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10CFR50.55a(a)(3)(I)

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555 - 0001

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Subject: Risk Informed Inservice Inspection Program
Request for Additional Information

References: (1) Letter from C. G. Pardee (EGC) to the US NRC, "Alternative to the ASME Boiler and Pressure Vessel Code Section XI Requirements for Class 1,2 and 3 Piping Welds Risk Informed Inservice Inspection Program," dated May 18, 2001

(2) Letter from W. A. Macon, Jr. (US NRC) to O. D. Kingsley (EGC), "LaSalle County Station, Units 1 and 2 – Request for Additional Information (TAC Nos. MB 1982 and MB 1983)," dated September 28, 2001

Exelon Generation Company (EGC), LLC, in Reference 1, submitted to the NRC, a proposed alternative to the existing American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," requirements for the selection and examination of Class 1 and 2 piping welds. The NRC in Reference 2, requested additional information to complete their review. The EGC responses to the questions are contained in Attachment 1.

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Rec'd 01/14/02

Should you have any questions concerning this letter, please contact Mr. William Riffer, Regulatory Assurance Manager, at (815) 415-2800.

Respectfully,

Charles G. Pardee
Site Vice President
LaSalle County Station

Attachment 1) Response to Request for Additional Information

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - LaSalle County Station

ATTACHMENT 1
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

NRC Question #1:

The Individual Plant Examination (IPE) staff evaluation report dated March 14, 1996, noted that review of the LaSalle County Station (LaSalle), Units 1 and 2, IPE was different from the review of most IPEs, as the LaSalle IPE was developed from previous staff studies. However, based on the IPE submittals from Zion, Quad Cities, and Dresden, the staff evaluation report had technical concerns regarding your methodology for common cause and human reliability analysis. The technical concern regarding common cause from the other IPEs was that the common cause failure parameters were lower than normally used. There were several human error concerns including, the treatment of pre- and post-initiator human actions and the treatment of time and other performance shaping factors. Please describe how these concerns have been addressed in the current LaSalle Probabilistic (PRA).

Exelon Generation Company, LLC (EGC) Response:

The issues raised by NRC staff in their safety evaluation for the LaSalle County Station Individual Plant Examination (IPE) regarding common cause failure (CCF) and human reliability analysis (HRA) are discussed below.

<u>CCF Issue 1</u>

The common cause failure analysis is not specific for LaSalle County Station. Generic common-cause failure databases were used in the development of the common cause factors for the Risk Methods Integration and Evaluation Program (RMIEP) analysis.

<u>Response</u>

The amount of plant-specific common cause data is quite small and alone is insufficient to provide a basis for the derivation of CCF probabilities. The latest common cause data compiled for the NRC by Idaho National Engineering and Environmental Laboratory (INEEL) in NUREG/CR-5497, "Common-Cause Failure Parameter Estimates," dated October 1998 were used in the LaSalle County Station Probabilistic Risk Assessment (PRA) update. This "generic" data is considered to be the best available generic data source for CCFs. Where available, plant-specific data has been used to supplement the generic data in the LaSalle County Station PRA updates for selected PRA important components.

<u>CCF Issue 2</u>

The Beta Factor common cause analysis is too conservative.

<u>Response</u>

The original PRA developed for the IPE used a simplified Beta Factor method for addressing common cause. However, the current CCF analysis is based on the more realistic Multiple Greek Letter (MGL) method and uses the latest INEEL work on common cause parameters from NUREG/CR-5497. This is considered to be a significant improvement in the PRA methods for LaSalle County Station.

HRA Issue 1

The RMIEP HRA results are expected to be non-conservative. Comparisons to other EGC PRAs show that the LaSalle County Station Human Error Probability (HEPs) are generally lower. It is expected that the difference in HEPs is due in large part to the methods used to incorporate the use of plant procedures in performance of the actions.

Response

The human reliability analysis in the current LaSalle County Station PRA has been completely modified as part of the update to incorporate extensive operator and training personnel interviews, as well as procedure reviews to verify the HEP assessments.

