ILLINOIS POWER COMPANY

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L17-82(09-03)-6

September 3, 1982

Mr. Thomas M. Novak
Assistant Director for Licensing
Division of Licensing, ONRR
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Novak:

Submittal of LRG-II Position Papers
and Summary of Outstanding Issues

The LRG-II Working Group has submitted previously five volumes of position papers that discussed each of the 56 designated LRG-II issues. Your August 17, 1982 letters to the LRG-II Executives provide a tabular summary of the status of those 56 issues. As your letter indicates, 40 issues have been resolved, 11 remain unresolved, and five have been determined to be non-generic (to be addressed in individual LRG-II plant reviews). We are in complete agreement with your tabulation with respect to the issues which are indicated as resolved.

The purpose of this letter is to discuss each of the outstanding issues from the standpoint of whether the next action is that of the LRG-II or of NRR, and to transmit Volume VI of the position papers covering those issues that required LRG-II action. In addition, Volume VI contains two new common issues, one identified by the staff and one identified by the Working Group. These resultant 13 issues are the following:

1. 9-RSB - Long-Term Operability of ECCS Pumps - It is our understanding that review of this issue has been transferred from the Reactor Systems Branch to the Equipment Qualifications Branch. The LRG-II position on this subject has not been discussed in the Clinton or Perry SER or SSER/1 but the subject is briefly discussed in the Grand Gulf SSER/2 (Section 3.10, p.3-9). Our position on this issue was provided in Volume V of the Position Papers (May 17, 1982), and we believe it is sufficiently complete to resolve this issue. Following EQB review of this information we are prepared to meet with the reviewers, if necessary, to complete resolution of the issue.

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2. 2-CPB - Seismic and LOCA Loads on Fuel - General Electric has just completed a special study, sponsored by LRG-I and LRG-II members, to resolve this issue.

The report, NEDE-21175-P, Supplement 3, "BWR/6 Fuel Assembly - Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss of Coolant Accident (LOCA) Loadings", has been submitted to the staff. The LRG-II position paper in Volume VI references and incorporates this report.

3. 3-CPB - Channel Box Deflection - Our position on this issue was outlined in Volume V, and, we understand, has been reviewed by the staff, along with EPRI NP-2483, "An Assessment of BWR Fuel Channel Lifetimes." We believe no further staff or LRG-II action is required, and we are awaiting the documentation of the issue resolution. This is a very significant issue, and we appreciate the staff review efforts to effect resolution.

4. 6-CPB - Inadequate Core Cooling - Information regarding this issue was provided in Volume V. In that position paper it was indicated that the LRG-II participants were members of the BWR Owners Group for TMI Activities and that this larger organization would direct the study of inadequate core cooling instrumentation and obtain closure for this issue. Since the LRG-II is not directly in charge of this program, it is believed that for the purpose of documenting the status of this issue, it should be stated that LRG-II will no longer address this issue.

5. 2-CSB - Hydrogen Generation and Control - Information regarding this issue was provided in Volume V. In that position paper it was indicated that the LRG-II participants were members of the Hydrogen Control Owners Group for Mark III Containments and that this organization would direct the hydrogen control program and support closure for this issue through plant-unique submittals. Since LRG-II is not directly in charge of this program, it is believed that for the purposes of documenting the status of this issue, it should be stated that LRG-II will no longer address this issue.

6. 1-ICSB - Failure in Vessel Level Sensing Lines Common to Control and Protection Systems - Information relating to this issue was provided in Volume V. It is our understanding that this information has been reviewed and found acceptable, and that resolution can be documented in the next LRG-II plant SSER.

7. 2-HFS - ATWS Emergency Operating Procedure and GE Reactivity Control Guidelines - Information relating to this issue is provided in Volume VI. It references

Revision 2 of the Emergency Procedure Guidelines for BWR/1-6 which has been submitted to the NRC by the BWR Owners Group. It is our understanding that approval will be forthcoming, and this should resolve the issue for LRG-II.

8. 3-HFS - Common Reference for Reactor Vessel Level Measurement - This issue has been resolved for Clinton. The information resolving this issue for LRG-II is provided in Volume VI.

