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August 23, 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 Core Performance Branch Request for Additional Information Concerning TMI Item II.F.2 (SLI-8211 and SLI-8218)
Reference:	PECO and NRC Conference Call dated July 19, 1983
File:	GOVT 1-1 (NRC)

Dear Mr. Schwencer:

As requested by the Core Performance Branch Reviewer in the reference conference call, the attached additional information provides a description of the Limerick specific review of the reactor water level instrumentation, a discussion of the improvements to this instrumentation, and a discussion of the need for additional inadequate core cooling instrumentation.

Sincerely,

John S. Kemper

RJS/gra/i6

Copy to: See Attached Service List

A002
1/1

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ELECTRICAL ENGINEERING DIVISION
N3-1, 2301 Market Street

August 18, 1983

SUBJECT: Item II.F.2 of NUREG-0737
Response to NRC Request for Information
Limerick Generating Station

The BWR Owners' Group has submitted two reports to the Nuclear Regulatory Commission (NRC) to respond to the requirements of item II.F.2 of NUREG-0737, "Clarification of TMI Action Plan Requirements." The first report, SLI-8211, "Review of BWR Reactor Vessel Water Level Measurement Systems," analyzes the current water level measurement system and concludes that there are problems with the instrument design in some plants. The second report, SLI-8218, "Inadequate Core Cooling Detection in Boiling Water Reactors," places the reactor water level instrument problems into perspective through the use of a probabilistic risk assessment. SLI-8218 also examines if any new instruments must be added to measure the adequacy of core cooling.

This report concludes that:

a.) Reactor water level measurement conclusively indicates the adequacy of core cooling.
b.) The risk of core melt associated with failures of the existing reactor water level instrumentation is small. This risk can be made smaller with improvements in the existing instrumentation and by the implementation of water level validation techniques.

c.) There are several new instruments that are viable for measuring the adequacy of core cooling. However, the addition of new instrumentation is not cost beneficial.

Philadelphia Electric Company has participated in the BWR Owners' Group during the development of these reports and agrees with the generic conclusions contained in them. The following is a description of the Limerick specific review of the reactor water level instrumentation and a discussion of the improvements to this instrumentation.

The Limerick Generating Station (LGS) reactor pressure vessel (RPV) water level measuring system has four (4) divisions of instrumentation. The instrumentation uses cold reference legs and differential pressure measuring devices. These devices are connected to the vessel through parallel lines; that is, the reference leg and the variable leg sensing lines have the same vertical drop (12 feet at LGS) in the drywell. We also include in our design analog transmitters and electronic trip units. The LGS design does <u>not</u> include mechanical trip units or Yarway heated reference column devices.

Prior to the issuance of the S. Levy, Inc., study SLI-8211, "Review of Reactor Vessel Water Level Measurement Systems", Philadelphia Electric Co. reviewed the LGS RPV water level measuring system and found that some of the instrument lines were non-parallel. We have made piping changes to our system using existing penetrations to ensure that the lines are as parallel as possible, thus reducing the errors in level indications caused by elevated drywell temperature.

When SLI-8211 was issued, we reviewed the study and examined the concerns and made improvements when they were justified for LGS. We have added additional instrumentation to monitor reference leg temperature as an aid to detecting a potential reference leg boil-off condition. As we have described in previous communications, the reference leg temperature monitoring system consists of one dual element, type "[illegible]", 0.5%, surface-mounted thermocouple installed on each of the reference legs. The outputs of these thermocouples provide inputs to the Emergency Response Facility Data System (ERFDS). In the ERFDS, these signals are used to generate displays useful to the reactor operator in following the LGS emergency procedures and identifying the potential for reference leg boil-off. Displays and information available on the ERFDS include reference leg temperature (both individual temperatures and trend displays), compensated water level, and a reactor pressure vs. reference leg temperature plot useful in determining conditions leading to reference leg boiling.

Section 5.4 of SLI-8211 summarizes the major areas of concern with the water level instrumentation for the various BWR designs reviewed.

The concerns and how they relate to LGS are as follows:

Concern 1:

Yarway temperature compensated reference legs could flash at intermediate pressures during the course of plant cooldown.