Evaluation of Post-Initiating Event HEPs
The Cause-Based Decision Tree (CBDT) method, developed by Garreth Parry, and others for the Electric Power Research Institute [EPRI], was used to quantify the likelihood of errors in detection, diagnosis, and decision-making. The diagnosis and decision-making HEP estimate was supplemented by also using the Time Reliability Correlation from NUREG/CR-1278, "Handbook of Human Reliability Analysis With Emphasis On Nuclear Power Plant Applications, Final Report," and Accident Sequence Evaluation Program (ASEP). Finally, the Technique for Human Error Rate Prediction (THERP) described in NUREG/CR-1278 was used to quantify errors associated with task execution. Compared with the method used in the base IPE for LaSalle County Station, the combination of the CBDT, Time Reliability Correlation, and THERP methods provides a more realistic basis for assessing post-initiator human actions.

Evaluation of Pre-Initiator HEPs
Pre-initiator HEPs were screened and evaluated using the ASEP Human Reliability Analysis Procedure, NUREG/CR-4772, "Accidents Sequence Evaluation Program Human Reliability Analysis Procedure," dated February 1987. Most of the pre-initiator HEPs were added to the PRA subsequent to the IPE.

The complete update of the HRA using the above methods, operator staff interviews and training observations was judged to resolve all issues related to inconsistency with other EGC sites and to remove the suspected non-conservatisms. The Dresden Nuclear Power Station, LaSalle County Station, and Quad Cities Nuclear Power Station had their HRAs updated to use the same methods and approaches for consistency.

HRA Issue 2

The HRA should reflect the content and employment of the plant procedures that are in effect at LaSalle County Station. The RMIEP analysis process did not specifically attempt to include all the operator actions in the LaSalle County Station Emergency Operating Procedures (EOPs) as implemented during the progression of an accident.

Response

The station has performed updated HEPs from the IPE based on plant-specific review of operating, abnormal, and emergency procedures plus operator interviews and training observations. Identification of post-initiator actions in the current LaSalle County Station PRA is based on review of past PRAs and current LaSalle County Station procedures, including LaSalle County Station General Abnormal (LGAs) and LaSalle County Station Operating Abnormal (LOAs), and interviews with operators and trainers. All key actions to accident sequence progression are modeled. The two primary reasons that not all LGA directed actions are specifically modeled are: 1) the action is subsumed by a broader action, and 2) the action does not impact the modeling of the progression of the accident sequence.

NRC Question #2:

Please provide a version number or other reference identifying the version of the PRA used to support the submittal.

EGC Response:

The LaSalle County Station PRA used to support the Risk-Informed Inservice Inspection (RI-ISI) submittal is referenced as follows:

LaSalle Nuclear Stations PRA Model 2000A Update:

Exelon Risk Management Document SA787, "LaSalle 2000A/B/C Core Damage Frequency (CDF) Models"

LaSalle Nuclear Stations PRA Model 2000A LERF Update:

Exelon Risk Management Document SA796, "LaSalle 2000A Large Early Release Frequency (LERF) Model"

NRC Question #3:

Please provide the baseline core damage frequency (CDF) and large early release frequency (LERF) of the version of the PRA used to support the submittal.

EGC Response:

The LaSalle County Station PRA used to support this submittal calculates a Core Damage Frequency (CDF) total of 5.9E-06 and a Large Early Release Frequency (LERF) total of 1.0E-06. These results are based on a Unit 2 model. The two units are essentially symmetrical and therefore the CDF and LERF values are considered appropriate for Unit 1 also.

NRC Question #4:

Page 18 of 32 states that, "[t]he RI-ISI risk impact calculations took credit for the IGSCC and FAC program inspections in that no change in risk was considered for those welds when a ASME Section XI examination was removed." The staff has found this acceptable as long as the elements are inspected under the intergranular stress corrosion cracking (IGSCC) or flow-accelerated corrosion (FAC) program. In other words, if an IGSCC Category B weld is not being inspected in the IGSCC program, and a ASME Section XI inspection on the weld is being discontinued in the risk informed – inservice inspection (RI-ISI) program, the impact of discontinuing that inspection should be reflected in the change in risk calculations. The failure frequency used may exclude the IGSCC or FAC degradation mechanism. Please confirm that the change in risk calculation reflect the impact of all discontinued inspections (in medium and high segments) or provide results demonstrating that the change in risk, including these discontinued inspections continue to meet all the Electric Power Research Institute (EPRI) change in CDF and LERF criteria.

