ENCLOSURE

GE Nuclear Energy

SAFETY EVALUATION OF LIMERICK 1, CYCLE 3

REVISED CORE LOADING PATTERN

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Introduction and Summary

Outage exams at the end of cycle 2 resulted in identification of 17 leaking bundles, 13 from fuel inserted in reload 1, cycle 2 and 4 from the initial core fuel. The failure mechanism was crud-induced localized corrosion (CILC). Based on visual exams of the initial core and reload 1 fuel, 48 initial core bundles and two reload 1 bundles were reconstituted, and 42 reload 1 bundles were inspected and cleared for reinsertion into the core for cycle 3. The other initial core and reload 1 bundles which were in cycle 2 will be discharged. The remainder of the core will be comprised of 152 previously discharged initial core fuel bundles, 224 fresh fuel bundles previously identified in Reference 1, and an additional 296 lower enriched fresh fuel bundles which were to have been loaded in the Limerick 2 initial core.

The purpose of this report is to provide a safety assessment of the revised core loading. The results of this assessment are that no fuel failures are expected to occur during cycle 3 and the revised core loading licensing analyses are bounded by the Reference 1 document. Therefore, the revised Limerick 1, cycle 3 core loading does not result in an unreviewed safety question or require changes to the plant Technical Specifications other than those required by the Reference 1 analysis and submitted to the NRC by Philadelphia Electric Company by their letter dated January 27, 1989.

Revised Core Loading Pattern

The revised core loading pattern for cycle 3 is shown in Figure 1. The cycle 1 initial core fuel to be inserted includes 116 0.94% enriched bundles and 36 1.63% enriched bundles previously discharged from the core and 48 reconstituted 2.48% enriched fuel bundles. Forty-four fuel bundles from reload 1, cycle 2 will be reinserted into the cycle 3 core. Two of these reload 1 bundles were reconstituted. The rest of the fuel to be loaded are fresh bundles.

Fuel Performance Evaluation

The reconstituted 2.48 wt% enriched initial core bundles and two reload 1 reconstituted bundles are comprised of fuel rods which were fully inspected to a corrosion visual standard of 2 or better. A sampling of rods from each rod lot in the other initial core and reload 1 fuel bundles to be reinserted also resulted in a visual standard of 2 or better. Corrosion visual standards indicate the amount of rod corrosion observed, with a lower number indicating less corrosion. Previous experience has shown that fuel with a visual standard of 4 or better will operate successfully for another cycle without failure.

By using a visual standard of 2 or better for the reinserted initial core and reload 1 fuel, Philadelphia Electric has conservatively provided additional margin to fuel failure. This margin is increased even further for the 44 reload 1 bundles to be reinserted by placing these bundles in the low duty region near the periphery of the core. Corrosion on the initial core and reload 1 fuel has been determined to be primarily due to the synergistic effect of less than expected corrosion resistance of the cladding and early cycle chemical transients in conjunction with increasing levels of copper input to the reactor water. Water chemistry for cycle 3 will be closely monitored to preclude recurrence. In addition, the fresh fuel has a more corrosion resistant cladding. Therefore, all acceptable fuel design limits are met and General Design Criterion 10 in 10CFR50 Appendix A is satisfied.

Reconstituted Fuel Nuclear Characteristics

The reconstituted 2.48 wt% initial core bundles and the two reload 1 reconstituted bundles incorporate donor rods which closely match the nuclear characteristics of the replaced rods in the original bundle. The donor rods, except in some cases the D4 rod, have the same initial U-235 enrichment and Gadolinia concentration as the replaced rods with rod exposures allowed to deviate by as much as 4 GWD/ST above the replaced rod to 2 GWD/ST below the replaced rod. Some D4 rods were replaced with the same U-235 enrichment in the majority of the rod but no Gadolinia. However, this rod is similar to the rod it replaced in nuclear characteristics, considering burnup. Lattice calculations performed for the hot operating state using the NRC approved GEMINI nuclear methods documented in Reference 2, Section 3 indicate that the nodal K-infinity of the reconstituted bundles is within 0.01 ΔK of the original bundle except for the top two nodes in the Gadolinia rods when these rods are exchanged. The K-infinity for these nodes is within 0.06 ΔK. The reconstituted bundles have local peakings which do not exceed 3% above the original local peaking. This difference is insignificant, therefore these bundles have the same nuclear characteristics as the original bundles and do not require separate enrichment and Gadolinia concentration distribution sheets. Therefore, the reconstituted bundles are considered to be the same as the original bundles in the technical discussions below and do not impact the conclusions presented in this safety evaluation.

Core Nuclear Analyses Results

The revised core nuclear analyses were performed with the GEMINI nuclear methods documented in Section 3 of Reference 2. The parameters of importance analyzed with this model for the revised loading pattern are the void reactivity coefficient, core cold shutdown margin and standby liquid control shutdown margin. The void reactivity coefficient is discussed in the section on MCPR operating limit; the other two are addressed below.

