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ComEd

July 10, 1998

JMHLTR: 98-0199

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Subject: Dresden Nuclear Power Station Units 2 and 3
Plant Specific ECCS Evaluation Changes - 10CFR50.46 Annual Report
DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Reference:

1) "Dresden Nuclear Power Station Units 2 and 3, Plant Specific ECCS Evaluation Changes - 10CFR50.46 Report DPR-19 and DPR-25, NRC Docket Nos. 50-237 and 50-249," letter to USNRC from J. Stephen Perry (ComEd), JSPLTR #97-0131, July 10, 1997.

2) "Dresden Nuclear Power Station Units 2 and 3 Evaluation of Methods To Address ECCS Flow and Pressure Measurement Uncertainties, NRC Docket Nos. 50-237 and 50-249," letter to USNRC from J. Stephen Perry (ComEd), JSPLTR #97-0059, March 21, 1997.

This letter fulfills the annual reporting requirement of 10CFR50.46(a)(3) for Unit 2 and Unit 3 of Dresden Nuclear Power Station. Reference 1 provided the Nuclear Regulatory Commission last year's PCT data for Dresden Station. Attachments 1 and 2 provide Peak Cladding Temperature (PCT) information for the limiting Loss of Coolant Accident evaluations for Dresden Station, including all assessments as of July 1, 1998. The assessment notes (Attachment 3) provide a detailed description for each change or error reported.

In March 1997, Dresden Station committed to perform an evaluation of methods to address Emergency Core Cooling System (ECCS) flow and pressure measurement uncertainties (Reference 2). The PCTs reported for both Unit 2 and Unit 3 include the results of the instrument loop inaccuracies. The ECCS measurement uncertainties were treated as a degradation to the overall ECCS response modeled in the 10 CFR Appendix K LOCA analyses. Inclusion of these ECCS measurement uncertainties in the LOCA analyses fulfills the requirements of the Reference 2 commitment.

ADD'l.

If there are any questions or comments concerning this letter, please contact Frank Spangenberg at (815) 942-2920, extension 3800.

Respectfully,

Preston Swafford for

John M. Heffley
Dresden Site Vice President

Attachment 1: Dresden Unit 2 10 CFR 50.46 Report
Attachment 2: Dresden Unit 3 10 CFR 50.46 Report
Attachment 3: Dresden Unit 2 and Unit 3 PCT Assessment Notes

cc: Regional Administrator - RIII
L. W. Rossbach, Project Manager - NRR
Senior Resident Inspector - Dresden
Office of Nuclear Facility Safety - IDNS

Attachment 1

Dresden Unit 2 10CFR 50.46 Report (Continued)

PLANT NAME: Dresden Unit 2
ECCS EVALUATION MODEL: EXEM BWR
REPORT REVISION DATE: 7/10/98
CURRENT OPERATING CYCLE: 16

ANALYSIS OF RECORD

Evaluation Model: Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, ANF-91-048(P)(A), dated January 1993.

Calculations:

1. "Dresden LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM-9B and 9x9-2 Fuel – Reduced LPCS Runout Flow," EMF-98-007(P), Supplement 2, Siemens Power Corporation, dated January 1998 (see Note 1).

2. "LOCA Break Spectrum Analysis for Dresden Units 2 and 3," EMF-97-025(P), Revision 1, Siemens Power Corporation, dated May 1997.

