07/28/80

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 1)	)	

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

The NRC Staff responds, in part, as follows to the fifteenth 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

2. How does a control rod "drift out", as in Abnormal Occurrence 75-93 of Brunswick Steam Plant, Unit #2 of report 10/14/75?

Response

Control rods in BWR's may drift when something interferes with the latching mechanism which seats the drive in its notch. The direction and speed of the drift depends on the pressure differential across the drive piston. Drifts are usually very slow. A drifting rod is detected by activation of switches intermediate to the normal notch positions. Drifting rods are declared inoperable and deactivated.

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3. Were any estimates or calculations made of any of this the missing material from the fuel channels of the Vermont Yankee BWR (50-271) as set forth in the utility's Abnormal Occurrence 73-30, of 10/12/73? (25a)

Response

This question seems confused. We have not made calculations of the missing material from any fuel channels.

4. In an LER dated 9/25/74 for the Duane Arnold Energy Center, (number DPR-331/74-4, what was decided the cause of the control rod movement? (46)

Response

According to NUREG/CR-1331 (EGG-EA-5079) "Data Summaries of LER's of Control Rods at U.S. Commercial Nuclear Power Plants, January 1, 1972 to April 30, 1978," the Duane Arnold Event was attributed to a probable cause of "noise spikes in the Rod Manual Control System or Timer". The text of the LER confirms this diagnosis.

5. In an LER of 11/24/76 for the Millstone I Nuclear Power Plant (RO-76-34/IT) reactivity was inserted during shutdown apparently due to the Rod Worth Minimizer or Rod Control system of another type (RPCS?). Was the amount of reactivity ever determined?

Response

The Millstone Unit 1 reactor does not have a Rod Sequence Control System but only a Rod Worth Minimizer (RWM). The RWM was not programmed to provide guidance for this test which was a routine shutdown margin determination. Thus the RWM was shut off during reloading. The alternate of two licensed operators was in effect.

As to the amount of reactivity inserted, information presented at the December 10, 1976 meeting of the ACRS permits an estimate. The IRM trace shows that a period (not asymptotic) of about one second was reached. This corresponds to a reactivity insertion of about 0.5% $\Delta k/k$ or less.

6. In the conclusions of the Document IN 1370, "SPERT Project Report, 10/68-9/69, on page 84, it states, "Follow-on experimental programs, equivalent to the proposed Phases II and III will be required to furnish additional information before a justified level of confidence can be established with respect to the adequacy of space-time analytical techniques." What programs of research were done since 1969 to supply this want? (15)

Response

Phases II and III of the experimental series described in IN-1370 were not carried out. An example of a program to investigate large core transient response has been cited in the response to Interrogatory 5 (Contention 15) of the interrogator's fourteenth set of interrogatories. Again this reference is EPRI-NP-564 "Transient and Stability Tests at Peach Bottom Atomic Power Station, Unit 2, at End of Cycle 2, June 1978". Space-time techniques (not WIGLE) have been satisfactorily compared to these tests.

7. In WASH 1146, it states on page III-92,

   The degree of success obtained in analyzing the SPERT transient experiments indicates the development of a verified calculational model for small cores to be well in hand, so that additional work in this field would only serve to refine a sufficiently satisfactory calculational capability.

   (A) Is the WIGLE code the "verified calculational model"?
   (B) If the NRC currently holds this position, how does it justify doing so in light of the quotation below from page 86 of IN 1370?

> The fact that the WIGLE calculational scheme consistently gave a nonconservative estimate of the observed space-time effects in these simplified assemblies poses an important safety question, since WIGLE is widely use for excursion analysis. (15)

Response

(A) The code referred to in the quoted statement is evidently the PARET Code, a point kinetics code with thermal-hydraulic feedback.

(B) Does not require answer.

8. What are the currently accepted uncertain bounds in the resonance parameters of U_{238}? (33)

Response

For the important resonance parameters - neutron width and gamma width, BNL-325, Third Edition, Volume 1 shows that for the important low-energy resonances the uncertainty in the neutron width varies from one to ten percent depending on the resonance averaging about 4 percent. The uncertainty in the gamma width varies from about 4 to 15 percent with an average of about ten percent.

