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FEB 19 1982

50-244

MEMORANDUM FOR: Gus C. Lainas, Assistant Director for Safety Assessment, DL

FROM: Themis P. Speis, Assistant Director for Reactor Safety, DSI

SUBJECT: DSI INPUT TO CONGRESSMAN UDALL'S LETTER OF 2/5/82

Per your request, we have addressed the concerns identified in Congressman Udall's letter of February 5, 1982, and which were assigned to DSI in your letter of February 11, 1982.

Specifically, the Reactor Systems Branch addressed questions 6, 11, 15, and 17 part II; and the Accident Evaluation Branch addressed question 10. Because we have an interest in a number of other questions raised by Mr. Udall (e.g., questions 2, 7, 8, 9, and parts of 13, 14 and 18), we would be happy to review the drafts to these questions at your pleasure.

Original Signed By
Themis P. Speis

Themis P. Speis, Assistant Director
for Reactor Safety,
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Enclosure:
As stated

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QUESTION 6 — What would the course of the incident have been had the PORV block valve failed to close partially or fully following failure of the PORV to close fully?

ANSWER

Had the block valve failed to close after the PORV stuck open, the additional coolant loss from the primary system would have caused the primary system pressure to continue to decrease below 900 psig. As the pressure in the reactor system decreased, the combined leak flow through the valve and the rupture would decrease and safety injection flow would increase until the flows were approximately equal. Analyses by Westinghouse in WCAP-9600 (Ref. 1) indicate that the reactor system pressure would stabilize at approximately 700 psia. The pressure would then remain relatively constant until the operator took action to depressurize the plant with the intact steam generator. If the block valve were only partially closed, the combined leak flow and safety injection flow would equalize at a pressure between 700 psia and the 1300 psig pressure which was reached at Ginna after the block valve was fully closed. Additional leakage out of the reactor system through the broken steam generator tube would not occur for the case of the block valve stuck fully open since the primary system pressure would be less than the affected steam generator pressure. If the block valve were only partially closed, the reactor system might repressurize so that some leakage out the broken tube could occur; however, it is expected that the leakage would be less than that which occurred with the block valve fully closed.

The effect on core coolant inventory of a combined PORV leak and steam generator tube leak would be similar to a postulated break in the reactor coolant hot leg with an equivalent break size of about 2½ square inches. The consequences of this event on core cooling would be bounded by the spectrum of small break analyses performed for Ginna (Ref. 2). These analyses demonstrated that the core is adequately protected by the Emergency Core Cooling System in the event of a small break LOCA.

The staff concludes based on the discussions above that the effect of the block valve failing to close or leaking during the event at Ginna would have been a decrease in coolant loss through the steam generator tube and an increase in coolant loss through the PORV. Since coolant loss through the PORV is confined within the containment building and coolant loss through the broken tube may be released through the secondary system safety valves, offsite doses would probably have been lessened had the block valve stuck open at Ginna. Small break LOCA analyses for Ginna indicate that the core would be adequately cooled had the block valve failed to close.

REFERENCES

1. Report on Small Break Accidents for Westinghouse NSS System, WCAP-9600, Westinghouse Electric Corporation, June 1979.

2. Letter from LeBoeuf, Lamb, Leiby & MacRae, Attorney for RG&E, to L. Muntzing US AEC, transmitting small break LOCA analyses for Ginna, September 6, 1974.

QUESTION 10. What consideration has been given the potential for radioactivity escaping PWRs via a path including breaks in steam generator tubes and a stuck open safety valve?

ANSWER

Steam generator tube rupture accidents are one of the class of design basis accidents considered by applicants and staff in each review of PWR license applications. The staff's Standard Review Plan, NUREG-0800, describes the criteria and procedures used at Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure (PWR)" (copy attached).

The analysis focuses on the potential release of radioactive noble gases and radioiodine both pre-existing in the reactor primary and secondary coolant, and generated concurrently with the accident. The former case uses the maximum activity levels permitted by the plant's proposed Technical Specifications. The latter case postulates activity released from the fuel as a result of the accident, including the potential for fuel failures.

