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10 CFR 50.46

Docket Nos. 50-348
50-364

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Joseph M. Farley Nuclear Plant
Peak Clad Temperature (PCT) Calculation

Gentlemen:

10 CFR 50.46(a)(3)(i) states:

> Each applicant for or holder of an operating license or construction permit shall estimate the effect of any change to or error in an acceptable evaluation model or in the application of such a model to determine if the change or error is significant. For this purpose, a significant change or error is one which results in a calculated peak fuel cladding temperature different by more than 50°F from the temperature calculated for the limiting transient using the last acceptable model, or is a cumulation of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50°F.

10 CFR 50.46(a)(3)(ii) requires a significant error be reported to the NRC within 30 days and a proposed schedule for reanalysis be provided or take other action to show compliance.

Alabama Power Company recently became aware of two Small Break Loss of Coolant Accident (SBLOCA) evaluation model errors in the calculation of PCT for Farley Nuclear Plant Units 1 and 2. The errors resulted in increases in the SBLOCA PCT of 3[illegible]°F and 40°F for both Units 1 and 2. Additionally, a 39°F SBLOCA PCT penalty concerning an auxiliary feedwater enthalpy change delay time input error has been identified. Although no single change is greater than 50°F, the sum of the changes is greater than 50°F. Attachment 1 provides the current SBLOCA PCTs for both Unit 1 and Unit 2. It should be noted that neither unit is limited by SBLOCA PCT; both are limited by large break LOCA PCTs. Attachment 2 provides additional details on the model changes developed by Westinghouse.

The current evaluation model for SBLOCAs for both Farley units is the Westinghouse WFLASH model. Evaluations of the WFLASH SBLOCA model errors and change have been completed and quantified. These evaluations demonstrate that the analysis of record PCT remains valid and that the total resultant PCTs, i.e., analysis of record PCT plus all other PCT penalties due to plant modifications and PCT assessments due to LOCA model changes or errors, for both units continue to remain in compliance with



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'0 CFR 50.46 requirements. It should be noted that in a recently submitted :ense amendment for use of VANTAGE-5 fuel, a Westinghouse NOTRUMP evaluation has been provided. All three of the above model error issues have been accounted for in the new NOTRUMP analysis.

In summary, the SBLOCA PCTs for both Farley units have recently increased due to Westinghouse ECCS model changes and due to an auxiliary feedwater enthalpy change delay time input error. The SBLOCA is not the limiting design basis accident for either Farley Unit 1 or Unit 2 with respect to PCT. The resulting PCT continues to comply with the 2200°F limit.

If there are any questions, please advise.

Respectfully submitted,

ALABAMA POWER COMPANY

J. D. Woodard

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JDW/REM:map 0519

Attachments

cc: Mr. S. D. Ebneter
Mr. S. T. Hoffman
Mr. G. F. Maxwell

ATTACHMENT 1

PEAK CLAD TEMPERATURE CHANGES FOR
SMALL BREAK LOCA ANALYSIS

| | Unit 1 SBLOCA | Unit 2 SBLOCA |
|---|---|---|
| Resultant PCT from ECCS Evaluation Model (°F) | 1712* | 1712* |
| Modifications to ECCS Evaluation Model (°F) | -2* | -2* |
| PCT Change Due to Downflow to Upflow Conversion of Reactor Vessel Baffle/Barrel Region Coolant Flow (°F) | 117* | N/A |
| PCT Change Due to Spilling the Broken Loop Safety Injection to Containment Backpressure (°F) | 46* | 46* |
| PCT Change Due to Loose Parts in the Reactor Coolant System (°F) | 32* | N/A |
| PCT Change Due to Increased Uncertainty in Reactor Coolant System Temperature (°F) | 2* | 2* |
| PCT Change Due to Auxiliary Feedwater Enthalpy Delay Modelling (°F) | 39 | 39 |
| PCT Change Due to Evaluation Model (EM) Changes for Fuel Rod Model Initial Condition Revision (°F) | 37 | 37 |
| PCT Change Due to Evaluation Model (EM) Revision for Rod Internal Pressure Initial Condition (°F) | 40 | 40 |
| Total Resultant PCT (°F) | 2023 | 1874 |

*Reported in 10 CFR 50.46 Annual Report dated March 21, 1991.

