RELATED CORRESPONDENCE

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

'83 MAR 23 A11:42

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 NRC STAFF'S SECOND SET OF INTERROGATORIES AND DOCUMENT PRODUCTION REQUESTS

CESG responds herewith to NRC Staff's interrogatories and document production requests of Dec. 15, 1982. This is filed after the 14 day period for a sequence of reasons including the press of other obligations and an illness on the part of the responder. The matter has been discussed with Mr. Johnson, Staff counsel, who indicated that it would be satisfactory to respond as soon as able. As appropriate, the responses to interrogatories will be made in six parts per the Staff request.

By "reactor degradation" CESG means a reduction in the capacity of the reactor vessel to withstand the range of internal pressures to which it will be subject in routine and faulted operation. Crack formation in welds is an early step in degradation. Crack growth is driven by stresses. Internal pressure, which increases the reactor periphery, results in stress concentration at the region of minimum radius, the leading edge, of the crack. A temperature gradient in which the interior of the reactor is cooler than the exterior, contributes stress due to a differential in thermal expansion, further increasing stress concentration in the leading edge of the crack. Crack propagation occurs more readily in brittle than in ductile materials. A high nil ductility temperature means that crack propagation can occur under conditions

DS03

which would be encountered of injection of cooling water into an appreciably pressurized reactor. The combination of cracks and incipient cracks with an increased nil ductility (brittle) reference temperature constitutes reactor degradation.

It would be pointless and burdensome to list texts, articles, news items, and 10CFR items well known to the parties.

J. L. Riley provided the foregoing response and will provide the subsequent responses. If CESG introduces a witness in this area it will probably be Riley. If another witness is secured the parties will be promptly advised.

Part (5) does not apply. See response to part (1) for (6). For the remaining interrogatories response to these items will be made only if it differs from the foregoing.

2. What are the physical or chemical mechanisms through which such "reactor degradation" is contended to occur?

See the response to 1 foregoing for the mechanical model by which the physical capability of the reactor degrades. The writer cannot speak to the mechanism by which neutron fluence alters the morphology of steel so as to increase the nil ductility reference temperature other than to note that ductile structures tend to be, for crystallizing materials, associated with small crystallite size. Fatigue, which is also a factor in embrittlement, is associated with growth in crystallite size.

3. Do you contend that such "reactor degradation" will occur at Catawba?

CESG contends that reactor degradation will occur at Catawba.

4. If your answer to the previous interrogatory is affirmative, explain fully the basis for drawing conclusions about Catawba "reactor degradation" based on the experience of other reactors?

I doubt that there is any contest that the Catawba reactors will degrade. The key question is--will the degradation be of a magnitude that has safety significance. While it is advanced by Staff that the reduced copper content of the steel in the Catawba reactors and weld materials will reduce the rate of degradation, in my view there has not been sufficient experience to make this a reliable conclusion. Just as the state-of-the-art at the time of fabricating the Oconee generation reactors was not adequate to provide an acceptably low rate of degradation, I am unaware of an evidentiary basis that the present state of the art can reliably predict the low rates of embrittlement which Staff posits.

5. What do you mean by "much more rapid increase in reference temperature than had been anticipated," as used in CESG Contention 18 (Palmetto 44)?

I have not seen the predictions made for the Oconee generation of reactors of the rate of nil ductility reference increase. However news items I have read indicated that the rate was greater than expected. It is also a matter of common sense that it would be unlikely that a large investment would be made in a plant which would, in a fraction of its lifetime, degrade so as to fail to meet specific safety criteria.

6. Please explain the conditions, timing, events, properties, through or by which this "much more rapid increase in reference temperature", etc., has occurred?

I am only able, as in the foregoing response, to indicate that the changes in embrittlement rate occurred more rapidly than had been

expected. I reject the burden of providing an explanation for a phenomenon which, to my knowledge, remains, in a fundamental, theoretical sense, unexplained but which is present as fact.

7. What do you contend to be the "anticipated" increase in reference temperature at Catawba?

I do not contend a specific increase in reference temperature at Catawba. I do contend that knowledge about rate of embrittlement was deficient at the time of the Oconee generation of reactors and that the validity of present knowledge has yet to be demonstrated.

8. What is the basis for your answer to the previous interrogatory?

See the preceding response.

