JUL 12 1971

Samuel W. Jensch, Esq., Chairman
Atomic Safety and Licensing Board
U. S. Atomic Energy Commission
Washington, D. C. 20545

Mr. R. B. Briggs
Molten Salt Reactor Program
Oak Ridge National Laboratory
P. O. Box Y
Oak Ridge, Tennessee 37830

Dr. John C. Geyer, Chairman
Department of Geography and
Environmental Engineering
The Johns Hopkins University
Baltimore, Maryland 21218

In the Matter of Consolidated Edison Company of New York, Inc.
Indian Point Nuclear Generating Unit No. 2
Docket No. 50-247

Gentlemen:

The responses of the AEC regulatory staff to the questions asked by the presiding Atomic Safety and Licensing Board at the hearing session on May 13, 1971 are attached herewith.

Sincerely,

Myron Karman
Counsel for AEC Regulatory Staff

Enclosures
As stated

cc w/encl: J. Bruce MacDonald, Esq.
Angus Macbeth, Esq.
Anthony Z. Roisman
Honorable William J. Burke
Paul S. Shemin, Esq.
Leonard M. Trosten, Esq.
Algie A. Wells, Esq.
Mr. Stanley T. Robinson, Jr.

bcc: OGC Files - Beth/GT/Docket
REG Files
DRL - Dr Morris
Mr D. Muller



14

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE ▶ | OGC | OGC | DRL | | | hearing |
| SURNAME ▶ | MKarman/rmh | JBKnotts | Dr PAMorris | | | |
| DATE ▶ | 7-12-71 | 7-12-71 | 7- -71 | | | |

RESPONSES OF THE DIVISION OF REACTOR LICENSING TO THE QUESTIONS OF THE ATOMIC SAFETY AND LICENSING BOARD AT THE HEARING SESSION DATED MAY 13, 1971

(Tr. 749-750)

Question 1

"I think that at the time of cross-examination I would appreciate some further information from this witness as to whether he has assumed a uniform mixing of the containment spray in the containment atmosphere and whether there are any so-called dead spots in the containment area which may affect the assumption of uniform mixing.

I think as I understand from the several times that this subject has been under consideration in some proceedings that it is with some difficulty that the assumption can be entertained that there will be uniform and perfect mixing, and furthermore there may be a problem of extrapolating from a small experiment to a larger experiment, although as I understand it in a recent instance of a small experiment it has been asserted with some certainty that you cannot extrapolate from a small experiment to a larger area. If it doesn't have to do with containment spray, but it may have to do with another safety mechanism. So I have the problem that if it's difficult in one instance to extrapolate for a certain sort of mixing to a larger site, I wonder how we can assume that we can extrapolate from these experiments to the containment atmosphere, which I think is a separate consideration."

Answer

Please refer to our comments in the response to Question 3 relative to the containment spray tests performed at Zion Units 1 and 2.

(Tr. 750-751)

Question 2

"As I recall some of the previous discussions about containment spray the staff made a statement I think Indian Point 3 as to this effect, that the research and development program relating to the drop size spectrum, the drop coalescence and possible effect of the liquid phase mass transfer resistance is not in itself sufficient to resolve the present uncertainties, and I wondered just what has been done to remove those uncertainties.

This may be directed to the staff in inquiry for presentation in that regard when their presentation of evidence is made, but the Applicant, likewise may desire to address himself to those factors."

Answer

The position of the staff with respect to resolution of uncertainties in its current evaluation model, stated at the Indian Point 3 hearing, was directed primarily to the point that the theoretical equations are not in themselves sufficient to adequately define performance. Thus comparison with experiments designed to evaluate the overall iodine removal performance of chemical additive spray systems is required. The definition of individual uncertainties in the theoretical equation is therefore only one link in the chain of reasoning which ultimately results in assigning an overall factor of conservatism, or "design margin," to the expected performance of these iodine removal spray systems under design basis accident conditions.

