Commonwealth Edison
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December 23, 1991

Dr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attn: Document Control Desk

Subject: Dresden Station Units 2 and 3
Quad Cities Station Units 1 and 2
LaSalle County Station Units 1 and 2
Topical Report for Neutronics Methods for BWR Reload Design
NRC Docket Nos. 50-237/249, 50-254/265, and 50-373/374

References:

1. M.H. Richter (CECo) Letter to T.E. Murley (NRC), dated December 12, 1990, submitting CECo Topical Report NFSR-0085.

2. M.H. Richter (CECo) Letter to T.E. Murley (NRC), dated May 8, 1991, submitting Supplements 1 and 2 to CECo Topical Report NFSR-0085.

3. Conference Call between M. Chatterton (NRC) and R.J. Chin (CECo) and J.E. Wieging (CECo) on October 8, 1991

Dear Dr. Murley:

Commonwealth Edison Company (CECo) submitted the BWR Neutronics Methods Topical Report NFSR-0085 in Reference 1, and submitted Supplements 1 and 2 of Topical Report NFSR-0085 in Reference 2. A conference call (Reference 3) was held on October 8, 1991, to discuss the Reference 1 and 2 submittals. As a result of that call, CECo was requested to transmit responses to several of the reviewer's questions. These responses are attached.

Please note that in anticipation of completion of Staff review of NFSR-0085, a related administrative change to the Technical Specification Section 6 methodology references for Quad Cities and LaSalle County Stations is being prepared with submittal expected in January, 1992. No changes are required for the Dresden Technical Specifications as CECo is not currently requesting approval to perform any analysis for Dresden which are required for the Core Operating Limits Report, as referenced in Section 6 of the Technical Specifications.

Please contact this office should further information be required.

Respectfully,

Peter L. Piet
Nuclear Licensing Administrator

Attachment: Commonwealth Edison Response to NRC Questions on Edison Topical Report NFSR-0085

cc: A. Bert Davis-Regional Administrator, RIII
B.L. Siegel-Dresden/LaSalle Project Manager, NRR
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/scl:55:6

Commonwealth Edison Response to NRC Questions on

Edison Topical Report NFSR-0085

Main Report

Page 1-2, Paragraphs 3 and 4
How do Edison results compare with GE for physics data, loading patterns, and neutronic parameters?

Edison used the GE NRC-approved code package, namely the lattice physics code TGBLA and the core simulator code PANACEA, to determine the physics data for the benchmark. Validation tests were performed using Edison's Quality Assurance procedures to ensure the same physics results were obtained for specific test cases. Although side by side comparisons of Edison and GE-generated lattice physics values are not available for each lattice in the benchmark, the accuracy of the physics data is demonstrated by the accuracy of the benchmark results.

The as-loaded-core configuration, or loading pattern, and core conditions throughout the cycles in the benchmark were taken from historical site data.

A comparison of the neutronic parameters in Table 1-1 is discussed in response to the next question.

Table 1-1
Please provide the results.
Which items will be done for which plants?
Please provide comparisons of data previously calculated for each item.
What is the uncertainty for each item and what is the justification for using that uncertainty?

Edison does not plan to take over all aspects of the design process listed in Table 1-1 immediately upon NRC approval of the Topical Report. Rather, Edison is phasing in the methodology, performing many of the analyses in parallel with the fuel vendor, or with the review and concurrence of the fuel vendor, for several reloads. This process of gradually incorporating analyses allows Edison to ensure that the results obtained with in-house methods, in-house procedures, and in-house engineers are correct. For instance, Edison is currently performing the loading pattern development for all upcoming cycles. This includes the development of the loading pattern itself, along with target rod patterns, core axial power distributions, hot excess reactivity, and shutdown margin calculations. The resulting analyses are transmitted to Edison's fuel vendor for review, concurrence, and incorporation as the basepoint for all vendor-performed licensing calculations. Once the Edison Topical Report has been approved and the phase-in process completed, Edison will no longer request review and concurrence of this part of the design process on a routine basis. Exceptions to this may exist, as would be the case if a new fuel product line was being loaded, in order to learn how to fully utilize the new design features.

Edison's Topical requests approval to evaluate all parameters listed in Table 1-1 of the Topical Report for operation, testing, and surveillance, with the exception that Edison does not currently anticipate using GE methodology to provide process computer input for those units not fueled with GE product lines. Hence Edison is not requesting approval for providing process computer input for those units. Currently, Quad Cities and LaSalle County Stations are loaded with GE fuel; Dresden Station has Siemens Nuclear Power (SNP) fuel. (SNP was previously named Advanced Nuclear Fuels.)

Items in Table 1-1 are specifically addressed as follows:

Items 1 and 2 in Table 1-1, the calculation of "R" and shutdown margin calculations, are discussed in Edison Topical Report NFSR-0085, Supplement 2.

Item 3, hot excess reactivity, is the difference between the eigenvalue calculated assuming a hot all-rods-out condition and the projected hot critical eigenvalue. The projection of the hot critical eigenvalue is discussed in response to later questions. The calculation of hot excess reactivity is not a licensing-basis calculation, but is evaluated to ensure the plant can reach full power while concurrently meeting the thermal limits.

The data required for operation, Item 4, are those parameters that are typically calculated as part of the loading pattern development, which was discussed previously. Specifically, target rod patterns are developed for each reload to determine the cycle energy and to ensure that thermal margins will be met during the cycle. Thermal margins are also checked via daily surveillances during the cycle to ensure compliance to the Technical Specification limits. Hot reactivity anomalies are the difference between actual critical eigenvalue and the predicted hot critical eigenvalue, which will be discussed in a later section of this response. The core axial power distribution is that distribution which results from using the target rod patterns. The axial power distribution is of interest primarily at the end of the cycle, where it is used as the basis for the pressurization transients. Although Edison currently has in-house transient and thermal-hydraulic capability using EPRI codes (RETRAN and VIPRE) for which a CECo Topical for PWR applications is currently under NRR review, BWR reload licensing applications of the EPRI codes are not planned for four to five years. In the interim, the analysis of limiting plant transients such as pressurization events for BWR reload licensing will continue to be performed by Edison's fuel vendors using neutronic input provided by Edison (upon NRC approval).