Perhaps more pertinent to what the interrogator wants, the uncertainty in the infinite dilution resonance integral, according to the above reference, is less than two percent. Further, the uncertainty in the "recommended" Hellstrand correlation, to which Doppler calculations are normally compared is, according to BNL-NUREG-23500, "Evaluation of Temperature Dependent Resonance Integrals Using the Hammer Code", about $\pm$ 3.5%.

11. Please clarify the following taken from P 4-15 of NUREG 0152, S. 4.3.3, "The gadolinia is uniformly distributed in the fuel pellets, but is axially distributed within the fuel rods", with particular attention to the underlined portion.

Response

What is meant by the quoted statement is that the gadolinia is uniformly distributed within each pellet, but that not all pellets contain gadolinia and different pellets may have different gadolinia concentrations. A fuel assembly will contain only a few fuel rods which have gadolinia in them. The rods which contain gadolinia may have the same concentration per pellet throughout the length of the rod as pellets containing gadolinia may be present over only a portion of the length of the rod. The distribution of gadolinia in the fuel assemblies is information proprietary to General Electric.

12. Over the past 15 years, would staff agree that the Water-to-Fuel Volume Ratio of reactor cores have increased? (15)

Response

The staff does not keep track of water-to-fuel ratios as a routine matter. However, it is our impression that changes in this quantity over the past 15 years have been small and that, if there has been a trend it has been in the direction of increasing water-to-fuel ratio. Several specific events have affected this quantity including:

1. replacement of first cycle curtains by gadolinia
2. Changes in channel thickness
3. Inclusion of water holes in the fuel assembly design
4. Reduction in control rod thickness
5. Rod diameter changes.

13. Page 15-15 of NUREG-0152 (GESSAR - SER) states conditions for the rod drop analysis. There is no xenon factor. What changes will be made due to this factor in this determination?

Response

The Xenon condition is implied by the statement that the accident occurs 30 minutes after shutdown from full power operation. Xenon concentration would be at about the full power value.

14. What justification has G.E. presented for setting a 0.060 fuel rod deflection limit? (See: P. 4-5 NUREG-0152, please)

Response

The clearance limit of 0.060 inch was based on thermal-hydraulic tests and analyses, which indicated that even with rod-to-channel clearances of only 0.030 inch that there was only a slight difference in critical power performance. See "BWR/6 Fuel Design Amendment No. 1," General Electric Licensing Topical Report NEDO-20948-1, November 1976.

15. Has there been an operating license review for an application which referenced as GESSAR-238 NSSS? If so, what were the results of the fuel surveillance program mentioned on page 4.4 of NUREG-0152 on cladding strain? (39)

Response

No, there has not been an operating license review on GESSAR-238 plant. However, in response to the second part of this question the results of fuel surveillance programs on lead 8x8 fuel assemblies are discussed in periodic reports such as NEOM-23609, "8x8 Fuel Surveillance Program, Quad Cities 1

Fuel Assembly GEH-24, Second Interim Measurement of Precharacterized Fuel Assembly," August 1977. Such reports are General Electric Company proprietary. Measurements provided in such reports generally include rod profilometry, bowing, and length. Thus "information on cladding strain" is periodically updated and fully documented, as noted on referenced page 4-4 of NUREG-0152 (Safety Evaluation Report on GESSAR-238).

16. What is fuel rod pitch? (39)

Response

Fuel rod pitch is the spacing distance between the center of a fuel rod and the center of the nearest adjacent fuel rod. For the ACNGS fuel design, the fuel rod pitch is 0.640 inches.

18. Has G.E. provided a topical report on the full power scram reactivity function, as mentioned on Page 4-21 of NUREG-0152? (15)

Response

The requirement for a topical report on scram reactivity referenced in NUREG-0152 has been satisfied by a discussion presented as part of the ODYN code verification in NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, Vols. 1, 2, and 3". Our consultant, Brookhaven National Laboratories, has also done audit studies of the BWR scram characteristics as reported in BNL-NUREG-50584, "A Dynamics Analysis of BWR Scram Reactivity Characteristics".