9. Do you contend that the "much more rapid increase in reference temperature than anticipated" which you state has occurred at "a number of PWR's" will occur at Catawba?

I do not contend that the magnitude of the change will be the same as for high copper steels. I contend that the rate is uncertain; that the kinetics of change are not established; that the rate is not necessarily linearly time dependent.

10. Please explain fully the basis for drawing conclusions about increase in Catawba reference temperature based upon the experience of other reactors.

The experience with other reactors indicates that some years ago the experts in the embrittlement field were not able to make accurate predictions of embrittlement rate. I have seen no evidence that current expert predictions are reliable. In view of the fact that without neutron fluence there is fatigue and thermal stress embrittlement, there is every reason to expect an increase in nil ductility reference temperature, but at a rate

which will be revealed by experience but which is not presently known.

11. Please identify the "PWR's" referred to in CESG Contention 18 (Palmetto 44).

Fort Calhoun, Robinson-2, San Onofre-1, Maine Yankee, Palisades, Yankee-Rowe, Oconee-1.

12. Do you base your statements that "[t]he license should not issue" and "Catawba should not be permitted to operate" upon failure of Catawba to meet any regulatory requirement?

No. Rather on the exemptions granted by the NRC for plants not meeting regulatory requirements.

13. If the answer to the previous interrogatory is affirmative, please identify each such regulatory requirement and explain your basis for applying it to Catawba.

Response not required.

14. If the answer to Interrogatory 12 is negative, please explain fully the authority and the basis for your position that "Catawba should not be permitted to operate" and that [t]he license should not issue".

The granting of exemptions from Appendix H, 10CFR 50, as for Oconee-1, -2, and -3 presages the grant of similar exemptions for Catawba if they will permit operation after the plant falls outside the scope provided by regulation.

15. Do you contend that the Oconee plant was shut down for half a year because of "much more rapid increase in reference temperature than had been anticipated"?

Among other reasons, yes.

16. If the answer to (a) above is affirmative, please explain fully the basis for your contention.

It took time to remove and reload the full core in order to make an ultrasonic inspection, as well as to make the inspection. However I have not contended anywhere that the reactor was shut down

half a year for this cause.

17. Please clarify the meaning of your statement that "there was a series of cracks and welds shown...".

Typographical error. The reporter should have written " . . . cracks in welds . . . "

18. What is the relevance of the quoted statement in Interrogatory 17 to your contention that there has been "a much more rapid increase in reference temperature than had been anticipated."

None. See response to 17.

19. What is the relationship between your statement concerning "cracks and welds shown..." and the adjusted reference temperature of the Catawba reactor vessels?

None. See respons to 17.

20. Do you contend that the Catawba reactor vessels were manufactured or otherwise "provided" by Babcock & Wilcox?"

No.

21. Do you contend any unit of Oconee was shut down because of "reactor degradation" or "reactor embrittlement"?

Yes. Temporarily to examine the coupons for change in nil ductility reference temperature and to examine welds for the number and dimensions of cracks.

22. If your answer to the previous interrogatory is affirmative, explain fully the relationship, if any, between the Oconee shutdown and such "reactor degradation" or "reactor embrittlement".

The shutdown was made to monitor the magnitude of reactor degradation and embrittlement.

23. Do you contend that the Catawba reactors will be shut down because of "a much more rapid increase in reference temperature than anticipated"?

The Catawba reactor, under the regulations, will receive periodic shutdowns for monitoring purposes. The duration of the shutdown

will depend on a variety of factors including the degree of embrittlement, the number, location, and size of cracks, the regulations in force at the time, the willingness of the NRC to provide an exemption, if that seems to be required, and the representations of Duke to the NRC re need for power, need for earnings, etc.

24. If your answer to the previous question is affirmative, explain each and every basis for your position.

It is generous to suggest that I can "explain each and every basis for my position". I freely admit making no claim to omni-prescience. The considerations given foregoing should suffice to indicate my line of thought.

25. If your answer to Interrogatory 23 is affirmative, how often and how long do you contend the Catawba reactor will be shut down?

I cannot make any such explicit prediction. I will resort to the standard probabilistic refuge of the staff and say that there are finite probabilities that the reactor may be shut down for varying periods of time. Absent an empirical basis for assigning such probabilities I will answer, au Judge McMillan, I'm not a bookie.