Item 5 of Table 1-1 discusses the data required for startup. These calculations use the same modules of PANACEA that are used to determine the cold critical eigenvalue, which is discussed later in this response.

However, the application is slightly different in that the critical eigenvalue has been projected, and hence is the known variable, and the rod pattern is allowed to change. Tables 1 and 2 compare Edison and GE results for control rod worths during the approach to criticality for previous cycles of Quad Cities and LaSalle County Station, respectively. The GE data for Quad Cities and LaSalle are contained in References 1 and 2, respectively. These comparisons demonstrate that the Edison results are comparable to the GE results for the approach to criticality calculations.

Edison is requesting approval to use the NFSR-0085 methodology to provide process computer input, Item 6, only for those units loaded with GE fuel product lines. The process computer input for the GE core monitoring code is basically the same input, namely a PANACEA restart file, as that required for the Traversing Incore Probe (TIP) and power distribution comparisons which were performed for the benchmark. Additional information required on the restart file for core monitoring includes items such as control blade and Local Power Range Monitor (LPRM) depletion factors. These factors are determined by the manufacturer of the components and are typically empirical constants as a function of component exposure and are easily added to the restart file. Since the additional information required on the process computer restart file does not affect the monitoring of Technical Specification parameters, the accuracy of the TIP comparisons in the Topical Report demonstrate that Edison can provide the process computer input data, which is essentially a PANACEA restart file, to the station.

This question also addressed uncertainties. Uncertainties are discussed in response to the next question.

<u>*Page 2-1, #2, Second one.*</u>
Please justify this statement, namely "Commonwealth Edison is justified in its application of calculational uncertainties identical to those applied by GE".

BWRs explicitly measure in-core power distributions on a frequent basis (at least daily) during operation; therefore, there are very few uncertainty calculations involved in the design process of BWR reloads. An exception to this is the power distribution uncertainty used as an input to the MCPR Safety Limit calculations. Edison requested GE to evaluate the TIP standard deviation results in the Edison Topical Report to ensure that the MCPR Safety Limit which GE evaluates generically on a product line basis remains applicable once the Edison Topical Report is approved and Edison performs the analyses of record for the units. In Reference 3, GE concurred that the power distribution uncertainties in the Edison Topical Report are bounded by the value used by GE in calculating the MCPR Safety Limit. This value is 8.7%, as reported in GESTAR, Reference 4. The uncertainties on other plant parameters which are inputs to the calculation of power distribution uncertainty, such as feedwater flow, are not affected by the transition to in-house reload design.

Page 4-2, Sections 4.2.1
Hot Critical Eigenvalue

An important parameter in the design process is correctly projecting the hot critical eigenvalue for the cycle being designed. This projection can use information from past cycles of the unit, known trends from a sister unit, and estimated differences resulting from other changes, such as the implementation of a new fuel product line.

For instance, a projection of the LaSalle 2 Cycle 4 hot critical eigenvalue would be performed using the information from Figures 4.2-17 and 4.2-18 in NFSR-0085 and the following steps: LaSalle 2 Cycle 2 and LaSalle 2 Cycle 3 both show a linear decrease in eigenvalue as a function of exposure until approximately mid-cycle, and then the eigenvalues are relatively constant until the end of cycle. Additionally, both cycles have approximately the same beginning of cycle eigenvalue, 1.002, and end of cycle eigenvalue, 0.996, within approximately 0.002 delta k. Therefore, it is projected that the hot critical eigenvalue for LaSalle 2 Cycle 4 would be about the same as that in LaSalle 2 Cycle 3. The values from Cycle 3 have a greater weight in this evaluation than those in Cycle 2 because Cycle 3 information is more recent, and the Cycle 2 eigenvalues are affected to a greater extent by the presence of the initial core fuel, which has different neutronic characteristics than reload fuel, such as a significantly lower enrichment, namely 0.71 to 2.2 w/o U235 for the initial core fuel compared to approximately 3.0 w/o U235 for the reload fuel.

The above discussion assumes that there were no other changes between Cycles 3 and 4, but that was not true. The fuel product line GE9 was loaded in Cycle 4. This fuel product line is different from that previously loaded, since it has more internal water moderation and axial enrichment variations. Additionally, the fuel temperature used in the lattice physics calculations is lower than that used for previous fuel product lines. This change was made to be consistent with the fuel vendor's recommendations. These differences must be evaluated to determine the effect on the projected eigenvalue. GE had determined that the additional moderation and axial enrichment would have little difference in the projected critical eigenvalue, but the different fuel temperature, relative to the database, would result in a slight downward shift. A downward shift in the hot critical eigenvalue is considered conservative in the design of a cycle, since this would result in more cycle energy than that projected if the Cycle 3 end of cycle hot critical eigenvalue was assumed. In this case, the engineer would typically assume that the end of cycle hot critical eigenvalue for Cycle 4 is approximately the same as the eigenvalue for Cycle 3, as this would be conservative in regards to cycle energy.

Comparisons of the GE and Edison results for the hot critical eigenvalues for Quad Cities 1 Cycle 10 and LaSalle 2 Cycle 3 are contained in Figures 1 and 2. The GE data is documented in References 5 and 6, respectively. This data could not be presented in tabular form since the incremental exposure intervals chosen for core tracking were different. The mean and standard deviation of the two cycles are summarized as follows:

	Edison Data		GE Data	
Unit/Cycle	Mean	Standard Deviation	Mean	Standard Deviation
L2C3	0.9984	0.0015	0.9985	0.0016
Q1C10	1.0017	0.0010	1.0023	0.0011

These values demonstrate that the Edison results are approximately the same as the GE values. The difference in mean eigenvalue for Q1C10 is consistent with the data in Figures 1 and 2.