19. Has the AEC or NRC ever rejected or otherwise disproved the research work under contract with Idaho Nuclear Corp. under the prime contract of the AEC by Diaz and Ohanian of the University of Florida, reported in Hetrick, Dynamics of Nuclear Systems, of 1972, which showed the WIGLE code was not conservative because it underestimated the delay time of neutrons moving through a reactor core? (15)

Response

To our knowledge neither the AEC nor the NRC has taken a position on the soundness of the referenced University of Florida research. What has been done, in response to Contention 15 by Mr. Doherty, is to argue that the results are irrelevant to the Rod Drop Accident.

The doubt cast on the suitability of WIGLE apparently stems from some experiments performed at the University of Florida and reported in "Dynamics of Nuclear Systems", D.L. Hetrick, Editor, published by the University of Arizona press in 1972. This publication is the record of a symposium held in March of 1970 at the University of Arizona. The relevance of these experiments to the RDA is tenuous. The accident is a localized event and the reactivity feed back necessary to terminate the initial pulse is produced in the immediate vicinity of the dropped rod. Thus, propagation effects are not important for the initial pulse. The second phase of the excursion, involving the removal of the heat from the fuel takes place over seconds of time while the neutronics pulse traverses large distances rapidly (for example, seven feet in four milliseconds) in the Florida experiments. Further the energy transferred by the pulse is essentially the same for both WIGLE calculation and experiment.

20. Does Staff generally agree with this statement, taken from vol 34 Transactions of American Nuclear Soc. P482., "Fission-gas-induced swelling increases with fuel burnup."?

Response

Yes.

21. Because of the below statement in NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis", I have questions.

The quote: Because of BWR spray cooling and large amount of heat transfer by radiation (to the massive channel boxes), partial flow blockage is less important in BWRs than in PWRs, and GE does not employ a flow blockage model in their ECCS analysis.

A. What percent of heat is trasferred by the "massive" channel boxes during a LOCA?

B. Is the lower pressure of the BWR the main reason the flow blockage model is not required by the NRC?

C. Why would not slow depressurization with 25°C/sec. temperature ramp not produce swelling in a BWR-6?

D. Referring to C. above, can the pressure within a BWR rod become high enough that depressurization of the reactor will produce sufficient stress to cause ballooning?

Response

A. Heat is transferred to the channel boxes from the fuel rods to increase the thermal energy storage in the channel boxes. The heat transferred is a significant fraction of the heat transferred from the fuel rods. By contrast there is no such heat transfer path in a PWR. The actual amount varies with time.

B. No, during a BWR LOCA there would exist a large heat flow from the fuel rods to the massive channel boxes, which act as heat sinks. Consequently, temperature gradients would exist across the fuel bundle and locally around the circumference of the fuel rod claddings. These temperature gradients would result in non-uniform creep deformation rates of the cladding surfaces, thus limiting the degree of ballooning at rupture and therefore the degree of flow blockage. This phenomenon in conjunction with the unique BWR capability for spray cooling to the upper portion of fuel bundles would ensure that coolant will reach the rods. Consequently, NRC does not require flow blockage due to fuel rod swelling to be explicitly included in BWR analyses.

C. A slow depressurization rate with a 25°C/sec. cladding temperature ramp rate could produce fuel cladding swelling in a BWR-6 provided that both the cladding temperature and the fuel rod internal pressure are large enough. Swelling (ballooning) and rupture (but not blockage) are included explicitly in BWR LOCA analyses.

D. Yes, the internal pressure in a BWR-6 fuel rad can be large enough to cause ballooning during depressurization events. Ballooning is explicitly considered in BWR LOCA analyses.

22. Please list any decisions or uses made of the figure Applicant states appears in error. The figure (an amount) is on page 20-1 of the SER Supp #2. There, the amount $1,055,000,000 appears and Applicant claims by Amendment #2, this figure should be $1,372,042,000. (Applicant's statement is on page #16 of its Response to Doggett & Perrenod Interrogatory Set #1, served 7/1/80).

Response

The staff acknowledges that the cost estimate in SER Supplement No. 2 did not include interest charges. However, its conclusion in this proceeding will be based on a revised cost estimate and financing plan.

