12/6/79

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 RESPONSES TO JOHN F. DOHERTY'S
FIRST SET OF INTERROGATORIES

The NRC Staff responds as follows to the first set of interrogatories propounded by John F. Doherty to the Staff in the captioned proceeding:

1. What is the progress of the Advisory Committee on Reactor Safeguards on ATWS?

Response

a. Documents issued since publication of Volumes 1 and 2 of NUREG-0460 in April 1978 are listed in Appendix G, "Bibliography" of NUREG-0460, Volume 3. ATWS meetings since publication of Volumes 1 and 2 of NUREG-0460 are listed in Appendix G (page G-3) of NUREG-0460, Volume 3. An extension of those lists since publication of NUREG-0460, Volume 3, is as follows:

January 4, 1979 ACRS discussion on ATWS

January 31, 1979 ACRS ATWS Working Group meeting

March 2, 1979 ACRS ATWS Working Group meeting

March 8, 1979 ACRS discussion on ATWS

August 10, 1979 ACRS discussion on ATWS

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The issue is still under deliberation by the ACRS.

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b. Not applicable (N/A).

c. C. Moon, A. Thadani, and/or W. Brooks are the Staff members with expert knowledge with regard to all ATWS related questions. Mr. Moon is the Licensing Project Manager for the A'lens Creek application. The other gentlemen are Staff technical reviewers.

d. No, except as directed by the projected rulemaking discussions.

e. One or more of the Staff members listed in part c. above will testify on the ATWS related contentions.

f. See a. above.

g. N/A.

2. Provide a copy of the subcommittee report made available to the committee on January 4, 1979.

Response

a. A copy of page 1 and pages 4 through 9 of the transcript for the 225th General Meeting of the Advisory Committee on Reactor Safeguards is enclosed. The Subcommittee's report begins on line 14, page 4.

Parts b. through g. N/A.

3. What is the staff's estimate of the probability of an ATWS in a BWR plant?

Response

a. The Staff's estimate of the frequency of an ATWS in a BWR is approximately 2×10^{-4}/Reactor Year. The bases for this estimate are given in NUREG-0460, Volume 1. Since the actual operating experience with commercial power reactors is limited, this ATWS frequency estimate is believed to have many uncertainties.

b. NUREG-0460, Volume 1, Volume 2, and Volume 3 and the documents referenced therein. NUREGs are available for purchase from the National Technical Information Service, Springfield, Virginia 22161, or Staff will make available for inspection and copying at offices in Bethesda.

c. See 1 c.

d. No, except continued monitoring of the performance of the scram system.

e. See 1 e.

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f. See a. above.

g. NUREG-0460, Volumes 1, 2 and 3. See b. above.

4. How does the staff justify moving ahead with hearings on applicant's ATWS system when the June 1976 review by the Environmental Protection Agency says,

> As incidents occur in the nuclear power industry, their significance relative to reactor safety should be evaluated and placed into meaningful perspective.(?)

> a - Specifically, the incident at Three Mile Island, Unit 2, beginning March 28, 1979.

> b - Reserved.

Response

a. Evaluations of postulated ATWS events have been conducted for several years. With the issuance of NUREG-0460, Volume 3, the Staff believes that the significance of postulated ATWS events relative to reactor safety has been evaluated and placed in meaningful perspective. In addition, the Staff believes that in NUREG-0460, Volume 3, it has delineated specific changes in licensing requirements that are fully appropriate in response to the results of the evaluations. The Applicant has not only committed to implement those changes in the Allens Creek design, but has committed to implement alternative and additional changes that may be reflected in any regulations that the Commission may promulgate in response to the Staff's proposed rule changes. This commitment provides the basic justification for proceeding with the construction permit hearing on Allens Creek. In addition, the Staff knows of no facts that would preclude implementation of the changes prior to a decision on issuance of an operating license for Allens Creek.

