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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

NRC STAFF SUPPLEMENTAL TESTIMONY OF RALPH O. MEYER RELATIVE TO FUEL SPECIFIC ENTHALPY, GAP CONDUCTANCE, AND CLADDING SWELLING

[Doherty Contentions 3, 20(a) and 39]

Q. Please state your name and position with the NRC.

A. My name is Ralph O. Meyer. I am the Section Leader of the Reactor Fuels Section in the Core Performance Branch.

Q. Have you prepared a statement of educational and professional qualifications?

A. Yes. It is attached to this testimony.

Q. What is the purpose of this testimony?

A. The purpose of my testimony is to respond to Doherty Contentions 3, 20(a) and 39, which concern certain performance characteristics of the fuel to be used in the proposed Allens Creek Nuclear Generating Station (ACNGS). I will respond to each of these contentions separately below.

DOHERTY CONTENTION 3

This contention basically asserts that the Applicant's design limit on specific fuel enthalpy of 280 cal/g is too high and does not provide a suitable margin for protection against accidents resulting in the dispersion of fuel into the coolant through cladding ruptures produced by internal gas pressure. Mr. Doherty also asserts that the rupture of the fuel cladding will result in the following:

(a) Fuel fragments being released into the coolant.

(b) Pressure pulses due to fuel contacting the coolant water.

(c) Further degradation of cladding strength.

(d) Jamming of control rods.

Q. What is the asserted basis for this contention?

A. Mr. Doherty asserts that tests on General Electric fuel rods show that the cladding will rupture at enthalpy values of between 147 cal/gram and 175 cal/gram.

Q. What fuel enthalpy criteria were used in the Allens Creek PSAR?

A. Two enthalpy criteria were used. (1) A 170 cal/g value was used to define rod perforation (sometimes referred to as failure or rupture) which is used in the dose analysis, and (2) a 280 cal/g value was used as a limit to preclude pressure pulses and loss of coolable geometry.

Q. Which of these criteria is related to the four concerns expressed by the Intervenor?

A. All four of the above concerns are related to the 280 cal/g limit. However, the two tests referred to by the Intervenor were measurements of fuel rod perforation, which is related to the other criterion of 170 cal/g. Fuel rod perforation is a very docile phenomena that does not lead to the expressed concerns which pertain to cladding rupture leading to loss of coolable geometry.

Q. Is a radially averaged limit of 280 cal/g adequate to preclude these ill effects?

A. No, but it is not far off. We recently discovered that the 280 cal/g value, which was derived from SPERT data, was a total energy insertion rather than a radially averaged fuel enthalpy as used in the Allens Creek PSAR. The difference is about 50 cal/g so that a more correct value might be closer to 230 cal/g on a radially averaged basis.

Q. Is the NRC revising this 280 cal/g limit?

A. No, we have concluded that a revision would not be worthwhile. While it would be interesting from a scientific point of view to revise these limits, a sufficient data base does not exist for a very refined modification. And the very expensive PBF research program that was supporting this effort has recently been given a low priority so that work will not continue. The reason for stopping this work is that recent refined neutronic calculations performed for the NRC by Brookhaven National Laboratory show that reactivity insertions for the BWR will be much smaller than the values calculated by General Electric with its simpler, more conservative analysis. Enthalpy values, according to the Brookhaven study, cannot be greater than 50 to 100 cal/g--far below the 280 cal/g (or even 230 cal/g) limit.

Q. Does the Allens Creek analysis need to be redone using a lower enthalpy limit?

A. No. Based on the Brookhaven study just mentioned, it is clear that neither pressure pulses nor serious fuel rod damage (e.g., fragmentation) will occur due to a rod drop accident in a BWR.

DOHERTY CONTENTION 20(a)

Q. What does this contention allege?

A. In essence, this contention alleges that the Applicant has underestimated fission gas release from the fuel during a LOCA and that this underestimation will result in higher peak cladding temperature. Therefore, Mr. Doherty contends that the Applicant should not be permitted to use fuel rods above 24,000 MWd/t, the threshold where significant fission gas release occurs.

Q. How do you intend to address this contention?

A. I will show that the Applicant will be required to properly account for fission gas release from the fuel and its effects on the operation of Allens Creek Nuclear Generating Station Unit 1, and that any accommodation that might be needed would be made in operating limits and not in design changes.

Q. Is it possible that the Applicant has underestimated fission gas release from the fuel in the safety analysis of ACNGS?

A. Yes, the Applicant's safety analysis for the loss-of-coolant accident utilizes the General Electric GEGAP-III code to calculate initial fuel conditions. The NRC Staff believes that this code may underpredict the release of fission gas from the fuel at burnups in excess of 20,000 MWd/t.

Q. If fission gas release has been underpredicted, will this cause the peak cladding temperature to exceed the 2200°F limitation imposed by 10 C.F.R. § 50.46?

A. No, that will not be permitted. We will require the Applicant to properly account for fission gas relese at the OL stage, and any indicated increases in peak cladding temperature would be compensated by reductions in the operating limit (MAPLHGR) or by other improvements in the analysis that would eliminate the need for an actual reduction in the operating limit.

Q. Can greater fission gas release be accommodated without a fuel design change?

A. Yes. Based on generic calculations, GE results indicate that its worst-case plant would experience an increase in peak cladding temperature of about 85°F due to enhanced fission gas release at 33,000 MWd/t. This would require a reduction in allowable power (MAPLHGR) of only about 4% at 33,000 MWd/t. Since fuel with these higher burnups has lost significant reactivity anyway, such reductions in allowable power for those fuel bundles should not result in any overall plant power reductions. Therefore, fuel design changes are not needed.