9. 1-CHEB - Reactor Coolant Sampling - The information supplied in Volume IV addresses the narrow LRG-II issue of sampling points. However, as we understand the view of CHEB, it would be more meaningful to evaluate reactor coolant sampling in light of the system description and operating characteristics. The information in 3-CHEB (Volume VI) provides an indication of what will be done with the samples taken at the locations described in 1-CHEB and 2-CHEB in further support of the resolution of those two issues. Beyond that, we believe the issue becomes plant-specific.

10. 2-CHEB - Suppression Pool Sampling - This issue has the same status and considerations as 1-CHEB above.

11. 3-CHEB - Estimation of Fuel Damage From Coolant and Pool Sampling - Information on this issue, based on the interim acceptance of the Fermi-2 procedure, is contained in Volume VI. We believe this should resolve 3-CHEB on the same basis as was acceptable for Fermi-2.

12. 5-ASB - CRD System Vessel Inventory Make-UP Rate - This is a new issue proposed by the LRG-II. Section 8.1 of NUREG-0619 recommends a special two pump CRD make-up test to demonstrate additional fire protection capability. The LRG-I presented (letter dated March 26, 1982) a special discussion showing that such capability was not appropriate or necessary. The staff accepted this position as documented in the Susquehanna SSER/3 (Sec. 4.6.2). The same considerations apply to LRG-II plants, and the conforming position paper to resolve this issue is presented in Volume VI.

13. 4-MEB - Kuosheng Incore Instrument Tube Break - This is a new issue proposed by the staff. As a consequence of an instrument tube failure at Kuosheng, GE has proposed fixes for all BWR/6 plants. Information relating to this issue is presented in Volume VI. The resolution was accepted for Grand Gulf, as inferred from staff presentations at the August 11 ACRS Subcommittee meeting and August 12 ACRS full Committee meeting.

In summary, Volume VI of the LRG-II Position Papers presents information on Issues 2-CPB, 2-HFS, 3-HFS, 3-CHEB, 5-ASB, and 4-MEB. We believe there is presently enough information submitted and reviewed on 9-RSB, 3-CPB, and 1-ICSB for documenting resolution.

There are two additional matters that relate to the updating of the table in your August 17 letter. Item 1-HFS, Special Low-Power Testing Program, was resolved (as indicated in the table), and the documentation has been provided in the Perry SSER (Sec. 14). Secondly, Items 1-SEB and 2-SEB, relating to Combination of Loads and Fluid/Structure Interaction do not have clear Perry SER references. Both the staff and LRG-II agree that these issues have been resolved, but existing SER/SSER documentation does not clearly acknowledge resolution. We had been advised that such acknowledgement would be in the Perry SSER/1, but it apparently was not available at publication time. For purpose of accountability we would appreciate these items being closed vis-a-vis specific LRG-II referencing at the next SSER opportunity.

We appreciate the accurate status of LRG-II efforts presented by your August 17 letter, and would hope to be in a position to receive an update on these remaining 14 issues by the first part of October. To that point, if it would be useful for another meeting with the LRG-II plant project managers and particular reviewers prior to that time, please let us know.

Sincerely,

D. L. Holtzscher

D. L. Holtzscher
Chairman
LRG-II Working Group

DLH/wp/lt

Attachments: LRG-II Position Papers, Volume VI

cc: C. O. Thomas, Chief-Standardization & Special Projects Branch, NRR
E. M. Buzzelli, CEI
W. J. Reed, GSU
R. Artigas, GE
R. S. Boyd, KMC
J. D. Richardson, MP&L
J. Palomo, Hydroelectrica Espanola
P. Powell, WPPSS

SEPT. 3, 1982

LRG-II POSITION PAPERS

VOLUME VI

TECHNICAL DISCUSSIONS AND RESOLUTIONS OF SIX LRG-II ISSUES. THE POSITIONS TAKEN IN THESE PAPERS WILL BE REFERENCED IN LRG-II PLANT OL APPLICATIONS.

NOTE: VOLUME VI CONTAINS REVISED RESPONSES ON FOUR STILL-OUTSTANDING LRG-II ISSUES, AND PROVIDES POSITION PAPERS ON TWO NEW ISSUES.