The significance of the incident at Three Mile Island, Unit 2, beginning March 28, 1979, is being evaluated by the Staff and by other gorups, including investigative groups and Congressional committees. The perception of the Staff based on its reviews to date resulted in a delineation of followup actions in its letter of October 10, 1979, to Houston Lighting and Power Company. Their response for Allens Creek was transmitted by letter dated November 14, 1979. The results of the Staff's review of this response will be provided to the hearing board and to all parties, likely by a supplement to the Safety Evaluation Report.

As other TMI-2 review results become available, Staff responses will be developed and any changes applicable to Allens Creek will be required to be implemented by Houston Lighting and Power Company even if a construction permit has been issued. On the basis of evaluations to date, including the followup actions of October 10, 1979, the Staff has no reason to expect that the TMI-2 accident will suggest a need for any change in the Staff's ATWS position as stated in NUREG-0460, Volume 3. This is so because an anticipated transient at TMI-2 evolved into an accident as the result of reductions of heat removal capability and not as a result of a failure to scram.

b. Documents referenced in a. above. The October 10 letter has been sent to all members of the service list and the Applicant's response should be in the LPDR.

c. See 1 c.

d. No.

e. See 1 e.

f. See a. above.

g. See b. above.

5. How much credit is assigned the Doppler increase in an ATWS where the:
   a. Main Steam Isolation Valve trips.
   b. Turbine trips.

Response

a. No credit has been assigned for the Doppler increase because there currently is no regulatory requirement to consider main steam isolation valve trips and turbine trips without scram. As noted in response to Question 18 in Appendix B to NUREG-0460, Vol. 3, nominal values of input parameters (with the exception of the moderator temperature coefficient) have been used in analysis performed by the vendors and the Staff.

It should be noted that the Doppler effect is not generally a highly significant feedback reactivity factor in the BWR ATWS overpressure transient with recirculation pump trip (RPT). The transient is dominated by the reactivity effect of voids and the Doppler plays only a relatively minor role. The details of that role depend on the details of the reactor. However, the relative void and Doppler effect can be seen in the BNL BWR ATWS studies presented in BNL 17608 (RP-1022) and BNL 18577 (RP-1031). It may be noted from these studies that the Doppler reactivity is a small fraction of the void reactivity and the overpressurization in these events is not significantly affected by significant changes in the Doppler coefficient.

b. Page 32 of BNL Report # 17608, February 1973.
   Page 48 of BNL Report # 18577, January 1974.

c. See 1 c.

d. No.

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e. See 1 e.

f. See a. above.

g. See b. above.

6. Has the staff taken the position (and is it the current position) that the recirculation pump trip feature combined with the ability to manually shut down the reactor (by inserting control rods or by initiating the Standby Liquid Control System (SLCS)) provide sufficient protection in view of the very low probability of complete failure to scram? (This is a paraphrase of 15.1.28 of the PSAR.)

Response

a. The Staff has not taken the position that the recirculation pump trip feature combined with manual reactor shutdown provides the necessary long-term protection from ATWS events. Because of the limited number of plants operating today, the Staff has required that all operating BWR plants incorporate the recirculation pump trip feature and train operators to take corrective measures in the event of an ATWS until the Commission acts on this matter. We plan to recommend to the Commission next spring that the ACNGS be required to implement alternative #4 (as described in NUREG-0460, Vol. 3) solution to ATWS before it is permitted to begin operation. The requirements of the alternative #4 solution to ATWS go beyond the installation of a recirculation pump trip combined with manual reactor shutdown. At this stage we have the Applicant's commitment that the construction process would not restrict the ability to implement alternative #4 hardware modifications if required by the Commission.

b. NUREG-0460, Volume 3.

1591 030

c. See 1 c.

d. No.

e. See 1(e).

f. See (a) above.

g. See (b) above.

7. Does NRC take the position that all scrams are reported to it because licensee recording devices cannct be fudged, charts hidden, or other subterfuges won't evade NRC surveillance?