Since the Indian Point 3 construction permit hearing, Westinghouse has performed analyses designed to more accurately define the effect of drop coalescence, steam condensation, liquid phase resistance, and consideration of a drop size spectrum in the calculational model. We have concluded that this work has increased the confidence level of the model, and we are currently evaluating the possible modification of the uncertainty correction used in the calculation of spray effectiveness for the Indian Point 2 plant.

(Tr. 752)

Question 3

"I wonder, and this will involve the staff as well, as to how the compliance inspector will determine if the spray system meets the performance specifications, and of course that raises the assumption or raises the question what are the performance specifications for the containment spray and how can it be determined to be performable. I just happened to be going over some phases of the construction permit decision on Indian Point 3 that I think has some relevance to Indian Point 2, since the same types of containment spray systems are used. Is that correct?

In connection with that I think we got into some considerations of plateout factors and all those matters and if they could be discussed, composition on the containment wall, and what is the temperature of the containment wall, and I think there was some question about whether there will be TID 14844 assumption plateout factor or whether the containment wall is going to be higher so that that portion is no longer realistic..."

Answer

Those tests performed on the components of the containment spray system include air flow testing of the individual spray nozzles and water flow testing of the recirculation system. The compliance inspector verifies that these tests have been performed. The results of those tests and inspections indicated in the applicant's response are also available to the inspector.

The Regulatory Staff has not required that a full flow test of containment spray systems be performed on Indian Point 2. The ACRS has, in one instance, in its July 24, 1968 letter on Zion Station Units 1 and 2, recommended that "the applicant give further consideration to testing the containment spray systems with full flow to the spray nozzles at least once at an appropriate time during construction." The applicant, Commonwealth Edison Company, voluntarily performed containment spray tests, with water only, to verify spray coverage within the containment. Commonwealth Edison concluded that the test objectives were met to their satisfaction. The letter from Commonwealth Edison to the Director of the Division of Reactor Licensing, dated March 10, 1971, and summarizing the results of the test, is enclosed for your information.

In the evaluation of potential offsite doses for the Indian Point 3 plant, the staff has relied on the source term and plateout assumptions stated in TID-14844. The Indian Point 3 decision questioned the validity of using these plateout assumptions in conjunction with a containment spray which both removed a significant fraction of the airborne iodine and decreased the driving force to the walls and interior surfaces.

(Tr.752)
(Continuation)

In the calculational model suggested in TID-14844, instantaneous plateout of 50 per cent of the halogens released from the primary system is assumed. We have assessed the degree of conservatism associated with this assumption by use of a realistic time-dependent plateout model. The results indicate that, in the absence of sprays, the airborne iodine concentration calculated to be available for leakage from the containment is less for the more realistic model than would be obtained by applying the TID-14844 assumptions.

For the combination of containment sprays and the plateout process acting simultaneously as iodine removal mechanisms in the containment, the overall iodine removal calculated with a realistic time-dependent plateout model, dependent only on iodine transport to surfaces by natural convection processes and neglecting steam transport, exceeds that derived from the instantaneous plateout model unless the spray removal rate is relatively high (spray removal constant of 10-15 $hr^{-1}$ or greater depending on containment volume). We therefore have concluded that use of the TID-14844 plateout assumption for the Indian Point 3 plant is both reasonable and conservative.

(Tr. 756)

Question 4

"I might mention the subject to the Staff, if I may, I think in the course of one of these conference hearings we had some reference to the statements by the Advisory Committee on Reactor Safeguards enumerating items of concern for certain water reactors and I think the Staff enumerated what those concerns were as reflected in the communications from the ACRS in the course of the last three or four years. The inquiry was what updating we could have respecting those concerns and it should be perhaps noted on the record that the Staff did send to us a document which is of some size and entitled WASH 1146, entitled Water Reactor Safety Program Plan. I have tried to give at least a cursory review to that document, which is over a hundred pages long, and it outlines, as I review it, what is planned for certain R & D work, and it's in some detail for each of the several matters set forth in there, and there are several references to information, status, and needs, current and planned programs for many items. That seems to be the general division for each of the programs and plans.

