UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

1/10/80

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 RESPONSE TO JOHN F. DOHERTY'S
EIGHTH SET OF INTERROGATORIES

The NRC Staff responds as follows to the eighth set of interrogatories propounded by John F. Doherty in this proceeding.

8-1. The Staff has provided a Feb. 15, 1979 letter and enclosure from R. Mattson of the NRC to G. G. Sherwood of General Electric. On page 54, the enclosure requires G. E. to provide a list of "all systems, sub-systems, and/or components required for ATWS mitigation for an alternative 4 plant.

a/ What items were on that list?

b/ Does staff require more than these for ATWS mitigation?

c/ Has G. E. or applicant resisted or proposed alternatives or in any way indicated it will not be able to comply with additional requests for this list from Staff?

d/ If the answer to c/ above is yes, describe what these systems, sub-systems, etc. are, please.

e/ What items does the staff require in b/?

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Response

a. A complete list of plant equipment relied on to perform the major functions during ATWS events is given in Section 3.4 of a GE Proprietary Report, "Assessment of BWR Mitigation of ATWS (NUREG-0460 Alternate #3), May 1979." Some of the systems relied on to mitigate the consequences of ATWS events are listed below:

i) Automatic Recirculation Pump Trip (RPT)

ii) High Pressure Core Spray System

iii) Reactor Core Isolation Cooling System

iv) Safety/Relief Valves

v) Residual Heat Removal System

vi) Standby Liquid Control System

vii) Feedwater Runback Feature

viii) Suppression Pool and Containment

b. If the Staff recommends that alternative 4 solution be applied to the ACNGS, the following hardware modifications would be implemented. (The plant design already incorporates the RPT feature.)

i) Upgrading of the current standby liquid control system charging rate and incorporating automatic actuation circuitry.

ii) Automatic actuation circuitry to runback the feedwater flow to a lower rate upon receipt of an ATWS signal.

Besides the above described hardware modifications, the Staff would require that all automatically actuated systems satisfy the criteria described in Appendix C of NUREG-0460, Vol. 3.

Since the anticipated plant hardware modifications are based on preliminary generic analyses, the Staff has required (2/15/79 letter from Mattson to Sherwood) analyses to verify the appropriateness of these conclusions.

c. The Staff recommendation to the Commission will be to require implementation of Alternative #4 solution to ATWS on the ACNGS. Although the industry has maintained that the implementation of Alternative #4 hardware modifications are not necessary from the standpoint of safety, the Applicant has committed (Section 15.2, Supplement No. 2 to the Safety Evaluation Report, NUREG-0515) to implement Alternative #4 or any other plant modifications that may be required by the Commission.

d. Not applicable.

e. See b. above.

8-2. Referring to the document described in 8-1, on page 43 of it, this document says, "For all BWR plants . . . the suppression pool temperature limit remains an open item for ATWS."

a/ How much difference between the local temperature and bulk pool temperature does G. E. propose, and what are the maximums for each of these measurements?

b/ Does G. E. plan any changes in the quencher hole patterns?

c/ Has G. E. proposed any other methods of controlling the temperatures since this request from the NRC?

d/ Has the NRC remained firm in a 200°F temperature limit for the suppression pool for any transient?

Response

a. Based on G. E.'s interpretation of supporting data, G. E. proposed: (1) 14°F for local to bulk temperature difference and (2) no limit on local pool temperature, i.e., pool temperature can be raised up to the saturation temperature corresponding to the total pressure inside the suppression chamber (containment). The NRC Staff has had discussions with G. E. on this issue. G. E. indicated that additional experimental and analytical evidence will be submitted to support the proposed temperature difference and temperature limit on the suppression pool during safety/relief valve operation. In addition, in-plant tests will be conducted on a protypical MARK III plant. Results of the tests should provide a better definition of pool temperature difference to be used for Allens Creek before an operating license for Allens Creek is issued.

b. G. E. has not proposed any changes in the quencher hole pattern.

c. G. E. has proposed that installation of circuitry diverse from the electrical portion of the current scram system in conjunction with the recirculation pump trip would assure that the containment temperature is maintained below 200°F.

d. The NRC is maintaining the position on local pool temperature limit of 200°F on the basis of data base currently available to the Staff. This limit will be used until new data base justify otherwise.

