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PEACH BOTTOM ATOMIC POWER STATION STEAM GENERATORS
50-171

REFERENCE: Memorandum of November 25, 1964 by Holt & Impara to Levine & Goller

We have looked at the probability and the possible consequences of a sudden, large failure in a Peach Bottom steam generator which would release steam into the primary coolant gas system. Consideration of this failure mode is important in order to determine whether it could result in over-pressurizing and rupturing the primary coolant system and then the containment building.

The only credible failure mode which we could identify which would produce a large flow path from the steam-side into the primary coolant gas side is that involving a failure of the superheater outlet tubesheet. This is a stainless steel tubesheet which is attached to the main carbon steel tubesheet by a cylindrical connection piece and a dissimilar weld. The probability of such a failure occurring will be minimized by high quality steam system water chemistry and by periodic inspections of the steam generators, with particular attention to the superheater outlet tubesheet area, as required by the technical specifications.

As an upper limit consideration, we evaluated the possible consequences of a hypothetical failure which would expose the entire steam and water inventory of a steam generator unit to the primary coolant system. The analysis assumed adiabatic flashing of the water in the steam generator to attain instantaneous pressure equilibrium. It was assumed that no significant amount of liquid water would be carried over into the primary system by slugging or entrainment. For these conditions, the maximum pressure that would result in the primary system was calculated to be about 525 psig. The combined capacity of the system's two dump valves and three safety valves were evaluated and found to be sufficient to reduce this pressure to less than the 450 psig design pressure of the primary system in less than 11 seconds. This short period of pressurization at less than 17% over design pressure should not cause the primary system to fail. Pressures up to 120% of design are permitted for short periods by the ASA Piping Code. Also, the primary system will have been pressure tested to 125% of design pressure during preoperational testing.

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A-392

Actually, for any realistic steam flow rate into the primary system from a failed steam generator the dump valves would start to relieve pressure as soon as it exceeded 390 psig, so that the maximum equilibrium pressure attained would not be as high as that indicated above. Our evaluation of the combined capacity of the relief valves indicates that it is approximately equal to the maximum steam flow that could reasonably occur. The maximum pressure attained should; therefore, only very slightly exceed the highest set pressure of any of the relief valves, which corresponds to the 450 psig design pressure of the system.

During the initial pressure transient, there would be very little time or opportunity (flow paths to relief valves bypass reactor core) for steam to react with core graphite to form non-condensible gases before the steam is relieved to the containment atmosphere. The steam remaining in the system after the initial pressure transient has been relieved may react with the graphite, but the gases and pressure generated would easily be relieved by the dump and safety valves. Within 30 seconds after the start of the accident, the isolation valves in the loop containing the failed steam generator should be closed, the primary system should still be pressurized to about 390 psig, and the containment building will probably have been isolated by the ventilation radiation monitors and therefore contain a small pressure due to the helium and steam released from the safety valves. Little, if any, of the water in the failed steam generator will be dumped because the dump is automatically terminated when the steam pressure is reduced to approximately 50 psi above the primary system pressure. The water left in the failed steam generator should, however, be isolated from the reactor core by the loop isolation valves. Even if one or more of the loop isolation valves do not close properly, a significant amount of this water could not react with core graphite to form noncondensible gases because the core would be cooled below the reaction temperature by the unfailed steam generator and circulator in the other loop before a significant amount of this water could vaporize and enter the pressurized primary system. If, as is quite probable, one of the safety valves fail to seat properly, the pressure in the primary system would slowly equilibrate with the containment atmosphere at a pressure less than the 8 psig design value of the containment building. With any of these various possibilities the total release of radioactivity to the containment would be small, consisting mainly of the 4225 curie gaseous activity inventory normally permitted in the primary coolant system. The doses at the site boundary from such an accident would be negligible.

As a result of our inquiry, Philadelphia Electric arranged to have their nuclear designer, General Atomics, perform a detailed digital computer calculation on this problem. The results reported to us by telephone (refer to telecon of December 10, 1964) were in very good agreement with those indicated above. The maximum pressure calculated to occur in the primary system was 465 psig, occurring 3-1/2 seconds after the steam generator failure, and is reduced to less than 450 psig

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within 10-1/2 seconds. The calculation assumed that the superheater outlet tubesheet was suddenly and completely removed so that the entire inventory of superheated steam in the steam generator flowed into the primary system and then saturated dry steam from the steam drum until equilibrium pressure was attained. The eight inch diameter steam pipe between the steam drum and the steam generator was assumed to be the only restriction to flow. The calculation allowed for relief valve operation as their set points were reached and the generation of non-condensible gases by the reaction of steam with core graphite at rates in accordance with experimental data.

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