23. Which of the Contentions of this Intervenor will Staff recommend be postponed for consideration until the Operating Licensing stage for the Allens Creek plant with no consideration at the Construction Licensing?

Response

None have been identified at this time.

24. Is Applicant currently required to be able to reach cold shutdown in 24 hours following a DB accident?

Response

As noted on page 5.9 of Supplement No. 2 to the Staff Evaluation Report, NUREG-0515 the ACNGS can be brought to cold shutdown within 36 hours. If a lower time period should become a requirement prior to a decision on issuance of construction permits the applicant and other parties to the proceeding will be advised.

25. Is Applicant currently required to have its entire RHR system of safety-grade equipment?

Response

Criteria 1, 2 and 34 of the General Design Criteria are the principal regulatory requirements for the residual heat removal system (RHR). Further guidance by the staff is provided by Regulatory Guides 1.26 and 1.29 which delineate acceptable provisions for meeting the requirements of these criteria. These guides include quality assurance provisions and seismic design criteria associated with the "safety-grade" designation.

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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 28thday of July, 1980.

Elizabeth Ann Tyler

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 FIFTEENTH 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, by deposit in the Nuclear Regulatory Commission internal mail system, this 28th day of July, 1980:

Sheldon J. Wolfe, Esq., Chairman *
Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dr. E. Leonard Cheatum
Route 3, Box 350A
Watkinsville, Georgia 30677

Mr. Gustave A. Linenberger *
Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, DC 20555

J. Gregory Copeland, Esq.
Baker & Botts
One Shell Plaza
Houston, Texas 77002

Jack Newman, Esq.
Lowenstein, Reis, Newman & Axelrad
1025 Connecticut Avenue, N.W.
Washington, DC 20037

Carro Hinderstein
8739 Link Terrace
Houston, Texas 77025

Richard Lowerre, Esq.
Asst. Attorney General for the State of Texas
P.O. Box 12548
Capitol Station
Austin, Texas 78711

Hon. Jerry Sliva, Mayor
City of Wallis, Texas 77485

Hon. John R. Mikeska
Austin County Judge
P.O. Box 310
Bellville, Texas 77418

Mr. John F. Doherty
4327 Alconbury Street
Houston, Texas 77021

Mr. and Mrs. Robert S. Framson
4822 Waynesboro Drive
Houston, Texas 77035

Mr. F. H. Potthoff, III
7200 Shady Villa #110
Houston, Texas 77055

D. Marrack
420 Mulberry Lane
Bellaire, Texas 77401

Texas Public Interest
Research Group, Inc.
c/o James Scott, Jr., Esq.
13935 Ivymount
Sugarland, Texas 77478

Brenda A. McCorkle
6140 Darnell
Houston, Texas 77074

Mr. Wayne Rentfro
P.O. Box 1335
Rosenberg, Texas 77471

Rosemary N. Lemmer
11423 Oak Spring
Houston, Texas 77043

Leotis Johnston
1407 Scenic Ridge
Houston, Texas 77043

Atomic Safety and Licensing *
Appeal Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Atomic Safety and Licensing *
Board Panel
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Docketing and Service Section *
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Mr. William J. Schuessler
5810 Darnell
Houston, Texas 77074

The Honorable Ron Waters
State Representative, District 79
3620 Washington Avenue, No. 362
Houston, TX 77007

Margaret Bishop
11418 Oak Spring
Houston, Texas 77043

J. Morgan Bishop
11418 Oak Spring
Houston, Texas 77043

Stephen A. Doggett, Esq.
Pollan, Nicholson & Doggett
P.O. Box 592
Rosenberg, Texas 77471

Bryan L. Baker
1923 Hawthorne
Houston, Texas 77098

Robin Griffith
1034 Sally Ann
Rosenberg, Texas 77471

Elinore P. Cummings
926 Horace Mann
Rosenberg, Texas 77471

Mr. William Perrenod
4070 Merrick
Houston, TX 77025

Carolina Conn
1414 Scenic Ridge
Houston, Texas 77043

U.S. Nuclear Regulatory Commission
Region IV
Office of Inspection and Enforcement
611 Ryan Plaza Drive
Suite 1000
Arlington, Texas 76011

Stephen M. Sohinki
Counsel for NRC Staff