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The Honorable Edward J. Markey
Chairman, Subcommittee on Oversight
and Investigations
Committee on Interior and Insular Affairs
United States House of Representatives
Washington, D.C. 20515

Dear Mr. Chairman:

Thank you for your letter of September 28, 1982 regarding the steam generators for the Clinch River Breeder Reactor Plant (CRBRP). The NRC staff is currently reviewing the CRBRP construction permit application. Although the review is not complete, the information and preliminary conclusions contained in the enclosed answers to your questions should remain valid. If relevant new information becomes available during the remaining course of the review, I will provide you an update of our response. The staff Safety Evaluation Report, which will document the review of the steam generators as well as all matters important to safety, is currently scheduled to be issued in March, 1983.

I hope you will find the enclosed information useful in your consideration of the CRBRP steam generators. Please let me know if I can be of further assistance.

Sincerely,

(Signed) Jack W. Roe for

William J. Dircks
Executive Director
for Operations

Enclosure:
As stated.

cc: Rep. Ron Marlenee

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Retyped in OEDO to correct address, see attached ORC for prior concurrences.

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| OFFICE | CRBRPO:NRR | CRBRPO:NRR | CRBRPO:NRR | CRBRPO:NRR | NRR | EDO for | OCA |
| SURNAME | RBecker/er* | BMorris* | WFoster* | PCheck* | HDenton* | W Dircks | |
| DATE | 10/29/82 | 10/28/82 | 10/29/82 | 10/29/82 | 11/1/82 | 11/3/82 | 11/16/82 |

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William J. Dircks
Executive Director for Operations

Enclosure:
As stated

10/28

| OFFICE | CRBRPO:NRR | CRBRPO:NRR | CRBRPO:NRR | CRBRPO:NRR | NRR | | |
|---|---|---|---|---|---|---|---|
| SURNAME | RBecker | BMorris | WFoster | PCheck | HRDenton | | |
| DATE | 10/29/82 | 10/28/82 | 10/29/82 | 10/29/82 | 11/1/82 | | |

ANSWERS TO QUESTIONS
FROM REP. E. J. MARKEY TO CHAIRMAN N. PALLADINO

QUESTION 1. What is the potential for leaks or ruptures in the steam generators planned for use at Clinch River?

ANSWER

A design goal of the steam generators for the Clinch River Breeder Reactor Plant (CRBRP) is that they be leak free. From our review of the steam generator design and of the testing program we believe that the principal historical causes of steam generator tube leaks and failures on liquid metal reactors have been appropriately taken into account. (The experience of LWRs is, of course, generally inapplicable because the steam generators are of quite different design.) Therefore, our preliminary conclusion is that the potential for leaks and ruptures will be acceptably low.

QUESTION 2. What would be the range of consequences for a steam generator tube leak or rupture at Clinch River and how would it differ from those in pressurized water reactors?

ANSWER

The CRBRP is arranged so that intermediate sodium coolant loops separate the radioactive sodium in the primary coolant system from the steam generators. The intermediate system pressure is higher than the primary pressure to prevent radioactive contamination of the intermediate system sodium. Therefore, in contrast to PWRs, steam generator leaks at CRBRP will have no significant radiological consequences.

In the event of a steam generator tube failure at CRBRP, the plant will be manually or automatically shut down. If the leak is small, high pressure steam and water will be forced into the intermediate sodium system. The reaction between sodium and water liberates hydrogen, but not in sufficient quantities to create a significant pressure disturbance. The concern with small leaks, however, is that they may cause a reaction front which could impinge on the surrounding tubes and cause additional and possibly large failures. Hydrogen detectors are provided to sense the presence of hydrogen in the sodium and alert the operator. The operator then has sufficient time to establish the location of the leak and to shut the plant down in the normal manual fashion.

If the failure is large, the steam and water being forced into the intermediate sodium will cause a sufficiently energetic reaction to result in a pressure pulse. The pressure increase will be sensed, the plant will be automatically shut down and the Sodium-Water Reaction Protection System will be activated by breaching a rupture disc which provides a path from the steam generator to quickly remove the sodium-water reaction products including hydrogen. A mixture of reaction products, sodium and water is forced into a system which separates the hydrogen and allows the hydrogen to be harmlessly flamed in the atmosphere. The water side of the steam generator is quickly isolated and depressurized to minimize the available steam and water and limit the reaction.

