UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

07/28/80

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

NRC STAFF'S PARTIAL RESPONSE TO
JOHN F. DOHERTY'S FOURTEENTH SET OF INTERROGATORIES

The NRC Staff responds, in part, as follows to the fourteenth set of interrogatories propounded by John F. Doherty in the captioned proceeding. By agreement with Mr. Doherty, the remaining responses will be filed as soon as the necessary Staff reviewers complete current review assignments.

GENERAL INTERROGATORIES

1. Relevant to Contention #45, in an enclosure to a memo from R.O. Meyer, Leader Reactor Fuels Section, NRC to R.L. Tedesco, (NRC) 7903130590 of Feb 7, 1979, "Enclosure 3", Background for Jan 10 Strategy Meeting, it states:

   II. The North Anna Criteria
   1. Calculated forces were multiplied by a Safety Factor of 1.75 (we backed down from an original requirement of 2.0 because they could not make it).

   (a) Will the North Anna Criteria apply to ACNGS
   (b) What is the criteria 1.75 or 2.00? (I assume the units are gravitational)
   (c) If the answer to (b) is "no" or "neither", how will the ACNGS safety factor be calculated?

Response

(a) No.

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(b) The so-called Safety Factor was the ratio of the measured grid deformation strength to the caluculated impact load on the grids for a combined earthquake and LOCA. It is a dimensionless parameter.

(c) The North Anna criteria have been superseded by procedures set forth in Appendix A of proposed Revision 2 of Section 4.2 of the Standard Review Plan (NUREG 75/087). This proposed revision was noticed in the Federal Register February 27, 1980 and is available in the PDR.

2. Relevant to Tex PIRG Contention #33 - Pressure Vessel Intetrity

(a) Is neutron irradiation the only source of increase of RT_{NDT}?

(b) What percentage of embrittlement is believed cause by the neutrons of between 0.1 Mev and 1.0 Mev by staff calculations due to neutrons of this energy in: (1) Piping, (2) Reactor vessel head, (C) the reactor vessel.

Response

The shift in RT_{NDT} as a function of fluence at operating temperatures is the principal property monitored in the reactor vessel surveillance program. Neutron fluence is the only source considered in the increases of RT_{NDT}.

The shift in RT_{NDT} due to neutron fluence is calculated from Regulatory Guide 1.99. The calculated value is a conservative estimate of the shift and is based on the use of all available data, including data obtained at all neutron energies. The shift is greater in the vessel beltline region and is relatively unimportant (in terms of toughness degradation) in pipe and the vessel head.

4. Kindly correct me if I am incorrect, but I believe the limit on average rate of reactor coolant temperature per hour is 100°F. What conservatisms were built into the determination of the limit described above? (Relevant to TexPIRG 39)

Response

This question is related to the conservatism built into the reactor vessel and the pressure-temperature operating limits imposed on the reactor vessel.

Appendices G and H of 10 CFR Part 50 require that the reactor vessel pressure-temperature limits provide operating safety margins at least equivalent to those recommended by Section III of the ASME Code. The Code requires fracture toughness testing of ferritic materials in the reactor pressure boundary and provides a basis for determining the limits for normal, upset, and test conditions. The Welding Research Council Bulletin 175 provides the technical basis for the Code requirement.

The principles of linear elastic fracture mechanics are used to determine the reactor vessel safe operating conditions. At all times during heat-up and cooldown, the referenced stress intensity factor must be greater than the sum of twice the stress intensity caused by pressure plus the stress intensity caused by thermal gradients. The stress intensity caused by thermal gradient is less than that caused by pressure and conservatively given in WRC 175.

5. On page 5.2-20 of the PSAR, it states:

> Charpy V-notch tests as defined in BS2321-2 are to be conducted on both unirradiated and irradiated ferritic materials; however, the special beltline longitudinally oriented Charpy speciments required by the general reference NB-2300 and specifically NB2322.2(a)(6) will be included in the surveillance program.

(a) Has Staff agreed to this?

(b) Won't allowing this cause loss certainty in determining the upper shelf energy?

(c) Why is it necessary to grant this omission?