8-3. In Appendix D of NUREG-101, the SER for Phipps Bend, in its ACRS review, it says, "We (ACRS) were unable to concur with some conclusions" . . . among them that ". . . the G. E. Co. had performed reliability analysis for alternate conceptual designs for the automatic depressurization system which included on-line testing capabilities of the pilot solenoid actuation valves. We (ACRS) further stated we were unable to concur with their conclusion because some assumptions used in their analysis had not been adequately justified."

G. E. was required to develop a program to:

(e) Establish the reliability of the pilot solenoid valves that will be used in the ADS system and the pressure relief system of the GESSAR-238.

a/ Referring to the above, does ACRS believe that G. E. has committed themselves to such a program now?

b/ What are the general results for now that their effect for the future on GESSAR-238 pressure relief systems from the program developed by G. E.?

c/ Specifically have any pilot solenoid actuation valve manufacturers been found to make unacceptable valves?

d/ And, specifically have any manufacturers of pressure relief valves been found to make unacceptable valves?

e/ If the answer to c/ or d/ is yes, state the manufacturer's name(s) and which component they manufacture, please.

Response

a. As reported in Section 7.3.2.2 of the GESSAR-238 Safety Evaluation Report, NUREG-0152, G. E. has committed to a program to establish the reliability of the pilot solenoid valves that will be used in the automatic depressurization system and pressure relief system. As stated there, the results of the program will be used to demonstrate that the required test interval for the pilot

solenoid valves is compatible with the planned refueling outage for the GESSAR-238 NSSS design.

b. As stated in Section 7.3.2.2 of the GESSAR-238 Safety Evaluation Report, we consider the G. E. program to be acceptable for the Preliminary Design approval and will report the results of our evaluation of the program at the final design stage of review. As we state there, we conclude that there is reasonable assurance that the General Electric Company program can demonstrate that the failure rates and test interval used in the analyses are acceptable. If the General Electric Program should fail to provide the required demonstration, the Applicant has committed in Section 5.2.2 of Supplement No. 2 to the SER to use safety/relief valves which will meet the criteria presented in the GESSAR.

c. See a.

d. See a.

e. N/A.

8-4. Are all SRV's in the GESSAR-238 power operated?

Response

Yes. However, as noted in Section 5.2.2 in evaluating overpressure protection half of the valves are assumed not to open until the valve spring set pressure is reached.

8-5. If the answer 8-4 is "no", how many are not power operated?

Response

See the response to 8-4.

8-6. At the start of the Harrisburg (Three Mile Island -II) accident of March 28, 1979 were there any other instruments present and functioning to measure radiation other than thermo luminescent dosimeters which do not distinguish the elements causing the gamma radiation but only the amount of gamma radiation? (D-40)

Response

Yes. Report NUREG-0558, "Population Dose and Health Impact of the Accident at the Three Mile Island Nuclear Station," May 1979 includes a discussion in Section 2 of the nature of the radioactive materials releases. Appendix B to the report describes the environmental surveillance activities of the Department of Energy which measured the radionuclides in the environment from the release. As stated in Appendix B, the environmental monitoring activities started at 4:00 p.m. on March 28, 1978, and included some in situ radionuclide identification by gamma spectrum analysis and gamma spectrum analysis of environmental samples. As stated on page B-6 of Appendix B, the conclusion regarding the predominant radionuclides in the airborne discharges was supported by information received from the NRC Licensee (Metropolitan Edison) concerning the measured composition of stack discharges, and analyses of the airborne radioactive material in the containment.

8-7. Have any test safety relief valves ever been subjected to the following condition: For a duration of 14 seconds, subjected to a pressure of 3,200 psi at a flowrate of 575 lbs/sec a steam-water mixture in supercritical state at a temperature of 650°F? (See 8-8)

Response

The conditions described in this question (3200 psi, 650°F) are not expected to occur during an ATWS event in a BWR design. Fur further discussion, see response to Question 8-8.

