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April 27, 1983

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Limerick Generating Station, Units 1 and 2 Response to Reactor Systems Branch Draft Safety Evaluation Report (DSER) on TMI Items

Reference: A. Schwencer to E. G. Bauer, Jr. letter dated March 11, 1983

File: GOVT 1-1 (NRC)

Dear Mr. Schwencer:

The attached documents are draft responses to the Reactor Systems Branch DSER, Item 2, relating to TMI items. The response to II.K.3.18, which was submitted in the March, 1983 revision of the FSAR, is being revised and will be formally incorporated into the FSAR revision scheduled for May, 1983. The responses to II.K.3.30 and II.K.3.31 will be formally incorporated into the FSAR revision scheduled for April, 1983. The responses to II.K.3.13, II.K.3.15, II.K.3.16, II.K.3.22, II.K.3.24, II.K.3.25, and II.K.3.45 will be formally incorporated into the FSAR revision scheduled for May, 1983. Please note that the response to II.K.3.21 has been submitted via FSAR revision in March, 1983.

Sincerely,

Eugene J. Bradley

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cc: See Attached Service List

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Judge Richard F. Cole	(w/ enclosure)
Judge Peter A. Morris	(w/ enclosure)
Troy B. Conner, Jr., Esq.	(w/ enclosure)
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Response to II.K.3.13

Analysis performed by the BWR Owners' Group (NEDO-24951) has concluded that changing the initiation setpoint of the HPCI/RCIC is unwarranted*. The same NEDO report did recommend a modification to the RCIC circuitry to permit auto-restart of RCIC on low level after a high level trip. Therefore, modifications to the RCIC circuitry are currently underway to delete the high water level turbine trip and to apply this signal to the auto-close circuit of the steam supply valve. This will provide automatic operation of the RCIC system to trip at high water level and auto restart at low water level. This will be implemented prior to fuel load.

*The NRC has accepted this generic report.

Response to II.K.3.15

The HPCI/RCIC steam line isolation logic is currently being modified to address the spurious isolation of these systems due to the pressure spike which accompanies their start-up. The modification consists of adding a time delay to the high flow trip logic of HPCI/RCIC. This will prevent the instantaneous pressure spike from causing a system isolation. This will be implemented prior to fuel load.

This design change was submitted by the BWR Owners' Group and has been accepted generically by the NRC.

Response to II.K.3.16

PECO endorses the BWR Owners' Group generic response to Item II.K.3.16 for Limerick. This response is described in NEDO-24951, "BWR Owners' Group NUREG-0737 Implementation: Analyses and Positions Submitted to the USNRC," June, 1981. The following recommendations from NEDO-24951 will be implemented at Limerick in order to reduce the challanges to relief valves by approximately an order of magnitude:

(1) Low Water Level Isolation Setpoint (Reference Section 6.3.1.1.1 of NEDO-24951). The RPV water level isolation setpoint for MSIV closure is being lowered from Level 2 to Level 1 as part of the ATWS modifications for Limerick.

(2) Low-Low Set Relief or Equivalent Manual Actions (Reference Section 6.3.1.3.1 of NEDO-24951). This recommendation assures that following the initial pressurization the pressure will be relieved by one valve alone, and the remaining safety relief valve will not experience any subsequent actuation. At Limerick this will be accomplished manually as described in "BWR Emergency Procedure Guidelines," Revision 1 (prepublication form submitted January 31, 1981).

(3) Reduce MSIV Testing Frequency (Reference Section 6.3.1.4.4 of NEDO-24951). A number of isolation events occur when the MSIV closure tests are being conducted. Reducing the MSIV testing frequency would result in a reduction in the number of isolation events. Appropriate reductions will be made to the frequency of testing for the Limerick MSIV's.

Response to II.K.3.18

The BWR Owners' Group has submitted a report in NEDO-24951 to the NRC in which they propose five options to address this concern. Limerick will take steps to make the required modifications to the ADS logic when the NRC rules on the acceptability of the proposed options. This modification will be implemented during the first refueling outage, in accordance with NUREG-0737 Implementation Requirements.

Response to II.K.3.21

Philadelphia Electric Company endorses the BWR Owners' Group position for Item II.K.3.21 for Limerick. This position was forwarded to the NRC by letter from D. B. Walters (BWROG) to D. G. Eisenhut (NRC) dated December 29, 1980. The conclusion of this position is that automation of the restart of the LPCI and CS will result in a net decrease in safety because of the complexity of the logic required. Logic modifications to the LPCI and CS systems are therefore not warranted at Limerick.

Response to II.K.3.22

Modifications are now underway to modify the RCIC system suction valve logic to automatically switch suction from the Condensate Storage Tank to the Suppression Pool on low Condensate Storage Tank level. This will be implemented prior to fuel load.