An evaluation of Figures 1 and 2 also show relatively good agreement. The LaSalle 2 comparison is better than the Quad Cities comparison primarily because the operational strategies at the units are different. Hot critical eigenvalue data is most consistent if the unit is operating in steady-state. Quad Cities load follows, both with nightly load drops and daily system Economic Generation Control (EGC), to a much greater degree than LaSalle. Therefore, it is more difficult to select state conditions and exposure increments which approximate the unit operation for a given interval due to xenon transients which results from power changes. The exposures which show the greatest difference for the Quad Cities eigenvalues are those intervals in which the unit was load following significantly.

Page 4-2, Section 4.2.2
Cold Critical Eigenvalue

The cold critical eigenvalue for upcoming cycles is projected with the same methodology as that described for the hot critical eigenvalue. In that hot critical eigenvalue example, a change in the design process resulted in a potential decrease in the eigenvalue relative to the historical database. A decrease in the cold critical eigenvalue is non-conservative for shutdown margin calculations if not taken into account in the projection. If the eigenvalue is lower than that projected based on the historic database, criticality is reached sooner, or at a higher control rod density. In this case, the shutdown margin may be over-predicted, a non-conservative scenario. Therefore, the engineer selects a cold critical eigenvalue which is slightly lower than that justified by the historical database. Alternatively, the engineer may increase the shutdown margin design goals slightly to accommodate any deviations.

The following compares the Edison and GE-calculated cold critical eigenvalues at the beginning of cycle for several recent cycles. The GE data is contained in Reference 7. Other cycle exposure points were not available for comparison.

Unit/Cycle	Edison Cold Critical Eigenvalue	GE Cold Critical Eigenvalue	Difference, Delta k
L1C3	1.0020	1.0012	0.0008
L1C4	0.9998	0.9997	0.0001
L1C5	1.0026	1.0027	-0.0001
L2C3	1.0011	1.0028	-0.0017
L2C4	1.0013	1.0012	0.0001
Q1C10	1.0067	1.0049	0.0018
Q1C11	1.0053	1.0050	0.0003
Q1C12	1.0048	1.0040	0.0008
Q2C10	1.0053	1.0045	0.0008
Q2C11	1.0042	1.0043	-0.0001

These differences demonstrate that the Edison and GE generated cold critical eigenvalues are approximately the same, and hence Edison is adequately calculating the cold critical eigenvalue.

Page 4-2, Section 4.2.3
Power Distribution (TIPs)

Edison has evaluated the nodal and radial standard deviations of the calculated to measured TIP readings on an on-going basis to ensure the code package continues to adequately predict the core behavior. The current design criteria being used at Edison for power distribution standard deviations are 10% nodally and 6% radially. If the standard deviations approach these values, they are beyond what has been historically seen for this analysis, and signal that the engineer should determine what is causing this discrepancy. For instance, this acceptance criteria has been violated during a TIP evaluation since the benchmark was completed. The engineer evaluated the individual string readings and discovered that readings from one TIP machine, covering one part of the core, were consistently overpredicted by PANACEA. However, readings from symmetric strings to the strings on this TIP machine were consistently higher than the readings on the suspect TIP machine. Therefore, it was concluded that this was probably an instrument error, since the readings from the next set of TIP traces were evaluated, and the standard deviations were once again within the range of the historical values.

Edison plans on performing periodic evaluations of the TIP readings to ensure the power distribution is being adequately predicted by PANACEA. This data will be used to determine the ongoing adequacy of the code. Sufficient redundancy is provided in the nuclear instrumentation, namely the TIPs and LPRMs, to determine whether any deviations are caused by code deficiencies or are actually occurring at the plant. Therefore, unless deviations such as described above for the single TIP machine are seen, the plant data is considered accurate, and the calculated data is considered suspect.

Comparisons to available GE data are as follows:

Unit/Cycle	Cycle Exposure, MWd/ST	Edison Nodal Standard Deviation, %	GE Nodal Standard Deviation, %	Difference
L1C4	3721	6.3	6.0	0.3
L1C4	5405	6.6	6.4	0.2
L1C4	6857	6.5	6.7	-0.2
L1C4	7657	6.3	6.7	-0.4

The GE data is contained in Reference 8. GE could not provide Quad Cities data for this comparison.

The comparison of these standard deviations of measured to predicted core power distribution demonstrates that Edison can predict power distribution to the same degree of accuracy as GE.

Supplement 1

Page 3-2, Second Paragraph
Please discuss and explain.

This paragraph is discussing the fact that the gamma scan measurements are not symmetric across the diagonal axis, while the lattice physics calculations, which assume infinite lattice boundary conditions, are symmetric. The measurement asymmetries are caused by instrument error and by the actual flux tilt which the bundles would experience during operation. The second of these factors cannot be accounted for in the analysis, as the detailed pin-by-pin calculations are not performed in the core simulator. Some previous BWR topicals performed by other utilities have averaged the readings of the symmetric fuel pins and compared the lattice physics calculations to this average. This paragraph was included to indicate that the measured readings were not averaged in this way for the Edison Topical Report, and hence the standard deviations may be slightly greater in this analysis.

Assembly Gamma Scans
What do similar GE curves look like?

See GE Document NEDE-30130-P-A Pages 5-19 and 5-20 for the relative barium-140 concentrations as a function of axial height. See GE Document NEDE-30130-P-A Pages 5-23 through 5-25 for the percent error in assembly integrated power as a function of core position. A copy of these pages is attached for your use.

A comparison of the Edison and GE results (NEDE-30130-P-A) for the standard deviations for the assembly gamma scans are as follows:

Cycle	GE Nodal Values, %	Edison Nodal Values, %	GE Radial Values, %	Edison Radial Values, %
2	4.2	4.8	2.1	2.4
4	5.5	4.7	2.6	3.1
5	6.4	5.3	3.0	4.2

These results are comparable; the Edison results are slightly better for the nodal standard deviations, while the GE results are slightly better for the radial standard deviations.

Pin Gamma Scans
How do the Edison results compare to similar GE data?

GE did not present results similar to those contained in Tables 3.4-1 and 3.4-2 in NEDE-30130-P-A, therefore a comparison to GE data could not be made for the pin gamma scans. GE used a different database for this calculation, as they evaluated a mixed oxide assembly, while Edison evaluated only uranium dioxide assemblies.