EGC Response:

In the LaSalle Risk-Informed Inservice Inspection (RI-ISI) analysis, welds having flow-accelerated corrosion (FAC) as the only damage mechanism or intergranular stress corrosion cracking (IGSCC) as the only damage mechanism were removed from the population for element selection, i.e., no RI-ISI exams were selected for any of these elements. The FAC-only and IGSCC-only elements were assumed to be addressed by their respective augmented inspection programs. In the original RI-ISI risk impact calculations, removal of American Society of Mechanical Engineers (ASME) Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," exams from these welds was not included in the delta risk calculations. The results of these delta risk calculations for both the Markov model and a bounding methodology, as discussed in Section 3.8.1 of Reference 1, were provided in Tables 9 and 10 of Reference 1.

As a result of this RAI, new calculations have been performed which include the impact of eliminating the FAC-only and IGSCC-only welds from the RI-ISI program. Those Section XI examinations eliminated would result in minor changes in risk for those specific welds and would contribute to the overall delta risk that was quantified for the systems. The new delta risk calculations include the impact of all discontinued ASME Section XI inspections in all high, medium and low risk segments. Both Markov model calculations and bounding calculations were made for the systems that had FAC-only or IGSCC-only welds within the RI-ISI scope. The results of these new calculations are presented in Tables RAI-1 and RAI-2. In comparison to Tables 9 and 10 of Reference 1, the ECCS, FW, HPCS, RCS and RWCU systems' ΔCDF and ΔLERF values are revised for Unit 1 and the CRD, ECCS, FW, RCS and RWCU values are revised for Unit 2. The individual changes in CDF and LERF for each impacted system, as well as the overall total ΔCDF and ΔLERF were found to be negligible with respect to the acceptance criteria. Tables RAI-1 and RAI-2 show that all the incremental CDF and LERF acceptance criteria are met.

NRC Question #5:

Code requirements listed on page 2 of 5 of the submittal need clarification. Examination Category B-F should include Items B5.130, B5.140 and B5.150. The submittal lists Items B5.10, B5.130 and B5.150. Thus Item B5.10 is included instead of B5.140. Similar clarifications are needed for Items in Category B-J, C-F-1 and C-F-2.

EGC Response:

The correct examination item numbers are B5.10, B5.130, B5.140, B5.150, B9.11, B9.12, B9.21, B9.22, B9.31, B9.32, B9.40, C5.11, C5.12, C5.41, C5.42, C5.51, C5.52, C5.70, C5.81 and C5.82.

NRC Question #6:

Please clarify that the RI-ISI program be updated every 10 years and submitted to the NRC consistent with the current ASME XI requirements.

EGC Response:

The RI-ISI program is an alternative to the requirements of ASME Section XI requirements for Class 1 and 2 piping welds implemented through the use of a relief request in accordance with 10 CFR 50.55a(a)(3)(i). Therefore, a relief request for implementation of a RI-ISI program during subsequent 10-year inservice inspection intervals will be submitted concurrent with the update to the latest edition and addenda of the Code every ten years in accordance with ASME 10 CFR 50.55a(g)(4)(ii).

NRC Question #7:

Please describe under what conditions will the RI-ISI program be resubmitted to the NRC before the end of any 10-year interval.

EGC Response:

It is our intent to resubmit the RI-ISI program to the NRC before the end of a 10-year interval if there is a significant change in the RI-ISI methodology. For other program changes it is not our intent to resubmit the program.

The RI-ISI program will be maintained as a living program and updated consistent with EPRI TR 112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure." Changes that could impact the RI-ISI program include major changes to the LaSalle County Station PRA or changes to weld selection. Our Risk Management program requires a review of past applications following a PRA update. This requirement will be applied to the RI-ISI program. If the review determines that a change to the RI-ISI program is required, the change would be performed consistent with the EPRI methodology. Likewise, a change to the welds selected would cause a revision to the RI-ISI program consistent with the EPRI methodology. These changes to the RI-ISI program would not be resubmitted to the NRC.

Additionally, requirements for RI-ISI program maintenance are being developed by EPRI. The EPRI "Living Program Criteria" document is expected to be published by the end of 2001.