A. Core Cold Shutdown Margin

The core cold shutdown margin is determined by calculating the core effective multiplication factor, K-eff, with all control rods fully inserted except for the highest worth rod. The core is assumed to be in a cold, xenon-free condition to ensure that the calculated values are conservative. The Technical Specifications require that the analyzed margin to criticality be 0.38% ΔK/K. The analyzed minimum margin for the revised core loading was approximately 1.86% ΔK/K; therefore, the Technical Specifications requirements were met.

B. Standby Liquid Control System

The standby liquid control system (SLCS) is designed to provide the capability of bringing the reactor from a full power, minimum control rod inventory to a subcritical condition at any time in the cycle with the reactor in the most reactive xenon-free state. To conservatively bound design basis conditions, the SLCS shutdown margin analysis is performed at a cold, all-rods-out condition. The results of the analysis show that the minimum analyzed SLCS shutdown margin for the revised core loading is about 6.7% ΔK with a boron concentration of 660 ppm. Therefore, the licensing bases for the SLCS are met.

<u>MCPR Safety Limit</u>

The MCPR Safety Limit is the result of an NRC approved statistical, core-wide bounding analysis described in Section 4 of Reference 2. The result of this analysis is a value at which 99.9% of the rods in the core are expected to avoid boiling transition. As shown above, the fuel bundles to be loaded are approved standard fuel designs with standard manufacturing tolerances. This fuel will not impact instrument measurement or correlation uncertainty. Therefore, the current bounding analysis applies and the limit for the revised core loading is 1.07, which is the same as the limit for the Reference 1 core loading.

<u>MCPR Operating Limit</u>

The MCPR operating limit is determined by adding the change in CPR of the limiting anticipated operational occurrence to the MCPR safety limit. The limiting events identified in Reference 1 for Limerick 1 are Load Rejection Without Bypass (LRw/oBP), Feedwater Controller Failure (FWCF), Inadvertent HPCI Activation (IHA), and Rod Withdrawal Error (RWE). The impact on those events for the revised cycle 3 core loading pattern relative to the results documented in Reference 1 are given below. Both LRw/oBP and FWCF are pressurization events and are considered together.

A. Pressurization Events

Pressurization events are strongly dependent upon the dynamic void coefficient and the scram reactivity. Both of these parameters have been analyzed with the NRC approved GEMINI methods for the revised core loading and compared to values for the reference core loading.

The dynamic moderator void coefficient is a function of the void fraction in the moderator. The void distribution depends on the power and flow distribution and average core pressure. A comparison of the revised loading pattern relative to the loading pattern documented in Reference 1 shows that the void coefficient for the revised loading pattern is less negative throughout the cycle.

The scram reactivity is strongly dependent upon the axial power distribution existing prior to the scram. A more bottom peaked axial power distribution will result in an improved scram response. A conservative power-exposure iteration is used to determine this power shape throughout the cycle. For the revised cycle 3 loading pattern, the power distribution for the EOC-2000 MWD/ST and the EOC exposure points are significantly more bottom peaked than the Reference 1 loading pattern resulting in an increase in the scram reactivity.

Both the less negative void coefficient and enhanced scram response for the revised core loading will improve the results of the LRw/oBP and FWCF relative to the results reported in Reference 1.

B. Inadvertent HPCI Activation (IHA)

For this event, it is assumed that the high pressure coolant injection pumps are inadvertently started and the cold water injection results in a decrease in inlet subcooling and resultant increase in power. The most important parameter affecting IHA is the dynamic moderator void coefficient. As noted above, the dynamic void coefficient for the revised loading pattern is less negative than Reference 1. Therefore, the results for this event reported in Reference 1 are bounding for the cycle 3 revised core loading pattern.

C. Rod Withdrawal Error (RWE)

This event assumes that while operating in the power range in a normal mode of operation, the reactor operator makes a procedural error and withdraws the maximum worth control rod to its rod block position. The positive reactivity insertion results in a local power increase. This event was analyzed for the revised core loading pattern using GEMINI analysis methods. The response to the RWE for the revised core loading is bounded by that documented in Reference 1.

D. Summary

All of the limiting anticipated operational occurrences for the revised cycle 3 core loading pattern are bounded by the Reference 1 results.

Reactor Vessel Overpressure Protection

To assure that the reactor vessel upset condition pressure limit is not exceeded with new core loadings, a closure of a main steam line isolation valve (MSIV) with failure of the direct MSIV position switch scram is analyzed. Scram results from a signal of high neutron flux. This pressurization event has the same dependencies as the pressurization events analyzed to determine the MCPR operating limit. The lower negative void coefficient and enhanced scram response will also improve the pressure response of this event for the cycle 3 revised core loading relative to the Reference 1 result.

