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May 25, 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
Draft Safety Evaluation Report (DSER)

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 Core Performance Branch Draft Safety Evaluation Report items 1 through 4 and 6 through 11. Item 5 is closed per the attached letter from Mr. J. S. Kemper to Mr. R. L. Tedesco dated December 22, 1981, concerning Channel Box Deflections. Chapters 4, 5, 6 and 15 of the FSAR will be revised to incorporate the latest approved revision of the "General Electric Standard Application for Reactor Fuel" (GESTAR II) which incorporates the responses to all eleven Core Performance DSER items. This revision is scheduled for incorporation into the FSAR in June, 1983.

Sincerely,

E. J. Bradley

RJS/mjb 5/18/83-1

Copy to: See attached Service List

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1/1

cc:	Judge Lawrence Brenner	(w/o enclosure)
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Question (1) - Fuel Rod Pressures

Section 4.2 of NUREG-0800 identifies excessive fuel rod internal pressure as a potential fuel system damage mechanism. In this sense, damage is defined as an increased potential for elevated temperatures within the rod as well as an increased potential for cladding failures. Because analytical methods for fuel performance analysis may not adequately treat the effects of net outward stress on the cladding, and because these effects (such as, unstable high fuel temperatures and ballooning during departure from nucleate boiling (DNB) events) might be important, NUREG-0800 calls for rod pressures to remain below nominal system pressure during normal operation unless otherwise justified. Since the applicant has not yet proposed the use of this rod pressure limit, or justified an alternative, we are unable to conclude that the analysis of fuel rod pressure has been performed in an acceptable manner. This is an open item.

Response

During the first cycle of plant operation at Limerick, fuel rod internal pressure remains below nominal system pressure. For subsequent cycles, this issue will be addressed in a manner that will satisfy the intent of Section 4.2 of the Standard Review Plan, before the start of the second cycle.

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Question (2) - Fuel Rod Mechanical Fracturing

The term "mechanical fracture" refers to a cladding defect that is caused by an externally applied force such as a hydraulic load or a load derived from core plate motion. These loads are bounded by the loads of a LOCA and SSE, and the mechanical fracturing analysis is usually done as a part of the seismic-and-LOCA loads analysis (see Section 4.2.3.3(4) of this report). Because that analysis has not been completed for Limerick, it is not clear what design limit will be used for the mechanical fracturing analysis. This is an open item.

Response

The analyses documented in NEDE-21175-3-P, "BWR Fuel Assembly Evaluation of Combined SSE and LOCA Loadings, (Amendment No. 3)", July 1982, are applicable to Limerick. In particular, evaluations of the fuel rod are discussed in Section II.2.2 of that document.

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Question (3) - Fuel Assembly Structural Damage From External Forces

Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. NUREG-0800, Section 4.2, and Appendix A to that section state that fuel system coolability should be maintained and that damage should not be so severe as to prevent control rod insertion when it is required during these low probability accidents. Because GE has not completed this analysis for Limerick, it is not clear what the exact design limits will be, but they must follow the guidelines of NUREG-0800 Section 4.2, Appendix A. This is an open item.

Response

Fuel assembly structural evaluations are discussed in Section II.2 of NEDE-21175-3-P. The analyses presented in that document are applicable to Limerick.

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Question (4) - Fuel Rod Bowing

GE has stated (NEDE-21660-P) that BWR fuel operating experience, testing, and analysis indicate that there is no significant problem with rod bowing even at small rod-to-rod and rod-to-channel clearances. GE stated that (a) no gross bowing has been observed (excluding the rod-bowing-related failures in an early design), (b) a very low frequency of minor bowing has been observed, (c) mechanical analysis indicated deflections within design bases, and (d) thermal-hydraulic testing has shown that small rod-to-rod and rod-to-channel clearances pose no significant problem. The staff is reviewing a GE generic topical report (NEDE-24284-P) that is intended to update the GE rod bowing experience and document the overall GE rod bowing safety analyses. The staff's review is scheduled for completion in February 1983. Based on GE's statements that there is no significant problem and the staff's belief that rod bowing effects in BWRs are smaller than in PWRs, where the effects have been compensated for with available margins, fuel rod bowing is now considered to be a confirmatory issue.

Response

The NRC has concluded that the GE fuel rod bowing topical reports NEDO-24284 and NEDE-24284-P are acceptable for reference. These topical reports are applicable to Limerick.

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Question (6) - Fuel Rod Pressure

The applicant has indicated that the internal pressure is used in conjunction with other loads on the fuel rod cladding in the calculation of cladding stresses. The results of such calculations, as provided in NEDE-24011, show that the calculated stresses can be accommodated. We have examined these calculations and agree that the acceptance criterion for cladding stress (see NUREG-0800, Section 4.2.II.A.1.a) has been met, but that criterion is not the same as the one for rod internal pressure (see NUREG-0800, Section 4.2.II.A.1.f). Because the criterion for rod internal pressure involves more than the cladding mechanical limits, we consider the GE analysis described in NEDE-24011 to be insufficient for meeting the rod internal pressure criterion.

