ATTACHMENT B

COMMONWEALTH EDISON COMPANY TOPICAL REPORT

BENCHMARK OF BWR NUCLEAR DESIGN METHODS

NEUTRONIC LICENSING ANALYSES

(NFSR-0085, SUPPLEMENT 2, REVISION 0)

NFSR-0085
Supplement 2
Revision 0

Nuclear Fuel Services

COMMONWEALTH EDISON COMPANY

TOPICAL REPORT - SUPPLEMENT 2

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Abstract

This supplement to Commonwealth Edison Company Topical Report NFSR-0085, "Benchmark of 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 demonstrate 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-0085. The application procedures are consistent with those described in the General Electric Proprietary Document GESTAR II, which has 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-0085, "Benchmark of BWR Nuclear Design Methods", Reference 1, discusses the application procedures to be used by Edison in performing the neutronic licensing calculations listed in Table 1-1. The application procedures for the events listed in Table 1-1 are consistent with the procedures described in the Reference 2 GE document NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (GESTAR II), which are also summarized in Section 3 of this supplement. Because the same neutronic methods are used, together with consistent application procedures, the same uncertainties are also applied, as discussed in Reference 1.

Also included in this report are comparisons to General Electric (GE) results. GE is the fuel vendor for the four Edison BWRs at Quad Cities and LaSalle County Stations. Comparisons to vendor results are not available for all of the neutronic licensing events listed in Table 1-1 for both the Quad Cities and LaSalle County Stations. In some instances, GE did not perform cycle-specific analyses, as the events have been evaluated generically. However, a comparison to the vendor's results for one of the two stations was available in all cases and is included in Section 3.

As discussed in the Reference 1 Topical Report, Edison is requesting NRC approval for the evaluation of the events listed in Table 1-1 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 perform those analyses which determine the critical power ratio impact for units not fueled by GE (currently Dresden Units 2 and 3). This encompasses Items 3 through 7 in Table 1-1.

There are certain events in Table 1-1 which have been analyzed generically by GE. Edison will not perform these analyses on a cycle-specific basis providing the generic evaluation is applicable. This is discussed in more detail for the specific events in Section 3.

1.2 Overview of NFSR-0085, Supplement 2

A summary of the results and conclusions is contained in Section 2 of this supplement to Topical Report NFSR-0085. These results, which are detailed in Section 3, demonstrate that Commonwealth Edison can adequately perform the licensing calculations listed in Table 1-1.

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 are consistent with those employed by GE and outlined in Reference 2. 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.

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 GE 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 TGBLA/PANACEA code package contained in the Reference 1 Topical Report and as part of Edison's continuing core tracking calculations.

Several of the events listed in Table 1-1 are analyzed generically by General Electric, and need not be re-performed each cycle if no significant changes have occurred. This approach was reviewed and approved by the NRC in the review of GESTAR II, Reference 2. Since Edison has reproduced the GE results for the cases which are available and is using the same code package as GE, Edison does not intend to perform the calculations for these events on a per-cycle basis if no significant changes have occurred in the fuel loading or plant which would invalidate the generic analyses.

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 TGBLA/PANACEA code package, which was developed and initially benchmarked by GE 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 consistent with the GE standard methodology, which is outlined in GESTAR II, Reference 2. The limiting conditions used in the analyses are consistent with the extended operating domain documents previously NRC reviewed and approved for Quad Cities and LaSalle County Stations. The GE extended operating domain documents are contained in References 3 and 4 for Quad Cities and LaSalle County Stations, respectively.

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 PANACEA 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 20C, 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-0085, Reference 1, were used to develop the appropriate exposure-dependent eigenvalue bias for each unit.

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 elsewhere 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.

Comparisons to vendor analyses for the calculation of "R" for Quad Cities Unit 2 Cycle 10 and LaSalle County Station Unit 2 Cycle 3 are shown in Table 3-1. The vendor results for these cycles are contained in References 5 and 6 respectively. The comparisons in Table 3-1 demonstrate that Edison can adequately perform shutdown margin calculations, including the calculation of "R", as the maximum difference in the parameter "R" for the two cycles is 0.001 delta k.

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 20C, 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 system, PANACEA is used to add the negative reactivity effects of boron to the most reactive cold all-rods-out point in the cycle. The resulting eigenvalue is compared to the cold critical eigenvalue at that exposure point to determine the SLCS shutdown margin.