Supplement 2

Abstract, Page iii
Procedures are "consistent". What does this mean? What are the differences and how much effect do they make?

The same GESTAR licensing basis is maintained for these events. There are no differences between our procedures and the licensing basis in GESTAR.

Page 1-1
Please provide more justification for the uncertainties. What are the values? Why are they adequate?

"Uncertainties" may have been an imprecise term in this section. The uncertainties which are used in the licensing calculations are more precisely correction factors and include parameters such as the 0.02 delta R-factor adder required in the rotated bundle calculation. These correction factors are outlined in GESTAR and hence are part of the licensing basis of the analysis. Edison feels these values are adequate and can be applied in our analyses because the same codes and licensing basis, as defined in GESTAR, are being used.

Page 3-9
What happens if the criteria are not met on these events?

If the criteria are not met, a cycle-specific analysis of the Loss of Feedwater Heating event will be performed for Quad Cities using the same procedure as was used in the evaluation of the Loss of Feedwater Heating event for LaSalle as reported in the benchmark.

References

1. GE Letter REP:91-007, R. E. Parr (GE) to R. J. Chin (Edison), "QC1C12R11 - Cycle Management Report and Preliminary Core Operating Plan - Proprietary", January 8, 1991.

2. GE Letter REP:91-105, R. E. Parr (GE) to R. J. Chin (Edison), "LS1C05R04 - Cycle Management Report and Preliminary Core Operating Plan - Proprietary", April 17, 1991.

3. GE Letter REP:91-069, R. E. Parr (GE) to R. J. Chin (Edison), "MCPR Safety Limit for Edison BWR's - Response", March 22, 1991.

4. GE Document NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel".

5. GE Letter REP:90-043, R. E. Parr (GE) to R. A. Roehl (Edison), "Core Conditions Summary Report for QC1R09C10", February 16, 1990.

6. GE Letter REP:90-188, R. E. Parr (GE) to R. A. Roehl (Edison), "Cycle Summary Report for LS2R02C03", October 1, 1990.

7. GE Letter REP:91-262, R. E. Parr (GE) to R. J. Chin (Edison), "LaSalle and Quad Cities Cold Critical Data", December 5, 1991.

8. GE Letter REP:91-253, R. E. Parr (GE) to R. J. Chin (Edison), "LaSalle Unit 1 Cycle 4 TIP Comparison Data", November 20, 1991.

Table 1
Comparison of In-Sequence Control Rod Worth
Quad Cities 1 Cycle 11

Group One (24 Rods)	Group Two (20 Rods)	Group Three (24 Rods)	CECo Data Norm. Eigen.	CECo Data Delta k	GE Data Norm. Eigen.	GE Data Delta k	CECo-GE % Delta k
ARI	ARI	ARI	0.94982	N/A	0.94974	N/A	
ARO	ARI	ARI	0.98631	0.03649	0.98636	0.03662	-0.013
ARO	3@48	ARI	0.99022	0.00391	0.99028	0.00392	-0.001
ARO	6@48	ARI	0.99081	0.00059	0.99086	0.00058	0.001
ARO	14@48	ARI	0.99192	0.00111	0.99195	0.00109	0.002
ARO	17@48	ARI	0.99730	0.00538	0.99734	0.00539	-0.001
ARO	20@48(ARO)	ARI	0.99915	0.00185	0.99922	0.00188	-0.003
ARO	ARO	12@04,12@00	0.99921	0.00006	0.99927	0.00005	0.001
ARO	ARO	24@4	1.00012	0.00091	1.00021	0.00094	-0.003
ARO	ARO	12@08,12@04	1.00038	0.00026	1.00048	0.00027	-0.001
ARO	ARO	17@08, 7@04	1.00480	0.00442	1.00504	0.00456	-0.014
ARO	ARO	22@08, 2@04	1.00570	0.00090	1.00593	0.00089	0.001
ARO	ARO	24@08	1.00611	0.00041	1.00634	0.00041	0.000
ARO	ARO	12@12,12@08	1.00634	0.00023	1.00657	0.00023	0.000
ARO	ARO	17@12, 7@08	1.01162	0.00528	1.01187	0.00530	-0.002
ARO	ARO	22@12, 2@08	1.01230	0.00068	1.01254	0.00067	0.001
ARO	ARO	24@12	1.01272	0.00042	1.01296	0.00042	0.000

The eigenvalues in this table have been normalized to a cold critical eigenvalue of 1.0000.

The Delta k values reflect the reactivity worth of the control rod pulls described in Columns 1 through 3.

The last column compares the difference between the Edison and GE-calculated control rod worths.

The nomenclature is more fully described on the next page of this table.

Table 1, Continued
Comparison of In-Sequence Control Rod Worth
Quad Cities 1 Cycle 11

<u>Nomenclature:</u>

ARI = All Rods in the Group are at Position 00, the fully inserted position.

ARO = All Rods in the Group are at Position 48, the fully withdrawn position.

<u>Rod Position Indication:</u>

Unless noted by ARI or ARO, the rod positions in the first three columns in the table are interpreted as followed:

2@48 =	Two rods in the group are at Position 48 (fully withdrawn); the remainder are at Position 00 (fully inserted).
12@04, 12@00 =	Twelve rods in the group are at Position 04; Twelve rods in the group are at Position 00 (fully inserted). This nomenclature is only used if the position of all the rods in the group is specified.

