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# Nuclear Fuel Services

COMMONWEALTH EDISON COMPANY TOPICAL

SUPPLEMENT 1

## BENCHMARK OF CASMO/MICROBURN BWR NUCLEAR DESIGN METHODS

## NEUTRONIC LICENSING ANALYSES


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## Abstract

This supplement to Commonwealth Edison Company Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods", describes the application procedures which Edison will use in the evaluation of cycle-specific neutronic licensing events. Comparisons to vendor results are included, demonstrating Edison can adequately analyze the various abnormal neutronic licensing events, including the calculation of the impact on critical power ratio for these events. The neutronic codes used in this supplement are those benchmarked by Commonwealth Edison in NFSR-0091. The application procedures are equivalent to those described in the Siemens Power Corporation proprietary documents XN-NF-80-19(P), Volume 1, Supplements 1, 2, and 3, which have previously been reviewed and approved by the NRC.

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# SECTION 1 - INTRODUCTION AND OVERVIEW

## 1.1 Introduction

This supplement to the Commonwealth Edison Company (Edison) Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods" (Reference 1), discusses the application procedures to be used by Edison with the CASMO/MICROBURN methodology in performing the neutronic licensing calculations listed in Table 1-1. The application procedures for the events listed in Table 1-1 are equivalent to the procedures described in the References 2 and 3 Siemens Nuclear Power Corporation (SNP - formerly called Advanced Nuclear Fuels (ANF)) documents XN-NF-80-19, Supplements 1, 2, and 3. These application procedures are also summarized in Section 3 of this supplement. Because the same neutronic methods are used together with application procedures which ensure compliance with the licensing bases, the same uncertainties are also applied. These uncertainties are described in References 4, 5, and 6.

Also included in this report are comparisons to SNP results. SNP is the fuel vendor for the two Edison BWRs at Dresden Station.

As discussed in the Reference 1 Topical Report, Edison is requesting NRC approval for the evaluation of the events listed in Table 1-1 using the CASMO-3G/MICROBURN-B methodology with the following clarifications. Edison's fuel vendors will continue to perform the plant transient and accident analyses other than the neutronic events listed in Table 1-1. In addition, due to the differences in the critical power correlation and associated uncertainties, Edison is not currently requesting NRC approval to use this methodology to perform those analyses which determine the critical power ratio impact for units not fueled by SNP (currently Quad Cities and LaSalle County Units 1 and 2). This encompasses Items 3 through 7 in Table 1-1.

## 1.2 Overview of NFSR-0091 - Supplement 1

A summary of the results and conclusions is contained in Section 2 of this supplement to Topical Report NFSR-0091. These results, which are detailed in Section 3, demonstrate that Commonwealth Edison can adequately perform the licensing calculations listed in Table 1-1 with the CASMO-3G/MICROBURN-B methodology.

Section 3 includes the following for each event listed in Table 1-1:

* A short description of the event;

* An outline of the application procedure used to analyze the event; and

* Comparisons to available fuel vendor results.

## 1.3 Scope of Analyses

As stated in the Reference 1 Topical Report, Edison will perform all neutronic analyses required for the licensing, operation, testing, and surveillance of a BWR reload cycle. The key neutronic licensing analyses which are reevaluated each cycle are listed in Table 1-1. The methodology and core conditions employed by Edison to perform these analyses with the CASMO-3G/MICROBURN-B code package are equivalent to those employed by SNP and outlined in References 2 and 3. Edison's capability to perform this scope of analyses is justified by the benchmark results summarized in Section 2.

## 1.4 Vendor Interactions

Edison has implemented Design Interaction Procedures with each of its fuel vendors to ensure compatibility or equivalency with their methodology for future reload analyses as well as to administer the logistical details of data and information transmittals between Edison and its vendors. Additionally, Edison holds Technical Review Meetings with each fuel vendor for every reload to address any generic issues related to the reload.