Fuel: 9x9-2, ATRIUM-9B LFA and ATRIUM-9B
Limiting Fuel Type: 9x9-2
Limiting Single Failure: LPCI Injection Valve
Limiting Break Size and Location: 1.0 Double-Ended Guillotine (DEG) in a Recirculation
Suction Pipe

Reference PCT (see Note 2) PCT = 2018°F

MARGIN ALLOCATION

A. PRIOR LOCA MODEL ASSESSMENTS

None (see Note 3)

B. CURRENT LOCA MODEL ASSESSMENTS

Change in Unit 2 core spray leakage (See Note 4)	ΔPCT = 0°F
Unit 2 jet pump riser leakage (see Note 5)	ΔPCT = 0°F
HPCI upper bound delivery pressure (see Note 6)	ΔPCT = 0°F
LPCI min flow valve closing time (see Note 7)	ΔPCT = 0°F
ADS valve interlock pressure permissive (see Note 8)	ΔPCT = 0°F
Cycle 16 reload fuel (see Note 12)	ΔPCT = 0°F
Error in EXEM BWR evaluation model: HUXY strain model	ΔPCT = 7°F

Attachment 1

Dresden Unit 2 10CFR 50.46 Report (Continued)

modifications (see Note 9)	
Error in EXEM BWR evaluation model: Inappropriate capture ratio for high exposed fuel (see Note 10)	ΔPCT = 0°F
Error in EXEM BWR evaluation model: Accommodation density (SWMDEN) correction (see Note 11)	ΔPCT = -4°F
Error in EXEM BWR Evaluation Model: Fuel Grain Size (see Note 13)	ΔPCT = 0°F
Total PCT Change from Current Assessments	∑ΔPCT = 3°F
Cumulative PCT Change from Current Assessments	∑ |ΔPCT | = 11°F

NET PCT **PCT = 2021°F**

Attachment 2

Dresden Unit 3 10CFR 50.46 Report

PLANT NAME: Dresden Unit 3
ECCS EVALUATION MODEL: EXEM BWR
REPORT REVISION DATE: 7/10/98
CURRENT OPERATING CYCLE: 15

ANALYSIS OF RECORD

Evaluation Model: Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, ANF-91-048(P)(A), dated January, 1993.

Calculations:

1. "Dresden LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM-9B and 9x9-2 Fuel," EMF-98-007(P), Siemens Power Corporation, dated January 1998 (see Note 1).

2. "LOCA Break Spectrum Analysis for Dresden Units 2 and 3," EMF-97-025(P), Revision 1, Siemens Power Corporation, dated May 1997.

Fuel: 9x9-2 and ATRIUM-9B
Limiting Fuel Type: 9x9-2
Limiting Single Failure: LPCI Injection Valve
Limiting Break Size and Location: 1.0 Double-Ended Guillotine (DEG) in a Recirculation Suction Pipe

Reference PCT (see Note 2) PCT = 1920°F

MARGIN ALLOCATION

A. PRIOR LOCA MODEL ASSESSMENTS

None (Note 3)

B. CURRENT LOCA MODEL ASSESSMENTS

HPCI upper bound delivery pressure (see Note 6)	ΔPCT = 0°F
LPCI min flow valve closing time (see Note 7)	ΔPCT = 0°F
ADS valve interlock pressure permissive (see Note 8)	ΔPCT = 0°F
Error in EXEM BWR evaluation model: HUXY strain model modifications (see Note 9)	ΔPCT = 16°F
Error in EXEM BWR evaluation model: Inappropriate capture ratio for high exposed fuel (see Note 10)	ΔPCT = 0°F
Error in EXEM BWR evaluation model: Accommodation density (SWMDEN) correction (see Note 11)	ΔPCT = 0°F
Error in EXEM BWR Evaluation Model: Fuel Grain Size (see Note 13)	ΔPCT = 0°F
Total PCT Change from Current Assessments	∑ΔPCT = 16°F
Cumulative PCT Change from Current Assessments	∑ |ΔPCT| = 16°F

NET PCT **PCT = 1936°F**

Attachment 3

Additional Information for

Dresden Units 2 and 3 50.46 Report

1. Analysis of Record

The previous 50.46 reporting noted that the analyses of record were EMF-97-031(P), Revision 1 and EMF-97-031(P) respectively for Units 2 and 3. These reports are reissued as EMF-98-007(P), Supplement 2 and EMF-98-007(P) respectively.