Table 2
Comparison of In-Sequence Control Rod Worth
LaSalle 1 Cycle 5

			CECo Data		GE Data		CECo-GE
Group One (24 Rods)	Group Two (20 Rods)	Group Three (24 Rods)	Norm. Eigen.	Delta k	Norm. Eigen.	Delta k	% Delta k
ARI	ARI	ARI	0.95634	N/A	0.95655	N/A	-
ARO	ARI	ARI	0.99267	0.03633	0.99286	0.03631	0.002
ARO	2@48	ARI	0.99271	0.03637	0.99290	0.03635	0.002
ARO	4@48	ARI	0.99353	0.03719	0.99374	0.03719	0.000
ARO	6@48	ARI	0.99524	0.03890	0.99547	0.03893	-0.003
ARO	8@48	ARI	0.99779	0.04145	0.99802	0.04147	-0.002
ARO	10@48	ARI	0.99910	0.04276	0.99934	0.04279	-0.003
ARO	12@48	ARI	0.99971	0.04337	0.99995	0.04340	-0.003
ARO	14@48	ARI	1.00010	0.04376	1.00035	0.04380	-0.004
ARO	16@48	ARI	1.00465	0.04830	1.00483	0.04828	0.002
ARO	18@48	ARI	1.00731	0.05097	1.00747	0.05092	0.005
ARO	20@48(ARO)	ARI	1.01016	0.05382	1.01033	0.05378	0.004
ARO	ARO	12@04,12@00	1.01019	0.05385	1.01036	0.05381	0.004
ARO	ARO	24@4	1.01096	0.05462	1.01117	0.05463	-0.001
ARO	ARO	12@08,12@04	1.01116	0.05482	1.01139	0.05484	-0.002
ARO	ARO	17@08, 7@04	1.01258	0.05624	1.01286	0.05632	-0.008
ARO	ARO	22@08, 2@04	1.01399	0.05765	1.01432	0.05778	-0.013
ARO	ARO	24@08	1.01590	0.05956	1.01624	0.05969	-0.013
ARO	ARO	12@12,12@08	1.01619	0.05985	1.01655	0.06000	-0.015
ARO	ARO	17@12, 7@08	1.01854	0.06220	1.01896	0.06235	-0.015
ARO	ARO	22@12, 2@08	1.02022	0.06388	1.02061	0.06406	-0.018
ARO	ARO	24@12	1.02262	0.06628	1.02297	0.06642	-0.014

The nomenclature used in this table is the same as that described for Table 1.