ATTACHMENT 2

Details Concerning SBLOCA Model Changes For:

- Fuel Rod Initial Condition Inconsistency
- Rod Internal Pressure Initial Condition Assumption
- Auxiliary Feedwater Enthalpy Delay

## FUEL ROD INITIAL CONDITION INCONSISTENCY REVISIONS

During the review of the original Westinghouse ECCS Evaluation Model following the promulgation of 10 CFR 50.46 in 1974, Westinghouse committed to maintain consistency between future loss-of-coolant accident (LOCA) fuel rod computer models and the fuel rod design computer models used to predict fuel rod normal operation performance. These fuel rod design codes are also used to establish initial conditions for the LOCA analysis.

### Change Description

It was found that the small break LOCA code versions were not consistent with fuel design codes in the following areas:

1. The LOCA codes were not consistent with the fuel rod design code relative to the flux depression factors at higher fuel enrichment.

2. The LOCA codes were not consistent with the fuel rod design code relative to the fuel rod gap gas conductivities and pellet surface roughness models.

3. The coding of the pellet/clad contact resistance model required revision.

Modifications were made to the fuel rod models used in the LOCA Evaluation Models to maintain consistency with the latest approved version of the fuel rod design code.

### Effect of Changes

The changes made to make the LOCA fuel rod models consistent with the fuel design codes were judged to be insignificant, as defined by 10 CFR 50.46(a)(i). To quantify the effect on the calculated peak cladding temperature, calculations were performed which incorporated the changes.

### Small Break LOCA

The effect of the changes on the small break LOCA analysis peak cladding temperature calculations was determined using the 1985 small calculations for a typical three-loop plant and a typical four-loop plant. The analysis calculations confirmed that the effect of the changes on the small break LOCA ECCS Evaluation Model were insignificant as defined by 10 CFR 50.46(a)(3)(i). The calculations showed that 37°F would bound the effect on the calculated peak cladding temperatures for the three-loop plants.

## ROD INTERNAL PRESSURE INITIAL CONDITION ASSUMPTION

### Change Description

The Westinghouse small break loss-of-coolant accident (LOCA) emergency core cooling system (ECCS) Evaluation Model analyses assume that higher fuel rod initial fill pressure leads to a higher calculated peak cladding temperature, as found in studies with the Westinghouse large break LOCA ECCS Evaluation Model. However, lower fuel rod internal pressure could result in decreased cladding creep (rod swelling) away from the fuel pellets when the fuel rod internal pressure was higher than the reactor coolant system (RCS) pressure. A lower fuel rod initial fill pressure could then result in a higher calculated peak cladding temperature.

The Westinghouse small break LOCA cladding strain model is based upon a correlation of Hardy's data. Evaluation of the limiting fuel rod initial fill pressure assumption revealed that this model was used outside of the applicable range in the small break LOCA Evaluation Model calculations, allowing the cladding to expand and contract more rapidly than it should. The model was corrected to fit applicable data over the range of small break LOCA conditions. Correction of the cladding strain model affects the small break LOCA Evaluation Model calculations through the fuel rod internal pressure initial condition assumption.

### Effect of Changes

Implementation of the corrected cladding creep equation results in a small reduction in the pellet to cladding gap when the RCS pressure exceeds the rod internal pressure and increases the gap after RCS pressure falls below the rod internal pressure. Since the cladding typically demonstrates very little creep toward the fuel pellet prior to core uncovery when the RCS pressure exceeds the rod internal pressure, implementation of the correlation for the appropriate range has a negligible benefit on the peak cladding temperature calculation during this portion of the transient. However, after the RCS pressure falls below the rod internal pressure, implementation of an accurate correlation for cladding creep in small break LOCA analyses would reduce the expansion of the cladding away from the fuel compared to what was previously calculated and results in a PCT penalty because the cladding is closer to the fuel.

Calculations were performed to assess the effect of the cladding strain modifications for the limiting three-inch equivalent diameter cold leg break in typical three-loop and four-loop plants. The results indicated that the change to the calculated peak cladding temperature resulting from the cladding strain model change would be less than 20°F. The effect on the calculated peak cladding temperature depended upon when the peak cladding temperature occurs and whether the rod internal pressure was above or below the system pressure when the peak cladding temperature occurs. For the range of fuel rod internal pressure initial conditions, the combined effect of the fuel rod internal pressure and the cladding strain model revision is bounded by 40°F.

# AUXILIARY FEEDWATER ENTHALPY DELAY TIME INPUT

Change Description

In the small break LOCA analysis, values are inserted for Auxiliary Feedwater (AFW) delay time which would presume a certain length of time from signal for pumps to start up and initiate delivery (60 seconds). Another delay is input for the enthalpy change. This second delay accounts for the standing water in the pipe, at feedwater inlet temperature, to be purged, and for the pumped AFW water, at a reduced temperature, to finally arrive at the steam generator (SG) injection point before considering any cooling effect from AFW. In WFLASH, the enthalpy delay is given in user documentation to be a standard value of 132 seconds.

