12/6/79

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
JULY 17, 1979 SET OF INTERROGATORIES

The NRC Staff responds as follows to the July 17, 1979 set of interrogatories propounded by John F. Doherty to the Staff in the captioned proceeding:

1. Why has the NRC ceased publication of inspections results in NUREG-0300? The results were helpful in evaluating Applicant's performance at South Texas Project, which is relevant to this proceeding. Now, Intervenors must make a 180-mile trip on the chance there is some report at the Bay City Public Document Room.

Response

The NRC ceased to publish summaries of inspection results in NUREG-0030, "Construction Status Report--Nuclear Power Plants," because such information is not pertinent to the purpose of the report. We are enclosing a list of reports that are shelved in each Local Public Document Room. NUREG reports are available from "National Technical Information Service," Springfield, Virginia 22161.

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2. Which "Technical Specifications" would be changed in regard to underestimation if void collapse during overpressure transient as mentioned in the bottom line of page 4-7 of Supplement #2 of the SER, was found in the calculations?

Response

If the amount of reactivity inserted due to void collapse during an overpressure transient has been underestimated in the safety analysis due to an underestimate of the magnitude of the void coefficient, the change in critical power ratio resulting from the transient will have been underestimated. The operating value of the critical power ratio is established by adding the largest change in critical power ratio (ΔCPR) produced by an anticipated transient to the safety limit value. The sum then becomes the operating limit for the minimum critical power ratio and is included in the Technical Specifications (Specification 3.2.3 of the Standard Technical Specifications (NUREG-0123, August 15, 1976)).

3. Is there any other basis than that the point kinetics model contains additional conservatisms for stating change in technical specifications will accommodate suspected underestimations of reactivity insertion due to collapse of voids (steam bubbles) during overpressure transients?

Response

The additional conservatisms which may be present in the point kinetics code affect only the magnitude of the change which might be required in the technical specifications. The implication in the SER statement is that sufficient conservatism might be present to obviate the need for any change.

With respect to the calculation of overpressurization transients General Electric has recently developed and submitted for review an improved code* for use in the analysis of these transients. This code correctly treats the spatial void effects. The review of the code is essentially complete and approval is anticipated. Overpressurization transients will be recalculated with the approved code and operating limits established based on the results.

4. Does Staff maintain applicant's rod control and information system (RCIS) can be shown proven to function satisfactorily by any use in commercial BWRs since March of 1977?

Response

The rod control and information system to be used on Allens Creek has been designed for BWR/6's. Since no BWR/6 is yet operating no operating experience has been gained with the system. However, prior to the completion of construction of Allens Creek, operating experience will have been gained. The Grand Gulf reactor, for example, is in the operating license stage of review. Such experience will be available at the time of the operating license review for Allens Creek and any lessons learned from that experience will be applied to Allens Creek.

5. What is the name of the "one dimensional calculation referred to above" on p. 4-11 of the SER?

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* NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, Volumes 1 and 2," and NEDO-24154-P, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, Volume 3 (Proprietary)." The one-dimensional transient model described in NEDO-24154 (called ODYN) has been qualified against turbine trip tests performed at the Peach Bottom plant. The test results are reported in EPRI-563, "Core Design and Operating Data for Cycles 1 and 2 of Peach Bottom 2," June 197 and EPRI-564 "Transient and Stability Tests at Peach Bottom Atomic Power Station Unit 2 at End of Cycle 2" June 1978. The comparisons with ODYN calculations are reported in NEDO-24154.

Response

The name of the one-dimensional calculational code referred to in the SER is not known to the Staff. It is part of a code package used by General Electric and has been informally described to the Staff but has not been formally reviewed. It is similar to other such codes and is briefly described in Section 4.3.3.2.6 of the BWR/6 Standard Safety Analysis Report. The scram curve, which is obtained by their code, has been independently calculated by our consultant, Brookhaven National Laboratories, using a two-dimensional space time diffusion code, TWIGL-2. The agreement between the Brookhaven results and those of General Electric is excellent and is presented in BNL-NUREG-50584 as referenced in the SER.

The use of one-dimensional transient codes to obtain scram curves and the comparison of these to higher order calculations is discussed in NEDO-24154, Qualification of One-Dimensional Core Transient Model For Boiling Water Reactors, Volume 1, (PPQ2-1ff) and in a letter, E. D. Fuller to D. F. Ross, dated January 13, 1978, and numbered MFN 014-78. While this comparison is for the one-dimensional transient code used in ODYN, that code is similar to (if not identical to) that referred to in the SER. The results presented in these comparisons show that the one-dimensional code is conservative with respect to a three-dimensional one.

6. On the enclosed drawing, indicate where some (but not necessarily all) containment vacuum breakers are located with an "X", please.

Response

See attached drawing.

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7. On the same drawing indicate where some drywell containment breakers are located with a "Y", please.

Response

See attached drawing.