It occurred to me that perhaps my question wasn't clear. I'm not too much interested in the planning as I am in the results, and if you could take this document, 1146, which I think would be a good guide, and then fill in just what the results are, we will assume that these plans are still in effect; if they are not fully performed they are still being undertaken, but if you could give us documents that would show the results, or any other presentation of the factual data of what has been done, I am sure it would be more responsive to the question.

And if the Staff does not have these dates or the Staff does not have a witness who is intimately familiar with these programs and then a reasonable request might be to bring somebody from the departments that do have to do with the execution of this water reactor safety program as reflected in WASH-1146. That might involve the Director of Reactor Development Technology and if he would be available to present the matter directly under his supervision I am sure it would be a responsive presentation. He probably can give us a better overall picture than several witnesses from each of the several experimental programs.

By the way, isn't there a document at all in the Atomic Energy Commission that summarizes, say on an annual basis, what is being done on the research and development work other than what is reflected in the actual report to the Congress, which is, as I read it, quite general in nature?

For instance, I see this monthly publication, The Nuclear Safety Review, I believe it comes out of Oak Ridge, and I don't want to incorrectly or unfairly

(Tr. 756)
(Continuation)

describe it, but maybe for purposes of illustration let me use something that occurs to me that maybe it can indicate why I thought if there were a compendium of the research and development it would be helpful.

We get the separate component testing results. For instance, just to use the vernacular, there will be a report that the doorknob works, and then there will be a report that the hinge works, and then there may be a report that the paneling on the door is satisfactory. Now, what I have in mind, is there a report that says the door will be handled and the hinges will work as hung together? And I wonder don't we have anything that brings those things together. And, of course, that really isn't applicable here, but it's the type of thing I had in mind. When you put everything together in the containment will it work or when you put everything in the core vessel, will it work? And I think that rather than saying that the plastic cover for something has proven satisfactory, the Division of Reactor Safety, in fact I think it's set forth in the Indian Point 3 construction permit decision, reference was made to a Division of Reactor Safety announcement by the Atomic Energy Commission that the best test is in the assembled form, and that's the kind of data response I think would be helpful. And if the Commission Staff doesn't have a summary report as elaborate as this plan before us, maybe something like that could be developed for this proceeding and could be utilized in many, many cases.

But in any event, if we could have a data response."

Answer

The progress associated with the various water reactor safety contracts outlined in WASH-1146 is discussed in the enclosed tabulation. The listing includes all contracts which are sponsored by the Division of Reactor Development and Technology. The discussions in the tabulation are similar to those provided in the Bimonthly Technical Progress Review Nuclear Safety published by the Oak Ridge Nuclear Safety Information Center.

(Tr. 758)

Question 5

"The concern that I have also is reflected in the appropriations hearings. I believe these were last year but in many places the indication was given that certain experimental work could not go forward for lack of funds. And I wondered how that has affected or will affect the research and development work that may be pertinent for this proceeding. If some analysis of that could be made it would be appreciated."

Answer

The tabulation of water reactor safety program projects enclosed with the previous response states that the following projects have been terminated due to lack of funds.

Project No. 101 11 45 Incipient Failure Detection System Development
This program involved development and testing of an acoustic system for detecting and locating cracks in large complex vessels. This project was terminated in December, 1970, after the capability of the system was demonstrated.

Project No. 105 09 34 LOFT Assistance in Out-of-Pile Studies
The purpose of this program was assistance to the LOFT integral test program and involved investigation of the containment behavior of iodine and its reaction with stainless steel and various coatings. The project was canceled in November, 1970, following the completion of tests on the stainless and a series of tests on an Amercoat-66 liner.

Project No. 107 09 81 Fabrication Techniques for Advanced Uranium ShippingCasks
This project was terminated June 30, 1969, after the feasibility of using natural uranium as the radiation shielding material for large shipping casks was demonstrated.

(Tr. 758-759)

Question 6

"I have not had time to go through the Applicant's responses to the last questions by the Board except to look briefly at some statements that are made. As you know, I've asked several questions about the inspection program. Not yet have I seen the statement concerning the program that the Applicant is undertaking to assure that the inspection can be made. I have not had any indication of how much money, for instance, is involved or what the program is that the Applicant has undertaken. However, it says here: We are confident that the needed inspection equipment will be developed within the next ten years.

