RELATED CORRESPONDENCE

June 19, 1983

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
DUKE POWER COMPANY, et al.	)	Docket Nos. 50-413
	)	50-414
(Catawba Nuclear Station,	)	
Units 1 and 2)	)	

CESG'S RESPONSE TO APPLICANT'S INTERROGATORIES AND REQUESTS TO PRODUCE REGARDING CESG'S CONTENTION 18

Herewith CESG's response to Applicant's interrogatories and requests to produce dated Dec. 3, 1983.

CESG Contention 18/Palmetto Alliance Contention 44

7. Specify each regulatory guidance provision which you contend should be but has not been followed for the Catawba reactor vessels with regard to fracture toughness.

I know of none.

8. With respect to each guidance provision identified in your response to Interrogatory 7, specify the manner in which you contend the reactor vessel does not satisfy that guidance.

See 7.

9. What are your bases for your responses to Interrogatories 5 through 8? Identify all documents, testimony or oral statements by any person and legal requirements on which you rely in support of your position.

Not applicable.

11. Do you contend that Applicants will be unable to monitor adequately the fracture toughness of the reactor vessels during their operating life?

Probably.

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12. If your response to Interrogatory 11 is in the affirmative, please specify the measures which you contend must be taken to assure that changes in fracture toughness are identified.

I am not able to specify the measures.

13. Do you contend that data obtained from surveillance for reactor vessel fracture toughness cannot be utilized to determine the conditions under which the vessel can be operated safely?

I do.

14. If your response to Interrogatory 13 is in the affirmative, please specify the information you contend must be obtained and how that information must be utilized to determine the conditions under which the reactor vessel can be safely operated.

I cannot specify a solution to the problem. On the other hand there is clear evidence it exists. Referring to BAW Reports 1436. 1699 and 1697, Table 7-2, Observed Vs Predicted Changes in Irradiated Charpy Impact Properties, the four criteria considered do not give consistent nor concordant results. Weld Metal (WF-209-1) in BAW-1697 had a predicted Charpy of 206. Observed values ranged from 34 to 315°F change in transition temperature. The critical increase is 200°F, App. G IV C. Clearly the obtained numbers are meaningless. The fracture toughness has not been established.

16. For each mechanism identified in your response to Interrogatory 15, please specify the changes in material properties, including changes in molecular and granular properties, which you contend occur by that mechanism.

Not applicable.

17. For each change in material property identified in your response to Interrogatory 16, specify the consequences of that change under normal operation and anticipated operational occurrences.

Not applicable.

18. For each material property change identified in your response to Interrogatory 16, specify the consequences of that change under transient conditions.

If the numbers obtained indicative of nonductility are in error, then the Reactor Vessel Pressure-Temperature Limit Curves for a) Normal Operation Heatup, b) Normal Operation Cooldown, and c) Inservice Leak and Hydrostatic Tests will be in error. If the actual nil ductility temperature is higher than the limit curve, the probability of breach will greatly increase.

19. Specify the transient conditions under which you contend the consequences described in your response to Interrogatory 18 could occur.

Any pressure-temperature combination that falls above and to the left of the aforementioned, anpropriate limit curve.

20. What are your bases for your responses to Interrogatories 10 through 19? Identify all documents, testimony or oral statements by any person and all legal requirements on which you rely in support of your positions.

My basis for these responses is a professional career in the physical sciences including the properties of materials. I object to providing or identifying all those documents which qualify me to reach these conclusions as burdensome and irrelevant.

22. What provisions of the ASME Code do you contend are applicable to the Catawba reactor vessels? Specify the edition and addenda for each code provision.

The ASME code requirements are summarized in 10 CFR §50.55a (c)(2) or (3).

23. What provisions of the ASME Code do you contend the Catawba reactor vessels will not comply with?

I make no specific contention. Rather I contend that the experiemntal values on which judgments are to be made are highly variable and that procedures leading to presumably the same result fail in any meaningful and practical way to do so.

24. For each provision of the ASME Code identified in your response to Interrogatory 23, specify the measures which you contend must be taken to assure compliance with those provisions.

Not applicable.

25. What are your bases for your responses to Interrogatories 21 through 24? Identify all documents, testimony or oral statements by any person and all legal requirements on which you rely in support of your positions.

See response to 20.

26. Do you disagree with any of the information, data or analyses presented in the Catawba FSAR regarding fracture toughness? And subparts a., b., c., and d.

The FSAR presentation on fracture toughness is compatible with and derivative from Appendix G and and the related ASME material. I differ with the FSAR and its assurances in the same manner as indicated foregoing.

27. Please set forth each source of "stress" which you contend might result in brittle fracture of the Catawba reactor vessels.

The primary cause of stress is the difference in pressure inside the coolant system and the pressure of the containment atmosphere. All other things equal--which usually they aren't--the stresses in the various elements of the reactor will be simply proportional to this pressure difference.