8-8. Does staff concur with the conclusion of NUREG/CR-687, that no safety-relief valves have ever been subjected to ATWS conditions?

a/ If no, provide the names of any studies that indicate such valves survived such conditions, please.

b/ If yes, please mention any known facility for testing such conditions, such as LOFT, or PBTF.

Response

As noted in the introduction to NUREG/CR-0687, this report only addresses the operability of safety/relief valves under ATWS conditions for Pressurized Water Reactors. The report does not contain any information about safety-relief valves used in Boiling Water Reactor Service.

For most Boiling Water Reactors, under ATWS conditions, the Safety/Relief Valves are subjected to an operating environment consistent with that for which they are specifically designed, i.e., pressure, temperature, and type of fluid relieved-saturated steam. There is very little difference in

safety-relief valve operation between that which would occur for a normally anticipated plant transient resulting in safety/relief valve actuation, such as the simultaneous closure of all Main Steam Isolation Valves, for which reactor scram would normally occur, and the same transient if the scram did not occur, i.e., ATWS. The difference is that without scram all of the safety/relief valves would relieve steam somewhat longer than if the scram had functioned and one or more lower setpoint valves would remain open relieving steam for a somewhat longer period of time.

Specifically for BWR/6 reactors such as ACNGS, ATWS analyses reviewed to date by the Staff indicate that the valves would only have to relieve saturated steam at a system pressure of 1300 psi at the highest. This is well within the design capability of the safety/relief valves of the type to be utilized in the ACNGS.

The primary coolant system components for ACNGS including the safety/relief valves have a design pressure of 1250 psi or higher and in accordance with ASME Code requirements component pressures must be limited to no higher than 110 percent of design or 1375 psi for anticipated transients where credit, by Code rules, can be taken for the pressure mitigation capability of the reactor scram. ACNGS will have sufficient safety/relief valve capacity to maintain all component pressures below the Code 110 percent limit even when no credit is taken for the function of the reactor scram.

As additional assurance of safety/relief valve performance capability, on Boiling Water Reactors, it is normal practice to verify safety/relief valve operability during low power startup testing of the plant.

At low core power but with normal system operating pressure, each safety/relief valve is manually actuated from the control room causing steam to be relieved at a pressure only slightly lower than what the valves would be exposed to under ATWS conditions.

This test is useful as a confirmatory test of basic valve function, of the design relieving capacity, and of the capability of the valve to reclose after relieving at essentially design flow conditions.

Operating experience with BWR's has included a substantial number and variety of transients in which the safety/relief valves were called on and did function. These were not ATWS sequences, but (as discussed above) the valve operation is quite similar.

Taking these factors into account, the Staff concludes that there is adequate assurance of safety/relief valve operability under ATWS conditions should the ACNGS be exposed to this low probability event.

8-9. On page i of the executive summary of NUREG-0581, it states that the results of experiments at the Power Burst Test Facility (PBTF) indicate that licensing evaluations are using gap release assumptions that are conservative for Ce and I_2 by a factor of 2. (Data on PWR rods)

a/ With this in mind, why was the release of Kr much greater than the licensing evaluation for Three Mile Island?

b/ Was the gap release of Ce and I_2 less by a significant factor than expected?

(Note: Since gap release must be measured or inferred back from plant relase in the case of TMI-2, any assumptions in working backward need not be explained at this time because I am seeking only to find out if the NUREG-0581 observations held up at TMI, or not)

Response

a & b. A discussion of the conservatisms in gap release for fission products as compared to the observed release at Three Mile Island is mostly irrelevant. At TMI-2, the reactor experienced a small break LOCA. NRC has long recognized that an analysis of this accident based on gap inventory would be non-conservative. Therefore, as per the instructions given in Regulatory Guide 4 (1970) or 1.4 (1974), the radiological consequences of a PWR LOCA are analyzed on the following basis: (1) 25% of the equilibrium radioactive iodine inventory (core inventory not gap) developed from maximum full power operation of the core should be assumed to be immediately available for leakage from the primary reactor containment, and (b) 100% of the equilibrium noble gas (Kr, Xe) inventory (core) developed from maximum full power operation should be assumed to be immediately available for leakage. The release of other fission products during a LOCA would also be expected to be greater than for gap considerations.