Comparisons to vendor analyses for the calculation of shutdown margin after the initiation of the Standby Liquid Control System for Quad Cities Unit 2 Cycle 10 and LaSalle County Station Unit 2 Cycle 3 are shown in Table 3-2. The vendor results for these cycles are contained in References 5 and 6, respectively. Table 3-2 shows that Edison results agree very well with the GE results, thereby demonstrating that Edison can adequately perform shutdown margin calculations 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 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. Therefore, the 180 degree rotation of the assembly bounds the 90 degree rotated situation, and hence only the 180 degree misorientation is analyzed.

A misorientation error 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. This is consistent with GESTAR II, Reference 2, and the LaSalle Final Safety Analysis Report (FSAR), Reference 7.

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 misoriented fuel assembly is determined by depleting the assembly in a 180 degree rotated position using the lattice physics code TGBLA. The resulting exposure-dependent R-factors, which are a weighted average of the local pin powers, and the integrated power of the misoriented assembly are compared to the nominal, non-rotated case. Single channel operating limit CPR calculations and the 3D core simulator, PANACEA, are used to determine the relationship between delta R-factor and delta CPR. This method of analysis considers both changes in assembly reactivity and power peaking. Conservative adjustments are made to the calculated delta CPR to account for the infinite lattice assumption employed by TGBLA. This is described in more detail in GESTAR II, Reference 2.

Comparisons to vendor analyses for the misoriented assembly error for D-lattice assemblies loaded in Quad Cities Station are shown in Table 3-3. The delta CPR for a misoriented assembly is a function of the change in R-factor, together with the change in assembly power. GE furnished these base parameters to Edison for an assembly type loaded into Quad Cities Unit 2 Cycle 11 in Reference 8. Table 3-3 compares the maximum values of these base parameters, namely, the change in R-factor and assembly power, and the results demonstrate that Edison can accurately calculate these base parameters. Table 3-3 also compares the calculation of delta CPR from the base parameters of delta R-factor and delta power for an assembly type loaded into a previous cycle of Quad Cities, thereby demonstrating that Edison can determine the delta CPR given these base parameters. The GE base parameters and resulting delta CPR were furnished to Edison for this assembly type in References 9 and 5, respectively. Table 3-3 shows that Edison can determine both the base parameters and the resulting delta CPR for a misoriented assembly.

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

3.3.2 Mislocated Assembly

A mislocated assembly is an assembly which is loaded in an incorrect core position and not subsequently identified and corrected prior to core operation. The bundle is then monitored incorrectly, possibly resulting in a high reactivity, or limiting, assembly being modeled during the cycle as a low reactivity, or non-limiting assembly.

The probability that a mislocated fuel assembly will result in a CPR less than the safety limit is sufficiently small that cycle-specific analyses are usually not required. This is discussed in GESTAR II, Reference 2. An exception to this is the case in which the critical power correlation is different for the various fuel types in the core, leading to the possibility that a misloaded fuel bundle may be monitored with an incorrect and potentially non-conservative critical power correlation as well as with an incorrect set of lattice physics data. This could occur, for example, when a batch of the GE fuel product line GE9B is loaded into a core consisting of earlier GE fuel product lines. This is the case analyzed for the comparison to vendor analysis.

This event will not be evaluated on a cycle-specific basis, except in situations where changes in the fuel product line or critical power correlation require such an evaluation. As discussed in the previous section on the assembly misorientation error, an additional mitigating factor exists in that proper location of the fuel assembly in the reactor core is verified by visual observation and assured by verification procedures during and following core loading.

The 3-dimensional core simulator PANACEA is used to analyze a mislocated assembly which is uncorrected prior to operation. An adjustment to the results is made to account for axially varying R-factors, as required by GESTAR II.

A comparison to the vendor analysis for the determination of the effect of a mislocated assembly for Quad Cities 1 Cycle 12 is shown in Table 3-4. The vendor results are documented in the vendor analyses for Quad Cities 1 Cycle 12, Reference 10. Table 3-4 shows that Edison can adequately calculate the effects of a mislocated assembly analysis.

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. This is consistent with GESTAR II, Reference 2 and the individual plant Final Safety Analysis Reports.

Reference 2 states that a cycle-specific analysis of the CRDA event does not need to be performed if the Banked Position Withdrawal Sequence (BPWS) Rules (Reference 11) are used and certain key parameters, namely the Doppler coefficient, scram reactivity, and accident reactivity, are within the bounding values found in GESTAR II, Reference 2. Edison and Edison's fuel vendors have performed analyses which support the relaxation of BPWS constraints in the region between 75% and 50% control rod density for the Dresden and Quad Cities units. The NRC concurred with this revision in Reference 12. For these revisions to BPWS, a method consistent with GESTAR II, Reference 2, is used to determine the effect of the CRDA.