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LRG-II POSITION PAPERS - VOLUME 6

LRG-II Position Paper

September 3, 1982

Revision 1

2-CPB

COMBINED SEISMIC AND LOCA LOADS ANALYSIS ON FUEL

ISSUE:

The evaluation of the BWR/6 fuel assembly design for the combination of seismic and LOCA loads was presented in the General Electric Company document NEDE-21175-P*. The evaluation, prepared by GE and approved by the NRC, did not include the effects of containment hydrodynamic loadings and did not discuss the effects of potential fuel lift. As a result, this evaluation model has been approved for low acceleration loads only and is not approved for loads that may result from impact after fuel lift.

NRC consultants have performed a preliminary analysis which indicated that a lift condition may be possible. Applicants must describe the methods used to analyze fuel assembly dynamic response due to seismic and LOCA loads including hydrodynamic loads and potential loads from impact. Applicants must also demonstrate design margin.

LRG-II POSITION:

LRG-II participants and several other near-term operating license utilities have sponsored the preparation of Amendment 3 to the document NEDE-21175-P, "BWR/6 Fuel Assembly - Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings". Amendment 3 discusses both the potential for and impact of fuel lift during a combined SSE and LOCA event. This amendment is separated into two sections. Section 1 provides a detailed discussion of an analytical model developed by General Electric which is used to calculate fuel dynamic response during a combined SSE and LOCA event and gives the results of the bounding calculation for such a response. Section II of the amendment analyzes the capablity of the General Electric BWR 4/5/6 fuel assembly structurally to withstand the dynamic loadings, assuming bounding input accelerations. The results of these analyses show the General Electric BWR 4/5/6 fuel response is within acceptable limits and the dynamic and impact loading capability is conservatively in excess of the predicted loads. The results of the analysis documented in this report bound the results calculated for the individual plants for which the report was prepared. The plant unique results will be contained in the appropriate section of the plant FSAR.

Amendment 3 was submitted to the NRC via the letter from J. F. Quirk (GE) to H. Bernard (NRC) dated July 27, 1982.

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*NEDE-21175-P is dated November 1976

Amendment No. 1 is dated April 1977

Amendment No. 2 is dated September 1977

LRG-II Position Paper

September 3, 1982

Revision 1

2-HFS

EMERGENCY PROCEDURES REACTIVITY CONTROL GUIDELINES

ISSUE:

Develop a generic reactivity control guideline which can be utilized for preparing an emergency operating procedure for an anticipated transient without scram (ATWS) event.

LRG-II POSITION:

The LRG-II participating utilties have sponsored, through the BWR Owners' Group for TMI Activities, the development of emergency procedure guidelines which can be utilized for the preparation of emergency operating procedures as specified in TMI Action Plan Item I.C.1. Revision 2 to the Emergency Procedure Guidelines for BWR/1-6 includes "Contingency #7-Level/Power Control". This contingency provides the guidance for developing an emergency procedure for reactivity (power) control, including failure to scram.

Revision 2 of the Emergency Procedure Guidelines for BWR/1-6 was submitted to the NRC for review and approval via the letter from T. J. Dente (Chairman BWR Owners' Group) to D. G. Eisenhut (NRC) dated June 1, 1982.

LRG-II Position Paper

September 3, 1982

Revision 1

3-HFS

COMMON REFERENCE FOR REACTOR VESSEL LEVEL MEASUREMENT

ISSUE:

Provide a common reference level for all reactor water level instruments per the requirement contained in TMI Action Plan Item II.K.3.28.

LRG-II POSITION:

LRG-II participants jointly sponsored through the BWR Owners' Group an evaluation of providing a common reference level for vessel level instrumentation. This evaluation was submitted via the letter from D. B. Waters, Chairman of BWR Owners' Group, to D. G. Eisenhut, Director of Licensing (NRC), dated December 29, 1980. This evaluation concluded that the current BWR water level indication system is fully adequate to allow plant operators to respond properly under all postulated reactor conditions and that there are no required design changes based on any plant safety considerations.