8-10. On page 8 of NUREG-0581 in speaking of the properties of fuel cladding, it says, ". . . with respect to irradiation effects" . . . " until recently, most information was obtained at or below the operating temperatures of BWRs". The statement dates back no earlier than June, 1979.

a/ State references to any experiments where the effects of irradiation on BWR rods were tested by subjecting the rods to increasing power until rod failure, other than those shown in Figure 15 of NUREG-0581 (Page 24, Enclosed as Exhibit A)

b/ Admit or deny that there is no data on irradiation effects on BWR fuel rods when subjected to increasing power until clad failure, at:

(1) BWR operating temperature

(2) Above BWR operating temperature

c/ Please refer this Intervenor to any data on BWR fuel rods that show the effect of burn-up of between 15,000 MWd/t and 30,000 MWd/t on fuel rod integrity with energy deposit as the other variable. (Such data would be other points on Exhibit "A".

Response

a, b & c. Fuel rod response is based mainly upon the energy input (fuel enthalpy). The current PBF RIA series are intended to study LWR fuel behavior under coolant conditions which are typical of hot-standby for a BWR. In addition to the tests used for Figure 15 (NUREG-0581), two of these RIA tests designated RIA 1-1 and 1-2 have been performed in the PBF series. Preirradiated rods (approx. 5000 Mwd/tU) were used in both of these tests. Test results have been reported in two EG&G reports[1,2] and reviewed at an American Nuclear Society (ANS) Topical Meeting on fuel performance[3]. There are a few additional data points for irradiated fuel at the 15,000 and 30,000 Mwd/tU burn-up levels as shown in the enclosed figure. This data is from the previous SPERT series and should have appeared in Figure 15.

In the evaluation of the above-noted test results, one should keep the reactivity energies and burn-up effects in the proper perspective. The energy input for fuel rod disassembly is 280 cal/g and NRC requires that this limit never be exceeded in an accident analysis. Lower fuel rod cladding failure limits (170 cal/g for BWRs) may be exceeded but must be considered for the determination of fission gas release during the accident. For energy input, Allens Creek calculates a maximum of 135 cal/g at beginning-of-life and 60 cal/g at 10,000 Mwd/tU. This level decreases further with increasing burn-up. For cladding behavior, the damage mechanisms mostly saturate after approximately 5000 Mwd/tU. Ductility is diminished during this initial irradiation and then remains fairly constant throughout the rest of the design lifetime. The loss in ductility is accompanied by an increase in strength.

(1) Z. R. Martinson et al., "Reactivity Initiated Accident Test RIA 1-1," EG&G Quick Look Report, TFBP-TR-300, October 1978.

(2) Z. R. Martinson et al., "Reactivity Initiated Accident Test RIA 1-2," EG&G Quick Look Report, TFBP-TR-303, December 1978.

(3) P. E. MacDonald et al., "Light Water Reactor Fuel Response During Reactivity Initiated Accident Experiments," EG&G Presentation, Proceedings of ANS Topical Meeting on Light Water Reactor Fuel Performance, Portland, Oregon, April 29 to May 3, 1979.

8-11. At present is there any other source of data on gap conductance for thermal reactor fuel with maximum thermal output (kw/ft) other than the Halden experiments?

Response

Yes, see answer to 8-12.

8-12. If the answer to 8-11 is "Yes", please reference and explain.

Response

The following technical reports have long been the reference base for the development and calibration of gap conductance models. In most of these tests, the fuel operated at power levels in the range from 5 kw/ft. up through the normal power levels 10-15 kw/ft. with many of the tests performed well above 20 kw/ft.