Response

a. The NRC Staff's position on reporting of unit shutdowns and power reductions is included as Appendix D to Regulatory Guide 1.16, Revision 4, August 1975, "Reporting of Operating Information--Appendix A Technical Specifications." This guide provides reporting procedures that represent one way of satisfying the reporting requirements of Section 50.35, "Technical Specifications." Since the Applicant for Allens Creek has committed to comply with the position of Regulatory Guide 1.16 (PSAR, Appendix C, page C1.16-1), the Staff's position is that the Applicant will report all scrams even though in theory licensee recording devices could be fudged, charts hidden, and other subterfuges used to evade NRC surveillance. This position applies to other licensees committed to the positions of Regulatory Guide 1.16.

b. Reg. Guide 1.16 and 10 CFR §50.36. Reg. Guides may be ordered by writing to the U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Technical Information and Document Control.

c. See 1(c).

d. No.

e. See 1(e).

1591 031

f. See (a) above.

g. See (b) above.

8. NUREG-0460, Volume 3, page 4 states: The principal criterion for choosing among the alternative plant modifications is the level of safety judged by the NRC to be necessary. Another important consideration is the cost of the modifications.

   a. Isn't it against policy to consider cost in safety systems? (This intervenor recognizes that cost benefit analyses are a part of the Commission rules in regard to environmental, safety, such as with 10 CFR 20 guidelines on radwaste systems.)

   b. Will costs be a determining factor in the ATWS solution for Allens Creek (ACNGS)?

   c. If answer to "b" is "yes", explain how costs will be figured in. For example, how will the applicant be permitted to not use the full solution, assuming it is Alternative 4, NUREG-0460, volume 3, page 18, 19?

Response

a. NRC has not, in the past, clearly articulated policy regarding the role of costs in safety decisions. Value impact analyses are performed to select from various alternative solutions to a safety concern although a minimum level of safety is specified irrespective of the cost. For plants of ACNGS vintage, the proposed modifications in NUREG-0460 were specified first and value impact analyses were subsequently performed. The value impact analyses support implementation of Alternative 4 for ACNGS vintage plants.

It would be speculative to estimate the extent to which costs will be a determining factor in the Commission decision on this issue and, hence, in the resolution of this issue in the ACNGS solution. However, it is expected that the role of valve impact analyses will be considered in the context of the ATWS rulemaking proceeding.

b. NUREG-0460, Volume 3 and 11/09/79 Hendrie letter.

c. See 1(c).

d. As discussed on page A-3 of the 11/09/79 letter from Chairman Hendrie to Dr. Press, Director, Office of Science and Technology Policy, NRC intends to better define the role of costs in safety decisions.

e. See 1(e).

f. See (a) above.

g. See (b) above.

9. Will the NRC be able to present data from the Three Mile Island occurrence on the adequacy of Regulatory Guide 1.77 with regard to adequate conservation [sic] in assumptions relative to peak fuel enthalpy?

   a. Will the results of the December 3, 1978, LOFT tests be analyzed by that time, and available?

a. No. A schedule for availability of evaluations of data from Three Mile Island has not been established. Since the preliminary assessment is that fuel was damaged by temperature increases caused by loss of heat removal capability, it is not apparent that Three Mile Island evaluations will contribute to an understanding of the conservatism in peak fuel enthalpy resulting from reactivity excursions. Data from the December 9, 1978, LOFT tests have been reported in "Experimental Data Report", NUREG-CR-0492. Results of the analysis of those data have been discussed in several papers. Research Information Bulletin Number 63 presents a summary and cites references to more detailed papers. A copy of this bulletin is enclosed.

b. Reg. Guide 1.77 and NUREG-CR-0492.

c. See 1(c).

d. No.

e. We do not expect to offer any testimony on the relationship of TMI-2 to peak fuel enthalpy.

f. See e.

g. See e.

10. What is the staff's current progress in the analysis of General Electric Document, NEDO-10802, "Analytical Methods of Plant Evaluations of General Electric BWRs?