Extensive training in reload design activities has also been provided by SNP to Edison engineers at the SNP facilities in Richland, Washington. This training, which involved assignments of typically one year, included the performance of the full scope of neutronic calculations required for a reload cycle under the direct supervision and guidance of qualified SNP personnel using SNP standard procedures. These calculations included, but were not limited to, such activities as fuel assembly design, core loading pattern determination, control rod pattern development, shutdown margin determination, core monitoring code input generation, and performing neutronic licensing analyses such as standby liquid control system worth, fuel assembly loading error ΔCPR, rod withdrawal error ΔCPR, and rod drop accident enthalpy deposition. This training and design participation significantly enhanced Edison's expertise in these activities and provides additional supporting evidence that Edison can acceptably assume the full scope of reload design activities from the fuel vendors. In addition to this training, Edison has been generating the core loading patterns for the Dresden reactors which have been used for the fuel vendor's official analyses of record.

Table 1-1

Cycle Specific Neutronic Licensing Analyses

1. Shutdown Margin
2. Standby Liquid Control System
3. Fuel Loading Error - Misoriented Assembly
4. Fuel Loading Error - Misloaded Assembly
5. Control Rod Drop Accident
6. Control Rod Withdrawal Error
7. Loss of Feedwater Heating

## SECTION 2 - SUMMARY AND CONCLUSIONS

Comparisons to vendor results for the neutronic licensing activities listed previously in Table 1-1 are detailed in Section 3 of this report. These comparisons demonstrate that Commonwealth Edison can perform in an acceptable manner the analyses required for the Table 1-1 neutronic licensing events. The differences between the Edison and the SNP results are small in all cases and attributable primarily to small differences in the base, cycle-specific depletion cases. Edison generated the cycle depletions used as the basepoint for the neutronic licensing analyses, including the lattice physics input, as part of the benchmark of the CASMO-3G/MICROBURN-B code package contained in the Reference 1 Topical Report and as part of Edison's continuing core tracking calculations.

## SECTION 3 - NEUTRONIC LICENSING APPLICATION PROCEDURES

There are several neutronic events which are evaluated on a cycle-specific basis for each reload. This section describes these events and outlines the methods Commonwealth Edison Company will employ to analyze them. The CASMO-3G/MICROBURN-B code package, which was initially benchmarked by SNP and more recently benchmarked by Edison for its BWRs as reported in Reference 1, is used as the basis for the evaluation of these events. The application procedures used for these comparisons are equivalent to the standard methodology used by SNP, which is outlined in XN-NF-80-19 (References 2 and 3). The limiting conditions used in the analyses are consistent with the extended operating domain document previously NRC reviewed and approved for Dresden Station. The SNP extended operating domain document is contained in Reference 7.

A description of the method used to evaluate each of the cycle-specific neutronic events follows.

### 3.1 Shutdown Margin

The core must be capable of being made subcritical, throughout the operating cycle, by the margin specified in each plant's Technical Specification in the most reactive condition with the most reactive control rod fully withdrawn and all other control rods fully inserted.

The shutdown margin is determined by using the 3-D core simulator MICROBURN-B to calculate the core reactivity at selected exposure points with the strongest rod fully withdrawn. The core is conservatively assumed to be in the cold, xenon-free condition in order to ensure that the calculated values bound potential temperature and fission product poison conditions.

The most reactive condition at a particular exposure point in a cycle is when the moderator and fuel temperature are equal to 20°C, or ambient temperatures. Neutronic libraries generated at these conditions are input into the 3D core simulator to evaluate the shutdown margin. The shutdown margin is calculated at various exposures throughout the cycle to determine the minimum shutdown margin of the cycle.

The cold critical eigenvalue bias used to determine the shutdown margin is exposure-dependent, and is developed on a cycle-specific basis using historical data and known trends in cold critical eigenvalues. Specifically, the cold critical eigenvalues shown in NFSR-0091 (Reference 1) were used to develop the appropriate exposure-dependent eigenvalue bias.