1. G. Testa, et al., "In-Pile Fuel Studies for Design Purposes," Nuclear Applications and Technology, Vol. 7, December 1969.

2. M. G. Balfour, J. A. Christensen, and H. M. Ferrarri, In-Pile Measurement of UO_2 Thermal Conductivity, WCAP-2923, March 1966.

3. I. Devold, A Study of Temperature Distributions in UO_2 Reactor Fuel Elements, AE-318, 1968.

4. G. Kjaerheim, E. Rolstad, In-Pile Determination of UO_2 Thermal Conductivity, Density Effects and Gap Conductance, HPR-80, 1967.

5. I. Devold, Past, Present and Future Instrumented Fuel Assemblies at HBWR, HPR 35 1, section 2, 1964.

6. A. S. Bain, Microscopic, Autoradiographic and Fuel/Sheath Heat Transfer Studies in UO_2 Fuel Elements, AECL-2588, June 1966.

7. M. J. F. Notley, et al., Zircaloy Sheathed UO_2 Fuel Elements Irradiated at Values of d Between 40 and 83 w/cm, AECL-1676, (December 1962).

8. Jean-Claude Janvier, et al., Irradiation of Uranium Dioxide in a Resistant Cladding Effects of Initial Diametral Gap on Overal Behavior CEA-R-3358, (October 1967).

9. J. A. Ainscough, An Assessment of the IFA-116 and 117 Irradiations from Data Obtained form the In-Reactor Instrumentation, HPR-129, (April 1971).

10. M. J. F. Notley, R. Deshaies, et al., Measurements of the Fission Product Gas Pre-sures Developed in UO_2 Fuel Elements During Operation, AECL-2662 (1966).

In addition, the following non-Halden report also presents gap conductance data for fuel operating up to 12.8 kw/ft.

11. P. E. MacDonald et al., "Gap Conductance Test Series-2, Tests GC 2-1, 2-2, and 2-3, Draft Test Results Report", EG&G Topical Report, TFBP-TR-230, November 1977.

8-13. Have there been any new calculations of average uncertainty of gap conductance in the Halden BWR experiments? Report PNL-2494 (April 1978, indicated the average uncertainty was ± 19% for those experiments) Note: I am seeking any indication the uncertainty is not as great as this today.

Response

As noted in Figure 8 of NUREG-0581, the plot of gap conductance uncertainty is taken from PNL-2581, "Stored Energy Calculation: The State of the Art," May 1978. That report states on page 28--"It should also be kept in mind that the lower bound is of primary interest as lowering gap conductance increases fuel temperatures." Higher fuel temperatures lead to higher stored energy, thus a conservative position. Figure 8 shows the uncertainty of the lower bound to vary from 12 to 15% for the large gap case where the stored energy would be highest. The uncertainty for the lower bound of the BWR gap size (9 mils) is not significantly different. To the best of our knowledge, these uncertainty bounds reflect the current state-of-the-art.

8-14. Consulting pg. 14 of NUREG-0581, Fig. 8, gap conductance uncertainty is shown to increase as linear power increases. Is staff aware of any data on gap conductance where uncertainty is a function of fuel burn-up and linear power is constant?

a/ If "yes" please indicate where this data may be located.

b/ See Request for Document 10-8, where I have made a request for Document which would be appropriate if part a/ were "Yes".

Response

We are not aware of any data where the uncertainty for gap conductance as a function of power and burn-up is constant. The noted report above (PNL-2581) discusses various uncertainties, including gap conductance and stored energy. These uncertainties are not specifically constant in either case although the uncertainty for gap conductance comes fairly close in the large gap case as a function of power. However, the uncertainties are not expected to remain constant over the parameters that you have chosen, power and burn-up. Both of these parameters have a big effect upon the physical configuration between the fuel and cladding. For example, the gap size diminishes with increasing power or burn-up and the gas thermal conductivity decreases with increasing burn-up. Since the uncertainties for the component parts of the gap conductance model will be incorporated on a "weighted" basis, these changes with burn-up and power will lead to a different summation and hence, no constant value for the gap conductance uncertainty.