A comparison to the vendor analysis for the determination of the effect of a control rod drop accident in the area between 75 and 50% control rod density for Quad Cities Unit 2 Cycle 12 is shown in Table 3-5. The vendor results are documented in Reference 13. A comparison to a LaSalle cycle could not be made since GE does not perform the control rod drop accident analysis for those plants which apply unmodified BPWS rules, outlined in Reference 11.

Table 3-5 supports the conclusion that Edison can adequately calculate the effects of a control rod drop accident, as the GE fuel enthalpy deposition results are reproduced to within 5 cal/gm. Although this table provides a meaningful vendor comparison, the results should not be used as an indication of the degree of margin to the Technical Specification limit of 280 cal/gm for a CRDA while using the Edison revised sequencing rules. GE conservatively considered some out-of-sequence rods when determining the highest worth control rod in the region between 75 and 50% control rod density. For the revised sequencing rules, the dropped rod which was evaluated by GE would be an out-of-sequence rod, and hence the control rod drive could not be withdrawn by the control room operator due to rod worth minimizer restrictions. Edison evaluated the same out-of-sequence case as that used by GE to provide a consistent basis for comparison. However, the in-sequence control rods have lower rod worth than the rod which was evaluated, and hence result in a significantly lower fuel enthalpy deposition. Edison calculated the result of a CRDA in the region between 75 and 50% control rod density if the Edison sequencing rules were being followed without the conservative sequencing assumption made by GE. This yielded an energy deposition of less than 100 cal/gm in this region of the sequence, thereby demonstrating significant margin to the 280 cal/gm Technical Specification limit in the region where Edison is deviating from BPWS.

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 transient boiling and overheating.

The procedure used by Edison to analyze an RWE event follows that described in GESTAR II, Reference 2. The results of the RWE analysis are used to select a setpoint for the Rod Block Monitor 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. The LHGR limits have been evaluated generically for GE fuel, and need not be evaluated on a cycle-specific basis and demonstrated substantial margin to limits. This is consistent with approved amendments to GESTAR II.

Comparisons to vendor analyses for the calculation of the effect of an RWE for Quad Cities Unit 2 Cycle 10 and LaSalle County Station Unit 2 Cycle 3 are shown in Table 3-6. The vendor results for these cycles are contained in References 5 and 6, respectively. 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 PANACEA and procedure described in GESTAR II, Reference 2. 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 event has been, and is expected to continue to be, a non-limiting event relative to delta CPR.

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 string limit during such occurrences and hence are not a safety concern for the Loss of Feedwater Heating event. This is consistent with GESTAR II.

A comparison to the vendor analysis for the calculation of the loss of feedwater heating event is shown in Table 3-7. Table 3-7 demonstrates that Edison can adequately determine the effects of a loss of feedwater heating event, as the delta CPR is predicted to within 0.01. The vendor results for LaSalle County Station Unit 2 Cycle 3 are documented in Reference 6. A comparison to Quad Cities results was not made since GE no longer performs the Loss of Feedwater Heating event on a cycle-specific basis, as described in Reference 3. If the criteria in Reference 3 are met for the reload, Edison will not perform the Loss of Feedwater Heating analysis for Quad Cities on a per-cycle basis.

Table 3-1
Comparisons to Vendor Analyses - Calculation of "R"

Unit/Cycle	Parameter	Edison Results	Vendor Results	Difference, Delta K
Quad Cities 2 Cycle 10	BOC All-Rods-In K-effective	0.958	0.958	0.000
	BOC Strongest Rod Out K-effective	0.981	0.981	0.000
	"R"	0.006	0.007	-0.001
LaSalle County 2 Cycle 3	BOC All-Rods-In K-effective	0.960	0.960	0.000
	BOC Strongest Rod Out K-effective	0.983	0.985	-0.002
	"R"	0.006	0.005	0.001

Table 3-2
Comparisons to Vendor Analyses - SLCS Shutdown Margin

Unit/Cycle	Edison Results	Vendor Results	Difference, Delta k
Quad Cities 2 Cycle 10	0.044	0.043	0.001
LaSalle County 2 Cycle 3	0.037	0.037	0.000