Response to II.K.3.24

At Limerick, the HPCI and RCIC compartment unit coolers are powered by onsite emergency power and therefore continue to be available during a loss of offsite power. The unit coolers are described in Section 9.4.2.2. The emergency service water pumps which provide flow to the coolers are also powered from onsite emergency power. Adequate space cooling is therefore assured during a loss of offsite power. There are no other supporting systems that require offsite power such that operation of the HPCI and RCIC systems would be impaired should offsite power be lost. The current Limerick design is therefore acceptable.

Response to II.K.3.25

At Limerick, two systems are available for cooling the recirculation pump seals: The reactor enclosure cooling water (RECW) system and the recirculation pump seal purge system.

Recirculation pump vendor test data has shown that if either one of these seal cooling systems is operating, seal temperatures will remain within acceptable limits and excessive seal deterioration is not expected to occur.

The primary cooling for the recirculation pump seals is provided by the RECW system which cools the reactor water that flows to the lower seal cavity. After a loss of offsite power, the RECW pumps will be powered by onsite emergency power and will restart automatically. The service water system, which normally provides cooling water to the RECW heat exchangers, will not be available, but cooling water to the heat exchangers can be provided via manual realignment of the Emergency Service Water (ESW) system. If the RECW pumps do not restart or are unavailable for some other reason, the ESW can be manually routed directly to the recirculation pump seals for cooling by way of the RECW piping.

Backup cooling is provided by the recirculation pump seal purge system which injects cool water from the Control Rod Drive (CRD) system into the lower seal cavity. The CRD pumps are powered from the emergency diesels and can be manually restarted once onsite power is available. Hence, the CRD pumps provide an alternate method that is available for seal cooling during a loss of offsite power.

Even in the remote case where neither cooling source is reestablished and gross seal degradation occurs, the General Electric analysis (NEDO-24951) performed under the direction of the BWR Owners' Group and which is applicable to Limerick has shown that the maximum coolant loss would be limited to 70 gpm per pump. This loss is small enough to be compensated for by normal or emergency reactor water level controls.

Instrumentation for various parameters, including seal cavity pressure, seal staging and drain flows, drywell equipment drain sump pump flow and drywell floor drain sump pump flow, is available to the operator to indicate potential seal failure. In addition, gross seal failure may lead to changes in drywell pressure, temperature, or radioactivity, all of which are monitored and recorded in the control room.

It is therefore concluded that a total loss of recirculation pump seal cooling is not a problem at Limerick and modifications are not necessary.

Response to II.K.3.30

The response to the NRC small break model concerns was provided at a meeting between the NRC and GE on June 18, 1981. Information provided at this meeting showed that, based on the small break test results and sensitivity studies, the existing GE small break LOCA model already satisfies the concerns of NUREG-0626 and is in compliance with 10CFR50, Appendix K. Therefore, the GE model is acceptable relative to the concerns of Item II.K.3.30, and no model changes need be made to satisfy this item.

Documentation of the information provided at the June 18, 1981 meeting was provided via letter from R. H. Buchholz (GE) to D. G. Eisenhut (NRC) dated June 26, 1981.

Response to II.K.3.31

The small-break LOCA calculations included in the Limerick LOCA analysis are discussed in Section 6.3.3.7 and Table 6.3-5. The references listed in Section 6.3.6 describe the currently approved Appendix K methodology used to perform these calculations. Compliance with 10CFR50.46 has previously been established for that methodology. As stated in the June 26, 1981 letter from R. H. Buchholz (GE) to D. G. Eisenhut (NRC), no model changes are needed to satisfy NUREG-0737, Item II. K.3.31; therefore, there is no need to revise the calculations presented in Section 6.3.3.7.

Response to II.K.3.45

The Applicant endorses the BWR Owners' Group position on Item II.K.3.45 for Limerick. This position is presented in NEDO-24951. "BWR Owners' Group NUREG-0737 Implementation: Analysis and Positions Submitted to the USNRC," June 1981 and is summarized below.

An evaluation of alternate mode of depressurization other than full actuation of the ADS was made by the BWR Owners' Group with regard to the effect of such reduced depressurization rates on core cooling and vessel integrity.

Depressurization by full ADS actuation constitutes a depressurization from about 1050 psig to 180 psig in approximately 3.3 minutes. The alternate modes of depressurization that were evaluated considered vessel depressurization over the same range (1050 psig to 180 psig) within two different time periods (6-10 minutes and 15-20 minutes). The cases considered show that no appreciable improvement can be gained by a slower depressurization based on core cooling considerations. Since a full ADS blowdown is well within the design basis of the reactor pressure vessel and ADS is properly designed to minimize the threat to core cooling, no change in the depressurization rate is necessary, and no modifications to Limerick are needed for this TMI item.