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January 23, 1979

Letter No. D-3132
File Ref: N4M-2/12314
Ref Ltr: K-5020/4-27-78

MICHELSON
JAN 1978

Tennessee Valley Authority
400 Commerce Avenue
Knoxville, TN 37902

Attention: Mr. D. R. Patterson
Chief Mechanical Engineer

bcc: NSS 15/16 T1.2
R C Jones
B M Dunn

Bellefonte Nuclear Plant Units 1 & 2
Contract No. 71C62-54114-2
B&W Reference: NSS-15 & -16
Subject: Small Break LOCA Analysis

Gentlemen:

The attached report is in response to your reference letter. Please let us know if further discussion is required.

Very truly yours,

James McFarland
Senior Project Manager
Robert E. Lightle
By
Robert E. Lightle
Associate Project Manager

REL:dc
Attachment

cc: W. Brent Wade
J. L. Atchison
E. L. Logan

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Respon:e to TVA Letter K-5020, Emergency Core Cooling System — Sm 11 Break LOCA Analysis N4M-2-14(AR), April 27, 1978

Via TVA Letter K-5020, TVA transmitted to B&W a report entitled, "Decay Heat Removal During a Very Small LOCA for a B&W 205-Fuel-Assembly PWR," by C. Michelson, dated January, 1978. This report presents a simplified, hand calculation review of the small break transient and potential consequences for very small breaks not explicitly examined within the small break topical for the 205 FA plant, BAW-10074A, Rev. 1. Within this paper, the following concerns were expressed for the very small breaks:

1. How is decay heat removed?

2. Will system repressurization occur? If so, could a smaller case be a worst break?

3. If the operator isolates the break, will system repressurization occur? If so, will the pressure relief valves be subjected to slug or two-phase flow?

Responses to these concerns are developed in the subsequent paragraphs.

Before discussing these concerns, a general overview of the small break transient in a B&W 205 plant needs to be briefly discussed. Small LOCAs can be viewed as a slow transient during which the RCS can be described as a sealed manometer. Because of the internals vent valves, no extensive steam bubble will form within the reactor vessel while any significant liquid inventory remains in the loop. Many experiments have been run which show that so long as a fluid, with qualities less than 70% or so, covers the core, no adverse core temperature excursion will occur at decay heat power levels. Thus, any problems with small breaks will only occur after the RCS loops have depleted their inventory.

Decay heat removal from the core region is no problem as stated above. However, decay heat removal from the system as a whole needs to be examined further. There are two ways of removing decay heat from the system; via the break and/or via the steam generator. Both of these items are discussed in detail in the TVA letter. For the very small LOCAs of interest in this discussion, it was shown that the break alone is not capable of removing all the decay heat and heat removal via the steam generator is necessary. While the TVA-predicted break size that this occurs at was not checked quantatively, the actu l break size that it occurs at is inconsequential. Such a break size does exist where the steam generators are necessary.

The role of the steam generator as a heat removal source is basically as described in the letter. Initially, natural circulation will be maintained in the system and the necessary heat removal is easily accomplished. Once a steam bubble of sufficient size necessary to fill the U-bend at the top of the hot legs is formed, natural circulation will cease. The intermittant natural circulation discussed in the letter will not occur due to the slow nature of the small break transient. Once natural circulation ceases, the system will repressurize somewhat until the SG primary side liquid level drops below the SG secondary side level and condensation heat transfer is established. During this period between the natural circulation and condensation heat removal modes, the letter expresses concerns that the liquid inventory within the system will be depleted at a rate in excess of the rates for the breaks analyzed ~~by B&W~~ because of the partial repressurization of the system. TVA is concerned that this ultimately will result in more core uncovery than that shown in the small break topical report BAW-10074A. This is not the case.

During the natural circulation phase, it is obvious that the smaller the break, the slower the loss of system inventory and the longer the period of natural circulation. After natural circulation ceases, system pressure will be controlled by a "volume balance." That is, the system pressure will balance at a point where the volume of fluid discharged through the break equals the volume of steam being created in the core. Since the cold leg fluid enthalpy remains unchanged during a small break transient, the volume relief out the break increases with increasing system pressure and break size. The volume of steam being generated in the core decreases with increasing pressure. As the break decreases in size, the RC system will repressurize to a higher value; thus the volume relief out the break necessary to match the volume of steam being created decreases. Therefore, the system inventory will be lost at a slower rate as break size decreases. Once the SG becomes available for condensation heat removal, the primary system pressure will depressurize to approximately the SG secondary side pressure. Since the secondary side of the SG will respond in a similar manner to that of the 0.05 ft^2 break analyzed in the topical, the primary side pressure response, following the advent of condensation heat removal, will be similar to that of the 0.05 ft^2 break. Thus, for the smaller breaks, the system inventory will always be greater than that for the 0.05 ft^2 break and the core will always remain covered and will not undergo a temperature excursion.

