity cases using minor input changes indicated a PCT increase from 1968°F to 2062°F, a 94°F increase. Since the minor input changes and computer code version/methodology changes did not appear significant enough to account for the 94°F change, a non-conformance report was opened, and an investigation was undertaken to determine the cause of the PCT increase.

It was determined that the PCT increase was largely attributed to the change in the predicted location of the hot rod burst. Nodes just downstream of fuel assembly grids are typically non-limiting due to the enhanced cooling afforded by the grid. In the AOR, hot rod burst occurred just downstream of a spacer grid, hence the cladding temperature at the burst elevation remained relatively low. In the sensitivity case, the thermal-hydraulic results, though not significantly changed, were sufficiently different to move the location of the rod burst by one node (i.e., 3 inches), and the cladding temperature at the burst node was not influenced by the presence of a grid.

Investigation subsequently performed has concluded that there is no valid reason to discredit the 3-inch shift in the hot rod burst. Therefore, the plant-specific 94°F analytical PCT penalty will be assigned to the Byron Station and Braidwood Station cumulative PCT at this time.

This is not considered to be an error in the Evaluation Model nor an error in the application of the Model, but merely a consequence of the discretization of the thermal-hydraulic process that is fundamental to the Model. Westinghouse continues to consider the ramifications of the burst location behavior relative to the ability to obtain stable results and may introduce discretionary changes to the Model in the future.

Application of the 94°F analytical PCT penalty allows removal of the 15°F PCT penalty for SATAN/LOCTA Translation that was introduced in the 1997 10 CFR 50.46 Annual Report, submitted to the NRC by ComEd on April 23, 1997, because the computer code versions utilized in the sensitivity study incorporate the associated model corrections.

13. LOCBART Input Fuel Rod Outside Diameter (FOD) Error

An input calculational error in the Large Break AOR for the Byron and Braidwood Stations was discovered in the LOCBART computer code input FOD corresponding to the fuel rod outside diameter. LOCBART is the rod heatup computer code of the BASH Evaluation Model. This constitutes an error in the application of the Evaluation Model as defined in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors." A plant specific LOCBART sensitivity study was made which corrected the input FOD to its correct value, resulting in a 2°F PCT penalty.

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

The SPIKE computer code and the associated methodology are used to estimate fuel rod burst PCT penalties for SBLOCA analyses. The SPIKE code has been revised to reflect more recent data that was generated using the SBLOCA Evaluation Model and methodology. The SPIKE computer code was updated and validated to reflect the new database information.

Small Break LOCA analyses which include burst and blockage effects based on direct burnup studies are not impacted by the revision to SPIKE. Since the Byron Unit 2 and Braidwood Unit 2 SBLOCA analysis includes burst and blockage effects based on direct burnup studies, no SPIKE case was run and there is no impact on PCT.

Burst and blockage effects do not adversely impact SBLOCA analyses with PCTs less than 1700 °F. Since the Byron Unit 1 and Braidwood Unit 1 PCT is less than 1700 °F, there is no burst and blockage PCT penalty.

15. RCFC Start Time, Increase in Containment Spray Flow Rate, Storage of Lead Shielding Blankets and Scaffolding Materials Inside Containment, and Steam Generator (SG) Blowdown Valve Leakage

In the LBLOCA AOR the earliest start time of the RCFC was assumed to be 25 seconds. An evaluation was performed by Westinghouse to determine the impact of decreasing the start time from 25 to 15 seconds. The result was an increase in the PCT of 2 °F.

In the LBLOCA AOR the containment spray flow rate was assumed to be 8900 gpm. An evaluation was performed by Westinghouse to determine the impact of increasing the containment spray flow rate from 8900 to 9255 gpm. The result was an increase in the PCT of 1 °F.

The LBLOCA AOR does not assume any lead shielding blankets or scaffolding materials are stored inside containment. An evaluation was performed by Westinghouse to determine the impact of storing lead shielding blankets and scaffolding materials inside containment. The result was an increase in the PCT of 2 °F.