Q. Could the design differences between previous fuel types and that proposed for use in ACNGS lead to higher fission gas release?

A. No, the contrary is true. The major design differences between previous fuel types and that proposed for use in ACNGS are (a) lower linear power (8 x 8 vs. 7 x 7 geometry), (b) more stable fuel (densification resistant), and (c) higher initial fill gas pressure (3 vs. 1 atm.). All of these factors contribute to lower fuel temperatures and lower fission gas release at all burnups.

Q. Should the Applicant be permitted to use fuel rods above 24,000 MWd/t even though the rate of fission gas release might be increasing?

A. Yes. Since the fission gas is retained within the fuel rod cladding, operating limits can be adjusted if needed to accommodate the effects of fission gas release on fuel performance (although actual reductions in plant power are unlikely). As indictated above, the fuel design planned for Allens Creek is a modern improved fuel type that reduces fission gas release. However, the Applicant will be required to properly account for fission gas release in its analysis at the OL review stage.

DOHERTY CONTENTION 39

Q. What does Doherty contention 39 allege?

A. Doherty contention 39 alleges that the Applicant has not provided an adequate showing that the degree of swelling and incidence of rupture of the cladding are not underestimated.

Q. Would you describe the relevant part of Appendix K to 10 C.F.R. Part 50 that pertains to the fuel cladding swelling and rupture?

A. Yes. With regard to the Intervenor's Contention 39, the relevant Section of Appendix K is I.B., which is entitled "Swelling and Rupture of the Cladding and Fuel Rod Thermal Parameters." It states that "to be acceptable the swelling and rupture calculations shall be based on applicable data in such a way that the degree of swelling and incidence of rupture are not underestimated."

Q. Where is the Applicant's ECCS evaluation model that addresses swelling and rupture described?

A. The Applicant uses an ECCS evaluation model that is described in the General Electric topical report NEDO-20566, which is entitled "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 C.F.R. 50 Appendix K."

Q. Has the NRC Staff approved the GE ECCS evaluation model for licensing applications?

A. Yes. However, the cladding rupture temperature and cladding burst strain models in NEDO-20566 had been evaluated in NUREG-0630, "Cladding, Swelling and Rupture Models for LOCA Analysis," April 1980, and found to be non-conservative over some regions of applicability. Since revised cladding models have not yet been implemented, approval of NEDO-20566 was given with the following conditions:

1. Plant analyses performed with the ECCS evaluation model as described in NEDO-20566 are to be accompanied by supplemental analyses performed with the models of NUREG-0630.

2. Permanent revision to the GE cladding models will ultimately be required to update these models so that they conform to applicable experimental data.

Q. Has the Applicant performed the supplemental calculations for ACNGS?

A. No.

Q. Based on your understanding and experience with LWR ECCS model uncertainties and sensitivities, do you expect that the Applicant will be successful in demonstrating to the NRC Staff its ability to accommodate revised cladding models and comply with the LOCA specifications of 10 C.F.R. § 50.46?

A. Yes, because the present ECCS analysis of ACNGS indicates a margin to the LOCA limits of 10 C.F.R. § 50.46. This margin may be sufficient to offset the penalty incurred with the use of more restrictive cladding models. Additionally, the fuel vendors, including GE, are frequently updating their plant analyses to draw upon new credits and benefits that come available when research or analyses demonstrate that specific analyses are excessively conservative.

Q. Suppose that after the construction of Allens Creek the Applicant is unable to meet the LOCA criteria, even with credit taken for all excess conservatisms and new models that may be available. Would there be a means by which the Applicant could safely operate Allens Creek without first modifying some aspect of the plant design, such as the ECCS?

A. Yes, our experience has been that a slight reduction in the nuclear peaking factor results in a significant reduction in the calculated cladding peak temperature and degree of oxidation. Small reductions in peaking factor can probably be made without causing a reduction in overall design power of the reactor.

Q. Is it your conclusion that the Applicant can adequately demonstrate conformance to the LOCA acceptance criteria of 10 C.F.R. § 50.46 with cladding models that do not underestimate the incidence of rupture or degree of swelling as specified by Appendix K of 10 C.F.R. Part 50?

A. Yes, and this will be done at the OL stage as we have been doing for other plants.

PROFESSIONAL QUALIFICATIONS
OF
RALPH O. MEYER

U.S. Nuclear Regulatory Commission
Washington, D. C.

I am employed as the Section Leader of the Reactor Fuels Section in the Core Performance Branch. The Reactor Fuels Section has responsibility for reviews in the area of thermal, mechanical, and materials behavior of nuclear reactor fuel.

My general technical background is that of a reactor fuels engineer with experience in fission gas release, fuel densification, steady-state and transient fuel behavior, and fuel performance modeling. I am familiar with regulatory requirements related to reactor fuel design and performance.

I hold a B.S. degree in physics from the University of Kentucky and a Ph.D degree in physics from the University of North Carolina at Chapel Hill. I studied high-temperature and high-pressure effects on diffusion in metals as a Research Associate in physics at the University of Arizona.

From 1968 to 1973 I was employed as an Assistant Metallurgist in the reactor fuel development program at Argonne National Laboratory. My research included studies of gaseous fission product migration, segregation of fissile fuel material, and restructuring of oxide fuel pellets.

From 1973 to 1976 I was employed as a Reactor Fuels Engineer in the Reactor Fuels Section of the Core Performance Branch. My principal activities during that period were related to fuel densifications, fission gas release and the behavior of mixed-oxide fuels for the plutonium recycle program. Since 1976 I have been the Section Leader of the Reactor Fuels Section.

I am a member of the American Nuclear Society and have published more than 25 technical papers and topical reports.