We note that the rod internal pressures used in the GE cladding stress analysis are well in excess of the nominal coolant system pressure, and this has been the case for a number of other license applications which rely on the generic analysis in NEDE-24011. These calculations involve conservatisms employed to simplify the analysis and, as a result, do not provide an accurate estimate of rod internal pressure. In order to assess the impact of the fuel rod internal pressure on the safety analysis of other plants, we have relied upon other, more representative (but still conservative) analyses from General Electric (Sherwood, December 22, 1976). These calculations show that fuel rod internal pressures remain below system pressure for planar average burnups below 44,000 MWd/t. This conclusion remained unchanged for the newer prepressurized fuel design as well (NEDE-23785-1). On the basis of these calculations, we concluded that the rod internal pressure criterion, although not explicitly used by General Electric, has nevertheless been met.

Recent information from General Electric (Quirk, January 21, 1982) indicates that fuel rod internal pressure may exceed system pressure even on the basis of the more representative calculations. We attribute this change in conclusion to the higher burnup levels now being sought by the

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industry and by the advent of more appropriate fission gas release data and methods of analysis. Because we no longer have a basis from which we can conclude that the rod internal pressure criterion has been met, and because the applicant has not yet proposed an acceptable alternative to this criterion, we regard this as an open item.

Response

This question is addressed in the response to Question (1).

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Question (7) - Overheating of Fuel Pellets

Fuel melting temperature is discussed in Section 2.4.2.5 of NEDE-24011 as a function of exposure (burnup) and gadolinia content (of burnable poison rods). GE states in that report that fuel melting is not expected to occur during normal operation, and that statement is based on fuel temperature calculations performed with a model described in the proprietary supplement to Amendment 14 of GESSAR. Although limited melting during certain events such as an uncontrolled control rod withdrawal is permissible, such melting is not predicted to occur.

The staff has reviewed the UO_2 properties (thermal conductivity and melting point) that are important in reaching this conclusion and agree that UO_2 melting will not be a problem at Limerick during normal operation and anticipated transients. The effects of gadolinia concentration on thermal conductivity and melting temperature are addressed in a GE topical report on gadolinia fuel properties (NEDE-20943-P). However, NEDE-20943-P was withdrawn from the review. We subsequently reviewed selected GE gadolinia fuel properties as described in an appendix to the GESTR-LOCA report (Appendix B to NEDE-23785-1-P). That appendix has been approved. However, the thermal conductivity equation listed in this new report is not the one used in the Limerick fuel centerline melting analysis, which is described in NEDE-20943-P, which was the reference in NEDE-20411-P for GE gadolinia properties, and (b) GE performed a fuel centerline melting calculation only for urania fuel in Amendment 14 to GESSAR. This raises concern about the adequacy of GE's gadolinia fuel incipient melting calculations for Limerick (in particular, Table 2-4 of NEDE-24011-P). We thus request the applicant to confirm the adequacy of Table 2-4 in NEDE-24011-P or submit updated results for review. This is an open item.

Supplement

This question is classified as a confirmatory issue rather than as an open item. This was confirmed in a conversation wth C. H. Berlinger, Chief - Core Performance Branch, on April 6, 1983.

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Response

As previously discussed with the NRC staff in connection with a similar question on another docket, the General Electric Company does not license material properties for design analyses but, rather, maintains these analyses up-to-date. To fulfill our quality control obligations under 10CFR50, Appendix B, the latest property values are incorporated into design applications only after they are qualified in the design code. An improved fuel rod thermal-mechanical design code has recently been developed and qualified which includes the revised gadolinia fuel thermal conductivity relations. The results of the fuel centermelting analysis using this improved fuel rod design code verifies that gadolinia fuel melting is not expected to occur during normal steady-state operation or during the largest whole core anticipated operational transient.

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Question (8) - Fuel Rod Mechanical Fracturing

The mechanical fracturing analysis is usually done as part of the seismic-and-LOCA loads (see Section 4.2.3.3(4) of this report). Because that analysis has not been completed for Limerick, the information on mechanical fracturing is not available. This is an open issue.

Response

This question is addressed in the response to Question (2).

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Question (9) - Fuel Assembly Structural Damage from External Forces

An analysis must be provided by the applicant that shows that the Limerick fuel meets the structural requirements (including liftoff) of Appendix A to Section 4.2 of NUREG-0800. Because the review of a generic report (NEDE-21175-3-P) is not completed. (Question incomplete.)

Response

This question is addressed in the response to Question (3).

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Question (10) - Post-Irradiation Surveillance

A general post-irradiation surveillance program for fuel assemblies that meets the guidelines of Paragraph II.D.3 of Section 4.2 of the Standard Review Plan (NUREG-0800) should be provided. A plan similar to that proposed for WNP-2 and Perry would be acceptable.

Response

The Limerick fuel surveillance program will consist of interim and post-irradiation surveillance of both lead test assemblies and developmental BWR fuel. The schedule of inspection is contingent on both the availability of the fuel as influenced by plant operation and the expected value of the information to be obtained.