Response 1 (LGS):

At LGS, we do not have Yarway temperature compensated reference legs. The LGS design utilizes cold reference legs. Therefore, Concern 1 is not applicable to LGS.

Concern 2:

Position of orifices in both reference and variable legs.

Response 2 (LGS):

SLI-8211 recommends moving the instrument line flow orifices as close to the containment penetration as possible to prevent erratic level indications under boil-off and flashing conditions. We feel that it would be inappropriate to make such a modification at LGS. The following statement taken from SLI-8211, page 6-36, the second paragraph, provides the appropriate context for consideration of this concern:

> "It is apparent that considerable effort and expense would be incurred to implement this change in an existing plant, but implementation on a plant in early construction might be a nominal burden. On the other hand, the existing analytic basis that supports the motivation for the change is limited to a sample calculation resulting from an approximate analysis. Therefore, it would not be appropriate to make the change, if considerable expense results, without further detailed study of the phenomena and the particular installation."

At the time this study was completed and reviewed by PECO, LGS Unit 1 was approximately 85% designed and/or fabricated; now that figure has increased to 93%. To rework the present design of either unit would impose an undue cost for a modification whose need and benefit have not been fully justified at this time. The existing orifice locations are in accordance with ISA Standard S67.02, section 6.3.2. The significance of this potential problem is quite limited due to the following factors:

1) For those sequences in which water level measurement failures have the greatest contribution to core melt frequency, RPV depressurization is slow, resulting in significantly less error (about ½ foot per 100°F/hour cooldown) than for the limiting case (depressurization by ADS) considered.

2) The error in water level measurement due to the pressure gradient required to force the fluid through the orifice during flashing will occur only during the actual period of depressurization.

3) Operator training stresses the potential for inaccuracy in water level indication during rapid depressurization.

The other compensating measures employed at Limerick adequately address reference leg boil-off or flashing conditions. These measures are described in Response 4, below.

Concern 3:

Variable leg, Reference leg vertical drops not equal (non-parallel lines).

Response 3 (LGS):

The variable and reference legs of the reactor vessel level instrumentation are as parallel as possible at LGS, that is, the vertical drops within the drywell of both sensing legs are as equal as slope constraints will allow. This design minimizes indicated errors in level due to fluid density changes caused by drywell heatup (such as loss of drywell chillers).

The following chart illustrates the worst case differences in vertical drop:

Range	Reference Leg Vertical Drop	Variable Leg Vertical Drop	Difference
Wide	12.422'	12.159'	0.263'
Narrow	12.422'	12.155'	0.267'
Fuel Zone	12.208'	13.917'	1.709'

The wide range and narrow range instrument lines are clearly parallel. The parallelism of the fuel zone instrument lines may be questioned depending on the numerical acceptance criteria used. However, since the difference in elevation drop of the fuel zone instrument lines causes a conservative (i.e., lower than actual) indication in the control room during conditions of elevated drywell temperature, it is concluded that the intent of the recommendation which calls for parallel instrument lines is met.

Concern 4:

Reference leg boil-off and flashing.

Response 4 (LGS):

Since the LGS design includes a 12 foot vertical drop of the reactor water level reference leg inside containment, high drywell temperature can lead to boil-off and flashing of the water in the reference leg during RPV depressurization. This could cause errors in the indicated level. To determine if this error should be further reduced by repiping the reference leg, the following were considered:

1) The scenarios which could lead to a boil-off condition,
2) The information required by the operator under these scenarios,
3) The guidance provided to the operators in the plant emergency procedures, and
4) The cost of improvements.

It was determined that the events leading to substantial boil-off are very improbable and slow moving. In all cases, the operator will have sufficient time to consult his operating procedures and take the appropriate actions. Also, it was determined that even if considerable boil-off of the reference leg occurs, indication of level is still maintained on the wide range instruments, indicated level will change as actual level changes, and indicated level can be used to inform the operator about the adequacy of core cooling. Finally, it was determined that modification of the piping to reduce the error is prohibitively expensive relative to the potential benefit which could be obtained.