(d) Is the omission being granted for all BWR/6? (Rel. to TexPIRG39)

Response

Appendix H of 10 CFR Part 50 requires a surveillance program to monitor the change in fracture toughness from exposure to neutron and thermal environments. The staff reviews the surveillance program for compliance with Appendix H. The program for the Allens Creek Station complies with the regulation.

6. Is weld metal usually considered the limiting vessel material for determining RT_{NDT}? (Rel. to TexPIRG39)

Response

Weld metal is not necessarily the limiting material in a surveillance program, although the shift in RT_{NDT} is usually greater in weld metal than in wrought metal of similar composition.

8. The following five questions are relevant to Contention 15 (Doherty)

   A. How has G.E. justified setting a 0.50 inch fuel rod deflection limit? (see .2.1 B.4-5) of the GESSAR SER, NUREG 0152).

   B. What are some of the "Additional conservatisms" included in "offsetting this underestimate is the fact that the point-kinetics models contain additional conservatisms"? (See Sec. 4.3.2(2) of SER Sup. 2).

   C. What is the basis for the statement (Since) "any changes in the void coefficients for Allens Creek which result from the generic resolution of this effect can be accommodated at the Operating License stage of review, ..." (Sec. 4.3.2(2) of the SER Supp #2)?

   D. An article in Nuclear Safety by Glen O. Vright, 8(2) 116-27, states: "A rigorous analysis of the dynamic behavior of a nuclear reactor should include the simultaneous solution of the space-dependant neutron kinetics equations in the reactor core, together with the time-dependant conservation equations of mass, momentum and energy for the coolant flow through the reactor."

      (a) Has the Applicant or G.E. provided this for the ACNGS Core?

      (b) What among them has not been completed?

      (c) For any factors not completed, what is the principal reason for the incompleteness?

Response

A. We are unable to read the citation for GESSAR SER NUREG-0152. Note that in Section 4.2.1 of NUREG-0152 a 0.060-inch (rather than a 0.50-inch limit) fuel rod deflection limit is identified. See the Staff response to Interrogatory 14 of John F. Doherty's Fifteenth Set of Interrogatories to Staff for Staff comments on 0.060-inch limit.

B. The other conservatisms in the point-kinetics model include the use of conservative values for other input parameters. For example the scram reactivity input curve used is that for end of cycle (where the initial portion of the reactivity insertion is smallest) and is a generic curve (the so called "D" curve) chosen so that a large likelihood exists that it will be bounded by all similar reactors for all cycles. Finally, the "D" curve is multiplied by 0.8 to obtain the curve used in the analysis. The Doppler coefficient is multiplied by 0.95 before use in analysis.

C. Suppose that it is discovered that the void coefficient is actually more negative than has been supposed. The result of this would be that the change in critical power ratio which occurs during an overpressure transient would be underestimated. The change in CPR would be accommodated by changing the Technical Specification requirement on operating MCPR. This, in turn, would restrict operating freedom of the plant.

In connection with Parts B and C above it should be pointed out that the Staff has essentially completed the review of a Code - ODYN - that will replace the point kinetics code for use where spatial effects are important. It is anticipated that this code will be used for the analysis of such transients at the FSAR stage.

D. The quote from Nuclear Safety is correct. However a rigorous analysis of each transient which must be analyzed for a reactor design is not practical, even when it is possible. Recently, three-dimensional dynamics codes such as MEKIN, and its spin-offs, have become generally available, but they suffer from the lack of a computer with a large enough fast access storage to permit large core problems to be attempted. Even if a particular problem can be run with these codes the amount of time and effort required to do it is prohibitive for all but an occasional check problem.

The approach that is used for design analyses is to perform less sophisticated codes that are conservative with respect to a rigorous analysis. This is done by making the simplification in a conservative direction when this is feasible, i.e., when it is known what the conservative direction is. In many cases a particular simplifying assumption will have both conservative and non-conservative effects. In these cases engineering judgment is used to choose the dominant effect and make sure that it is conservative.

The simplified codes are then compared to more sophisticated codes for certain analyses to verify their conservatism. In certain limited areas they can be compared to experiment. For example, the General Electric Transient Codes, REDY and ODYN, were compared to transient tests performed in the Peach Bottom Reactor.

(a) Neither the applicant nor GE has provided any rigorous three-dimensional space-time kinetics calculations for ACNGS.

