6/30/80

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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
	§	

APPLICANT'S RESPONSE TO DOHERTY'S FIFTEENTH INTERROGATORIES

In response to the Fifteenth Set of Interrogatories propounded by John Doherty, Houston Lighting & Power Company ("Applicant") answers as follows:

1. Relevant to Contention TexPIRG 41, will the NSSS be in part controled [sic] as to pressure by the use of Barton Model 288 and/or Barksdale B2T-M1255 switches? (These are normally used in initiate opening of the SRVs in some reactors.)

Response: No.

2. Relevant to Doherty 33, what per-cent of neutrons is lost in the ACNGS core at full power at steady operating state?

Response: The number of neutrons lost has no relationship to Doppler effect. Data relevant to neutron population is in Section 4.3 of the PSAR.

3. Relevant to Doherty 33, what is the average peak SCRAM reactivity insertion rate for the ACNGS core?

Response: See PSAR figure 15.1.1-1.

4. Relevant to Doherty 47, does Applicant take the position this contention does not apply to ACNGS because the problems cited in the contention have not occured in any BWR so far?

Response: No.

5. Relevant to Doherty 39, has Applicant any data on fuel rod burn-up as a variable with clad swelling in conformance with 10 CFR 50.46 requirements? Is any of this data in NEDO 10, 329, a document available to this Intervenor?

Response: Calculations of clad swelling to show conformance with 10 C.F.R. 50.46 are contained in NEDO-20566.

6. What is the name of the manufacturer of Applicant's residual heat removal pump manufacturer? What is the model number?

Response: Byron-Jackson; Model No. $30DX20CKXH_2$-35TG.

8. Does Applicant take the position that it is not remotely possible for stored spent fuel to become unattended in the ACNGS?

Response: See response to #11-6 of John F. Doherty's 14th set of interrogatories to Applicant.

9. Does Applicant take the position the RCIS is such an improvement over the rod pattern control system in reliability that no accident possibilities need by considered?

Response: Since the RPCS is a subsystem of the RCIS, this question does not make any sense and Applicant cannot answer.

10. Does Applicant maintain a control rod drop accident is not possible for the ACNGS?

A. If "10" answer is "no", what is Applicant's estimate of the probability of such an accident, and how did Applicant obtain the statistic?

Response: No.

A. Applicant has not made such a probability estimate since a rod drop event is a Design Basis Accident (DBA) and, thus, ACNGS is designed to withstand such an event while protecting the health and safety of the public.

11. Does Applicant take the position it is impossible for there to be more than a single channel flow blockage? (DBA)

Response: No.

12. Does Applicant take the position that its reactor pressure vessel pedestal can withstand the effects of all accidents such that restart without repairs will still

provide a reactor vessel pedestal in full compliance with Commission regulations?

Response: The ACNGS RPV pedestal can withstand the effects of a DBA while protecting the health and safety of the public and is designed in full compliance with Commission regulations.

13[a]. Does Applicant take the position that any deformation of the reactor vessel would have to be the result of an event so catastrophic than [sic] ejection of a control rod, as a result of disattachment of the CRD housing from the deformed RPV would not lead to additional significant damage to the public or environment?

Response: Applicant has not analyzed in detail this speculative and extremely incredible scenario. Applicant's analysis of CRD housing failure may be found at PSAR Section 4.2.3.2.3.1.

13[b]. Does Applicant take the position a Control Rod Ejection accident is not possible with the ACNGS?

A. If the answer to "13" is "no", please give an estimate of the probability of a control rod ejection accident and the source of the estimate.

Response: No. Applicant takes the position that "possibilities" are totally subjective. Most anything is possible; a control rod ejection accident is not credible.

A. No probability estimate for a rod ejection event has been made. See PSAR Section 4.2.3.2.3.1.

14. Does Applicant take position its reactivity control system is not based in any way on the Doppler broadening effect to mitigate the effects of a transient caused over-power of the system?

Response: No. Doppler reactivity feedback effects are accounted for in the analysis of plant transients.

15. Does Applicant still maintain as in NEDO 20,964, that "The basic mathematical model in calculating void reactivity and reactivity coefficient for BWRs has been the same since 1961, "(P. 15.).

A. Does Applicant take the same position with regard to the Doppler effect?

Response: Yes.

A. Applicant has not made a historical study of Doppler effect methematical models.

16[a]. Does Applicant take the position the SPERT-I reactor did not use a powdered oxide of uranium in the experiments cited as backing mathematical models of power excursion data for G. E BWRs?

Response: No. G.E. does not base its models for predicting power excursion transients on tests at the SPERT-I reactor.