Core Stability

Core stability is a measure of the core oscillatory response to a perturbation. This parameter is strongly affected by the moderator void coefficient. The less negative void coefficient for the revised core loading improves the stability characteristics of this loading relative to the core loading given in Reference 1. In addition, the GE SIL-380 recommendations have been included in the Limerick 1 operating procedures.

Control Rod Drop Accident

Limerick 1 employs a banked position withdrawal sequence. As noted in Reference 3, an NRC approved statistical analysis for plants with this withdrawal sequence demonstrated that the peak fuel enthalpy in a RDA would be much less than the 280 cal/gm licensing limit even with a maximum incremental rod worth.

Loss-of-Coolant Accident

As documented above, the reconstituted bundles have essentially the same nuclear characteristics as the original bundles. Therefore, no change in MAPLHGR, peak clad temperature, or oxidation fraction is required for those bundles. The other bundles in the revised cycle 3 loading pattern have MAPLHGR values documented in the Technical Specifications and Reference 4. These values are reload independent.

Confirmation That the Proposed Technical Specifications Change Request Remains Valid for the Revised Core Loading

The revised core loading for Limerick 1, cycle 3 has been reviewed with respect to the criteria in 10CFR50.59, and it has been determined the new core configuration does not require a change to the plant Technical Specifications (other than those required by Reference 1 and submitted by Philadelphia Electric Company letter dated January 27, 1989) and does not constitute an unreviewed safety question for the following reasons:

1. As shown above, the Reference 1 report bounds plant operation with the revised loading pattern. Therefore, no additional Technical Specifications changes other than those required by Reference 1 and included in the January 27, 1989 submittal are needed.

2. The probability of occurrence and the consequences of an accident or malfunction of equipment important to safety is not increased above those analyzed in the FSAR due to the revised cycle 3 fuel loading configuration because, as described above, all of the fuel, including the reconstituted fuel bundles, to be loaded in the core are of standard fuel designs and meet all of the NRC approved licensing criteria.

3. For the same reason, the possibility of an accident or malfunction of a different type than analyzed in the FSAR does not result due to the revised cycle 3 loading pattern.

4. The margin to safety as defined in the Technical Specifications is not reduced by this change. As noted above, NRC approved methods were used to evaluate the cycle 3 revised core loading. The degree of conservatism in these methods has not changed. In addition, the Reference 1 analyses results will be used to establish the Limerick 1 reactor operating limits. As noted above, these limits are bounding for the revised core loading pattern.

<u>Conclusion</u>

The revised cycle 3 core loading pattern is bounded by the Reference 1 analytical results and does not result in an unreviewed safety question and 10CFR50 Appendix A General Design Criteria 10 is met for this core configuration.

References

1. "Supplemental Reload Licensing Submittal for Limerick Generating Station Unit 1 Reload 2, Cycle 3," 23A5926, Revision 0, October 1988

2. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-9, September 1988

3. "General Electric Standard Application for Reactor Fuel, United States Supplement," NEDE-24011-P-A-US-9, September 1988

4. "Basis of MAPLHGR Technical Specifications for Limerick Unit 1," NEDE-31401-P Errata and Addenda Sheet No. 1, October 1988

Figure 1
Limerick 1 Cycle 3
Revised Loading Pattern

60									1	1	1	1	1	1	1
58								1	1	8	4	8	4	1	4
56						1	1	4	8	5	8	8	5	5	8
54						1	4	8	5	2	8	2	6	8	2
52					1	1	8	6	7	5	8	6	7	6	8
50			1	1	1	4	5	8	6	9	6	8	6	9	6
48			1	4	8	5	3	8	8	6	8	3	7	6	3
46		1	4	8	6	8	8	3	6	8	3	8	6	8	8
44	1	1	8	5	7	6	8	6	3	6	7	6	9	6	7
42	1	8	5	2	5	9	6	8	6	9	5	8	6	9	6
40	1	4	8	8	8	6	8	3	7	5	3	8	7	6	2
38	1	8	8	2	6	8	3	8	6	8	8	2	6	8	8
36	1	4	5	6	7	6	7	6	9	6	7	6	3	6	7
34	1	1	5	8	6	9	6	8	6	9	6	8	6	9	6
32	1	4	8	2	8	6	3	8	7	6	2	8	7	6	3
	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29

<u>Fuel Type</u>	<u>Cycle Loaded</u>	<u>Number (Full Core)</u>
1 = BP8CRB094*	1	116
2 = BP8CRB163*	1	36
3 = BP8CRB248**	1	48
4 = BC320A	2	44
5 = BC318A	3	56
6 = BC322A	3	168
7 = BP8CRB163***	3	56
8 = BP8CRB248***	3	200
9 = BP8CRB094***	3	40
		764

* Reinserts
** Reconstituted
*** Originally scheduled for loading in Limerick 2, Cycle 1