The LBLOCA AOR does not assume any SG blowdown valve leakage. An evaluation was performed by Westinghouse to determine the impact of 10 gpm/SG blowdown valve leakage. The result was that there is no impact on the PCT.

The SBLOCA AOR is not affected.

Attachment 3

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

Assessment Notes Not Included in Rack-ups

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 rack-up sheets. These assessments, in all cases, resulted in benefits or zero penalty to the calculated Peak Cladding Temperature (PCT). However, we have conservatively chosen not to credit these PCT benefits, i.e., for each change a delta PCT of zero degrees Fahrenheit 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.

Emergency Diesel Generator (EDG) Underfrequency Evaluation

The safety analyses are performed assuming the EDG operates at the steady state frequency. Recent EDG loading sequence tests and modeling indicate that the underfrequency reduction can be as much as 2 Hz (i.e., from 55 to 57 Hz) for a period of 4 seconds. The impact of the frequency swing during EDG loading sequences was evaluated and was determined to have an insignificant impact on the PCT.

EDG Frequency Evaluation

Currently, all safety analyses are performed assuming the EDG operates at the steady state frequency. However, the Technical Specifications allow EDG frequency to be within $\pm$ 1.2 Hz of the steady state frequency of 60 Hz. In the letter from R. M. Krich (ComEd) to U.S. NRC, "Annual 10 CFR 50.46 Report," dated April 23, 1998, it was stated that Westinghouse would perform a formal assessment of any PCT impact associated with this EDG frequency band. After further review, evaluation, and discussion, it was decided that the current assumption of EDG operation at steady state frequency is appropriate and that it is not necessary for Westinghouse to perform a formal PCT assessment. As stated in the 1998 Annual 10 CFR 50.46," Acceptance criteria for emergency core cooling systems
for light-water nuclear power reactors," Report, dated April 23, 1998, the impact on PCT has been determined to be small. There is sufficient inherent conservatism in the Westinghouse LOCA Emergency Core Cooling System (ECCS) Evaluation Models to bound uncertainty associated with EDG frequency uncertainty, and therefore the overall change to PCT is expected to be zero. Therefore, the utilization of steady state EDG frequency in the LOCA analysis is judged to be appropriate.

Fuel Rod Design and 10 CFR 50.46 Acceptance Criteria

Westinghouse representatives recently informed ComEd representatives that fuel rod corrosion and its associated feedback effect on rod internal pressure and the pressure stress limit have led to a potential violation of fuel rod design criteria and 10 CFR 50.46 acceptance criteria. Violation of the "no gap reopening" fuel rod criterion does not automatically result in a 10 CFR 50.46 acceptance criterion violation. This issue was first addressed in Westinghouse letter NSD-NRC-97-97-5404, dated October 28, 1997. This letter concluded that a substantial safety hazard as defined in 10 CFR 21, Reporting of Defects and Noncompliance, does not exist, and that the same levels of safety as considered in the design basis evaluations were maintained

Plant specific evaluations are performed every reload cycle using Westinghouse methodology to ensure that the reload design does not violate the 17% total localized corrosion criterion of 10 CFR 50.46. Specifically, evaluations have been performed for Braidwood Station Unit 1 Cycle 9, Braidwood Station Unit 2 Cycle 8, Byron Station Unit 1 Cycle 10, and Byron Station Unit 2 Cycle 9. These evaluations have shown that the 17% total localized corrosion criterion of 10 CFR 50.46 is not violated.

LBLOCA Power Distribution

Appendix K to 10 CFR 50, "ECCS Evaluation Models," requires that the power distribution, which results in the most severe calculated consequences, be used in the ECCS Evaluation Model calculations. The current basis for all Westinghouse LBLOCA evaluations is the chopped cosine power distribution. Calculations were performed with BASH, which examined peak power locations and power distributions that were not considered in the original analysis. Under some circumstances, these evaluations lead to PCTs greater than those calculated with the cosine distribution. Previously, the Byron and Braidwood Stations included a conservative temporary PCT penalty of 100°F to bound the effects of other power shapes.