The Technical Specification basis for ensuring adequate shutdown margin uses the parameter "R". The value of "R" is the difference between the calculated shutdown margin at the beginning of the operating cycle and the calculated value of shutdown margin any time later in the cycle where it would be less than at the beginning. Therefore, the parameter "R" will always be greater than or equal to zero. The parameter "R" will be equal to zero if the minimum shutdown margin occurs at BOC, and will be greater than zero if the minimum shutdown margin occurs later in the cycle.

Additionally, as required by Dresden and Quad Cities Technical Specifications, an adjustment to "R" will be made for those units which contain original equipment GE control rods to account for residual, potentially inverted tubes in the control rods.

A comparison to the vendor analysis for the calculation of "R" for Dresden Unit 3 Cycle 13 is shown in Table 3-1. The vendor result for this cycle is contained in Reference 8. The comparison in Table 3-1 demonstrates that Edison can adequately perform shutdown margin calculations, including the calculation of "R", as the maximum difference in the parameter "R" for the cycle is insignificant.

## 3.2 Standby Liquid Control System

The Standby Liquid Control System (SLCS) is designed to be capable of bringing the core, at any point in the operating cycle, from a full power and minimum control rod inventory, which is defined to be at the peak of the xenon transient, to a subcritical condition with the reactor in the cold, or 20°C, xenon-free state without reliance on control rod insertion. The Technical Specifications for each plant indicate the degree of subcriticality required to be demonstrated by these calculations.

To determine the degree of subcriticality after initiation of the SLCS, MICROBURN-B is used with borated CASMO-3G cross section data to add the negative reactivity effects of boron to cold 20°C all rods out conditions at various points in the cycle. The resulting eigenvalues determined at various points in the operating cycle are compared to the cold critical eigenvalues at the corresponding exposure points to determine the margin to criticality achieved with the standby liquid control system worth. The reported margin to criticality reflects the minimum difference.

A comparison to the vendor analysis for the calculation of margin to criticality after the initiation of the Standby Liquid Control System for Dresden Unit 3 Cycle 13 is shown in Table 3-2. The vendor result for this cycle is documented in Reference 8. Table 3-2 shows that the Edison result agrees very well with the SNP result, thereby demonstrating that Edison can adequately determine the margin to criticality after the initiation of the standby liquid control system.

## 3.3 Fuel Loading Error

There are two types of fuel loading errors - the misorientation of an assembly and the mislocation of an assembly. These are discussed in Sections 3.3.1 and 3.3.2, respectively.

A significant fuel loading error is a low probability event, as it is estimated to occur much less than once in a plant lifetime. Additionally, multiple errors are required to first misload, fail to detect during core verification, and then operate the plant without identifying and then correcting the loading error.

### 3.3.1 Misoriented Assembly

A fuel assembly is misoriented if it is loaded and operated in a position that is rotated 180 degrees from its proper orientation. The 180 degree rotation bounds the 90 degree rotation because a BWR lattice is designed symmetrically about the diagonal axis, and the narrow-narrow corner of the lattice has the highest enrichment fuel rods due to the lower neutron thermalization in this area. All other corners of a D-lattice have lower enrichment fuel rods since these areas have greater neutron thermalization in their nominal positions. Therefore, the limiting condition is when the fuel rods, which are expected to operate under the lowest thermalization conditions, actually experience the highest thermalization conditions. The effect of the 180 degree rotation has been demonstrated in Reference 3 to bound the 90 degree rotation, and, therefore, only the 180 degree misorientation is analyzed.

A fuel assembly misorientation is of minimal concern for C-lattice plants, such as LaSalle County Station, due to the uniformity of the water gaps around the assembly. Therefore, misorientation analyses are not performed for C-lattice plants since a misorientation of an assembly results in an insignificant change in local peaking or Critical Power Ratio and hence is not a limiting event.

However, D-lattice plants, such as Quad Cities and Dresden Stations, have non-uniform water gaps. An undetected and uncorrected misorientation of the fuel assembly may result in larger than anticipated local peaking on the wide-wide side of the fuel assembly, since the wide-wide side has the larger water gap, and hence greater neutron thermalization. This may lead to a degradation of MCPR margin.