16[b]. Does Applicant take the position there will not be a shortage of trained welders for construction of the ACNGS?

A. If "yes" please cite any source of information adequate to show that this shortage will not occur.

Response: Applicant takes the position that there is no information indicating that there will be a shortage of trained welders.

A. Not applicable.

17. What is the largest increase in outside diameter Applicant maintains can occur as a result of an abnormal operating transient?

A. Give the source of any reference giving this largest increase.

Response: Applicant objects to this interrogatory on the grounds of vagueness. No system or component is identified which might experience an increase in outside diameter.

18. What aspects of the Three Mile Island - II fuel rods differ from the ACNGS to the extent that Applicant argued in its Reply to the filing of this contention that it was inappropriate to consider the contention in the construction permit hearings? (Include any metallurgical, dimensional or quantitative difference regarded as significant.)

Response: The fuel rods for ACNGS are described in Sections 4.2 and 4.3 of the PSAR. The fuel rods used in Three Mile Island - II reactor are no doubt described in the publicly available licensing documents for that project. Intevenor may draw his own conclusions as to differences.

19. Does Applicant maintain the higher operating pressure of PWR fuel rods is sufficient to indicate that BWR rods and PWR rods will act sufficiently different in swelling behavior such that BWR rods are certain not to swell enough to:

(1) Violate 10 CFR 50, App. K?

(2) Violate GDC 35?

Response: ACNGS fuel rods will comply with 10 CFR 50 Appendix K and GDC 35. Applicant has not compared BWR fuel rods and PWR fuel rods for these aspects.

20. Does Applicant maintain that lateral support of the ACNGS reactor core is sufficient to withstand the lateral force due to a flashing near the end of the subcooled blowdown portion of a LOCA transient?

Response: All lateral forces incurred during a LOCA will be designed for in the ACNGS reactor core.

21. Doe [sic] Applicant maintain there is no flashing near the end of the subcooled blowdown portion of a local transient?

Response: No.

22. Will Applicant apply the "North Anna Criteria" (See Interog. 1, Set #14, this Intervenor to Staff). In units there presented what is the ACNGS safety factor?

Response: The ACNGS core internals and core support structures will be designed to the loadings specified in PSAR section 4.2, tables 4.2-6 through 4.2-14. The core support structure will be designed in accordance with ASME Section III. Design margins are discussed in the referenced PSAR tables. Applicant does not intentionally or directly employ the "North Anna Criteria." Intervenor may make his own comparisons and draw his own conclusions concerning the "North Anna Criteria."

23. Do Applicant's calcuations [sic] research indicate ductile fracture toughness decreases as the length of a crack increases in SA 533 Gr. B., metal? (Rel to TexPIRG #39)

Response: No.

24. What would be the effect predicted of the use of higher burnup of fuel (that is "burning"-up the fuel for longer duration) on the RT_{NDT}? (Rel to TexPIRG 39)

Response: No effect of the type described is predicted.

25. What effect is expected on the void faction of being required to increase coolant temperature at the same pressure to avoid exceeding the RT_{NDT} based Technical Specifications? (Rel to TexPIRG 39)

Response: No effect of the type described is expected.

26. Relevant to Doherty 14, in the event the crew must leave the facility with the reactor in operation or uncertain status, will they be instructed, required, or have as standard operating procedure, to leave the RHR pumps in reactor cooling mode or back up to the fuel pool cooling system?

Response: See response to #8 above.

27. The following questions are relevant to TexPirg - 39

A. Are there any automatic start or stop actions of the reactor when the limit on the average rate of reactor coolant temperature change during normal heat-up or cooldown is exceeded? Include alarms in your answer if appropriate? [sic]

B. On 5.2-19 of the PSAR, it state [sic] the reactor vessel pressure retaining components comply with requirements of NB-2300 in Summer 1972 Addenda of ASME code. On 5.2-20, it states "Charpy V-Notch tests as defined in

NR2321-2 [sic] are to be conducted on both unirradiated and irradiate [sic] ferritic materials; however, the special beltline longitudinally oriented Charpy specimens required by the general reference NB-2300 and specifically NB2322. 2(a)(6) will not be included in the surveillance program?

(1) Why omit the specimens?

(2) Has NRC permitted this? If "yes" give reference in PSAR or make available copy of this authorization, please.

(3) How will this lack of information be regained?

(4) What is the academic training, and title of persons who advised this departure from the ASME code to Applicant?

C. In 5.4.3(c) of PSAR, how is the average reactor coolant temperature determined? If you took temperatures at different parts of the core, piping etc., explain the locations and how wieghted? (I presume this a simple question)

Response:

A. There are no automatic actions based on heatup or cooldown rate.

B.