Response

a. In NUREG-0460, Vol. 2, Appendix XVI, page 18, we stated that the current model is only valid for use in analyzing ATWS events wherein no significant pressure pulses are formed in the steam line during the event. The Staff has since received General Electric Company's ODYN Computer Code Model developed for use where a significant pressure pulse is generated in the steam line. By letter of March 20, 1979, to General Electric Company the Staff stated its conclusions that: (1) the technical details of the ODYN program are satisfactory, and that no modifications to the program are necessary; (2) ODYN calculations are acceptable for those transients proposed by General Electric in its letter to NRC dated March 31, 1978. The Staff also stated that use of the REDY code, as described in NEDO-10802, for the transients proposed by General Electric in its letter of March 31, 1979, will be reviewed by the Staff at a later date.

General Electric performed additional ATWS analyses in a report entitled "Assessment of BWR Mitigation of ATWS," dated May 1979 wherein they analyzed ATWS events for Alternate #3 using the REDY code and presented a comparison with the ODYN code for the short-term behavior of the plant. The Staff has requested additional confirmatory analysis for the long-term behavior. The Staff review will continue upon receipt of these analyses.

The NRC's Regulatory Requirements Review Committee has completed its review and concurred with the Staff approach insofar as it applied to ACNGS.

b. NUREG-0460, Vol. 2 and the March 20, 1979 letter to General Electric.

c. See 1 c.

d. Independent audit calculations are being performed at BNL.

e. See 1 e.

1591 035

f. See a.

g. See b.

11. Referring again to NUREG-0460, V. III, page 4, how many dollars per chance of ATWS accident was the limit on modifying standard design Mark III, BWR-6 plants? (Or answer in another way if convenient this cost-benefit type question).

Response

a. As stated on page 46 of NUREG-0460, Vol. 3, all standard designs receiving a preliminary design approval after January 1, 1978, would be required to meet Alternative #4. Although, Allens Creek is a Mark III, BWR-6 plant it does not reference a standard design. Therefore, Alternative #4 applies because a construction permit, if granted, will be granted after January 1, 1978, independent of any value-impact balancing specifically for Allens Creek.

b. NUREG-0460, Vol. 3.

c. See 1 c.

d. See part d. of response to Interrogatory #8.

e. See 1 e.

f. See a. above.

1591 036

g. See b.

12. How will the risk of an ATWS stay acceptably small in view of the fact that the limited number of operating reactors will increase? (See NUREG-0460, v. III, P. 43).

Response

a. As stated on page 43 of NUREG-0460, Vol. 3, it is the Staff judgment, considering the increasing number of nuclear power plants, that actions should be taken to require specific hardware changes that diminish the future contribution of ATWS to the overall societal and individual risk arising from nuclear power plants.

Therefore, the Staff has recommended in NUREG-0460, Vol. 3, that a mixture of preventive and/or mitigative features be implemented on various classes of plants that will provide the desired level of reduction in risk from ATWS events. For ACNGS the Staff does not intend to permit its operation to begin until appropriate level of protection for ATWS events is implemented as required by the Commission rulemaking on this issue.

b NUREG-0460, Vol. 3.

c. See 1 c.

d. No.

1591 037

e. See 1 e.

f. See part a. above.

g. See b.

13. How much more 13.4% sodium pentaborate solution in gallons will be require of licensees to meet the "high capacity liquid poison injection" requirement of Alternat in Table 1 on Page 18 of NUREG-0460, v. III?

Response

a. The required quantity of sodium penaborate solution will be determined by analysis performed in accordance with the generic resolution, e.g., rulemaking decisions.

It should be noted that the problem of interest (e.g., that indicated by the use of "high capacity" in Alternative 4) is not one of amount, but rather of rate of input, particularly early in the transient recovery period, so that the residual power level from RPT action can be quickly reduced.

b. N/A.

c. See 1 c.

1591 038

d. No.

e. See 1 c.

f. See a. above.

g. N/A.