Verification of the proper orientation of the fuel assemblies is one of the checks of Edison's BWR core loading procedures. All of the following items provide indication of proper orientation:

a) The channel fastener must be located at the corner of the assembly which is placed next to the center of the control blade.

b) The identification boss on the assembly handle points toward the adjacent control rod. In addition, the assembly serial number is engraved on the top of the handle in a standard orientation; specifically, it is readable looking from the center of the control cell.

c) The channel spacer buttons are adjacent to the control rod passage area.

d) There is cell-to-cell replication, meaning the above elements form a repeating pattern as a whole and the handles form a square in each cell.

The effect of a fuel assembly misorientation is measured in terms of its impact on steady state operating MCPR. The delta CPR for this event is determined by calculating the difference between the core minimum MCPR from the MICROBURN-B depletion of a correctly loaded core and a core containing a misoriented assembly. Target control rod patterns are used for the depletions.

Before a core containing a misoriented assembly can be depleted using MICROBURN-B, CASMO-3G lattice physics calculations are performed to neutronically characterize each misoriented assembly chosen for evaluation. These calculations model the misoriented assembly along with the three other correctly oriented assemblies in the control cell. The CASMO-3G calculations model a 180 degree rotation of the lattice including changes in the narrow-narrow and wide-wide water gaps associated with the rotation. When misoriented, an assembly has the proper wide-wide water gap dimension at its bottom relative to the correctly oriented configuration. However, the wide-wide water gap dimension at the top is reduced relative to the correctly oriented configuration because the channel spacer buttons at the top of the assembly lean against the core upper guide plate. This axial variation in the water gaps is modeled in the CASMO-3G calculations.

A comparison to the vendor analysis for the determination of the effect of a misoriented assembly for Dresden Unit 3 Cycle 13 is shown in Table 3-3. Results are shown for typical beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC) statepoints. The vendor results are documented in the vendor analyses for Dresden Unit 3 Cycle 13 (Reference 8). Table 3-3 demonstrates that Edison can adequately calculate the effects of a misoriented assembly.

This event is analyzed for each unique fresh assembly type which is loaded into a D-lattice reactor, such as Dresden, to confirm that the MCPR Safety Limit is not exceeded. Substantial margin to the limit is typically demonstrated.

### 3.3.2 Mislocated Assembly

Analysis of this event assumes that a fuel assembly is placed in an incorrect location while refueling and that this error is not discovered and corrected during the subsequent core verification process. The reactor is then assumed to operate during the cycle with the misloaded fuel assembly.

If a high reactivity assembly happens to be misloaded into an area of low reactivity, a decrease in the CPR of fuel assemblies in the immediate vicinity of the mislocation may occur. The CPR may decrease enough such that it may become the core minimum CPR (MCPR). The fuel assembly mislocation calculations are performed to quantify the MCPR difference ($\Delta$CPR) between operation of the correctly configured core and a core assumed to contain a fuel assembly in an incorrect location.

Scoping analyses are performed to identify potentially limiting fuel assembly mislocation positions. These analyses use 3D MICROBURN-B cross sections to generate localized power estimates for each postulated mislocated assembly in candidate core locations. After identifying potentially limiting fuel assembly mislocation positions, the MICROBURN-B 3D simulator code is used to deplete the cycle assuming the misloaded assembly is undetected and hence not corrected prior to operation. The resulting MCPRs are compared to those for the correctly configured core.

A comparison to the vendor analysis for the determination of the effect of a mislocated assembly for Dresden Unit 3 Cycle 13 is shown in Table 3-4. The vendor results are documented in the vendor analyses for Dresden Unit 3 Cycle 13 (Reference 8). Table 3-4 demonstrates that Edison can adequately calculate the effects of a mislocated assembly.