2. Reporting of Different Peak Cladding Temperatures for Each Unit

Dresden Unit 2 and Unit 3 are being maintained under separate analyses of record (EMF-98-007(P), Supplement 2 and EMF-98-007(P) respectively) as a result of a degraded Core Spray runout flow condition that exists at Dresden Unit 2. This flow condition is lower with respect to the LOCA analysis assumption for Dresden Unit 3. The following table lists the Core Spray runout flows assumed for both Units 2 and 3 in the analysis of record for each unit.

Dresden Units 2 & 3 Core Spray Runout Flow

Current Unit 2 CS Runout Flow (GPM)	Current Unit 3 CS Runout Flow (GPM)	Currently Unit 2 Analyzed CS Runout Flow (GPM)	Currently Unit 3 Analyzed CS Runout Flow (GPM)
5400(1)	5650(2)	5300(1)	5650(2)

(1) Core Spray runout flow tests at Dresden Unit 2 show that at least 5400 gpm of runout flow per loop would be available from the Core Spray system. Based on this information the Core Spray flow was conservatively modeled as being 5300 gpm per loop in the analysis of record for Unit 2 (EMF-98-007(P), Supplement 2)

(2) Core Spray runout flow tests at Dresden Unit 3 show that at least 5650 gpm of runout flow per loop would be available from the Core Spray system. Based on this information the Core Spray flow was modeled as being 5650 gpm per loop in the analysis of record for Unit 3 (EMF-98-007(P))

3. Prior LOCA Model Assessment

The prior LOCA model assessment submitted by the referenced letter was a new baseline analysis for Dresden Units 2 and 3. Therefore, there is no PCT change.

(Reference: J. S. Perry (ComEd) letter JSPLTR#97-0131 dated July 10, 1997, "Dresden Nuclear Power Station Units 2 and 3, Plant Specific ECCS Evaluation Changes - 10CFR50.46 Report DPR-19 and DPR-25, NRC Docket Nos. 50-237 and 50-249.")

4. Change in Unit 2 Core Spray Leakage

During the Dresden Unit 2 R15 Outage In-Vessel Visual Inspection (IVVI), the core spray piping weld crack leakage was recalculated to be 14 gpm which is increased from 2 gpm. Even with this change, the total calculated leakage is still bounded by the analysis value. The following table lists the leakages assumed for both Units 2 and 3 in the analysis of record for each unit.

Dresden Units 2 & 3 Leakage Currently Calculated and Analyzed for Loss-of-Coolant Accidents

Source	Current Unit 2 Calculated Leakage (GPM)	Current Unit 3 Calculated Leakage (GPM)	Currently Analyzed Leakage Unit 2 (GPM)	Currently Analyzed Leakage Unit 3 (GPM)
RPV penetration assembly Design Leakage (2-Loop)	2 x 190 380 total	2 x 115 230 total	500(1)	500(1)
Upper T-box vent hole Leakage (2-Loop)	2 x 8 16 total	2 x 8 16 total	0(1)	0(1)
Core spray piping weld Cracks End of Cycle Leakage (2) (2-Loop)	14 14 Total	10+11 21 Total	0(1)	0(1)
Core spray piping weld Cracks with 48 months of postulated crack growth (2 Loop)	N/A	38+31 69	0(1)	0(1)
Core shroud weld cracks	184	0	184	184
Access hole cover	78	0	78	78
Bottom head drain line	295	295	295	295

(1) The 500 gpm of RPV assembly penetration leakage listed in the table is equivalent to 500 gpm of total leakage for the RPV assembly leakage, Upper T-box vent hole leakage, and the CS line postulated crack leakage. Since all of these leakages occur in the CS line between its entry into the vessel and the penetration of the core shroud, the distribution of these leakages is insignificant. Conservatively, none of the Core Spray leakage flow is credited to enter the vessel.

