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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
HOUSTON LIGHTING & POWER COMPANY	)	Docket No. 50-466
	)	
(Allens Creek Nuclear Generating Station, Unit 1)	)	

NRC STAFF'S RESPONSES TO JOHN F. DOHERTY'S SIXTH SET OF INTERROGATORIES

The NRC Staff responds as follows to the sixth set of interrogatories propounded by John F. Doherty to the Staff in the captioned proceeding:

1. Please give the name, if any of person or persons who will testify on the effect of poolswell on safety features in the ACNGS pressure suppression system, and supply other data on this person.

Response

Contentions dealing with containment systems will be addressed by J. Kudrick and/or M. Fields.

2. What is the structural design safety margin (that is: the safety margin above the pressure level that the suppression pool is designed to accommodate?

Response

The lower portion of the steel containment forms the boundary of the suppression pool. The suppression pool is, therefore, designed in accordance with ASME

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Section III, Division 1, Subsection NE, and NRC Regulatory Guide 1.57. The suppression pool has a factor of safety of 1.6 (margin of 0.6) against yielding and of 3.0 (margin of 2.0) against failure for internal pressure resulting from the postulated LOCA condition.

3. What is the largest loading force considered credible and what safety margin is there in the design of the pressure suppression system under that force?

Response

Pressure loads that may be experienced by the suppression pool boundary vary in both magnitude and duration. Without evaluating the structural response to each specific load in combination with other associated loads, a determination of the severest load is impossible. As a result, the containment system is evaluated for a number of different events which are identified in the PSAR. The steel containment will be designed in accordance with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. This Code requires a factor of safety of 1.33 (margin of 0.33) against yielding and 2.0 (margin of 1.0) against failure.

4. Must Applicant meet the largest credible or most probable force in the determination of #3 (above) in construction of ACNGS?

Response

The forces used by the Applicant to evaluate the structures are not considered "most probable." They are considered conservative and can be viewed as the largest credible. Pressure forces are combined with other loads as indicated in Section 3.8.2.3.2.1 of the PSAR.

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5. In the Mark-III, are "design" strengths or "Test" strengths used by the NRC to provide conservatism against defects in the fabrication and erection of the pressure suppression containment.

Response

The steel containment structure and reinforced concrete drywell will be pressure tested after completion of construction at pressure higher than the design pressure to see if the structures behave as designed so as to assure that the structures are conservatively designed and properly constructed.

6. Have the computer codes for predicting containment pressure and the temperature response of the BWR pressure containment been conservative with the Mark I and Mark II designs?

Response

The Contempt computer program is used by the Staff to calculate the temperature pressure response in both the drywell and wetwell. This program is used for all BWR containment designs, Mark I, Mark II and Mark III. The calculated results have been compared with test data for all BWR designs, and the results have proved to be conservative.

7. How will Applicant be able to inspect the beltline region of the reactor pressure vessel in view of limited access mentioned in Page A-6 and A-7 of NUREG-0474?

Response

A specific inspection method has not yet been selected. However, as stated in Section 5.2.6 of the GESSAR SER, GE has committed to develop methods "to facilitate the remote inspection of those areas of the reactor vessel not readily accessible to inspection personnel." This includes the beltline region. The Staff anticipates no difficulty in the implementation of such an inspection system for Allens Creek, especially in view of the fact that such systems have already been developed and implemented for PWRs. See, e.g., Nuclear News, November 1979.

8. Is the large pipe LOCA the maximum pressure and temperature strain placed on the Mark-III pressure system?

Response

The large pipe LOCA maximum pressure and temperature are combined with other loads to calculate strains experienced by the Mark III pressure suppression system. See PSAR Section 3.8.2.3.2.1.

9. On P. 13 of NUREG-0474, the Staff concludes the pressure suppression concept for containment design is safe and acceptable. State every "recently identified concern regarding pool dynamic loads for the Boiling water reactor pressure containment design", (Ibid.) and how these can be eliminated as sources of unacceptability and danger.

Response

Since the pool dynamic phenomena was discovered during the early pressure suppression test facility testing phase in 1972, no new related safety concern has been identified. Rather, the continued testing has been directed towards a better understanding of the phenomena and a more detailed load definition. The Applicant has incorporated this new information relative to load definition into the design of the power plant.

10. (Please refer to Interog. #6 above)
If not, what new factors have been introduced to make the codes conservative for the proposed Allens Creek Design?

Response

See response to Interrogatory #6.

11. Can all design based loss-of-coolant accidents be safely dealt with by the Mark-III when they occur in any room were (sic) components of the reactor water cleanup system (RWCS) are located?

    a. Which design based LOCAs?

Response

The plant has been designed to accommodate all design basis accidents (DBA). The Applicant has considered the pressure response to major pipe rupture within all rooms containing high energy lines including the RWCS rooms. These analyses have been used in the design of the containment structures.

12. What progress or effort is being devoted to mitigating or otherwise protecting the public or nuclear plants from multiple-sequential relief valve actuation, which is a valve actuation sequence which results in a new load combination on the Mark III system, reported on P. 64 of NUREG 0474?

Response

The General Electric Company has proposed modifications to the SRV electrical logic to eliminate this possibility. This was possible since the primary system pressure could be controlled by a single valve but the then existing logic would unnecessarily actuate more than one valve. The Staff has evaluated this proposed modification and has concluded that such a concept is feasible. The detailed review of the specific Mark III design is being conducted within task action plan (TAP) A-39. Scheduled completion of this task is March 30, 1980.

13. What progress has been made in modifying the relief valve control logic so that current load criteria can be maintained with regard to potential multiple-sequential relief valve actuations?

Response

See response to Interrogatory #12.

14. Why is a quencher design thought to obviate concern about pool temperature limits in the Mark-III? (Note: The Pool Temperature Limit occurs when there is continued blowdown in the pool bringing increased pool temperature causing steam condensation to be unstable and which introduces severe and sharp vibration and associated forces.)

Response

The proposed quencher design has demonstrated stable condensation throughout the tested pool temperature range. Tests have been conducted at temperatures approaching the boiling point. As a result, the Staff has concluded that this quencher design can be operated at local pool temperatures up to 200°F. Since the pool temperature transients have shown the bulk pool temperature will not exceed 175°F, the temperature concern does not appear significant for this design.

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Dated at Bethesda, Maryland,
this 6th day of December, 1979.