This issue was investigated for NOTRUMP following the discovery of a user documentation inconsistency, which, in effect, took credit for the cooler water instantaneously, ignoring the time required to clear out the line of the standing feedwater.

When confirmatory calculations for the subject plant were performed to determine a realistic delay time for the AFW enthalpy change, it was found that the old WFLASH standard of 132 seconds was in doubt. That is, main feedline piping actually installed in the plant had a volume great enough that, even assuming this volume of standing water to be swept out and precede the AFW lower enthalpy flow delivery, the time delay may in fact be greater than that assumed in the analysis. This determination prompted the evaluation for the WFLASH analyses as well. For the Farley WFLASH analysis, the enthalpy delay was input as 129.4 seconds. This value was determined to be inconsistent with values used in other safety analyses. The evaluation for WFLASH analyses was to ascertain actual enthalpy delay assumed and then to determine its adequacy in view of plant piping configurations.

There are two basic mechanisms for heat removal during a small break LOCA. These are 1) heat removal via discharge of high enthalpy fluid through the break, and 2) heat transfer from the primary to secondary across the SG tubes. Increasing the delay time for the delivery of low enthalpy AFW into the SG secondary leaves a higher total energy on the secondary side. This would tend to inhibit heat transfer from the primary to secondary and place greater reliance on break flow to remove energy from the RCS. Smaller breaks rely more heavily on heat transfer across the SG tubes due to limited break area resulting in limited break flow, and therefore would tend to be more sensitive to additional heat removal requirements. On the other hand, the smaller breaks exhibit a slower blowdown transient resulting generally in a more moderate clad heat-up rate. This trade-off suggests that the effects of delayed AFW enthalpy vary with break size with no absolute certainty of which break sizes will be most affected by the delay. Therefore, the evaluation considered not only the limiting break size, but also other break sizes which could potentially have a larger PCT effect which could result in a change in the most limiting break size.

Effect of Changes

In work being done for another plant, an inconsistency was discovered between the main feedwater purge time assumed in the small break LOCA analyses utilizing the NOTRUMP Evaluation Model and the time indicated by plant geometry and operating conditions. The purge time refers to the time required for the low flow, low enthalpy AFW to remove and replace stagnant high enthalpy main feedwater from the feedwater piping after AFW actuation.

The root cause of this error was determined not to be model-related in that computer coding was correct. It was an inconsistency in the Evaluation Model documentation provided to the user. By adhering to user instructions, these incorrectly based input values would be input into the Evaluation Model. The user documentation was subsequently corrected. However, inasmuch as the NOTRUMP treatment of this AFW enthalpy delay is similar to the approach taken with the older WFLASH code, a general survey of the delay time assumption in plant specific analyses was performed. As a result of this survey, it was found that the analyses of record for the Farley Units were also affected negatively by assumption of inadequate AFW enthalpy delay time. An evaluation was performed to assess the effect it might have on the reported small break analysis of record. That evaluation is described below.

Given that the small break LOCA analysis of record has the AFW enthalpy delay input as 129.4 seconds, an estimate of the actual purge volume and delay time was calculated. Data extracted from the analysis showed an input enthalpy for the main feedwater as 416.2 Btu/lbm., and the auxiliary feedwater enthalpy is input as 90.4 Btu/lbm. The auxiliary flow rate was 1050 gpm, or 350 gpm/loop. Based on a sweep volume consistent with that used as a basis for feedline break analyses, the sweep time once AFW is initiated was calculated to be 139.7 seconds. When considering the total delay time, then, the enthalpy change should not be credited until 199.7 seconds into the transient as opposed to 129.4 seconds assumed in the analysis of record. This calculation confirmed that the Farley WFLASH SBLOCA analysis would be affected by this issue.

The PCT effect of this increased delay was calculated by considering the difference in enthalpy between main feedwater and auxiliary feedwater and projecting the effect of removing this amount of cooling capacity from the secondary (effectively assuming this unwarranted cooling would remain in the RCS primary side). As noted above, break flow is the other prominent heat removal mechanism in the transient. The PCT effect could then be estimated by assuming this increase of energy could be removed from the primary side solely by break flow. The result was expressed in terms of an additional amount of transient time to remove this energy before the PCT conditions would otherwise be achieved. Assuming continual rod heat-up during this additional time of core uncovery, the projected PCT increase could then be extrapolated. The projected peak clad temperature penalty was calculated as 38.32 degrees which was rounded up to 39 degrees for reporting purposes.