(2) The end-of-cycle crack lengths (including unit specific projected crack growth) were used to calculate the leakages.

5. Unit 2 Jet Pump Riser Leakage

During the Dresden Unit 2 R15 Outage In-Vessel Visual Inspection (IVVI), new indication of a small crack was observed at the jet pump riser. The resulting leakage of the new flaw was calculated to be 11 gpm after two two-year cycles. The jet pump is in the injection path of the low pressure core injection (LPCI) system. Leakage in the jet pump will affect the results which assumes a single failure of the diesel generator (SF-DG). For SF-DG, the low pressure Emergency Core Cooling System (ECCS) available are one Core Spray (CS) pump and two LPCI pumps. Even with the additional leakage, the SF-DG case will not become more limiting than the limiting single failure of the LPCI injection valve where two CS pumps are available to mitigate the consequences of the LOCA. However, the PCT margin between the two single failures is reduced.

The following table list the leakages assumed for both Units 2 and 3 in the analysis of record for each unit.

Dresden Units 2 & 3 Jet Pump Leakage Currently Calculated and Analyzed for Loss-of-Coolant Accidents

Source	Current Unit 2 Calculated Leakage (GPM)	Current Unit 3 Calculated Leakage (GPM)	Currently Analyzed Leakage Unit 2 (GPM)	Currently Analyzed Leakage Unit 3 (GPM)
Jet pump bolted joint (total for 10 jet pumps)	582	582	582	582
Jet pump slip joints (total for 20 jet pumps)	225	225	225	225
Jet pump riser crack	11	NA	11	0

6. HPCI Upper Bound Delivery Pressure

The upper bound delivery pressure was changed from 1150 psid to 1120 psid to be consistent with the UFSAR and Design Specifications. The HPCI system is not credited for large break analyses, therefore, there is no impact on large break results. For small breaks, the analyses were performed with and without HPCI. SPC determined that the change in the upper bound delivery pressure will not cause the PCT of the small break with HPCI case to approach the PCT of the small break without HPCI case or PCT of the limiting large break. Therefore, there is no change in the limiting PCT.

7. LPCI Minimum Flow Valve Closing Time

Dresden Station determined that the LPCI minimum flow valve of the loop that is not selected by the loop select logic will not close during a LOCA. The analysis of record was performed with a closing time of 35 sec. The consequence of the minimum flow valve failing to close is a reduction of LPCI flow into the vessel during a LOCA. The reduction in LPCI flow rate will affect the results which assumes a single failure of the diesel generator (SF-DG). For SF-DG, the low pressure Emergency Core Cooling System (ECCS) available are one CS pump and two LPCI pumps. Even with the reduction in LPCI flow, the SF-DG case will not become more limiting than the limiting single failure of the LPCI injection valve where two CS pumps are available to mitigate the consequences of the LOCA. However, the PCT margin between the two single failures is reduced.

8. Automatic Depressurization System (ADS) Valve Interlock Pressure Permissive

The low pressure interlock setpoint for ADS actuation was changed from 50 psig to ≥100 psig and ≤150 psig. The analysis value used in the analysis of record was based on the Standard Technical Specifications value of ≥50 psig and ≤100 psig. (Dresden Technical Specifications has been upgraded through the Technical Specifications Upgrade Program.) The setpoint value listed in the current Technical Specifications is ≥100 psig and ≤150 psig. SPC determined that the change in setpoint will not impact the LOCA analyses results. The change in setpoint will tend to improve the likelihood that ADS will function as needed.