9. Relevant to Contention 12,: Other than bypassing a control rod known to be uncoupled from the drive or otherwise faulty, and held in fully closed position by the CR drive, what other reasons normally permit a licensee to bypass a rod with the Rod Control and Information System (RCIS)? (Sec. 7.7.1 of SER Suppl. #2)

Response

So far as is known to the staff only rods that are uncoupled or otherwise faulty are declared to be inoperable. Otherwise faulty includes such things as slow scram times, inoperable accumulators, etc. There is a limit of eight on the number of such inoperable rods. The analysis of the Banked Position Withdrawal Sequence (See NEDO-21231) has considered the effect of such inoperable rods and concluded that the consequences of the RDA are acceptable (232 cal/gm) even if the most adverse combination of bypassed rods (not permitted by procedures) is assumed.

10. At what energy are delayed neutrons calculated to be emitted in the space-time code used for G.E. cores?

Response

The space-time code used by G.E. lumps all the neutrons into a single group for the dynamics calculation. Thus the energies of the delayed and fast neutrons are specifically called out.

11. Briefly, how can a Doppler effect become positive in a reactor system? Such a thing appears impossible. (Rel. to Doherty 33)

Response

There are no known reactors in which the Doppler Coefficient is positive. It is, in principle, possible to have a positive Doppler if the fission resonances broaden more with increasing temperature than the capture resonances and the fission rate is greater than the capture rate. There was some concern at one time that that condition might prevail for fast reactors where a large fraction of the resonances occurred in the unresolved resonance region of U-235. The fear was groundless.

14. Rel to Doherty 33. On P. 20 of NEDO 20,964 (quoting an NRC request for information) G.E. quotes the NRC, "(s)ignificant discrepency exists between calculations and measurements for U_{238} resonance capture." Is this still true? How confident is NRC that its data which uses a figure for this measurement accurately measures resonance capture for this substance? Indicate please in the literature any conclusion with regard to this which has been used by the NRC in proceeding toward licensing various cores.

Response

The discrepancy between calculations and measurements mentioned in this interrogatory refers to the data in the Evaluated Nuclear Data File, particularly ENDF/B-III and ENDF/B-IV. However these data are not used in the GE calculations of resonance absorptions. (See response to the referenced question in NEDO-20964-1, July 1977) so the discrepancy has no relevance to the ACNGS).

15. In an overpressurization transient without temperature increase in the core what is the gross effect on Doppler reactivity? Relevant to Doherty 33.