The steam generator tube failure is assumed to be a double ended rupture of a single tube for purposes of calculating the rate of transfer of primary coolant to the secondary side of the affected steam generator. Flashing of the primary coolant is assumed to occur in this process with subsequent atomization and transfer of activity to the steam phase. Radioactivity leaving the steam generator is assumed to become airborne immediately and transported directly to the atmosphere via leakage paths not mechanistically specified. Such leakage could be through a stuck open safety valve, an open atmospheric dump valve, or through the condenser vent system. The release is assumed to be terminated when the primary and secondary coolant system pressures have equalized. For FSAR safety analyses, this is usually assumed to occur at about 30 minutes after the event initiation.

Exclusion area boundary and low population zone boundary doses are calculated and compared with the thyroid and whole body dose guideline values cited in 10CFR Part 100. Conservative values of site specific atmospheric dispersion characteristics are used in these calculations.

QUESTION 11. Is it generally agreed that if a leak had developed in both steam generators, the operators would have been able to institute the "feed and bleed" process described in Mr. Speis' January 28 memorandum?

ANSWER

Had a leak developed in the second ("A") steam generator at Ginna, the need to institute the "feed and bleed" process to assure continued core cooling would have depended upon the leak size and total leak rate of primary coolant out of the primary system. The primary concern associated with two leaking generators is that in order to use the steam generators to cool down the primary system to the residual heat removal (RHR) system entry level, the primary system pressure would have to remain slightly higher than the pressure in both faulted generator secondaries during cooldown. This would result in continued leakage of primary coolant to the secondary system. Primary coolant would have to be replaced by the high pressure injection (HPI) system which pumps water from the refueling water storage tank (RWST) into the primary system. Thus, the allowable leakage depends on the ability to cool the plant to RHR entry conditions prior to depleting the RWST. For small loss-of-coolant accidents in the primary system, the leaking water will accumulate in the containment sumps. Once the RWST level drops to a preset value, the pump suction is switched from the RWST to the sump and sump water is recirculated through the core. Decay heat is ultimately removed by the containment heat removal system.

For larger tube leaks in both steam generators, which might deplete the RWST inventory prior to RHR entry conditions being reached, the operators would be expected to open all PORVs to rapidly depressurize the primary system (as well as remove decay heat) to below the faulted steam generator secondary pressures, and isolate both steam generators. Primary coolant makeup would be accomplished with the HPI pumps.

At Ginna, a two-loop 1300 MWth Plant, there are two PORVs manufactured by Copes-Vulcan with a relief capacity of 179,000 lb/hr. steam. Although neither the staff nor the licensee has performed any detailed calculations, scoping estimates indicate that the Ginna plant can remove decay heat by the "feed and bleed" process.

It is noted that failures in both steam generators are presently not required in the design base for PWRs. Furthermore, existing emergency procedures, such as those at Ginna at the time of the tube rupture accident, do not provide the operators with explicit guidance on how to cooldown the plant with ruptures in multiple steam generators. However, as a result of the TMI accident, the staff's TMI Action Plan item I.C.1 requires the industry to upgrade emergency operating guidelines and procedures to cover multiple failure events. One of the specific events cited in NUREG-0737 is tube failures in multiple steam generators. Significant resources to the upgrading of guidelines and procedures have been allocated by both the industry and the staff. We anticipate approving the new emergency procedure guidelines by the end of FY 82. If this goal is met, upgraded procedures should be implemented at all operating plants by FY 83.

QUESTION 15 "How long did it take to reach cold shutdown? Is this a period longer than desirable? What was the reason for the period being longer than normal? What kind of malfunctions during the extended cooldown period might have led to a significant release of radioactivity to the environment?"

ANSWER

THe plant was in cold shutdown the day following the event (6.53 p.m.). The time from reactor trip to cold shutdown was 33 hours 25 minutes.

The period from reactor trip to cold shutdown was not longer than desirable. In fact, there was no urgent need to reach cold shutdown conditions after the steam generator tube leak had been terminated (equalizing primary pressure with the faulted steam generator) and the plant was in a stable shutdown condition. This stable safe shutdown was reached about two and half hours after the reactor trip.

In general, it is expected that cooldown with a ruptured tube in one steam generator would be significantly slower than a normal cooldown. This slower cooldown is because the reactor coolant system pressure is to be equalized to the pressure in the ruptured steam generator to minimize or terminate reactor coolant leak flow through the rupture. Since the direct release of steam from the ruptured steam generator is to be minimized (the steam would contain radioactive products from the primary system), depressurizing the faulted steam generator must be by other less direct means. Therefore, the rate at which the faulted steam generator can be depressurized is a limiting factor for the rate of reactor coolant system cooldown and depressurization.