NRC Question #8:

Page 10 of your submittal discuses the criteria for engineering evaluation and additional examinations if unacceptable flaws or relevant conditions are found during examination. Please clarify that the evaluation will include other elements in the segment or segments subject to the same root cause conditions. Please clarify how will the elements be selected for additional examinations. Specifically, please verify that the elements will be selected based on the root cause or damage mechanism and include high risk significant as well as medium risk significant elements (if needed) to reach the required number of additional elements, not limited to elements with the same or higher failure potential.

EGC Response:

EPRI TR-112657, Section 3.6.6.2 states the following regarding additional examinations, "Additional examinations will be performed on these elements up to a number equivalent to the number of elements required to be inspected on the segment or segments initially. If unacceptable flaws or relevant conditions are found similar to the initial problem, the remaining elements identified as susceptible will be examined. No additional examinations will be performed if there are no additional elements identified as being susceptible to the same service related root cause conditions or degradation mechanism." LaSalle County Station intends to use the additional examination criterion outlined in Subarticle-2430 of Code Case N-578-1. The Code Case does not deviate from the EPRI TR-112657 regarding additional examinations; rather, N-578-1 builds on the EPRI TR by providing additional details. Specifically, for High and Medium Risk category piping structural elements (i.e., Risk Group Categories 1 through 5 as defined in Table I-8 of N-578-1), LaSalle County Station will use the following criteria:

(a) Examinations performed that reveal flaws or relevant conditions exceeding the referenced acceptance standards shall be extended to include additional examinations. The additional examinations shall include piping structural elements with the same postulated failure mode and the same or higher failure potential.

  (1) The number of additional elements shall be the number of piping structural elements with the same postulated failure mode originally scheduled for that fuel cycle.

  (2) The scope of the additional examinations may be limited to those high safety significant piping structural elements (i.e., Risk Group Categories 1 through 5) within systems, whose material and service conditions are determined by an evaluation to have the same postulated failure mode as the piping structural element that contained the original flaw or relevant condition.

(b) If the additional required examinations reveal flaws or relevant conditions exceeding the referenced acceptance standards, the examination shall be further extended to include additional examinations.

  (1) These examinations shall include all remaining piping elements whose postulated failure modes are the same as the piping structural elements originally examined.

  (2) An evaluation shall be performed to establish when those examinations are to be conducted. The evaluation must consider failure mode and potential.

(c) For the inspection period following the period in which the original examination discovering the flaw or relevant condition was completed, the examinations shall be performed as originally scheduled.

LaSalle County Station believes that the rules for additional examinations described above are consistent with the intent of Code Inquiry IN00-010a which was discussed at the ASME Section XI Risk Based Working Group meeting and approved at the recent ASME Section XI Inquiry Session. However, it should be noted that the aforementioned Code Inquiry provides clarification for Code Case N-577/N-577-1 and not specifically for Code Case N-578/N-578-1.

Consistent with the intent of IN00-010a, if there are not enough high safety significant elements (i.e., in the same and higher "Risk Group Categories") with the same postulated failure mode, lower safety significant elements (i.e., in lower "Risk Group Categories"-other than Risk Group Categories 6 and 7) with the same postulated failure mode will be selected such that the number of additional elements is at least equal to the number of elements with the same postulated failure mode originally scheduled for that fuel cycle.

Thus, the description of additional examination requirements described in Section 3.5, page 10 of Attachment 2, is expanded by the additional examination requirements described in this response.

NRC question #9:

Page 19 of your submittal states that Unit 2 is in the second period of the second interval. The submittal further states that 43 percent of the ASME XI examinations have been completed. Please clarify whether the remaining 57 percent of the examinations will be from the ASME XI or the RI-ISI program. Please also specify, also, which 57 percent of the RI-ISI examinations will be performed as well as the basis for that selection. A similar question also applies to Unit 1.