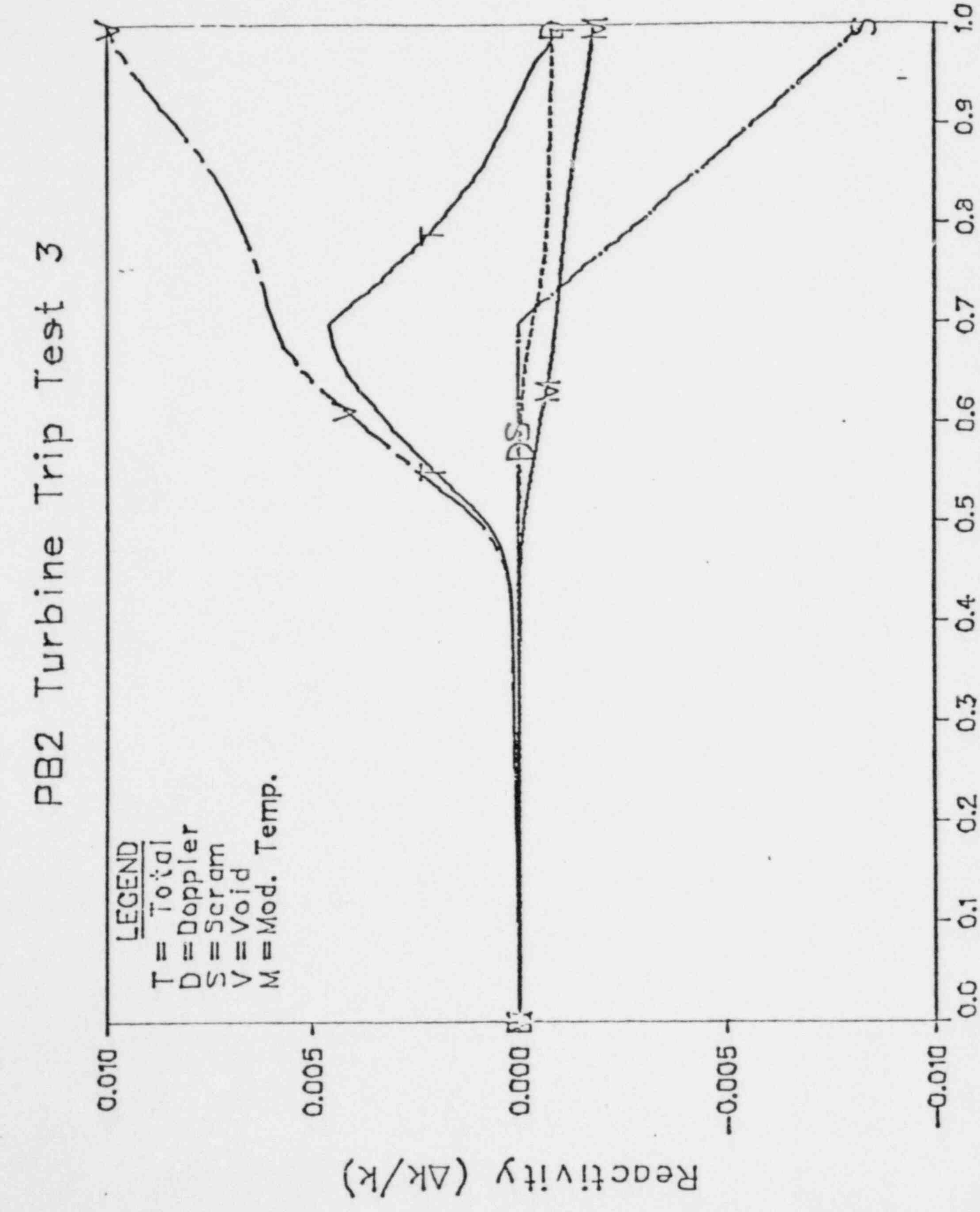
Response

In an overpressurization transient without temperature increase in the fuel there would, of course, be no reactivity feedback due to Doppler. However, no fuel temperature increase implies no power increase. In any event the Doppler effect plays a relatively minor role in overpressurization transients (See attached Figure, taken from BNL-NUREG-26689, Sept. 1979).

16. Relevant to Doherty 33. Is Doppler effect the principle shutdown mechanism under accident conditions?

Response

In accident conditions the Doppler effect serves to limit the increase in power due to a reactivity insertion (an increase in power raises the fuel temperature which gives rise to a negative reactivity to offset the externally inserted positive reactivity). Only in the case of very rapid reactivity insertions does the Doppler serve to "turn around" the power increase. In practical terms this means that the rod drop accident is the only one in which the Doppler coefficient plays a major role- and that only on the initial "spike". Shutdown ultimately depends on scram.



PB2 Turbine Trip Test 3
LEGEND
T = Total
D = Doppler
S = Scram
V = Void
M = Mod. Temp.
Reactivity (Δk/k)
0.010
0.005
0.000
-0.005
-0.010
0.0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0

Figure 34

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(Allens Creek Nuclear Generating Station, Unit 1)	)	

AFFIDAVIT OF CALVIN W. MOON

I hereby depose and say under oath that the foregoing NRC Staff responses to interrogatories propounded by John F. Doherty were prepared by me or under my supervision. I certify that the answers given are true and correct to the best of my knowledge, information and belief.

Calvin W. Moon
Calvin W. Moon

Subscribed and sworn to before me
this 28th day of July, 1980.

Elizabeth Ann Tyster
Notary Public

My Commission expires: July 1, 1982

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CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF'S PARTIAL RESPONSE TO JOHN F. DOHERTY'S FOURTEENTH SET OF INTERROGATORIES", and "AFFIDAVIT OF CALVIN W. MOON" in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class or, as indicated by an asterisk, through deposit in the Nuclear Regulatory Commission internal mail system, this 28th day of July, 1980.

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