In the paper concerns are raised relative to isolation of the break after natural circulation is lost. The scenario presented in the letter is reasonable. Should the break be isolated at that time, system repressurization to the pressurizer safety valve setpoint is probable. Two-phase or liquid flow through the safety valves will also probably occur. Once the system depletes sufficient inventory to establish condensation heat transfer across the SG, the system will depressurize and no further loss of inventory will occur. The core will remain covered for this secenario and no temperature excursion occurs. Should the pressurizer safety valves become damaged because of the two-phase flow out the valves, the response of the system would then be similar to that presented in the FSAR for the pressurizer safety valve stuck open accident and no core uncovery occurs. (Attached)

what if safety valve sticks open

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As far as the appropriateness of the operator using pressurizer level indication to trip the HPI pumps, B&W agrees that the level indication is not a reliable indication of the state of the RCS. However, use of the pressurizer level indication, along with system temperature and pressure measurements to ensure that the system is still in a substantially subcooled state, will provide sufficient guidance for operator action.

In summary, while the TVA paper raises valid concerns and gives a detailed examination of the small break transient, the small break topical report provides sufficient analyses to ensure the ability of the B&W 205 plant ECCS system to control small break in the RCS.

3. The RCS and main steam system pressure shall not exceed 110% of the design pressure.

4. Resultant doses shall not exceed 10 CFR Part 100 limits.

15.6.1.3.2 Methods of Analysis

The analysis of the effect of the inadvertent opening of the pressurizer safety valves was performed using the small break model documented in B&W topical report BAW-10074A, Revision 1, "Multinode Analysis of Small Breaks for B&W's 205-Fuel-Assembly Nuclear Plants With Internals Vent Valves." The small break model used in that topical was demonstrated to comply with the requirements of Appendix K to 10 CFR Part 50 in Appendix A of BAW-10104, "B&W's ECCS Evaluation Model." Flow through the open safety valve was calculated by the Moody critical flow correlation. The leak area is limited by the orifice area of the safety valve, which is 0.03 ft^2. In order to obtain rated flow through the valve at the valve rated pressure, it was necessary to use a discharge coefficient of 0.75. Other assumptions used for the analysis are as described in BAW-10074A, Revision 1.

15.6.1.3.3 Results of Analysis

Following the inadvertent opening of a pressurizer safety or relief valve, the design pressure of the RCS is never exceeded. The RCS depressurizes to approximately the secondary side pressure of the steam generator. The secondary side pressure is maintained at approximately 1250 psia by the safety valves on the steam generator. Since the design pressure of the main steam system is 1250 psia, the system is maintained at below 110% of its design pressure.

Figure 15.6.1-1 presents the inner vessel liquid volume (lower plenum, core, and upper plenum) for this accident. Since the core remains covered by liquid throughout the transient, pool nucleate boiling will be maintained and the cladding temperatures will remain within a few degrees of the water temperature. No metal-water oxidation is incurred, and no abnormal core geometries or fuel rod damage is caused by the low cladding temperatures.

By 3200 seconds, the injection rate from the one HPIS exceeds the combined leak rate and core steaming rate and will provide long-term cooling capability. Therefore, compliance with the acceptance criteria of 10 CFR Section 50.46 is ensured.

The discharge coefficient of 0.75 was chosen to match valve performance with the Moody correlation. If 0.75 is assumed, the rated flow through the valve at the valve's rated pressure is calculated by the Moody correlation. Although this matches the valve performance, it should be noted that the discharge coefficient assumed for this accident is not important. The system's response following the opening of a pressurizer safety valve is similar to that calculated for small breaks. B&W topical report BAW-10074A, Revision 1, discusses the consequences of small breaks and demonstrates that the core remains covered throughout the transient for breaks with leak areas less than 0.1 ft^2. Since the leak area for an open safety valve is only 0.03 ft^2, the core will remain covered thoughout the transient, and no fuel rod damage will occur regardless of the discharge coefficient assumed.

15.6.1.4 Barrier Performance

The inadvertent opening of a pressurizer safety valve does not result in fuel damage or excessive pressure in the Reactor Coolant System or main steam system since pool nucleate boiling maintains the cladding temperature within a few degrees of the water temperature and peak pressures never exceed the Code design limits.

15.6.1.5 Radiological Consequences

Since no fuel clad damage occurs, there will be no increase of radioactivity in the reactor coolant or in the steam. Because the doses are a function of the amount of steam released, the potential radiological consequences are bounded by the consequences of the loss of onsite and offsite ac power to the station transient (Section 15.2.6).

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

15.6.2.1 Identification of Causes

A break in fluid-bearing lines that penetrate the containment could result in the release of radioactivity to the environment. There are no instrument lines connected to the RCS that penetrate the containment. However, other piping lines from the RCS to the Makeup and Purification System and the Decay Heat Removal System do penetrate the containment. Leakage through fluid penetrations not serving accident consequence limiting systems is minimized by a double barrier design so that no single credible failure or malfunction of an active component will result in loss of isolation or intolerable leakage. The installed double barriers take the form of closed piping, both inside and outside the containment, and various types of isolation valves.

The most severe pipe rupture relative to radioactivity release during normal plant operations occurs in the Makeup and Purification System. This would be a rupture of the letdown line just outside the containment but upstream of the letdown control valves. A rupture at this point would result in a loss of