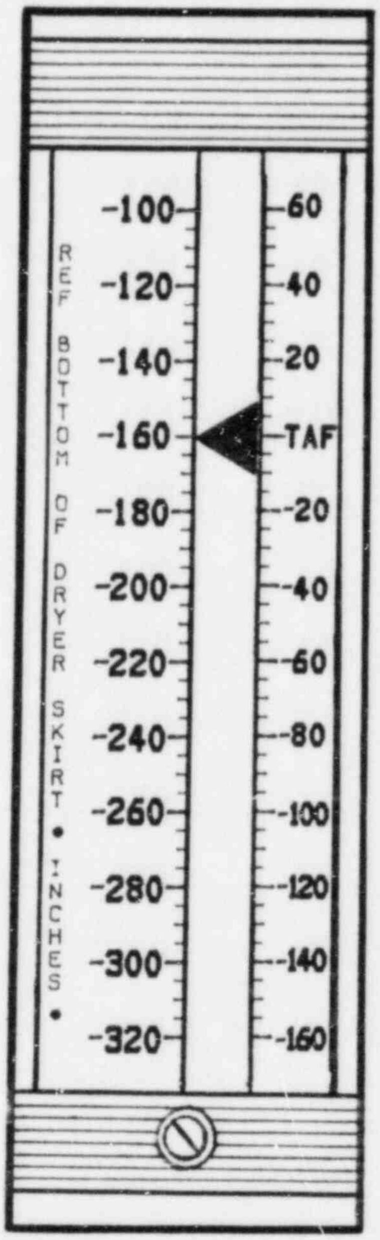
The above evaluation was rejected by the NRC as explained in the letter from D. G. Eisenhut to D. B. Waters, dated April 6, 1981. In this letter, the NRC stated its position that "...all level

instruments should be referenced to the same point. The selection of the reference point for any specific reactor has been left to the discretion of the licensee..."
In light of this situation, LRG-II has taken the position to conform with the guidance contained in the above referenced April 6, 1981 letter and has selected the common reference point to be at the bottom of the steam dryer skirt. This reference point was the reference point used for all RPV level ranges except the fuel zone instruments.

The fuel zone instruments will have dual numerical scales with the left-hand scale readings corresponding to the common instrument zero plane and the right-hand scale readings corresponding to the classical BWR fuel calibration which is referenced to the top of active fuel. A life-size copy of the dual scales mounted on a GE Type 180 instrument is provided on Figure 3-HFS.

This dual indicating scale for the fuel zone instrumentation is not confusing to the operator because it is secondary to the numerical scale which indicates "common" water level information. However, this arrangement also retains for the operator ready reference to actual fuel zone levels. Appropriate training for the use of the dual scale reactor water level indicators, including upgrading of training and maintenance documents and procedures, will be accomplished prior to fuel loading.

FIGURE 3-HFS



Main Control Room
Fuel Zone Water Level Indicator
Life-Size Dual Scales
Mounted on GE Type 180 Instrument

LRG-II Position Paper

September 3, 1982

Revision 1

3-CHEB

ESTIMATION OF FUEL DAMAGE FROM POST-ACCIDENT SAMPLES

ISSUE:

A procedure for relating post-accident radionuclide concentrations in reactor coolant and suppression pool samples should be developed.

LRG-II POSITION:

It is the LRG-II position to develop plant specific programs to estimate fuel damage based on the Enrico Fermi-2 Project procedure transmitted by a letter dated April 29, 1982 from Harry Tauber (Detroit Edison) to L. L. Kinter (NRC).

The estimation of core damage will be calculated by comparing the measured concentrations of major fission products in either gas or liquid samples, after appropriate normalization with reference plant data from a BWR-6/238 with a Mark III containment. (Reference: General Electric; Procedures for the Determination of the Extent of Core Damage Under Accident Conditions; RPE 8/CCL01 dated November 1981.)

The procedures will provide locations for obtaining the most representative samples (see 1-CHEB and 2-CHEB Position Papers) depending on accident severity and system conditions. Water samples (reactor coolant, suppression pool and RHR) and gas samples (containment and drywell) are analyzed by gamma spectroscopy for determination of I-131, Cs-137, Xe-133 and Kr-85 concentrations. The measured fission products are corrected for decay and the concentrations are normalized to the reference plant data appropriately for comparison to graphs to indicate percent cladding failure or percent fuel meltdown. Isotopic ratios for noble gasses and iodine are calculated for comparison with the ratios that are normally expected to be found in the core inventory and in the fuel gap.