8. Why will the NRC wait until after the Construction License is issued before reviewing the allowable limits and method chosen for monitoring slope and depth of sediment at the intake structure of the ultimate heat system?

Response

The Applicant has committed to design the ultimate heat sink such that failure of man-made structures in the ultimate heat sink will not impair the minimum functional capability of the ultimate heat sink. The Staff considers this commitment to be in accordance with the positions of Regulatory Guide 1.27, "Ultimate Heat Sink," and therefore acceptable at the construction permit stage of review. A review prior to the operating license stage of review has been agreed to by the Applicant to avoid later delays and complications, e.g., draining a filled lake, in the event the Staff does not find the Applicant's proposed final design to be acceptable. The commitment by the Applicant to provide information for an interim review is prudent and will allow sufficient time for review and establishment of requirements by the NRC technical staff to assure satisfactory implementation of the commitment to Regulatory Guide 1.27 positions.

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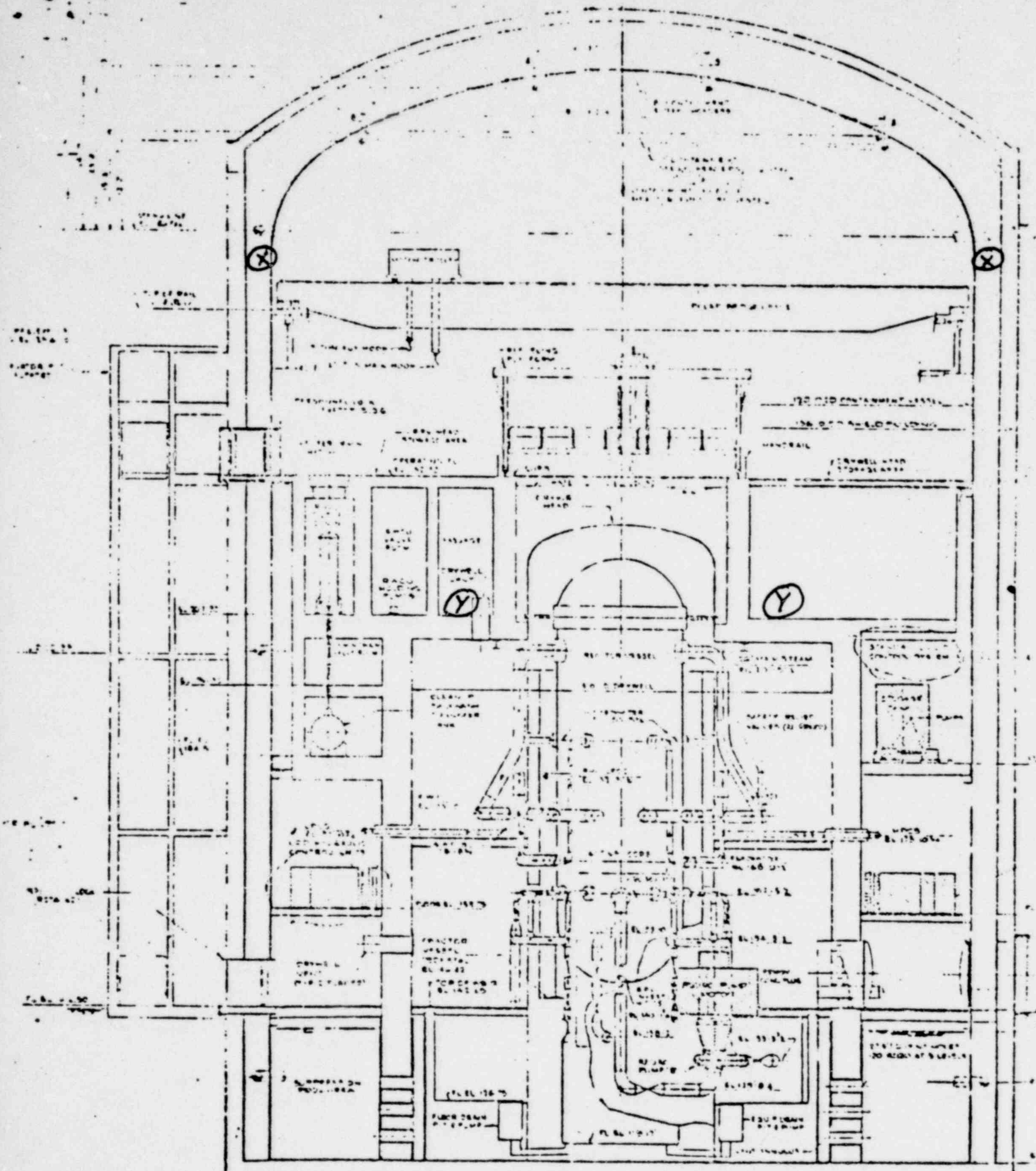
Dated at Bethesda, Maryland,
this 6th day of December, 1979.



SECTION A - A

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