9. Error in EXEM BWR Evaluation Model: HUXY Strain Model Modifications

In December of 1997, SPC reported an error on the application of the HUXY code. The HUXY code is used to perform heatup calculations during the entire LOCA and yields peak cladding temperature and local oxidation at the axial plane of interest. Calculations are made in HUXY to predict the rupture or failure temperature of the fuel cladding. One of the inputs to the cladding rupture temperature calculation is the internal rod pressure. Prior to rupture, the cladding experiences strain and the gas volume changes. A change in the internal dimensions of the cladding affects the internal gas pressure and consequently the predicted rupture temperature. It has recently been determined that the instantaneous strain is used in the calculation instead of the maximum strain. The instantaneous strain differs from the maximum strain in that it can fluctuate up and down based on the instantaneous stress. This fluctuation causes the rod internal pressure and the rod failure temperature to oscillate. This had the effect of prematurely predicting rod failures. SPC has determined that it is more appropriate to use the maximum strain for the rod failure temperature calculation. Using the maximum strain assumes the strain plastically deforms the cladding. Using the maximum strain is consistent with the strain calculations used in the calculation for the rod dimensions to determine the radiation heat transfer view factors. SPC has estimated the impact of the HUXY strain error to be 0°F because there is no fuel failure for the limiting case. For the non-limiting cases, the PCT would decrease.

SPC corrected their HUXY code and reran the limiting case for each Unit. Contrary to expectations, the PCT of the limiting case increased slightly even though there is no fuel failure.

10. Error in EXEM BWR Evaluation Model: Inappropriate Capture Ratio For High Exposed Fuel

During the Dresden Unit 2 Cycle 16 (D2C16) ATRIUM-9B heatup analysis, SPC discovered that the capture ratio used in the high exposure analyses is incorrect because it is based on beginning of life (BOL) exposure. For the high exposure analyses, a capture ratio based on BOL would result in lower cladding temperature. The analyses performed for D2C16 showed that the impact of using appropriate capture ratios for high exposure fuel was an increase in cladding temperature of less than 15 degrees F at 25-35 GWd/MTU and 35 degrees F at exposures greater than 35 GWd/MTU. The PCTs at these high exposures have more than 35 degrees margin to the limiting PCT. Therefore the limiting PCT does not change.

This issue is not applicable to the results for 9x9-2 fuel because the correct capture ratios were used in the analyses.

Attachment 3

Additional Information for

Dresden Units 2 and 3 50.46 Report (Continued)

11. Error in EXEM BWR Evaluation Model: Accommodation Density (SWMDEN) Correction

SWMDEN is an input to the RODEX2 computer code used by SPC to calculate fuel characteristics. It is defined as the asymptotic fuel density that occurs after full densification and full accommodation of the solid swelling by the as-fabricated fuel porosity. The value of SWMDEN is determined by the process used to fabricate the fuel. RODEX2 was benchmarked using a SWMDEN value of 0.9755, which was representative of fuel produced in the 1970s. SPC determined that a SWMDEN value of 0.995 is appropriate for the current fuel. However, LOCA analyses have been performed with a SWMDEN value of 0.9755. SPC reran the limiting case for each Unit using a SWMDEN value of 0.995.

12. Unit 2 Cycle 16 reload fuel

The LOCA analysis of record was performed for the fuel types in the Dresden Unit 2 Cycle 15 (D2C15) and Dresden Unit 2 Cycle 15 (D3C15) cores. The PCT for the new ATRIUM-9B fuel loaded into the D2C16 core is bounded by the PCT of the limiting fuel types in D2C15 and D3C15.

13. Error in EXEM BWR Evaluation Model: Fuel Grain Size

In October of 1997, SPC informed ComEd of an inconsistency between fuel grain size production and the grain size assumed in the mechanical analyses. The analysis of record used a pellet grain size of 16 microns; however, the most recent fuel productions have averaged values down to about 10 microns. SPC performed an evaluation for the impact on the LOCA analysis and concluded that the difference in fuel grain size will have no impact on the analysis of record. The fuel grain size can have a small affect on the internal rod pressure and stored energy at higher exposures. A smaller grain size would reduce the stored energy. However, since the limiting PCT occurs at very low exposures, there is no impact on the analysis of record.