It is indicated that there are four firms actively developing this type of equipment. I wonder whether some of the uncertainty might be removed if the technical specifications were altered to say that these inspections will take place; not that they will take place if the equipment is developed in time. I think that's part of our problem, that the technical specifications say that these inspections will take place if the equipment is developed. In the testimony that we get from the Applicant it says: We are confident that the inspections will take place and that we have committed ourselves to making the inspections. Maybe a large part of the problem could just be solved by modifying the Tech-Spec to take out any statements that this will be done if the equipment is developed. Possibly the staff and the Applicant could consider this and might have some change to suggest or some additional information to provide at the next session of the hearings that we have.

Answer

Section XI of the ASME Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Reactor Coolant Systems" was issued on January 1, 1970. The purpose of the code is to provide additional assurance for the long term integrity of primary coolant systems over their anticipated forty year lifetime of operation. Initial assurance of primary system integrity is provided by proper design and analysis; quality assurance including non-destructive testing during manufacture, and preoperational testing in place. In addition, verification of initial quality of the primary system is aided by base line inspections made as a part of the Inservice Inspection Code prior to initial operation.

The Inservice Inspection Code incorporates requirements for remote examinations for which efficient automated methods are currently underlying development. By including such requirements the need for such methods and

(Tr. 758-759)
(Continuation)

equipment is identified to the industry so that such equipment can be procured. The fact that further development is required is recognized in the Foreword to the Code rather than in the Code itself. The Foreword to the Code also indicates that the philosophy used to justify this approach was a belief that sufficient time was available subsequent to adoption of the code to permit the development of suitable equipment. Although the requirement for the use of this code as stated in AEC regulations is effective only for plants with construction permits issued on or after January 1, 1971, the inservice inspection program developed for Indian Point 2 as presented in Section 4.2 of the Technical Specifications has been updated to incorporate the inservice inspection requirements of Section XI. Since implementation of these requirements for Indian Point Unit 2 will be on a much more rapid time schedule than anticipated by the Code authors, the Technical Specifications include under remarks in Table 4.2-1 where appropriate to the performance of certain required examinations, the statement "These inspections are predicated on the development of remote mechanical ultrasonic examination devices."

We interpret this statement as correctly reflecting the current state of the art for remote inservice examinations rather than as a proviso for not performing the required examinations. We are encouraging the rapid development of this equipment in connection with our review of other plants and we currently expect that the long term quality of the Indian Point Unit 2 primary system can be assured by implementation of the requirements as stated in the proposed Technical Specifications.

Section 4.2.1(b) provides that "the results obtained from compliance with this specification shall be evaluated after five years and the conclusions of this evaluation shall be reviewed with the AEC." The purpose of this specification is to provide for review and amendment of Section 4.2 of the Technical Specifications so as to reflect additional new technology concerning primary systems and inspection methods.

(Tr. 816-817)

Question 7

"We might take this other one up, just one comment here, and it deals with the critical testing also. Once you start loading the fuel it becomes inconvenient to unload the fuel again and to take out the innards from the reactor. In connection with your motion, I would like to see a reply to a question, if you wish, by someone who is doing development work on ultrasonic testing of reactor vessels. I would like to see information concerning the effect of the surface roughness of the reactor vessel on the results that one can get from the inspection.

At one hearing I remember the manufacturer had chosen to change from inspecting the vessel from the inside to inspecting the vessel from the outside. The impression I have, or the understanding I have, was that the inspection could be done more satisfactorily because this way it was done when the vessel was fabricated. The vessel outside was smoother than the vessel inside, and that this would have some effect on the results of the inspection.

I'd like to be assured, before the reactor vessel becomes radioactive, that meaningful inspections by ultrasonic methods can be conducted from the inside without having to polish the surface, smooth the surface where the inspection is going to take place.

In other words, I wouldn't like someone to come back and say, 'Well now, we have made the plant radioactive; it's not convenient to get in and smooth the surface. The inspection isn't going to be as good as it would have been had we done this initially.'"

Answer

We have reviewed the applicant's response and have nothing to add.