## 3.4 Control Rod Drop Accident

The Control Rod Drop Accident (CRDA) is one of the design basis events for a BWR, resulting in a rapid insertion of reactivity due to the drop of a control rod from the core at low power. This has two potential effects: the fuel pin enthalpy deposition will suddenly increase, and a pressure spike occurs from the reactivity insertion. Low, or zero, power is the limiting condition for the evaluation of a CRDA. At higher powers, void feedback will mitigate the impact of a CRDA.

The CRDA assumes the highest worth rod in the core becomes stuck in the fully inserted position, becomes decoupled from its drive, and subsequently drops to the location of the control rod drive, which is assumed to have been moved without its control rod to the location of its group's current bank position in the withdrawal sequence. The CRDA can result in a rapid reactivity excursion and resulting fuel failure; therefore, sequences have been developed to ensure the reactivity insertion resulting from a rod dropping from the fully inserted position to the position of the control rod drive is less than that which would lead to gross fuel failure. The established threshold for precluding rapid fuel dispersal in the coolant is 280 cal/gm peak fuel enthalpy. The CRDA is initially mitigated by negative Doppler reactivity resulting from the temperature increase of the fuel, and is terminated by scramming all but the dropped control rod. The analysis does not assume any void feedback to mitigate the effects of the accident.

The increase in reactor pressure as a result of a CRDA is minimal, less than 15 psid, and therefore is not evaluated on a cycle-specific basis as part of the control rod drop accident.

Edison will perform CRDA analysis using the methodology developed by SNP as described in Reference 3. The CRDA analysis is performed at the most conservative reactor conditions for the event. The reactor is modeled in hot standby conditions with no voids and no xenon. The maximum fuel rod enthalpy is determined based on the worth of the dropped rod, the bundle peaking factor for the four assemblies surrounding the dropped rod, the core delayed neutron fraction, and the core doppler coefficient. These parameters are calculated using CASMO-3G and MICROBURN-B.

The worth of a dropped rod is dependent on the distance the control rod drops and the current control rod pattern. The reactivity of the control rod that is assumed to drop in the maximum fuel rod enthalpy calculation is determined through a series of MICROBURN-B calculations which model the control rod drops that could occur during the use of the startup control rod sequence. The control rod drop that causes the largest change in reactivity is used in the subsequent CRDA analysis. Consistent with the SNP methodology described in Reference 9, the worth of this dropped rod is increased by a fixed amount to account for possible notch position errors.

A comparison to the vendor analysis for the determination of the effect of a control rod drop accident for Dresden Unit 3 Cycle 13 is shown in Table 3-5. Results are shown for the maximum fuel enthalpy deposition that occurs as a result of the accident. Also, results are shown for the four parameters used in the calculation. The vendor results are documented in the vendor analyses for Dresden Unit 3 Cycle 13 (Reference 8). Note that the Edison results reflect a more precise interpolation between the parametric curves presented in Figure 7.1-3 of Reference 3, and thus the Edison Maximum Deposited Fuel Enthalpy is slightly less than that reported by the vendor. Table 3-5 demonstrates that Edison can adequately calculate the effects of a control rod drop accident.

## 3.5 Control Rod Withdrawal Error

The Control Rod Withdrawal Error (RWE) event is the inadvertent withdrawal of a control rod to its rod block position while the reactor is operating at rated thermal power. An RWE event will increase the local power in the region of the error and could potentially cause cladding damage due to the onset of transition boiling and overheating.

The procedure used by Edison to analyze an RWE event follows that described in Section 4.5 of XN-NF-80-19, Volume 1 (Reference 3). The results of the RWE analysis are used to select a setpoint for the RBM System to ensure that neither the MCPR safety limit nor 1% plastic strain limit on Linear Heat Generation Rate (LHGR) is violated during a postulated RWE. For the analysis, the reactor is assumed to be in the normal mode of operation and all reactor parameters within the Technical Specification limits and requirements. To maximize the worth of the inadvertently withdrawn control rod, the reactor is assumed to be xenon free. For added conservatism, the partially withdrawn control rods in the core are adjusted slightly to place the fuel in the vicinity of the error rod near the maximum allowed thermal limits.