The lead test assemblies are selectively inspected. Inspection techniques used include:

(1) Leak detection tests, such as "sipping".

(2) Visual inspection with various aids such as binoculars, borescope, or periscope, with a photographic record of observations as appropriate.

(3) Nondestructive testing of selected fuel rods by ultrasonic and eddy current test techniques.

(4) Dimensional measurements of selected fuel rods.

Unexpected conditions of abnormalities which may arise are analyzed. Examinations of selected fuel rods in Radioactive Material Laboratory (RML) facilities is undertaken when the expected value of the information to be obtained warrants this type of examination. Details of this surveillance program are documented in NEDE-24343-P, "Experience with BWR Fuel through January 1981", May 1981.

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In addition to the fuel surveillance program, offgas surveillance will be performed, and leak detection tests such as sipping are performed at the end of each cycle if it is warranted based on analysis of the offgas surveillance results. Offgas surveillance is a very sensitive measure of fuel performance.

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Question (11) - External Corrosion and Crud Buildup

Question 490.2 requested the applicant to provide:

(1) Allowable limits on external oxidation, hydriding, and crud buildup.

(2) Justification for those limits.

(3) Evidence that the Limerick fuel will not exceed those limits.

The response to this question addressed the stress analysis (SER Section 4.2.1.1(1) and 4.2.3.1(1)), cladding internal hydriding (SER Section 4.2.1.2(1) and 4.2.3.1(1)), and fretting wear (SER Section 4.2.1.1(3) and 4.2.3.1(3)). No information of consequence was provided on external oxidation, external hydriding, or crud buildup (also a cladding external phenomenon).

The effects of external hydriding are probably small and unimportant, but such a statement should have been made (and justified by suitable reference) by the applicant.

The effects of external oxidation (corrosion) and crud buildup are important, however, and may be either life limiting or result in a deviation from a SAFDL (see General Design Criterion 10 of 10CFR50, Appendix A). Therefore, specific design limits on corrosion and crud buildup could be provided. If these phenomena are addressed without using specific design limits, a simple statement to that effect should have been made here.

Response

This issue has been previously resolved on the GESSAR docket. For completeness, the GESSAR response is repeated.

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No separate design limit is required for corrosion and crud buildup because it is considered in the design analyses. Corrosion and crud buildup impact the calculated cladding temperature and material strength, and the ability of the clad to meet the stress limits prescribed in NEDE-24011-P-A-5. Thus, the amount of corrosion and crud buildup is limited by the stress limits on the clad.

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December 22, 1981

Docket Nos. 50-352
50-353

Mr. Robert L. Tedesco
Asst. Director for Licensing
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Limerick Generating Station
Resolution of Channel Box Deflection Issue
for Near-Term BWR OLs

Dear Mr. Tedesco:

Your letter of October 26, 1981 to E. G. Bauer transmitted Section 4.2.3.5 of Amendment No. 76 to the Zimmer FSAR, which proposes several steps that could be taken to mitigate the consequences of channel bowing, and requested Philadelphia Electric Company's commitment to a similar plan for Limerick Generating Station.

Philadelphia Electric Company proposes the following channel residence guidelines and testing procedures to mitigate potential fuel channel bowing and its effects.

Channel Residence Guidelines

1. Records will be kept of channel location and exposure for each cycle.*

2. When possible, channels will not reside in the outer row of the core for more than two operating cycles.

3. Channels that reside in the outer row for more than one cycle will be positioned in core locations that permit different channel sides to face the core edge on successive cycles.

4. Channels that reside in the outer row of the core for three or more cycles will not be shuffled inward.

5. At the beginning of each cycle, the combined (sum of) outer row residence times for any two channels in any control rod cell will not exceed four peripheral cycles.

* The term "cycle", as used in the following recommendations, is defined to be a nominal fuel cycle (12-18 months).

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Testing Procedures

Prior to beginning a new operating cycle, control rod motion testing will be performed for all core cells and therefore, for those cells that might be exceeding the above guidelines.

Rod motion testing currently performed by many operating BWR's as part of their Technical Specifications and Reload Startup Test Programs will adequately demonstrate the scrammability of the reactor.

The Standard Technical Specifications require that each control rod be moved at least one notch every 7 days. Technical Specifications also require periodic scram testing. In addition, Limerick Generating Station will perform a control rod motion test similar to that used by operating BWR's as part of Reload Startup Testing. This test will confirm that each control rod can move full stroke under normal control rod drive (CRD) operating pressures.

The tests discussed above demonstrate the mobility of the control rods at normal CRD operating pressure. The forces on the control blade during scram are orders of magnitude higher than those at normal operation. Therefore, control rods that are movable at normal CRD pressure will not prevent the scram function.

These tests will demonstrate that channel interference, if any, is not significant enough to affect the safety of the plant.

Should you require any additional information, please contact us.

Sincerely,

J. S. Kemper

RHL/mtk 14/2

bcc: E. J. Bradley
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