14. Has the applicant committed to a rapid fuel failure (less than one minute) detection system?

Response

a. The Applicant has committed to provide a main steam line radiation monitoring system (MSLRM) that consists of continuous identical gross gamma detectors in each of the main steam line loops of the reactor (SER Section 7.2). As shown in Table 11.6 of Supplement No. 2 to the SER, monitors are also provided in the off-gas system. A recent report, "Fuel Failure Detection in Operating Reactors, NUREG-0401, March 1978 discusses the sensitivities of these types of systems. The conclusions of the report in part are:

> The off-gas system radiation monitor (OGSRM) of a BWR has a response time of about two minutes and appears to be sensitive enough to detect the failure of several rods, if not a single rod failure. By contrast the main steam line radiation monitor (MSLRM), which is intended for use as a gross fuel failure detector, may be far less sensitive than anticipated. The number of failed rods required to cause the MSLRM monitor to alarm may vary from hundreds to thousands.
>
> The PWR letdown line radiation monitor is similar in response and sensitivity to the OGSRM. However, the background activity in the letdown line may be considerably greater than in the BWR off-gas system and this could increase the number of failures required for detection from a single rod to perhaps as many as ten.

> The Staff concludes that these BWR and PWR activity monitors, along with outer primary system sensors (pressure, temperature, level) that would detect accident conditions, are adequate to give an early warning and allow a timely response to degrading fuel conditions. A possible exception to this adequacy is associated with a postulated BWR flow blockage accident, which might proceed undetected in its early stages. More sophisticated detection system designs are possible, but they are either in the experimental stages of development or they are impractical for use in the large cores of commercial LWRs.

It should be noted that the possible exception relates not to detection but to timeliness of the detection. Considering the obvious resistance to flow blockage inherent in the Allens Creek fuel design as described in PSAR Section 4.2, the Staff has not perceived a need for a licensing requirement for earlier detection. As noted in Appendix D to Supplement No. 2 to the Safety Evaluation Report, one of the ACRS generic issues is designated "Instruments to Detect (Severe) Fuel Failures" [Group II (4), page D-3] and the issue is under generic review by the Staff. To date a licensing requirement for more rapid detection has not been identified.

b. SER, Supplement #2 to the SER, NUREG-0401. You should have copies of the SER and its Supplement #2.

c. See 1 c.

1591 040

d. No.

e. Seε 1 e.

f. See a. above.

g. See b. above.

15. State what consequences of ATWS events Alternative #4 (See NUREG-0460 v. III, pg 18) would not mitigate and indicate what you propose to do about the consequences?

Response

a. A design that implements the Alternative #4 hardware modifications has a calculated ATWS performance within the acceptance criteria, i.e., calculations using an acceptable evaulation model, for specified event sequences, and with specified values of parameters, given values of dependent variables within the criteria. The computer codes, parameter values, the event sequences and the acceptance criteria are described in Appendices XVI and IV of volume 2 of NUREG-0460. Because of conservatisms in criteria and based on parameter sensitivity studies, the uncertainties in codes and parameter values are not expected to have significant impact on consequences. The systems relied on to mitigate consequences of ATWS events are required to meet a high reliability criterion or co ider single failures as described in Appendix C of NUREG-0460, Vol. 3. Thi means that extremely low probability sequences (ATWS followed by non-sequential multiple failures) are not considered credible even though the consequences of such sequences could be significantly more severe than the acceptable criteria.

b. NUREG-0460, Vol. 1, 2 and 3.

c. See 1 c.

d. No.

e. See 1 e.

f. See a. above.

g. See b. above.

Dated at Bethesda, Maryland,
this 6th day of December, 1979.

1591 042

CALVIN W. MOON

PROFESSIONAL QUALIFICATIONS

LWR PROJECTS BRANCH NO. 4

DIVISION OF PROJECT MANAGEMENT

I am a Senior Project Manager in LWR Projects Branch No. 4, Division of Project Management, U. S. Nuclear Regulatory Commission. In my present position, I have overall responsibility for conducting the safety review of power reactor license applications assigned to me. This includes the responsibility for planning and coordinating the efforts of other technical personnel involved in the review.