Since the rate of power increase is slow compared to the time constants for heat transfer and delayed neutrons, this event is analyzed as a series of steady state calculations with the MICROBURN-B code. The MCPR during the event is compared to that existing prior to the event to determine a ΔCPR as a function of withdrawn rod position.

A comparison to the vendor analysis for the calculation of the effect of an RWE for Dresden Unit 3 Cycle 13 is shown in Table 3-6. The vendor result for this cycle is contained in Reference 8. Table 3-6 shows that Edison can adequately determine the effects of a rod withdrawal error.

## 3.6 Loss of Feedwater Heating

The loss of feedwater heating results in a core power increase due to the increase in core inlet subcooling and resulting void collapse. The decrease in core inlet water temperature can be gradual, which is consistent with the closure of a steam extraction line to a feedwater heater, or relatively rapid, which is consistent with bypassing feedwater around a heater. This event is analyzed using the steady-state 3D simulator MICROBURN-B. The steady-state 3D simulator can be used to analyze this event since core power increases at a very moderate rate; therefore, the steady-state assumption can be applied. Additionally, the Loss of Feedwater Heating (LFWH) event has been, and is expected to continue to be, a non-limiting event relative to ΔCPR.

Local and radial peaking factors remain essentially unchanged during core wide transients. Therefore, even though gross core power may increase significantly, local Linear Heat Generation Rates (LHGRs) do not closely approach the 1% plastic strain limit during such occurrences and hence are not a safety concern for the Loss of Feedwater Heating event.

Reference 10, which confirms the appropriateness of analyzing the LFWH event with a steady state 3D simulator, documents the SNP methodology. The effect of the LFWH event is measured in terms of the change in core MCPR. The initial MCPR is determined based on the reactor state conditions just prior to the postulated LFWH event, while the final MCPR is determined based on the reactor state conditions obtained from a heat balance calculation that assumes a decreased feedwater temperature.

Reference 10 shows that there is a linear relationship between the initial and final MCPR for the event. Also, the lowest initial MCPR results in the lowest final MCPR. Therefore, although the delta CPR increases as the initial MCPR increases, the consequences of the LFWH event are not as severe at a high initial MCPR as they are at a low initial MCPR.

Edison's analysis of the delta CPR for the LFWH event will assume an initial MCPR such that the final MCPR is maintained above the MCPR safety limit. The linear relationship between the initial and final MCPR demonstrated in Reference 10 confirms the conservatism of this approach. Edison will analyze the LFWH event at various statepoints from a depletion using target control rod patterns.

A comparison to the vendor analysis for the determination of the effect of a loss of feedwater heating for Dresden Unit 2 beginning, middle, and end of Cycle 11 is shown in Table 3-7. Results are shown for the delta CPR that occurs as a result of the event. The vendor results are documented in Reference 10. Table 3-7 demonstrates that Edison can adequately calculate the effects of a loss of feedwater heating event.

Table 3-1

Comparison to Vendor Analysis - Calculation of "R"

| Unit/Cycle | Parameter | Edison Results | Vendor Results | Difference Delta K |
|---|---|---|---|---|
| Dresden Unit 3 Cycle 13 | BOC Strongest Rod Worth ($\Delta K$) | 0.0335 | 0.0344 | 0.0009 |
| | "R" | 0.0022 | 0.0022 | 0.0000 |

Table 3-2

Comparison to Vendor Analysis - SLCS Shutdown Margin

| Unit/Cycle | Edison Result | Vendor Result | Difference Delta K |
|---|---|---|---|
| Dresden Unit 3 Cycle 13 | 0.048 | 0.048 | 0.000 |