Table 3-3
Comparisons to Vendor Analyses - Misoriented Assembly

Unit/Cycle	Parameter	Edison Results	Vendor Results	Difference
Quad Cities 2 Cycle 11	Delta R-Factor	0.014	0.014	0.000
	Delta Normalized Power	0.75	0.75	0.00
Quad Cities 2 Cycle 10	Delta CPR	0.14	0.14	0.00

Table 3-4
Comparison to Vendor Analysis - Misloaded Assembly

	Delta CPR		
Unit/Cycle	Edison Result	Vendor Result	Difference
Quad Cities 1 Cycle 12	0.09	0.09	0.00

Results are reported as the change in steady-state Critical Power Ratio, relative to the properly loaded configuration.

Table 3-5
Comparison to Vendor Analysis - Control Rod Drop Accident

Unit/Cycle	Conditions	Edison Results	Vendor Results	Difference
Quad Cities 2 Cycle 12	Cold			
	Bundle Beta	0.0064	0.0063	
	Rod Worth, % delta K	1.078	1.089	
	Enthalpy Deposition, cal/gm	153	157	-4
	Hot Standby			
	Bundle Beta	0.0063	0.0063	
	Rod Worth, % delta K	1.469	1.480	
	Enthalpy Deposition, cal/gm	252	257	-5

The Control Rod Drop Accident is evaluated in the range of 75 to 50% Control Rod Density, where BPWS rules are not being followed.

Table 3-6
Comparisons to Vendor Analyses - Rod Withdrawal Error

Quad Cities 2 Cycle 10:

		Delta CPR		
RBM Setting	Feet Withdrawn	Edison Results	Vendor Results	Difference
104	3.0	0.07	0.07	0.00
105	3.5	0.10	0.10	0.00
106	4.0	0.12	0.12	0.00
107	4.0	0.12	0.12	0.00
108	7.0	0.22	0.20	0.02
109	12.0	0.27	0.26	0.01
110	12.0	0.27	0.26	0.01

LaSalle 2 Cycle 3:

		Delta CPR		
RBM Setting	Feet Withdrawn	Edison Results	Vendor Results	Difference
104	4.0	0.15	0.15	0.00
105	4.5	0.18	0.18	0.00
106	5.0	0.19	0.19	0.00
107	6.5	0.23	0.22	0.01
108	12.0	0.23	0.23	0.00
109	12.0	0.23	0.23	0.00
110	12.0	0.23	0.23	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

	Delta CPR		
Unit/Cycle	Edison Results	Vendor Results	Difference
LaSalle 2 Cycle 3	0.103	0.106	-0.003

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

Section 4 - References

1. Commonwealth Edison Company Topical Report NFSR-0085, "Benchmark of BWR Nuclear Design Methods, November 1990.

2. GE Proprietary Document NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel", as supplemented.

3. GE Document NEDC-31449, "Extended Operating Domain and Equipment Out-of-Service for Quad Cities Nuclear Power Station Units 1 and 2", as revised.

4. GE Document NEDC-31455, "Extended Operating Domain and Equipment Out-of-Service for LaSalle County Nuclear Power Station Units 1 and 2", as revised.

5. GE Document 23A5846, "Supplemental Reload Licensing Submittal for Quad Cities Nuclear Power Station Unit 2, Reload 9, Cycle 10", January 1988.

6. GE Document 23A5841, "Supplemental Reload Licensing Submittal for LaSalle County Station Unit 2, Reload 2, Cycle 3", July 1988.

7. LaSalle County Station Final Safety Analysis Report.

8. GE Proprietary Letter GE:REP:91-045, R. E. Parr (GE) to R. J. Chin (Edison), "Rotated Data for Fuel Bundle EDB No. 1815", February 19, 1991.

9. Letter REP:90-061, R. E. Parr (GE) to R. A. Roehl (Edison), "Fuel Management Data", March 6, 1990.

10. GE Proprietary Document Design Record File J11-01399, "EE1 Cycle 12 FLE-Mislocated", April 20, 1990.

11. GE Document NEDO-21231, "Banked Position Withdrawal Sequence", January 1977.

12. Letter, B. L. Siegel (NRC) to T. J. Kovach (Edison), "Revised Control Rod Sequencing Methods for the Dresden and Quad Cities Nuclear Power Stations", September 21, 1990.

13. GE Proprietary Document Design Record File J11-01558, "RDA for EE2 Cycle 12", February 10, 1990.