8-15. What was the result of the investigation into the cause of the ATWS during pre-operational testing of the Monticello BWR and its date?

Response

During pre-operational testing at Monticello in 1970, four G. E. type HFA-51 relays failed to drop out when the associated switches were actuated. These failures occurred over a period of two months and did not result in an ATWS event since the failures were partial and were detected during pre-operational testing. The failure of the relays was due to adherence of the paint on the upper pole piece to the arnature plate while the relay was energized.

As a result, all reactor sites with this type of relays were instructed to energize the relays for 48 hours unless it could be positively established that a particular relay had been energized for that period or longer. In addition, all such normalized relays had to be deenergized once every 24 hours for a week following the 48-hour run. Subsequently, a more permanent remedy developed was to remove the paint from the face of the pole pieces for relays in the field and in the shop, and the pole face was to remain unpainted for future production.

8-16. Referring to 8-15, what steps have been taken if any to prevent this particular failure in BWR VI plants which will use the "Fast Scram" system such as ACNGS.

Response

Since the corrective actions described in response to Q 8-15 were instituted this failure mode has not recurred. Although these actions do not guarantee

that similar failure modes would not occur in the future, our stringent requirements on the design and operation of the scram system assure that the failure probability of the scram system is kept at a very low value and further the ACNGS design would be required to implement features to assure that the consequences from an ATWS event are within our acceptance limits.

8-17. Is the risk of ATWS considered greater with BWRs than PWRs?

Response

An automatic recirculation pump trip (RPT), a feature that has been incorporated in the ACNGS design, followed by prompt operator action could mitigate consequences of some ATWS events. Nevertheless, the consequences of an ATWS in a BWR are judged to be more severe than those resulting from an ATWS in a PWR. Therefore, the risk from ATWS events in a BWR is greater than that in a PWR, assuming that the frequency of an ATWS is approximately the same in the two designs.

8-18. In the Staff's knowledge, are there any other data on fuel clad failure for BWR rods to be added to this summary graph (Exhibit B enclosed) taken from Pg. 19, of NUREG-0581?

a/ If so, please provide a reference and if in accordance with the rules, a report or document showing such results.

Response

To the best of our knowledge, all available BWR fuel rod data in regard to post-DNB behavior are represented in Figure 11 of NUREG-0581.

8-19. As of October, 1979, have any of the flow blockage tests mentioned on Pg. 19 of NUREG-0581 been run?

Response

The flow blockage tests referred to in NUREG-0581 have not been performed.

These tests are scheduled for some time after 1983.

8-20. Has any contract researcher or Staff researcher concluded what factors are most likely to produce stress-corrosion cracking in stainless steel?

a/ If so, give the factors including the temperature, the flow of coolant, the pressure, and dissolved substances and their amounts in solution considered most likely to create the condition.

b/ Please refer this Intervenor to the document which presents the most recent staff position on Sensitization and stress-corrosion cracking.

Response

The factors that are likely to produce stress-corrosion in stainless steel have been identified by research and individual experience to be a combination of high total stress (including residual stresses from fabrication and welding processes), sensitization of the austenitic stainless steel by heat treating process, and a relatively high ozygen content of the coolant. The conditions are summarized in NUREG-0531, "Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants" as follows:

Stress-corrosion cracking is a condition of brittle cracking in metals caused by a combination of high stresses and corrosive environment. Three main conditions must be present for stress-corrosion cracking to occur:

1. Stress--only tensile-type stresses will cause stress-corrosion cracking; such stress may include residual stresses in the metal from fabrication as well as tensile stresses imposed by operating load conditions.

2. Corrodent--stress-corrosion cracking will proceed only if the environment is such that an electrochemical reaction can occur.

3. Susceptible material--A given alloy subjected to stress and an environment must also be in the appropriate susceptible condition for stress-corrosion cracking to occur.