Hot Critical Eigenvalues
Quad Cities 1 Cycle 10

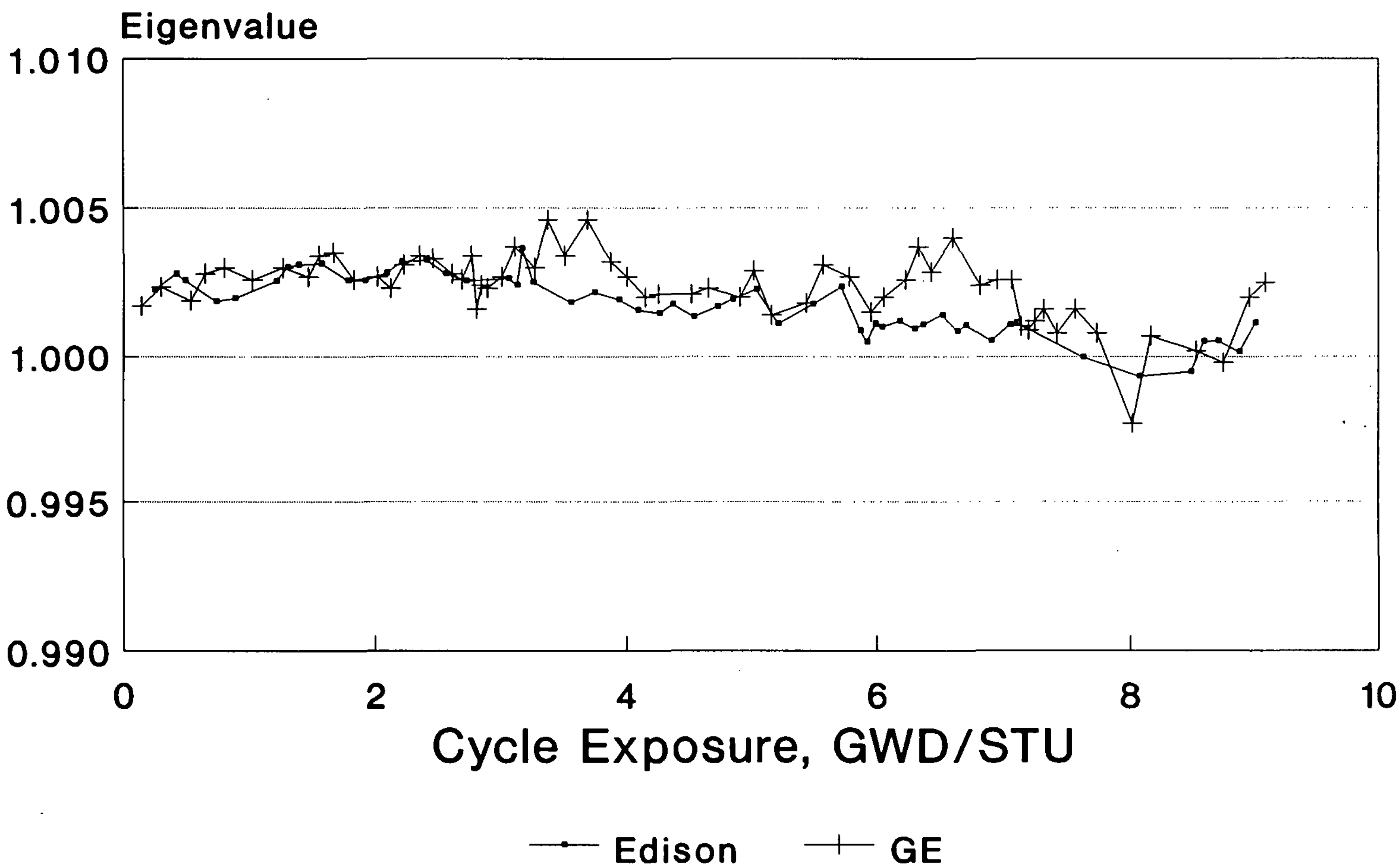

Figure 1
Comparison of Hot Critical Eigenvalues
Quad Cities 1 Cycle 10

Hot Critical Eigenvalues
LaSalle 2 Cycle 3

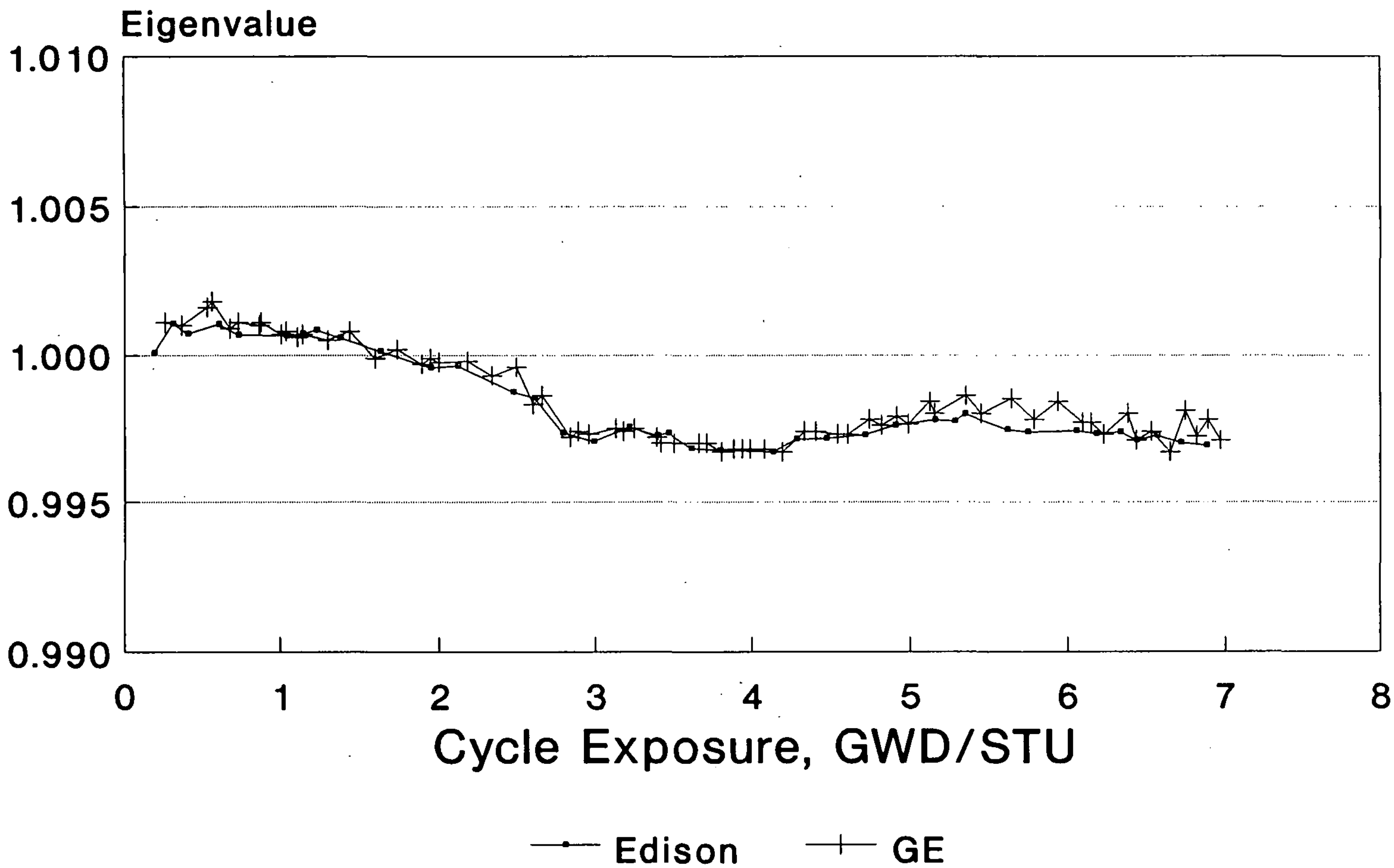

Figure 2
Comparison of Hot Critical Eigenvalues
LaSalle 2 Cycle 3

Attachment to

Commonwealth Edison Response to NRC Questions on

Edison Topical Report NFSR-0085

GE Document NEDE-30130-P-A

"Steady-State Nuclear Methods"

