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SUBJECT: Forwards suppl analysis addressing NRC concerns re possibility of break in auxiliary steam header blowing down both SGs,per agreement made in 940309 telcon re notice of deviation from listed insp repts.

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April 8, 1994

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Subject: Oconee Nuclear Station
Docket Numbers 50-269, -270, and -287
Additional Response to Notice of Deviation
Identified in Inspection Report 269, 270, 287/93-31

In a telephone conversation on March 9, 1994, Duke agreed to provide, by April 8, 1994, a supplemental analysis to address NRC staff concerns regarding the possibility of a break in the auxiliary steam header blowing down both steam generators. This analysis is attached. The analysis concludes that no unreviewed safety question exists.

If you have any questions regarding this analysis, please call Scott Gewehr at (704) 382-7581 or Gregg Swindlehurst at (704) 382-5176.

Very truly yours,

M. S. Tuckman

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IE01

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Evaluation of a Steam Line Break in the Auxiliary Steam Header Which Results In Both Steam Generators Blowing Down

Overview

A pipe break in the auxiliary steam header is the initiating event. This can result in both steam generators blowing down due to valves MS-24 and MS-33 being open. These valves can be remotely closed by the operator to isolate the steam generators from a break in the auxiliary steam header or the other steam generator, but are not supplied by emergency power. If valves MS-24 or MS-33 are not promptly closed, an overcooling transient will result which is essentially a small steam line break affecting both steam generators. FSAR Chapter 15 analyzes a double-ended rupture of the largest steam line, but that event only affects one steam generator following closure of the turbine stop valves. The concern to be addressed is whether the rupture of the auxiliary steam pipe causes any of the applicable acceptance criteria for FSAR Chapter 15 transients to be exceeded, or whether it constitutes an unreviewed safety question.

Transient Description

The pipe break in the auxiliary steam header initiates a blowdown of both steam generators. The total break area is that corresponding to two 6-inch diameter holes, or 0.393 ft^2. The initial break flow is a total of 710 lbm/sec, or approximately 24% of full power steam flow. This is only 10.5% of the break flow for a rupture of the largest steam line, and consequently a much slower overcooling transient will result. The depressurization resulting from the small steam line break causes a decrease in steam flow to the turbine and a loss of electrical generation. The Integrated Control System (ICS) responds by withdrawing control rods and increasing feedwater flow in order to regain the lost electrical generation. The increase in total steam flow causes a decrease in cold leg temperatures. Due to the negative moderator coefficient of reactivity, the decrease in cold leg temperature will cause reactor power to increase. Although the ICS will stop withdrawing rods at 101% power, the temperature feedback will continue to cause reactor power to increase. The reactor is likely to trip on high flux at 105.5% power. However, the reduction in the reactor vessel downcomer temperature will decrease the power indicated by the excore detectors, and the actual power level can increase in excess of the 105.5% setpoint without causing a reactor trip. For this size steam line break, and with the most negative moderator temperature coefficient at end-of-cycle, the core power can increase to as high as 118% of full power. However, this is possible only if the Condensate and Main Feedwater Systems can provide that feedwater flowrate, which is highly unlikely.

In summary, the short-term response to the small steam line break is an increase in power that may not result in a reactor trip on high flux. Provided that the feedwater flow can keep up with the increase in total secondary steam flow, an equilibrium will be established for some period of time. This higher equilibrium power level is of concern with respect to the DNBR limit.

Continuing on in the scenario, at some point in time the reactor will automatically trip or will be manually tripped. Then the transient evolves into an overcooling event, due to the mismatch between the primary heat source (decay heat and reactor coolant pump heat) and the uncontrolled secondary depressurization and energy release due to the break. Pressurizer level and RCS pressure will decrease due to the shrinkage of the primary inventory. This continues until the pressurizer empties and low pressure causes an Engineered Safeguards actuation of the High Pressure Injection System. The HPI System injection offsets the primary coolant shrinkage, and eventually begins to refill the RCS. The emptying of the pressurizer also causes a loss of subcooling. The operators are required to promptly trip the reactor coolant pumps on loss of subcooling. This action is required for small break LOCA mitigation, but must be performed regardless of the cause of the loss of subcooling. The operators are also required to isolate both main and emergency feedwater flow to a steam generator with a steam line break. In this situation, both steam generators would be isolated since both are blowing down. The hot water in the feedwater pipe will eventually flash into the steam generator as pressure continues to decrease. This source of feedwater cannot be isolated from the steam generators. Valves on lines connected to the main steam line are manually closed to attempt to isolate the steam line break, including MS-24 and MS-33. The operators are also required to throttle HPI flow once subcooling is restored. These actions complete the short-term mitigation actions, with additional recovery actions such as restoring feedwater flow to stabilize RCS temperatures, and restarting a reactor coolant pump for thermal mixing and pressurizer spray, to follow later.