I hold a Bachelor of Science degree in Mechanical Engineering from Iowa State University and a Master of Science degree from Stanford University. I am a registered professional engineer in the State of Iowa.

I have a total of 28 years of professional experience. For three years I was employed by the University of California at the Los Alamos Scientific Laboratory with responsibilities for the design and development of mechanical systems. For fourteen years I was employed by private industry in various staff and supervisory engineering positions working on the design and development of gas cooled reactors for application to military propulsion, marine propulsion, space propulsion and space auxiliary power systems.

In 1968, I accepted a position as Reactor Engineer with the Regulatory Staff of the Atomic Energy Commission. In this capacity, I participated in the development of reactor safety criteria. In my present position I have participated in the safety reviews of several power reactors by the Atomic Energy Commission and the Nuclear Regulatory Commission.

STATEMENT OF PROFESSIONAL QUALIFICATIONS OF

Walter L. Brooks

Present Employment - I joined the Nuclear Regulatory Commission (then the Atomic Energy Commission) in September of 1974. I am a member of the Core Performance Branch of the Division of Systems Safety with the title of Nuclear Engineer. In my position, I have primary review responsibility for core physics aspects of licensing submittals and, upon request from the Auxiliary Systems Branch, the criticality aspect of fuel storage facilities.

Education - B.A. in Mathematics, Lincoln Memorial University, 1943
M.S. in Physics, New York University, 1950
Ph.D. in Physics, New York University, 1953

Previous Employment - Gulf-United Nuclear Corporation and its predecessor companies, United Nuclear Corporation, Nuclear Development Corporation of America, and Nuclear Development Associates. My duties, during my employment from 1953 to 1974, included the following:

- Performance and evaluation of critical experiments for D_2O moderated latices
- Performance and evaluation of light water moderated lattice critical experiments
- Performance and evaluation of fast reactor critical experiments
- Development of calculation methods for D_2O moderated reactors
- Verification and mofification of a nodal calculation technique for light water reactors.

Publications

Numerous reports on the results of critical experiments and methods development.

1591 044

Ashok C. Thadani

PROFESSIONAL QUALIFICATIONS

I am a principal Reactor Engineer in the Reactor Systems Branch, Division of Systems Safety, Office of Nuclear Reactor Regulation. I am responsible for coordinating the evaluation of the Anticipated Transients Without Scram for Light Water Reactors and other safety aspects of the reactor coolant, emergency core cooling, and other auxiliary systems which are assigned to me during the review of nuclear power reactor license applications.

I received a Bachelor of Science Degree in 1965 from the University of Tennessee and a Master of Science Degree in 1967 from the Catholic University. Both of these degrees are in Chemical Engineering.

From 1967 to 1968, I was employed by Melpar Co. where I performed Research and Development studies on coal utilization and air pollution control.

From 1968 to 1969 I continued my studies towards a Doctor of Philosophy Degree in Chemical Engineering at the Catholic University of America.

During 1969 to 1972, I was a Senior Engineer with Westinghouse Electric Corporation where I was initially involved in developing mathematical models for the cooldown of the NERVA reactor. Subsequently I was involved in code development efforts and accident analysis.

From 1972 to 1974, I was a staff engineer at the Singer Simulation Products where I was involved in real time simulation of nuclear power plant behavior during normal and accident conditions. I was also responsible for directing Research and Development activities in modeling techniques.

In February 1974, I accepted employment with the Atomic Energy Commission (now the Nuclear Regulatory Commission) in the Reactor Systems Branch. I have been responsible for reviewing and coordinating the review of the Anticipated Transients Without Scram for Light Water Reactors. In addition I have reviewed and am still reviewing the safety systems of both PWR and BWR designs.

1591 046