The LGS emergency operating procedures are structured to avoid conditions where boiling and/or flashing of the reference legs would occur and to direct appropriate actions should such conditions develop. The following discussion of high energy line breaks (previously sent to you as a response to RAI421.16) is illustrative of this approach:

High energy line breaks are breaks occurring in a fluid system whose normal plant conditions are either in operation or maintained pressurized under conditions where either or both of the following are met:

a. Maximum operating temperature exceeds 200°F
b. Maximum operating pressure exceeds 275 psig.

High energy line breaks inside containment can be categorized into three sizes: a) large, b) intermediate, and c) small. Each break has its own temperature and pressure effects on the vessel, drywell, and suppression pool. However, it is the small break that imposes the most severe long-term temperature conditions on the vessel level instrumentation reference legs.

Under a design basis small break accident condition, the maximum temperature in the drywell could approach 255°F. Given this elevated drywell temperature and the gradual depressurization of the reactor vessel, a potential boil-off condition could occur when the reactor vessel pressure falls to the saturation pressure corresponding to the actual drywell temperature (18 psig for a 255°F drywell temperature). It should be noted that in instruments with cold reference legs, the reference leg temperature can only increase to the drywell ambient temperature. Thus, under the least favorable conditions, a cold reference leg will not boil until the system pressure falls to 55 psia.

For the purpose of this illustration, it is assumed that the reactor and containment are at the maximum normal conditions when a small break occurs that allows blowdown of reactor steam to the drywell. The resulting pressure increase in the drywell leads to a high drywell pressure signal that scrams the reactor and initiates the high pressure emergency core cooling systems and containment isolation. Drywell temperature would increase and reactor vessel pressure and level will decrease at a slow rate due to the small size of the break. The automatic depressurization system (ADS) will not initiate until low RPV level and high drywell pressure signals have been received.

The reactor operator is alerted to the incident by the drywell high pressure and temperature alarms and by the reactor scram. It is at this point that the operator will start to follow the emergency procedures. The operator will be aware of the elevated drywell temperature and pressure and is required to monitor and control these parameters as well as RPV level and pressure.

During any accident or transient condition, the operating procedures instruct the operator to control drywell temperature with any available means of drywell cooling (restart tripped coolers, start backup coolers, utilize backup cooling water source) and to begin normal RPV depressurization (100°F/hr cooldown). If available drywell cooling is insufficient to prevent a further increase in temperature, then, before the drywell design temperature is reached (340°F), the operator is required to shut down the drywell fans, initiate drywell sprays, rapidly depressurize the vessel, and initiate low pressure makeup systems. The sprays will lower drywell temperature and prevent a reference leg boiling condition from occurring.

Throughout each of these operations, the operator will also monitor RPV pressure and reference leg temperature and compare these parameters against a curve to determine the potential for reference leg boiling. This comparison is rapidly performed on one of the ERFDS displays. If the plant conditions fall into the area above the curve where boil-off is possible or if RPV water level cannot be determined, the operator is instructed to depressurize the vessel and initiate low pressure makeup systems. This injection will continue until the reference leg temperature is reduced and level indication is available. Flooding the vessel ensures that adequate core cooling is maintained throughout the transient.

If the reactor operator allows conditions to deteriorate, reference leg boiling could occur. Level indication will be erratic during boiling but will stabilize once boiling stops. If conditions deteriorate to the point where the total reference leg of the wide range level transmitter boils off inside containment, the operator will still have an active indication of level. The wide range indicator will indicate approximately +16 inches (on a scale of -150 to +60 inches) when the actual water level is at the lower instrument tap. This means that if the reactor operator maintains level indication above +16 inches on the wide range indicator, he will have an indication that water is above the lower instrument tap and hence will know that the fuel is covered. The emergency procedures and operator training will emphasize this fact. The operator will be made aware of reference leg flashing long before reaching this point by differing indications on the various water level instruments and by the ERFDS displays previously described.