The continued overcooling introduces the concern that the reactor may return to power. This may or may not occur for steam line breaks depending on the rate of the overcooling (size of the break), the promptness of feedwater isolation, and the boration effect of the HPI and core flood tank injection. The smaller the break size, the less likely it is that a return to power can occur. The return-to-power situation is of concern due to the stuck rod assumption. An assumed stuck rod will cause very high core power peaking, which in combination with the power level may challenge the DNBR limit.

The loss of subcooling and the tripping of the reactor coolant pumps introduces a concern regarding the potential for primary-to-secondary heat transfer to be interrupted. The primary inventory shrinkage will be offset by HPI injection, but if the HPI cannot match the shrinkage rate, then additional steam voids will form in the locations where the highest temperature coolant exists. Due to the hot leg design of the B&W NSSS, a void at the top of the hot leg can interrupt loop flow. Another likely location for a void to form is in the upper head of the reactor vessel. A void in this location cannot lead to a loss of natural circulation. Eventually HPI will refill the primary and reduce the void size by compression. This may induce a restart of natural circulation which will condense the void. However, if a large void forms it will require mitigation actions such as venting or bumping reactor coolant pumps for complete removal. A sustained interruption of primary-to-secondary heat transfer due to a large hot leg void will cause an overcooling transient to evolve into an undercooling transient. Maintaining isolation of all feedwater will also cause an overcooling event to evolve into an undercooling event. This sequence of events could then lead to challenging the pressurizer PORVs and safety valves. Operator action would then be required to restore primary-to-secondary heat transfer, or to align HPI feed-and-bleed cooling. Either course of action will maintain core cooling. However, this potential sequence of events is considered to be a safety concern due to it evolving from an initiating event of a failure in a non-safety component, the auxiliary steam pipe.

Results of Analysis

In order to address these issues and concerns, several transient thermal-hydraulic analyses were performed to quantify the results of the overcooling event initiating from a failure of the auxiliary steam pipe with valves MS-24 and MS-33 open and not isolated during the event.

The reactor power level can increase to as high as 118% due to the positive reactivity from the cooldown. The analysis of the thermal-hydraulic statepoint resulted in a minimum DNBR greater than the limit. Therefore, the pre-trip transient response was acceptable for this small steam line break.

The post-trip overcooling of the reactor did not result in a loss of the shutdown margin, and no return-to-power occurred. The rate of overcooling and positive reactivity addition resulting from a small steam line break of this size was more than offset by the rate of negative reactivity provided by boration from the HPI flow. This result remained valid regardless of whether or not offsite power was lost, or with or without the operator tripping the reactor coolant pumps on the loss of subcooling. For larger break sizes, such as the large steam line break analyzed in the FSAR, this result would be different. Therefore, there is no post-trip DNBR concern for this small steam line break.

The analyses also showed that the magnitude of voiding in the loops was dependent on the number of HPI pumps in operation. Voiding was more likely to occur in the reactor vessel head than in the loop. A total of >112 ft^3 of void volume evenly split in both hot legs would be required to interrupt loop flow in both loops. This void volume did not occur in the analyses with more than one HPI pump in operation. However, with one HPI pump operating the loop void volume may be large enough to interrupt natural circulation. Repressurization due to HPI refilling of the RCS would compress the voids and possibly restore natural circulation had it been interrupted. For all HPI pump combinations natural circulation flow would be lost once the steam generators boiled dry. Operator action to restore feedwater in order to stabilize RCS temperatures is always required. Reactor coolant pump bumps or hot leg venting may also be necessary to reestablish natural circulation flow if a large hot leg void remained. These actions are only necessary for a severely degraded HPI System.

Therefore, the concern regarding an interruption of natural circulation flow, and the possible need for initiating HPI feed-and-bleed cooling, is not substantiated for this small steam line break provided that more than one HPI pump is operating. With only one HPI pump operating additional operator actions to induce loop flow may be necessary .

Summary

In summary, due to the small size of a rupture of the auxiliary steam line connecting the main steam lines, the resulting overcooling is much less than the large steam line break analyzed in the FSAR, and the results are less significant. There is no potential for fuel failure due to DNB for either the pre- or post-trip core response. There is no loss of shutdown margin and return to power, such as occurs for large steam line breaks. Also, the volumetric shrinkage due to the overcooling is slow enough to be compensated for by the expected HPI injection flowrate, and so only minimal voiding occurs outside of the pressurizer. Operator action is necessary to restore primary-to-secondary heat transfer. As an alternative, HPI feed-and-bleed cooling mode can be implemented for core cooling. It can therefore be concluded that a failure of the auxiliary steam pipe which results in a blowdown of both steam generators remains bounded by the FSAR main steam line break analysis, and that no unreviewed safety concern exists.