To address the power shape issue, Westinghouse has developed an alternate axial power shape methodology, ESHAPE (i.e., Explicit Shape Analysis for PCT Effects). The ESHAPE methodology is based on explicit analysis of a set of skewed axial power shapes. The NRC as part of the Westinghouse LBLOCA Evaluation Model has previously approved the explicit use of skewed power shapes. Westinghouse has performed evaluations for the Byron and Braidwood Stations using ESHAPE and has determined that the cosine power shape used in the Analysis of Record remains limiting. Therefore, the PCT penalty of 100°F was removed.

LUCIFER2 Downcomer Azimuthal Flow Path Calculations

The LUCIFER2 computer code generates component databases that are used by the SATIMP, BASHER, and SPADES input processors to develop plant-specific input models for the LBLOCA and SBLOCA analyses. An error was discovered in LUCIFER2 whereby a reactor vessel diameter below the hot/cold leg elevation was used for calculations that apply above the hot/cold leg elevation, resulting in incorrect values for various downcomer azimuthal flow path parameters.

For the SBLOCA analysis, this error has no impact on the calculated PCT since the downcomer azimuthal flow paths defined in LUCIFER2 are not used.

For the LBLOCA analysis, this error only affects the SATAN6 computer code, since the downcomer azimuthal flow paths defined in LUCIFER2 are not used in BASH computer code. Westinghouse calculations using the SATAN6 computer code showed that this error correction has no impact on PCT.

BASH Vapor Film Flow Regime Heat Transfer Error

As discussed in WCAP-9561-P-A, "BART-A1: A Computer Code for the Best Estimate Analysis of Reflood Transients," the Berenson model for film boiling is used in the BASH computer code to calculate the cladding-to-fluid heat transfer coefficient for conduction across the vapor film in the vapor film flow regime. An error was discovered in the

BASH computer code, resulting in an underprediction of the cladding-to-fluid heat transfer coefficient (i.e., a lower heat transfer coefficient).

Westinghouse calculations using the BASH computer code showed that this error correction had a negligible effect on the core inlet flooding rate during reflood and as such has no impact on PCT.

BASH Broken Loop Accumulator Empty Time Logic Error

An error was discovered in the BASH computer code that resulted in immediate emptying of the faulted accumulator upon entry into the reflood phase of the transient.

For cases where the faulted accumulator empties prior to entry into the reflood phase of the transient, this change has no effect on PCT. For cases where the faulted accumulator empties during the reflood phase of the transient, this error would have a negligible effect on the containment pressure during the reflood phase and therefore would have a negligible effect on the core inlet flooding rate. There is no impact on PCT as a result of this error.

BASH Pumped Injection Spill Logic Error

An error was discovered in the BASH computer code that resulted in an underprediction of the spilling flow (i.e., a lower spilling flow) to containment for dry containment plants with accumulator/Safety Injection (SI) interaction that use interactive COCO computer code.

Westinghouse calculations using the BASH computer code showed that this error would have a negligible effect on the containment pressure during the reflood phase and therefore would have a negligible effect on the core inlet flooding rate. There is no impact on PCT as a result of this error.

LOCBART Pellet Diameter Adjustment Error

To account for small differences in pellet average temperatures between the LOCA models and the fuel rod design models (i.e., PAD), part of the initialization process for the LOCTA-IV program, (i.e., WCAP-8301, "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," F.M. Bordelon et. al., June 1974) fuel rod model involves making small adjustments to the fuel pellet diameter such that the pellet average temperature at steady-state full-power operation matches the data from the PAD code. To compensate for this adjustment to the pellet diameter, a factor is applied to the core power such that the ratio of core power to uranium dioxide (UO_2) mass remains constant. During review of the LOCBART code, which is based upon the LOCTA-IV model, it was discovered that a second compensating adjustment also existed, whereby a similar factor was being applied to the pellet density to achieve the same purpose. In order to avoid double-counting for the pellet diameter adjustment effect, the code was corrected to use only the power adjustment in accordance with WCAP-8301. There is no impact on PCT as a result of this error.