26. Do you contend that a "change in the capacity factor" should be reflected in the environmental cost-benefit analysis for Catawba?

It would be reasonable to alter the capacity factor, but I doubt that the Staff is in any better position than I to be quantitative. It should be born in mind that Duke initially anticipated a capacity factor of 80% for the sister McGuire units. Experience has shown that 50% is a better reflection for 1000 MW plus units. Considering its present steam generator problems at McGuire, and prospectively at Catawba 1, Duke would be delighted to reach 50% without embrittlement considerations.

27. If your answer to the previous question is affirmative, explain in detail what cost, if any, should be "entered" and the basis for such costs.

See preceding response re quantifiability.

28. Do you contend that the environmental cost-benefit evaluation for Catawba's operating license is inadequate or incomplete unless it reflects a cost for shut down time attributable to either "much more rapid increase in reference temperature", or "reactor embittlement", or both?

Yes.

29. If your answer to the previous question is affirmative, explain fully each and every factual and legal basis for your answer thereto.

Factual: a conservative position would provide a lower limit of electrical production based on the experience indicated in response to interrogatory 11. An upper limit would be no effect. The lower limit would be estimated excluding exemption from the 100°F terminal value for nil ductility reference temperature. Legal: compliance with the NEPA requires a defensible and realistic definition of the alleged benefit.

30. Describe fully any "reactor degradation" which you contend has occurred at Oconee, including the sources for the description provided.

A report on an NRC draft letter, dated July 28, 1981, reported in INSIDE NRC. - August 10, 1981, p. 5 referring to "significantly reduced fracture toughness"; letter from an NRC employee who witnessed the Oconee UT; IE Bulletin No. 83-02 indicating serious unanticipated stress corrosion cracking in large diameter stainless steel recirculation piping at BWR plants; paragraph II.C.3.a of Appendix H, 10CFR part 50 vs. Staff response of Dec. 15, 1982 to PA interrogatories 3--" . . . an RT_{NDT}, as of December 31, 1981, at the vessel ID greater than 150°F," and 35; and Duke's

response to same interrogatory 35.

31. Do you contend that Applicants will not be in compliance with any provision of 10 CFR, Part 50, Section 50.55a, and Appendices G and H?

It has happened at Oconee, see response to interrogatory 35 (b). If present forecasts of embrittlement rate are in error it may happen again.

32. If your answer to the previous interrogatory is affirmative, what specific paragraph do you contend will not be met, and what provision thereof? When do you contend it will not be met?

Appendix H, §II C 3 a re adjusted reference temperature not exceeding 100°F. Foregoing makes clear there is no present empirical basis for projecting a date.

33. If your answer to Interrogatory 31 is affirmative describe fully the factual basis for your position.

NA.

34. Do you challenge the sufficiency of any standard contained or referenced in Appendix G and H of Part 50?

Yes.

35. If your answer to the previous interrogatory is affirmative, please identify each standard which you contend to be insufficient, and explain the basis therefor.

I am dubious about the entire construct. If the standards were really adequate there would be no need for a surveillance program. If they had been adequate the reactor degradation problem would not have occurred; IE would not be issuing bulletins on stress corrosion cracking or the failure of fasteners. It would not be necessary to further school persons in UT crack detection. The specification of reactor safety is an uncompleted art.

36. If your answer to Interrogatory 37 is affirmative, what do you contend would be sufficient standards for purposes of Appendices G and H?

I would obviate the need for such standards by denying licenses to all nuclear plants, in view of the cost of a major failure.

37. Explain the basis and reasons for your answer to the previous interrogatory.

See response to 36.

38. Do you contend that the method of calculating adjusted reference temperature specified in Reg Guide 1.99 is in any way insufficient?

The sample size is inadequate, the variance excessive.

39. If your answer to the previous interrogatory is affirmative, identify each way you contend such method is insufficient?

See response to 38.

40. Do you challenge the end of life adjusted reference temperature for the Catawba reactor pressure vessels, calculated pursuant to the methodology specified in Reg Guide 1.99?

Yes.

41. If your answer to the previous interrogatory is affirmative, identify each and every basis for your answer.

See response to 38. A linear time function is assumed. The method is nonconservative in that no effort is made to find the weakest sample. Failure will, of course, occur at the weakest point. It is optimistic to assume the complete homogeneity of weldments and, for that matter, of base stock for alloys.