Although the operator does have water level indication that is accurate enough to indicate the adequacy of core cooling under all conditions, further improvement in the accuracy of water level indication may be possible. One method of improving the accuracy would be to reduce the reference leg vertical drop inside the drywell as discussed in SLI-8211. To implement this modification at LGS while still maintaining parallel reference and vertical legs would involve at least eight new drywell penetrations. Analysis of the containment will be necessary to show if this modification is technically feasible from a containment integrity viewpoint. As noted at the NRC/BWR Owners' Group meeting held on May 19, 1983, in Bethesda, MD, such changes would cost on the order of $4 million per unit. Limerick unique cost estimates have not been developed but are estimated to be similar or slightly higher. In addition to the direct cost, there will be cost associated with the slippage in the construction schedule caused by such a major modification.

The cost of making this modification must be compared to the potential benefit to be obtained. A methodology to estimate the potential benefit (i.e., core melt frequency reduction) associated with water level measurement improvements on a generic basis was developed and utilized in SLI-8218, "Inadequate Core Cooling in Boiling Water Reactors." SLI-8218 recognizes that the benefit to be derived for a particular plant will be dependent on plant specific features and analyses. In relating this work to LGS, we have found a number of differences between the generic assessment and the current design of LGS that substantially reduce the contribution to core melt frequency at LGS due to the reactor water level instrument design. The major differences are:

a) Improvements to the reactor water level instrumentation at LGS have been made as described above (reference leg parallelism and reference leg thermocouples utilized for ERFDS compensation /display). These features were not considered in the generic analysis.

b) There are four electrical divisions of water level instrumentation at LGS for the control of the low pressure emergency core cooling systems. The generic analysis assumed only two electrical divisions (Table 4-2 of SLI-8218). The Limerick design is considerably more tolerant of multiple failures.

c) The LGS drywell cooling system is substantially more reliable than the generic plant due to redundancy and a backup cooling water source. The LGS design includes redundant fan/coil units, piping, and chillers and an automatic backup heat sink (reactor enclosure cooling water). The significance of these differences is discussed in Sections 4.5.1.1 and 4.5.1.8 and Appendix H of SLI-8218. It is noted that such features would "...result in a core melt frequency contribution reduction... of about an order of magnitude."

d) The LGS design includes additional features to mitigate the consequences of an anticipated transient without scram (ATWS) that are not found in the generic plant (Section 4.1.1. of SLI-8218).

e) The LGS design includes analog transmitters and trip units in lieu of the mechanical devices assumed to be employed for the generic analysis (Section 4.2 of SLI-8218).

These differences lower the contribution of water level measurement system failures to core melt frequency to a value which is significantly lower than that shown in the SLI-8218 generic analysis. Hence, it is concluded that it would not be cost/beneficial to further modify the reactor water level instrumentation at LGS.

We believe that the design of the LGS drywell cooling system and the improvements we have made to the LGS water level measuring system are responsive to the findings and conclusions of SLI-8211 and 8218. The reliability of the measurement system itself has been improved and the operator's ability to recognize potential problems and failures has been substantially improved. The improved system will provide the reactor operators with accurate, reliable, water level indication. It is recognized that an error in indicated level due to reference leg boiling and flashing could result under certain, low probability events. However, the emergency procedures have been written and equipment has been provided to facilitate the operator's recognition of this event and to assure that appropriate steps to mitigate the effects of such accidents are taken.

Concern 5:

Logic configurations may lead to situations and transients that have not been previously considered.

Response 5 (LGS):

The LGS logic design is made up of four (4) divisions. Of the scenarios investigated in SLI-8211, section 5.2.2, pages 5-44 through 5-48 and 5-51, none presented any challenge to fuel design limits or danger of core uncovery. This concern is not applicable to LGS and plants of similar logic design.

Concern 6:

Mechanical Instrumentation Failures

Response 6 (LGS):

The LGS design calls for electronic transmitters and solid state trip units. Therefore, this concern is not applicable to LGS.

The above discussion shows that the design of the LGS reactor water instrumentation has been improved as recommended in SLI-8211 where such improvement has proven to be cost beneficial. This will provide the reactor operators with accurate, reliable, water level indication. As shown in SLI-8218, this improved instrumentation will enable the operator to verify the adequacy of core cooling.