The degree to which one of these ingredients must be present for stress-corrosion cracking to occur is variable and depends on the degree of the other two incredients present. Stress-corrosion cracking is enhanced by crevices or crevice-like conditions where stresses and corrodents can concentrate.

The recent Staff position on stress-corrosion cracking is presented in NUREG-0313, Rev. 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," October 1979.

8-21. Is ultrasonic testing still the best way to inspect for cracks in piping according to the NRC?

Response

Ultrasonic examination is the basic volumetric inspection method used to comply with the inservice inspection requirements of Section xi of the ASME Code. NRC has recommended volumetric inspection coupled with leak detection procedures.

8-22. Does acoustic emission data always need validation by ultrasonic testing the case of reactor coolant piping?

a/ What is the NRC position if the acoustic emission data indicates cracking requiring repair and ultrasonic testing does not indicate the presence of cracking?

Response

The feasibility of flow detection by on-line acoustic emission surveillance has as yet to be demonstrated. The application of acoustic emission procedures and data analysis requires development and proof testing before specifying them in operating nuclear systems. A primary area of development is validation and correlation of crack growth by the two methods. The NRC position will be stated after acoustic emission is proven to be an acceptable method for indicating the presence and growth of structural defects.

8-23. What methods are being developed and expected to be available at the time of ACNGS operation which will give greater detection sensitivity of pipe cracks? (Note: According to NUREG-0581, pg. 41 this is being done)

Response

NUREG-0581, "Summary of NRC LWR Safety Research Programs on Fuel Behavior, Metallurgy/Materials and Operational Safety," describes research programs in process. There is no assurance that the programs will result in development of methods applicable to operating nuclear systems. There are a number of research programs in process to improve the sensitivity of inservice inspection procedures. The Staff's position has been stated in the documents cited above.

8-24. If (according to NUREG-0581, P. 41) extrapolation to real structures of acoustic emission tests data anlaysis have not been completely proof tested, what remains to be done?

Response

The feasibility of acoustic emission to the detection of flaws in operating nuclear systems should be proven as the first step.

8-25. May Interrogatories and requests for documents be filed, based on the Contentions of John R. Shreffler?

Response

Please refer to 10 CFR §2.740, which sets forth the scope of permissible discovery requests.

Dated at Bethesda, Maryland,
this 10th day of January, 1980.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

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
Licensing Project Manager

Subscribed and sworn to before me
this 9TH day of JANUARY 1980.

Nadine R. Sidle
Notary Public

My Commission expires: July 1, 1982.

1758 350

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
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HOUSTON LIGHTING & POWER COMPANY	)	Docket No. 50-466
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CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF RESPONSE TO JOHN F. DOHERTY'S EIGHTH 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 10th day of January, 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

R. Gordon Gooch, Esq.
Baker & Botts
1701 Pennsylvania Avenue, N.W.
Washington, DC 20006

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 Alcanbury Street
Houston, Texas 77021

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

Mr. F. H. Potthoff, III
1814 Pine Village
Houston, Texas 77080

D. Marrack
420 Mulberry Lane
Bellaire, Texas 77401

Texas Public Interest
Research Group, Inc.
c/o James Scott, Jr., Esq.
8302 Albacore
Houston, Texas 77074

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

Charles Andrew Perez
1014 Montrose Blvd.
Houston, Texas 77019

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

Margaret Bishop
11418 Oak Spring
Houston, Texas 77043

Glen Van Slyke
1739 Marshall
Houston, Texas 77098

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
1118 Montrose
Houston, Texas 77019

Robin Griffith
1034 Sally Ann
Rosenberg, Texas 77471

Elinore P. Cummings
926 Horace Mann
Rosenberg, Texas 77471

Mrs. Connie Wilson
11427 Oak Spring
Houston, Texas 77043

Patricia L. Streilen
Route 2, Box 398-C
Richmon, Texas 77469

Carolina Conn
1414 Scenic Ridge
Houston, Texas 77043

Mr. Robert Alexander
10925 Briar Forest #1056
Houston, TX 77042

Colleen P. Woodhead
Counsel for NRC Staff