Table 3-3

Comparison to Vendor Analysis - Misoriented Assembly

| Unit/Cycle | Parameter[1] | Edison Results | Vendor Results | Difference |
|---|---|---|---|---|
| Dresden Unit 3 BOC 13 | Delta CPR | 0.030 | 0.023 | 0.007 |
| | Minimum CPR | 1.917 | 1.915 | 0.002 |
| Dresden Unit 3 MOC 13 | Delta CPR | 0.197 | 0.210 | 0.013 |
| | Minimum CPR | 1.574 | 1.572 | 0.002 |
| Dresden Unit 3 EOC 13 | Delta CPR | 0.260 | 0.266 | 0.006 |
| | Minimum CPR | 1.569 | 1.579 | 0.010 |

[1] Delta CPR = Difference between Minimum CPR for core correctly loaded and core with misoriented assembly

Minimum CPR = Minimum CPR for core with misoriented assembly

Table 3-4

Comparison to Vendor Analysis - Misloaded Assembly

| Unit/Cycle | Parameter[1] | Edison Results | Vendor Results | Difference |
|---|---|---|---|---|
| Dresden Unit 3 BOC 13 | Delta CPR | 0.000 | -0.001 | 0.001 |
| | Minimum CPR | 1.934 | 1.925 | 0.009 |
| Dresden Unit 3 MOC 13 | Delta CPR | 0.174 | 0.177 | 0.003 |
| | Minimum CPR | 1.621 | 1.623 | 0.002 |
| Dresden Unit 3 EOC 13 | Delta CPR | 0.124 | 0.134 | 0.010 |
| | Minimum CPR | 1.710 | 1.725 | 0.015 |

[1] Delta CPR = Difference between Minimum CPR for core correctly loaded and core with misloaded assembly

Minimum CPR = Minimum CPR for core with misloaded assembly

Table 3-5

Comparison to Vendor Analysis - Control Rod Drop Accident

| Unit/Cycle | Parameter[1] | Edison Results | Vendor Results | Difference |
|---|---|---|---|---|
| Dresden Unit 3 Cycle 13 | Dropped Control Rod Worth, $\Delta k$ | 0.0105 | 0.0109 | 0.0004 |
| | Doppler Coefficient, 1/k dk/dT(°F) | $-10.87 \times 10^{-6}$ | $-10.50 \times 10^{-6}$ | $0.37 \times 10^{-6}$ |
| | Effective Delayed Neutron Fraction | 0.00538 | 0.00535 | 0.00003 |
| | Four-Bundle Local Peaking Factor | 1.24 | 1.26 | 0.02 |
| | Maximum Deposited Fuel Enthalphy, cal/gm | 151 | 162 | 11 |

Table 3-6

Comparison to Vendor Analysis - Rod Withdrawal Error

Dresden Unit 3 Cycle 13

| RBM Setting | Feet Withdrawn | Delta CPR Edison Results | Delta CPR Vendor Results | Delta CPR Difference |
|---|---|---|---|---|
| 105 | 4.5 | 0.14 | 0.15 | 0.01 |
| 106 | 5.0 | 0.16 | 0.17 | 0.01 |
| 107 | 6.0 | 0.19 | 0.20 | 0.01 |
| 108 | 7.0 | 0.21 | 0.22 | 0.01 |
| 109 | 8.5 | 0.23 | 0.24 | 0.01 |
| 110 | 9.0 | 0.24 | 0.25 | 0.01 |
| 111 | 12.0 | 0.26 | 0.26 | 0.00 |

Results are reported as the change in steady-state Critical Power Ratio.

Table 3-7

Comparison to Vendor Analysis - Loss of Feedwater Heating

| Unit/Cycle | Cycle Exposure (GWD/MTU) | Delta CPR Edison Results | Delta CPR Vendor Results | Delta CPR Difference |
|---|---|---|---|---|
| Dresden Unit 2 Cycle 11 | 0.12 (BOC) | 0.310 | 0.332 | 0.022 |
| | 4.47 (MOC) | 0.224 | 0.219 | 0.005 |
| | 8.56 (EOC) | 0.267 | 0.270 | 0.003 |

## SECTION 4 - REFERENCES

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