In addition, LRG-II plant-unique programs will address post-accident sampling system testing and operator training programs as required by Section 6.8.4.c of the Standard Technical Specifications. A third core damage category that is in between cladding failure and core melt, i.e. fuel overheating (metal-water reaction) will be included. Other plant indicators (e.g. reactor water level, hydrogen generation from zirconium-water reaction, containment monitors, etc.) will be factored into the program to aid in the interpretation of the extent of the core damage and cross check whether sampling is representative or sample analysis is reasonable.

5-ASB

CONTROL ROD DRIVE SYSTEM VESSEL INVENTORY MAKE-UP RATE TEST

ISSUE:

A flow test of the control rod drive (CRD) system is required b Recommendation No. 6 contained in Section 8.1 of NUREG-0619, "BW Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking". The purpose of the test is to assure that adequate flow into the reactor vessel from the CRD System would be available if the CRD System return line was cut and capped to eliminate nozzle cracking. LRG-II plants are required to perform this test or provide justification as to why such a test is not necessary.

LRG-II POSITION:

It is the LRG-II position that the CRD system make-up rate test recommended in NUREG-0619 not be performed. This position is based upon the evaluation prepared by the LRG-I and submitted via the letter from P. L. Powell (Chairman - Licensing Review Group) to H. Faulkner (NRC) dated March 26, 1982. This evaluation demonstrates that the requirement for a CRD system make-up flow test is no longer necessary since the intent of Recommendation No. 6 is met in other ways such as:

a. Plant fire prevention/protection and separation enhancements.

b. Development of symptom-oriented emergency procedure guidelines.

c. Post-TMI emergency core cooling system modifications.

The conclusions of this evaluation are also applicable to the LRG-II projects and are hereby adopted as an LRG-II position.

The NRC Staff found the above evaluation acceptable as documented in the Susquehanna Project Safety Evaluation Report (NUREG-0776) Supplement 3, Section 4.6.2

LRG-II Position Paper

September 3, 1982

4-MEB

KUO SHENG INCORE INSTRUMENT TUBE BREAK

ISSUE:

During a recent shutdown of the Kuo Sheng 1 plant, breakage of an incore instrument tube occured due to operation of the RHR/LPCI system in an abnormal mode for normal shutdown for an extended period of time. This resulted in LPCI injection directly into the core for an extended period of time, eventually causing fatigue failure of an incore instrument tube and subsequent leakage (approximately one gallon per minute) from the reactor pressure vessel. This situation can only occur in BWR/6 plants where the RHR/LPCI is connected to the core shroud below the top guide plate. This allows the LPCI flow to impinge directly on the upper end of the core and cause incore instrument tube vibration. Previous BWR designs have the RHR/LPCI connected to the shroud above the top guide plate.

BWR/6 plants should modify the reactor design to eliminate the RHR/LPCI flow impingement problem.

LRG-II POSITION:

It is the LRG-II position to install a flow deflector on each of the three LPCI inlets to prevent direct horizontal flow impingement upon the core and instrumentation. The flow deflectors will be in the shape of a rectangular plate, approximately 1' x 2' with a conical flow splitter which redirects the LPCI flow upward, downward and in the two horizontal directions tangential to the core. The flow deflector arrangement is shown in Figure 4-MEB. The deflectors will be fabricated of 316L stainless steel plate material and will be attached to the shroud wall by full penetration welds at the deflector legs and at the four corners of the deflector plate. The processes and procedures to be used for fabrication will be comparable to those used for other reactor internals.

The design, analysis, and testing of the deflectors, plus consideration of the effects of it on LPCI system parameters and plant safety, are addressed in the letter from Dr. R. Artigas (GE) to R. L. Tedesco (NRC) dated May 18, 1982. The acceptance of the deflector modification by the NRC was indicated at the Grand Gulf Project ACRS Committee meeting on August 12, 1982 (See Pages 26 & 27 of the transcript).

In addition to the use of flow deflectors, the Intermediate Range Monitor (IRM) instrument tube nearest to each LPCI inlet will be replaced by a strengthened tube. This modification would avoid IRM damage in the immediate vicinity of the LPCI injection point with the deflector installed and the LPCI operative for extended periods of time.

FIGURE 4-MEB

LPCI FLOW DEFLECTOR