SPADES Truncation Error

Various methods exist for entering input data into the SPADES computer code, which is used to generate the plant-specific input models for NOTRUMP. An error was discovered in the SPADES computer code whereby different methods of entering the input data could lead to minor differences in the resulting NOTRUMP input values, due to differences in the truncation methods. There is no impact on PCT as a result of this error.

NOTRUMP Array Boundary Error

An error was discovered that could potentially affect the data stored within arrays of the NOTRUMP executable computer code. Areas of the NOTRUMP computer code and the user externals were coded such that references to data locations beyond defined array boundaries could possibly have been utilized. To correct this problem, array range checking is now being enabled during execution via the use of specific compiler options. With these compiler options activated, attempts to use data outside of defined array boundaries result in code termination with the offending source code line identified to the user. To activate these compiler options, the dimensions on several dummy argument one-dimensional arrays were changed to utilize appropriate coding conventions. NOTRUMP models that were considered to encompass the range of array storage requirements were chosen and executed with both the erroneous and corrected code versions. From the results, there was no impact to PCT.

NOTRUMP Volumetric/Mass Based Consistency Error

NOTRUMP contains user input options for either mass or volumetric flow in the momentum conservation equations. The latter is used in NOTRUMP for the AP600 Evaluation Model (EM) due to the low pressures experienced in the AP600 SBLOCA transient. When evaluating the use of certain AP600 model features for potential use in the standard Appendix K Evaluation Model, it was discovered that undesirable numerical oscillations were occurring when flow direction changes were predicted in certain flow links, causing the code to abort. The cause of the problem was determined to be an inconsistent method of updating certain mass and volumetric rate variables during portions of the SBLOCA transient. When reviewing the details of this error, it was discovered that other code locations were also affected by this error, which meant that both the standard NOTRUMP EM and the AP600 EM were affected. To correct the problem, several subroutines were modified to correctly update volumetric and mass-based flow calculations on a consistent basis.

Westinghouse PWR plant calculations show that the nature of these changes leads to an estimated PCT impact of 0 °F.

LOCBART Transient Termination

Recent analyses using the BASH computer code predicted downcomer boiling to occur before the cladding temperature and/or oxidation transients have been conclusively terminated. A method has been developed to extend the transient beyond the onset of downcomer boiling by correlating the boiling-induced reduction in downcomer driving head to a corresponding reduction in the core inlet flooding rate.

Westinghouse experience indicates that the PCT will occur prior to the onset of downcomer boiling for the majority of anticipated applications. If the peak cladding temperature were to occur following the onset of downcomer boiling, the cladding temperature excursion will already have leveled off sufficiently that any increase in PCT that occurs following the onset of downcomer boiling would be expected to be insignificant. As such, this method of extending the transient beyond the onset of downcomer boiling will be implemented on a forward-fit basis.

NOTRUMP Inconel-690 Tube Properties

With the introduction of Alloy 690 tube material in the replacement Steam Generators (SGs), NOTRUMP tube material properties were updated to reflect the small differences between Alloy 600 (i.e., the material in the original SGs) and Alloy 690. The differences in material properties between Alloy 600 and Alloy 690 are expected to have a negligible effect on PCT, so this change will be implemented on a forward-fit basis.

Improved Code Input/Output (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 calculational results. In addition, various blocks of coding were rewritten to eliminate inactive coding, optimize the active coding, and improve code commenting, both for enhanced usability and to facilitate code debugging when necessary. The nature of these has no impact on PCT.