42. If you challenge the end of life adjusted reference temperature for either of the Catawba reactor pressure vessels, please specify the values you believe to be correct, and explain your method of arriving at the values so reached, including the reasons for using such methods.

The question incorrectly assumes that I think a correct value can be arrived at based on available knowledge and information.

43. Do you contend that either of Catawba's reactor pressure vessels can exceed the end of life adjusted reference temperature stated in your answer to the previous interrogatory during the operating life of the reactors.

See responses to 38, 41 and 42.

44. If your answer to the previous interrogatory is affirmative, please give the basis in full for your answer.

NA.

45. At what point in the life of the Catawba reactor pressure vessels do you contend the adjusted reference temperature calculated pursuant to Reg Guide 1.99 will be reached.

At a different point than the Staff thinks it will. Consider the cases of Oconee et al.

46. Explain the reasons for your answer to Interrogatory 48.

"48"? Typo?

47. Do you contend that all reactor pressure vessels have the same characteristics with respect to the increase during the operating life of the vessel in reference temperature?

No.

48. Explain the basis for your answer to the previous interrogatory.

There apparently is a scatter pattern of embrittlement rates for operating reactors.

49. Upon what variables do you contend the rise in reference temperature depends?

Neutron fluence, composition, stress history for routine heatup and cooldown, crack nucleation, stress due to inverted radial temperature gradients in ECCS operation.

50. How do each of the variables enumerated in your answer to the previous interrogatory affect the rise in reference temperature of the Catawba reactor pressure vessels? Explain the basis for your answer.

It is clear from the finding of cracks in welds as at Oconee that initiation and growth occurs under conditions in which the RT_{NDT} is acceptable to the NRC. It would seem reasonable to think that cracks would not start nor propagate under conditions of ductility and design levels of stress. But cracks have formed and have propagated. Once initiated, the stress for crack propagation is a small fraction of that for crack initiation and propagation in a whole piece. The Charpy V notch test utilizes stress concentration at a small radius to induce controlled cracking in the brittle regime. Nowhere in the Appendix H regulation do I see the matter of such crack initiation and growth dealt with. To read the material, an operation living up to technical specifications would not permit crack development. Having made it clear that there are unrecognized variables in the situation it is simple enough to remark that nil ductility reference temperature will rise with neutron fluence, rise more rapidly as trace elements including copper, p osphorus and nickel are present in increasing concentrations, rise with the fatigue associated with routine stress history, particularly the temperature gradients present during warmup and cooldown, rise with the increasing concentration of morphological crack initiation sites, and particularly with inverted temperature gradients due to ECCS cooling under pressurized conditions.

I affirm that the foregoing responses are true, correct, and complete insofar as I am able to provide them.

March 17, 1983

Jesse L. Riley for CESG

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIRMATION OF SERVICE

I hereby affirm that copies of "CESG's Response to NRC Staff's Second Set of Interrogatories and Document Production Requests" in the above captioned matter has been served on the following in the U.S. mail, first class, this 21st day of March, 1983.

James L. Kelley, Chairman
Atomic Safety & Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Dixon Callihan
Union Carbide Corporation
P.O. Box Y
Oak Ridge, Tennessee

Dr. Richard F. Foster
P.O. Box 4263
Sunriver, Oregon 97701

Atomic Safety & Licensing Appeal Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Atomic Safety & Licensing Board
Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docketing and Service Section
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Richard P. Wilson, Esq.
Assistant Attorney General
2600 Bull Street
Columbia, SC 29201

Michael McGarry, III, Esq.
Debevoise and Liberman
1200 17th Street, N.W.
Washington, D.C. 20036

William L. Porter, Esq.
Albert V. Carr, Esq.
Duke Power Company
P.O. Box 33189
Charlotte, NC 28242

George E. Johnson, Esq.
Counsel for NRC Staff
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Robert Guild, Esq.
Attorney for the Palmetto Alliance
P. O. Box 12097
Charleston, South Carolina 29412

Palmetto Alliance
2135½ Devine Street
Columbia, SC 29205

Henry Presler, Chairman
Charlotte Meck. Env't'l. Coalition
942 Henley Place
Charlotte, NC 28207

Jesse L. Riley

Jesse L. Riley for CESG