QUESTION 3 What studies, memoranda, or correspondence has NRC prepared or received on the consequences of steam generator tube leaks or ruptures in the Clinch River Breeder Reactor or other breeders?

ANSWER

The principal source of information on CRBRP steam generators is the Preliminary Safety Analysis Report. Chapter 5 describes the steam generator design and Chapter 15 presents the analysis of potential steam generator failures. Both chapters discuss international experience. The NRC staff's Safety Evaluation Report for CRBRP, which will document the review of all safety aspects of the plant, is being prepared and will be issued next March.

The following letters, memoranda and reports on this subject have been sent, received or reviewed by NRC's Clinch River Breeder Reactor Program Office:

a) Letter dated February 26, 1982 from P. S. Check, NRC to J. R. Longenecker, DOE. (Question CS 220.6)

b) Letter dated April 29, 1982 from J. R. Longenecker, DOE to P. S. Check, NRC. (Response to Question CS 220.6)

c) Letter dated June 8, 1982 from J. R. Longenecker, DOE to P. S. Check, NRC. (Description of CRBRP steam generator test program and an analysis of GAO Report)

d) Synopsis of Remarks, "BN350 Steam Generator Experience," B. I. Lukasevich, November 21, 1975.

e) Memo dated June 28, 1982 to Distribution from NRC's Office of International Programs. (Discussion of April 82 steam generator leaks at Phenix)

f) EPRI Report NP1972, "The Achievements, Findings and Lessons of Phenix LMFBR Power Plant Experience," August 1981.

g) GAO Report EMD-8275, "Revising the Clinch River Breeder Reactor Steam Generator Testing Program Can Reduce Risk", May 25, 1982.

With the exception of item (e) these documents are enclosed. Item (e) has been provided to us under an international proprietary information agreement. We are taking steps to get permission to release it to you.

QUESTION 4 What is the history of steam generator problems in domestic and foreign breeder reactors? Have there been corrosion, vibration, or other types of degradation detected in breeder reactor steam generators?

ANSWER

The history of steam generator experience in both foreign and domestic breeder reactors has been mixed. Problems with steam generators were experienced by the British and the Soviets almost immediately upon plant startup. The British and Soviet failures appear to have had a similar cause: stress-accelerated corrosion cracking resulting from lack of heat-treating welds and stressed components. This is difficult to confirm, however, because there is meager information from the Soviets. The French operated their demonstration plant, Phenix, for ten years before experiencing their first leak. The study of the Phenix leak has not been completed so no cause has as yet been identified for this failure. Although the Japanese have not built their demonstration plant, MONJU, they have experienced no failures in their small-scale model testing.

In the U.S., the actual experience with steam generators on fast breeder reactors has been limited. There has been, however, a considerable amount of steam generator experience with sodium-cooled nuclear reactors which represent similar service conditions. The following table summarizes this experience.

| Plant | Size-MWt | Number of Steam Generators | Years |
|---|---|---|---|
| EBR-I | 4 | 1 | 14 |
| EBR-II | 62 | 1 | 19 |
| SRE | 20 | 1 | 2 |
| FERMI | 143 | 3 | 10 |
| HALLAM | 85 | 3 | 2 |

The EBR-II steam generator has operated for 19 years without a water to sodium leak. EBR-I, SRE and Hallam were leak-free, but operated for shorter periods. The FERMI reactor experienced several tube leaks over the course of its operation. FERMI preoperational failures were attributable to stress corrosion cracking. FERMI operational failures were attributed in part to flow-induced vibration.

QUESTION 5 What assurances can the NRC provide that there will be no steam generator tube degradation at Clinch River once it is in operation?

ANSWER

It is impossible for NRC to warrant that there will be no steam generator tube degradation during operation. We do believe, however, based upon what we have learned thus far in our review of the steam generator design and test program, that the steam generators will operate safely. Should unexpected failures occur, we anticipate that the systems provided in the plant design for that contingency will serve acceptably to mitigate the consequences.