Pages 5-19, 5-20, 5-23, 5-24, 5-25

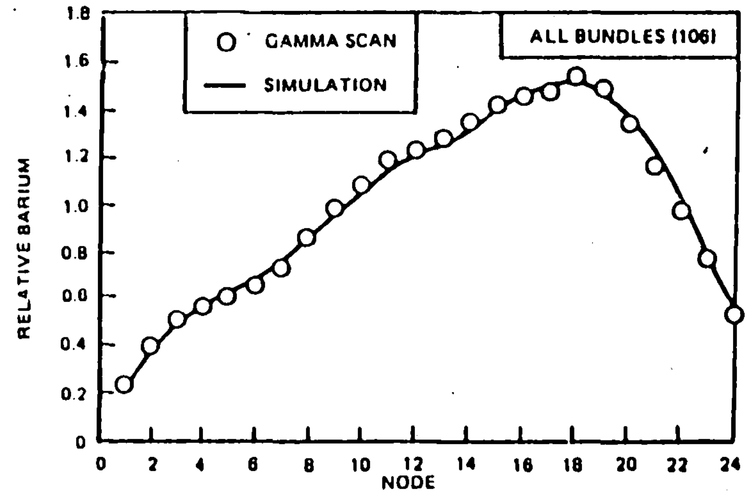

Figure 5-19. Hatch 1, Cycle 1: Axial Barium Distribution

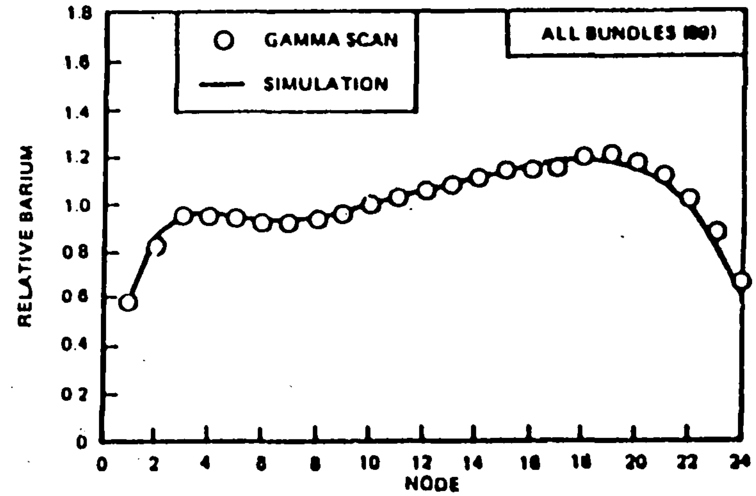

Figure 5-21. Quad Cities 1, Cycle 2: Axial Barium Distribution

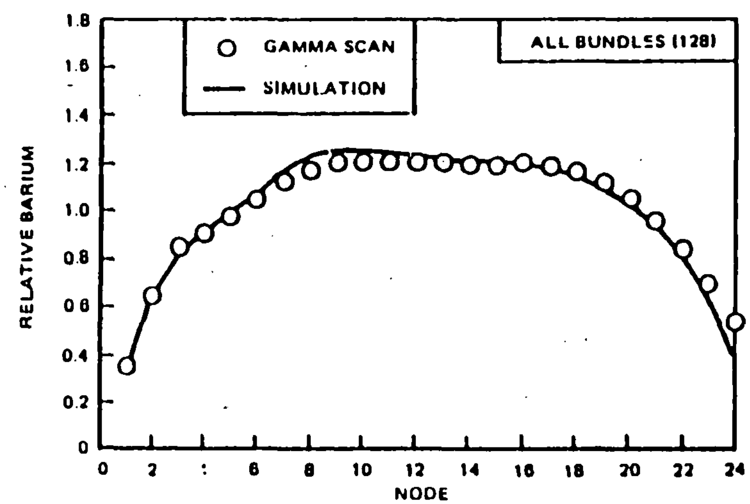

Figure 5-20. Hatch 1, Cycle 3: Axial Barium Distribution

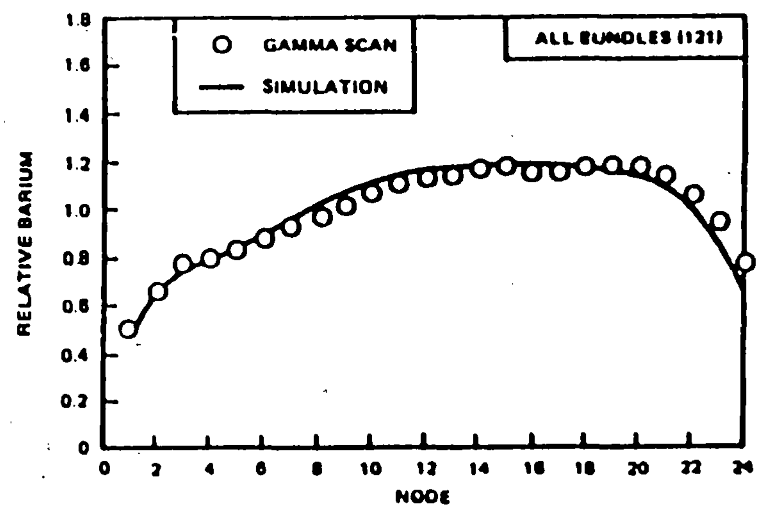

Figure 5-22. Quad Cities 1, Cycle 4: Axial Barium Distribution

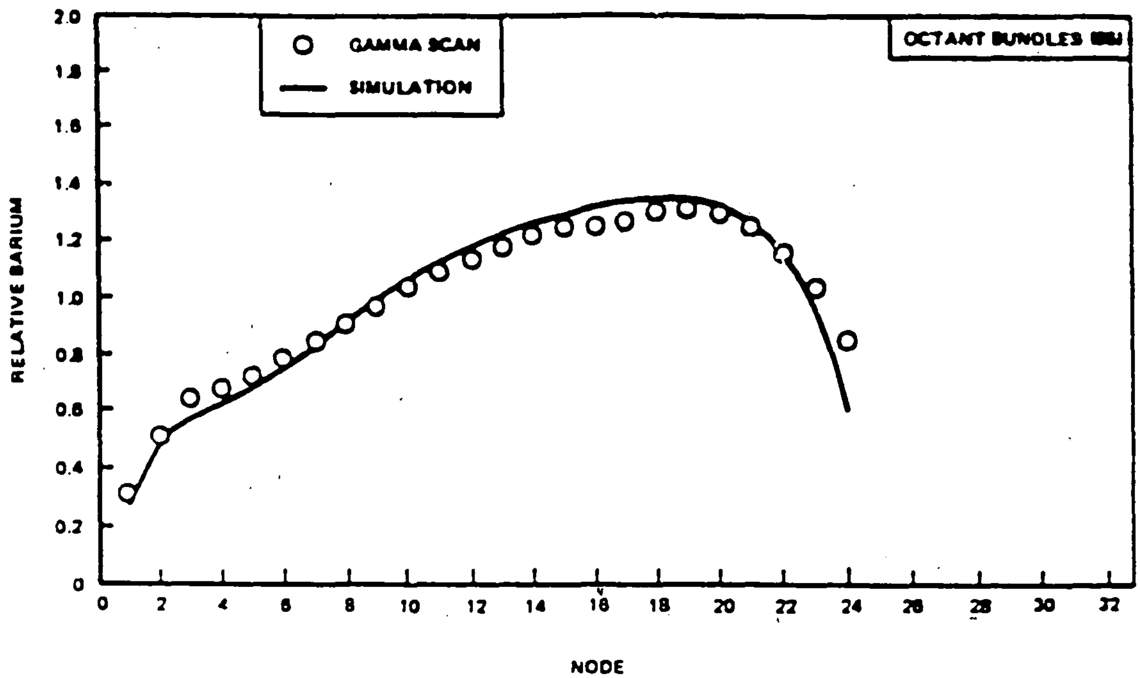

Figure 5-23. Quad Cities, Cycle 5: Axial Barium Distribution

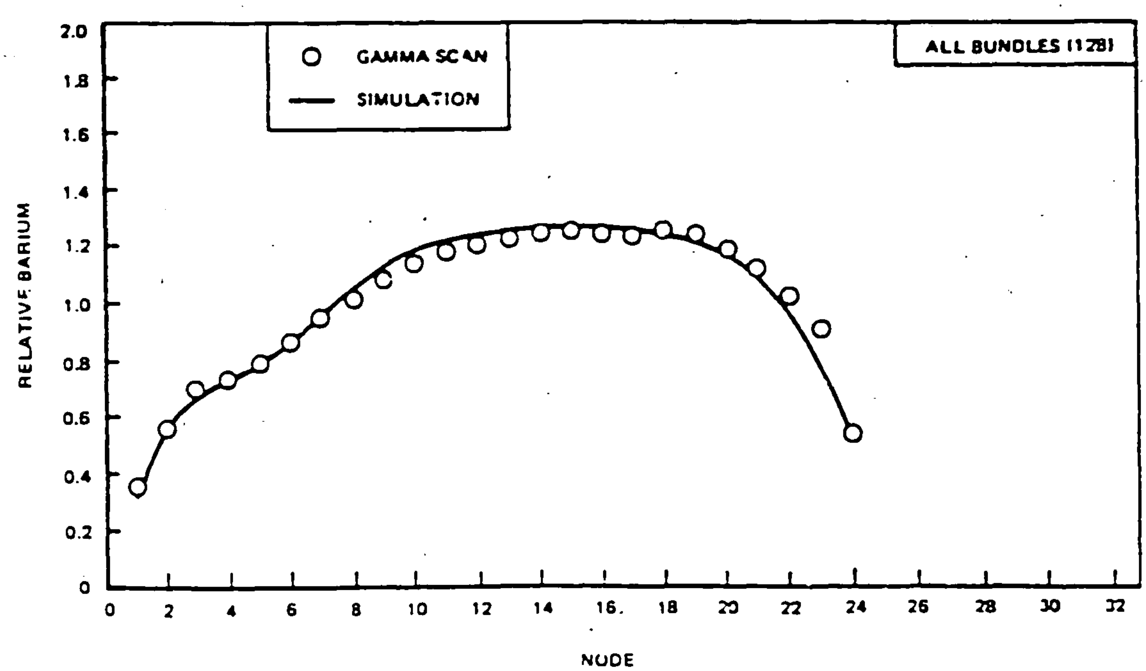

Figure 5-24. Millstone, Cycle 7: Axial Barium Distribution

				8.0*										
				8.2	3.0									
		0.8	5.7	4.3							3.6			
		6.7	4.3	3.5		·.9	0.0							
		7.7			0.3	−2.0		3.4						
	3.9		1.0	1.1			−1.0	−1.5	−1.8					
0.3	3.5		0.4	−2.8	−2.5	−0.4	1.1	0.7		2.4				
		−0.3		−1.4		1.3	−0.9		−0.8		0.3			
	0.7	−0.8		−4.4	−0.7	−1.1	−0.8	0.3	−0.8	−0.8		0.8		
2.8			−1.1	−1.2		−0.8	−2.8	−0.9			−1.1	−0.5	−0.3	
2.0	−0.2	−0.1	1.4	0.3	−0.9	−0.4	−1.0	−2.2	−1.3	−0.9	−0.5	−1.0	1.8	−1.5

*PERCENT DIFFERENCE BETWEEN CALCULATED AND MEASURED POWER

Figure 5-27. Quad Cities 1, End of Cycle 2: Bundle Power Comparisons

4.7*	4.6	4.8	5.6	1.3										
2.4	3.3	1.4	1.0	3.8	6.2									