(1) Transverse specimens will be tested. This testing is more conservative than testing the longitudinally oriented specimens.

(2) Yes. See GESSAR Safety Evaluation Report, Section 5.

(3) See (1) above.

(4) Applicant complies fully with the ASME Code.

C. See PSAR figures 5.1-3 and 7.7-14, as well as Section 7.7.

28. What is Applicant's estimate of the amound [sic] of channel deformation that could occur without interfering with control rod blade insertion and what is the source of the data? (Note: please answer with a linear measurement but feel free to describe in other ways.) Rel. to Doherty 45.

Response: This analysis is contained in NEDE-21354P.

29. Rel to TexPIRG 55, is the void fraction of greater percentage in the upper 1/4 of the core (upper 1/4 of the Fuel rod region than the bottom 1/4 of the fuel rod region?

Response: See PSAR Table 4.4-3.

30. Relevant to Contention 24, Doherty, does Applicant take the position that even if a rod drop occurs there will be no dispersal of fuel?

Response: Yes.

31. The following are Relevant to Contention 33

A. In NEDO 20,943, P 4.3-6, it states "a local Doppler feedback associated with a 3,000° F. to 5,000° F.

temperature rise is available for terminating the initial burst." How can this possibly assist in reactivity control unless there is a 3,000°F. to 5,000° F. change in fuel temperature in a rapid removal of a control rod?

B. What is the "intermediate resonance approximation", (last paragraph of p. 4.3-7 NEDO 20,948)? (It is unclear if this is referred to in the next sentence, please clarify.)

C. NEDO-10,527, P. 4, states that the accumulation of Pu_{240} and accident and scram reactivity shape function characteristics become more favorable as gadolinia depletion occurs. Wouldn't this mean that the consequences of rod drop accident due to the increased Dopple [sic] feedback at start after reload are most severe and become less so as the time between reload passes?

D. Is Doppler effect the principal shutdown mechanism under accident conditions?

E. In the core neutron calculations, how was the Doppler weighting factor determined? (See NEDO 20,964)

F. In Table I, NEDO 20,964-a, for 40% in channel void and 1,000 MWd/t fuel exposure, it says Doppler Coefficient = -0.6691? Is this an error or not? If not, give full particulars, please.

G. In a hypothetical overpressurization of the reactor, where pressure increases, but fuel temperature remains the same what is the gross effect on the Doppler reactivity of the system?

Response:

A. The design limit for a rod drop excursion is 280 cal/gm. This is equivalent to approx. 5150°F.

B. This complex approximation is explained in F. T. ADLER, G. W. HINMAN, and L. W. HORDHEIM, "The Quantitatine Evaluation of Resonance Integrals," Paper No. P/1938, Proceedings of the Second United International Conference of the Peaceful Uses of Atomic Energy, Vol. 16, p. 155-171, United Nationls, Geneva, 1958.

C. Since Doppler reactivity feedback increases with increased fuel exposure, the contribution of Doppler effect to reduce the severity of the rod drop accident also increases.

D. The contribution of Doppler effect as a shut down mechanism varies depending on the accident under study. For a loss of coolant DBA, it contributes very little. For a rod drop accident, it becomes a major contributor.

E. See NEDO-20964 sections 2-3.2 through 2-3.6 titled "Calculation of the Doppler Weighting Factor".

F. It appears that the number may be in error. General Electric has not identified the source of the apparent discrepancy.

G. If fuel temperature remains constant, Doppler reactivity remains constant.

32. Rel [sic]to Doherty 47, please identify the proposed turbine fully.

<u>Response</u>:

<u>NAME PLATE DATA</u>

Rating: 1211393 KW	No. 170X695 18 RPM	14 STAGES Quality 99.47%
Steam Conditions: Pressure 945 PSIG	Temperature 540F	Exhaust Pressure: 3.5" Hq. ABS.

Two-Stage Steam Reheat

Further information can be found in PSAR Section 10.2 or in documents previously requested by Intervenor.

33. What is reasonance capture of U_{238} used in Doppler calculations for the ACNGS core expressed in barns?

<u>Response</u>: 19.

34. How has General Electric justified setting a 0.060 inch Fuel Rod de lection [sic]limit (See 4.2.1 (P 4-5 of GESSAR-SAR, NUREG 0152) Relevant to Doherty 15).

<u>Response</u>: See NEDO-20948-P.

35. In the G.E. core neutronics code are delayed neutrons considered in the calculations? (Relevant to Doherty 15)

<u>Response</u>: Yes.