EGC Response:

The table provided in Section 4, page 19 of the submittal was intended to provide the current examination distribution of welds selected for the RI-ISI program. For Unit 2, twenty-six (26) of the welds that have been selected for examination in the RI-ISI program had been previously examined in the second interval under the current ASME Section XI program. It is not our intention to reexamine these twenty-six (26) welds during the second interval. The remaining welds selected for the RI-ISI program will be examined per the table provided in section 4, page 19 of the submittal, using the RI-ISI examination methods. Therefore by the end of the second period of the second interval, for Unit 2, a minimum of thirty (30) of the RI-ISI program selected welds, will have been examined. Twenty-six (26) of these welds will have been examined using the current ASME Section XI program and at least four (4) of these welds will have been examined using the RI-ISI program. At the end of the second interval all RI-ISI program selected welds will have been inspected, using either the current ASME Section XI program (26 welds) or the RI-ISI program (34 welds). Please note that the above discussion does not include the impact of four (4) socket welds included in the RI-ISI examination population. These four (4) welds require examination each refueling outage. Inspection of these welds will continue on a refueling outage frequency under the RI-ISI program.

This approach and methodology for selection of future welds inspection outlined above is also appropriate for Unit 1 using the Unit 1 specific values provided in the table in Section 4, page 19 of the submittal.

Table RAI-1

Impact of RI-ISI on CDF and LERF for LaSalle County Station Unit 1 Systems

System	Δ CDF Events/Reactor-Year			Δ LERF Events/Reactor-Year		
	Bounding ΔCDF for All Welds	**Realistic ΔCDF - Markov Model**	**Acceptance Criterion**	**Bounding ΔCDF for All Welds**	**Realistic ΔCDF - Markov Model**	**Acceptance Criterion**
CRD	5.16E-11	2.92E-11	<1.00E-07	4.64E-12	2.62E-12	<1.00E-08
ECCS	3.19E-11	-1.37E-10	<1.00E-07	-1.34E-10	-1.78E-10	<1.00E-08
FW	9.10E-10	1.94E-11	<1.00E-07	2.15E-10	-2.84E-10	<1.00E-08
HPCS	-6.27E-12	-9.15E-11	<1.00E-07	-1.55E-11	-5.90E-11	<1.00E-08
MS	3.68E-10	2.16E-10	<1.00E-07	2.70E-10	1.60E-10	<1.00E-08
RCIC	2.85E-10	1.60E-10	<1.00E-07	2.32E-10	1.43E-10	<1.00E-08
RCS	6.35E-09	3.44E-09	<1.00E-07	1.11E-09	5.92E-10	<1.00E-08
RWCU	-2.01E-11	-1.14E-11	<1.00E-07	-2.15E-12	-1.21E-12	<1.00E-08
Total	7.97E-09	3.62E-09	<1.00E-06	1.68E-09	3.75E-10	<1.00E-07

Table RAI-2

Impact of RI-ISI on CDF and LERF for LaSalle County Station Unit 2 Systems

System	Δ CDF Events/Reactor-Year			Δ LERF Events/Reactor-Year		
	Bounding ΔCDF for All Welds	**Realistic ΔCDF - Markov Model**	**Acceptance Criterion**	**Bounding ΔCDF for All Welds**	**Realistic ΔCDF - Markov Model**	**Acceptance Criterion**
CRD	1.03E-10	5.81E-11	<1.00E-07	1.14E-11	6.41E-12	<1.00E-08
ECCS	1.09E-10	-2.09E-11	<1.00E-07	-2.77E-10	-2.20E-10	<1.00E-08
FW	2.32E-09	8.92E-10	<1.00E-07	1.61E-09	6.39E-10	<1.00E-08
HPCS	1.40E-11	3.66E-12	<1.00E-07	-4.65E-14	-1.01E-12	<1.00E-08
MS	3.27E-10	1.91E-10	<1.00E-07	2.97E-10	1.79E-10	<1.00E-08
RCIC	2.43E-10	1.43E-10	<1.00E-07	2.35E-10	1.42E-10	<1.00E-08
RCS	3.73E-09	2.09E-09	<1.00E-07	7.47E-10	4.20E-10	<1.00E-08
RWCU	5.23E-10	2.96E-10	<1.00E-07	5.18E-10	2.93E-10	<1.00E-08
Total	7.16E-09	3.61E-09	<1.00E-06	3.41E-09	1.67E-09	<1.00E-07

For both tables above, a positive value indicates a risk increase, while a negative value denotes a risk decrease.