1.0	2.8	1.6		6.2	8.1	9.7	7.1	4.7						
−1.0	1.7				3.2		4.7	4.8	3.9					
−0.1		2.5		−0.1		3.6		6.5	6.4	6.3				
−0.3	2.7		1.8	−1.7	0.8		1.9	0.3	4.8					
−0.5	−1.6	0.8	1.5	−3.8	−1.7	−0.9		−0.9						
−2.2	2.2		1.5		0.5	−1.5	0.5							
−4.3		0.1		−0.9		−2.2								
−2.1	2.2		0.1	−2.5	−0.3									
−5.5	−0.5	−1.4		0.1	−1.0			−2.8						
−1.6	0.6		−0.7											4.5
−3.1		−1.7												
−3.8	−0.1									1.5				
−3.7	−1.3							−0.2						1.4

16 *PERCENT DIFFERENCE BETWEEN CALCULATED AND MEASURED POWER

Figure 5-28. Quad Cities 1, End of Cycle 4: Bundle Power Comparisons

6.0*	9.8	9.1	8.8	8.8										
1.9	3.7	2.4	2.8	3.3	7.9									
3.8	3.7	3.0	2.8	5.3	5.1	15.7	15.1	12.8						
−0.5	0.3	1.9	0.7		3.9	6.0	6.1	7.9	10.2					
−1.1	−0.7	0.2	1.0	1.2	1.4	3.8	3.8	8.8	5.2	10.6				
−2.5	0.2	0.1	0.3	0.2	0.5	2.0	2.7	1.4	6.8					
−2.7	−1.4	0.5	−0.8	−0.3	0.3	0.6	0.9	2.5	1.7					
−4.2	−2.4	−2.3	−1.5	−2.3	1.2	0.5	1.8							
−2.8	−2.3	−0.9	−2.7	0.7	−1.8	−0.2	−1.6							
−6.3	−2.2	−2.7	−2.0	−4.5	−3.2									
−6.0	−1.8	−3.4	−2.9	−4.0	−5.0				2.5					
−3.3	−2.9	−4.6	−2.2											
−1.4	0.4	−1.2	−2.5											
−2.0	−0.2													
−1.0	−4.3													

*PERCENT DIFFERENCE BETWEEN CALCULATED AND MEASURED POWER

Figure 5-29. Quad Cities 1, End of Cycle 5: Bundle Power Comparisons