A second major source of stress is internal temperature differentials. A properly annealed reactor vessel will be relieved of internal stresses. As long as it is isothermal, free of internal temperature differences, gradients, it will be free of internal stresses. Temperature gradients generate internal stresses. On reactor warmup the interior of the reactor thermally expands more than the cooler exterior. The result is that the interior is relatively undr compression, the exterior under tension.. If there were no pressure difference between inside and outside, there would be a locus of zero stress between inside and outside, a neutral surface. In normal operation a similar condition to that in heatup is present--the inner surface is, as a result, under slightly less tension than the outer surface. Heatup rate is limited to control the magnitude of the stress effect due to the differential thermal expansion. In a rapid, emergency cooldown the inner surface can become cooler than the outer. The temperature gradient is reversed, and the inner surface is, relatively speaking, under more tension. This is dangerous because it is a condition conducive to crack growth or propagation. Normal operatingtemperatures are hundreds of degrees lower than temperatures at which stress relaxation occurs. However if there is enough strain--extension--as with the outer surface of stud bolts which bend under load, there will be some stress relief by ductile slippage.

From the foregoing it should be clear that excessive pressures and/or inversions of the normal temperature gradients, move in the direction of brittle fracture in a reactor near or below the nil ductlity reference temperature.

36. For each of the measures described in your response to Interrogatory 35, set forth how those measures should be carried out, and when they should be taken.

No answer required.

37. What do you mean by "premature" in stating that premature reactor embrittlement is a hazard?

I read the intent of Appendix G as seeking to avoid a 200°F increase in nil ductility temperature at any time during the operating life of a reactor. Reaching such an increment during that period would be "premature".

38. What is the minimum water temperature to which you contend the Catawba reactor vessels should be subjected upon ECCS initiation?

I do not have a specific contention.

39. Describe in detail the accident scenario which leads to the minimum temperature specified in your response to Interrogatory 38.

Not applicable. But the most likely case is rapid cooldown from normal operation terminated by a LOCA sufficiently large to call for a high and continuing flow of emergency core cooling water; small enough to permit the operation of the injection pumps to build up a high pressure.

40. What are your bases for your responses to Interrogatories 33 through 39? Identify all documents, testimony or oral statements by any person and legal requirements on which you rely in support of your positions.

See response to 20. Additionally there is a discussion of this matter in, for example, BAW-1436 at p. 8-1.

41. What is the maximum reference temperature which you contend the Catawba reactor vessels could reach?

I make no specific contention.

42. Decribe in detail the factors which could cause the reference temperature to reach the maximum specified in your response to Interrogatory 41.

Not applicable.

43. Do you contend that Applicants' initial testing for fracture toughness of ferritic materials in the Catawba reactor vessels is inadequate?

It is more adequate than coupon testing for Oconee has been.

44. If the response to Interrogatory 43 is in the affirmative, please specify the inadequacies which you contend exist.

No answer required.

46. What are your bases for your responses to Interrogatories 41 through 45? Identify all documents, testimony or oral statements by any person and all legal requirements on which you rely in support of your positions.

See 20. Also discovery materials provided me by Mr. Al Carr concerning Oconee pre- and post-operational tests concerned with reactor integrity and ductility changes.

47. Do you contend Applicants should employ dummy rods instead of fuel in the outer core?

In the sense that Duke has overforecast, and overbuilt, it could reduce neutron fluence by employing dummy rods without problems in meeting peaks in use. However lacking knowledge of the magnitude of fluence reduction by this means I cannot make a judgment.

48. If your response to Interrogatory 47 is in the affirmative, describe the purpose and composition of those rods and the manner in which you contend they should be employed.

I do not have a basis for answering this question.

50. Do you contend measures should be taken to permit annealing treatment of the Catawba reactor vessels during their operating life?

My preference would be to succeed in having Catawba denied an OL. If successful this would moot the question.

51. If your response to Interrogatory 50 is in the affirmative, specify the measures which you contend should be taken.

No answer required.

52. What are your bases for your responses to Interrogatories 47 through 51? Identify all documents, testimony or oral statements by any person and all legal requirements on which you rely in support of your position.

The answers speak for themselves. No outside sources have been relied on.

Respectfully submitted,

Jesse L. Riley for CESG

June 20, 1983

Note: Those contentions to which no response has been given, 1, 2, 3, 4, 5, 6, 10, 15, 21, 29, 33, 45, 49, were, according to a telephone communication by Mr. Carr, regarded by Applicant as answered in CESG's response to NRC Staff interrogatories filed on April 1, 1983.

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AFFIRMATION OF SERVICE

I hereby affirm that copies of CESG'S RESPONSE TO APPLICANT'S INTERROGATORIES AND REQUESTS TO PRODUCE REGARDING CESG'S CONTENTION 18 in the above captioned matter have been served on the parties as listed in the U.S. mail this 20th day of June, 1983

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