In Ginna, the ruptured steam generator was drained to the reactor coolant system through the ruptured tube. Additional cooling and depressurization was provided by cold auxiliary feedwater which replaced part of the drained water.

If there has been no steam release from the ruptured steam generator in the early stage of the event, it is reasonable to expect that the cooldown period would have been longer. For a large initial steam space in the ruptured steam generator, a limiting factor for steam generator draining is the need to keep steam generator tubes covered. Should the steam come in direct contact with the tubes, rapid condensation would occur resulting in a rapid depressurization of the ruptured steam generator secondary side and re-initiation of reactor coolant leakage back through the ruptured tube.

During the extended cooldown period at Ginna, the ruptured steam generator was isolated and its pressure was significantly lower than the safety valve set pressure. All other steam valves from the steam generator were secured. The reactor coolant system was controlled similar to a normal cooldown, except for measures (increased letdown, boration) to accommodate the leak flow to the primary system coming from the secondary side.

As indicated in the response to question 10, potential releases of radioactivity to the environs during the short term or long term most directly relate to additional malfunctions in the faulted steam generator. Such leakage could be through a stuck open safety or rel valve flow path or through the condenser vent system. For FSAR radiological safety analyses, such releases are assumed during the first 30 minutes of the event, after which credit for operator correction is allowed.

QUESTION 17.
(Part II) Does the Commission believe that conditions might develop in PWRs calling for the use of remotely controlled valves for the purpose of venting steam?

ANSWER

In PWRs with inverted U-tube steam generators (i.e., Westinghouse and Combustion Engineering reactors), high point vents are required to be located on the vessel head. This requirement was added for the purpose of providing a vent path for non-condensible gases that could accumulate in the primary system under degraded core cooling conditions. Although these vents could be used to vent steam which might accumulate in the vessel upper head after saturation conditions are reached in parts of the vessel, it is not expected they would be used for this purpose, nor is it recommended that they be used to vent steam. Steam in the upper head of Westinghouse and Combustion Engineering reactors does not pose a direct threat to continued core cooling. If the steam bubble were to expand to the hot leg outlets, it would most likely condense as it came into contact with subcooled water exiting the core. If, for any reason, the water exiting the core was saturated, the steam would enter the hot leg pipes and travel to the steam generators, where it would be condensed.

For events such as the one at Ginna, the method we prefer for removing steam which accumulates in the upper head of the vessel is to restart a reactor coolant pump. The pump will force subcooled water into the upper head region and condense the steam bubble. The operators at Ginna demonstrated the capability to do this following the formation of a steam bubble in the upper head.

In PWRs with once-through steam generators (OTSGs) (i.e., B&W reactors), a steam bubble in the upper head of the vessel has the potential to temporarily interrupt natural circulation if it expands and is able to enter the hot leg outlets without condensing. These plants will have high point vents installed on the top of the hot leg inverted U-bends. In addition, some utilities with B&W reactors are installing vents on the top of the vessel head.

Analyses by B&W have indicated that interruption of natural circulation is a temporary phenomenon. The analyses show that system repressurization following the interruption of natural circulation will ultimately produce thermal-hydraulic conditions in the primary system which restore natural circulation. The staff is still reviewing the capability of the B&W analysis methods to properly predict the relevant thermal-hydraulic phenomena.

B&W has recently recommended use of the hot leg high point vents to vent steam which may accumulate during the recovery phase of a small break loss-of-coolant accident (SBLOCA).

During the accident phase of a SBLOCA, B&W has recommended the "bumping" of the reactor coolant pumps to sweep any steam trapped in the hot leg high points into the steam generator.

QUESTION 17. (Part II)

ANSWER (continued)

The use of the high point vents to vent steam in B&W reactors, as well as the acceptability of the B&W calculational models to properly predict the thermal-hydraulic behavior of the primary system under two-phase conditions, is under active staff review. At this point in the review, it is our preliminary conclusion that the use of the vents in B&W reactors to remove steam which accumulates at primary system high points may be the preferred method of steam removal if a reactor coolant pump cannot be restarted and run continuously.