36[a]. In the WIGLE code are all neutrons considered fast (prompt)? (Relevant to Doherty 15)

Response: G.E. does not use the WIGLE Code and can not comment on its content.

36[b]. Relevant to Doherty 48: In its minutes of a meeting on 1/16/79, the ACRS subcommittee on ECCS pointed out that a lowered rate of system depressurization can have a negative effect on the ECCS. The minute [sic] contain a graph which shows test data indicating fuel rod temperatures will oscillate near 860° K. instead of drop [sic] to 400° K. as predicted by the RELAP-4 code, for 500 seconds. Has Applicant or G.E., considered this finding when it decided it wished to eliminate the control rod drive return line? How did it decide to eliminate the CRD return line, then?

Response: The RELAP-4 Code is not used by GE for analysis of the ECC systems. Further, the contribution of any flow to the reactor from the control rod drive system is neglected in the analysis. The decision to remove the CRD return line was made based on IGSCC considerations.

37. What documents, studies, reports or other written material will Applicant use to disprove this Intervenor's contention that: (a) there is inadeuqate [sic] design and operating safeguards to protect the spent fuel pool during a DBA; (b) the RCIS is unreliable and foreign material may

impede control rod mechanisms; (c, the core neutronics codes undergenerate the true scram reactivity function; (d) reactivity insertion from a dropped rod will result in danger to life and safety; (e) flow blockage accident analysis should assure coolant flow blockage to more than one assembly; (f) concrete in the RPV pedestal will be weakened following LOCA and a hazard on restart; (g) ejection of a control rod is serious danger to the public; (h) unacceptable blockage of the submerged intake canal is possible from the design; (i) the shutdown capability of Doppler feedback is less than the core design indicates; (j) Applicant's plans with regard to welders and their training are inadequate; (k) Cold shut-down cannot be achieved in 24 hours; (l) ACNGS fuel rods are subject to swelling under certain accident conditions; (m) the information system for the PARVs will give false "closed" indications; (n) fuel rod holding system is inadequate against lateral loads at the end of the ECCS cycle; (o) a fall of one notch by a single control rod can cause fuel failure under conditions of high xenon when the reactor is in hot standby condition; (p) damage to critical components of that plant and to the steam train will result from cracking of turbine discs and turbine missiles; (48) removal of the CRD return line is hazardous because there is insufficient high pressure ECCS water in forseeable situations? Produce and make available each and every document which you will use to rebut or otherwise disprove the Contentions, as described above.

Response: Applicant can not state that each and every document which might be used in the presentation of its case with respect to the referenced contentions has been identified. If documents which have not already been made available to Intervenor are relied upon, they will be produced for inspection and copying when they are identified. Applicant also notes that some of the statements contained in (a) through (p) and (48) above are not correct representations of admitted contentions in this proceeding. Nothing in this response should be interpreted to mean that Applicant acquiesces in any attempt to alter the nature of admitted contentions.

38. Has Applicant reached any decision with regard to the problems which prevented reply to this Intervenor's Interrogatory 15-11 served Feb. 19, 1980? (Applicant is respectfully reminded of the provision of 10 CFR 2.740(e)(2), to seasonably update replys to interrogatories).

Response: Applicant did not receive an interrogatory identified as 15-11 served on February 19, 1980.

All documents referenced in these answers will be made available for inspection and copying at the Energy

Development Complex. No presently identified expert witnesses answered any of these inquires. Applicant recognizes its obligation to supplement or amend these answers in light of further "work" and will do so where and when appropriate.

Respectfully submitted,

C. Thomas Biddle, Jr.

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TB:08:A

STATE OF TEXAS §
§
COUNTY OF HARRIS §

BEFORE ME, the undersigned authority, on this day personally appeared Thomas E. Braudt, who upon his oath stated that he has answered the foregoing Applicant's Response to John F. Doherty's Fifteenth Set of Interrogatories to Houston Lighting & Power Company in his capacity as Project Licensing Engineer.

Thomas E. Braudt
Thomas E. Braudt

SUBSCRIBED AND SWORN TO BEFORE ME on this the 27th day of June, 1980.

Cheryl A. Southworth
NOTARY PUBLIC IN AND FOR
HARRIS COUNTY, T E X A S

My Commission Expires: 8/19/81
Cheryl A. Southworth

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 No. 1)	§	

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing Applicant's Response to John F. Doherty's Fifteenth Set of Interrogatories to Houston Lighting & Power Company in the above-captioned proceeding were served on the following by deposit in the United States mail, postage prepaid, or by hand-delivery this 30th day of June, 1980.

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