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10 CFR 50.4
10 CFR 50.46

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Browns Ferry Nuclear Plant, Unit 1
Facility Operating License No. DPR-33
NRC Docket No. 50-259

Subject: **Final Report of Emergency Core Cooling System Evaluation Model Changes**

References:
1. Letter from TVA to NRC, "30-Day Report of Emergency Core Cooling System (ECCS) Evaluation Model Changes," dated February 12, 2010
2. Letter from TVA to NRC, "30-Day Report of Emergency Core Cooling System (ECCS) Evaluation Model Changes," dated March 24, 2010
3. Letter from TVA to NRC, "Technical Specification Change TS-473 - AREVA Fuel Transition," dated April 16, 2010
4. Letter from NRC to TVA, "Browns Ferry Nuclear Plant Unit 1 - Non-acceptance of Utilization of AREVA Fuel and Associated Analysis Methodologies (TAC No. ME2451)(TS-467)," dated December 23, 2009

The Tennessee Valley Authority (TVA) is submitting this report in accordance with 10 CFR 50.46(a)(3)(ii) due to changes in the model used to determine compliance with Emergency Core Cooling System (ECCS) requirements.

During the NRC acceptance review of the Browns Ferry Nuclear Plant (BFN), Unit 1, license amendment request supporting the transition to ATRIUM-10 fuel (Reference 4), NRC requested that TVA address the single failure of one or all Automatic Depressurization System (ADS) valves in the Loss-Of-Coolant Accident (LOCA) analysis. TVA subsequently determined that a single failure of a 250VDC battery can

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disable the automatic functioning of the ADS. This battery failure impacts Reactor Motor Operated Valve (RMOV) Board B, which contains both trains of the ADS logic.

The previous LOCA analyses had only considered failure of the battery supplying RMOV Board A as the limiting single failure. The RMOV Board A scenario disables the High Pressure Coolant Injection (HPCI) system but leaves the automatic ADS function intact.

The initial assessment of the battery failure affecting RMOV Board B produced a limiting Appendix K Peak Clad Temperature (PCT) of 2168°F when calculated at Extended Power Uprate conditions, and was reported to the NRC in Reference 1. Subsequently, the assessment was re-performed at Current Licensed Thermal Power (CLTP) conditions and resulted in a limiting Appendix K PCT value of 1915°F, which was reported to the NRC in Reference 2. These evaluations were used to support operability assessments until a new baseline LOCA break spectrum analysis could be completed by GE Hitachi (GEH).

GEH has now completed the new baseline LOCA break spectrum analysis for BFN Unit 1 to address the failure of the battery which powers RMOV Board B. This new analysis addresses various break types (recirculation line, feedwater line, main steam line, and low pressure core spray line). Breaks in the HPCI injection line are bounded by the feedwater line break analysis, while breaks in the HPCI steam supply line are bounded by the main steam line break analysis. The evaluations in the new report bound both the GNF GE13 and GE14 fuel for BFN, Unit 1, and were performed at CLTP conditions. This report has been provided to the NRC in Attachments 20 and 21 of Technical Specification Change TS-473 (Reference 3). Reference 3 was submitted to support the introduction of AREVA ATRIUM-10 fuel into BFN, Unit 1.

It should be noted that the new GEH baseline LOCA break spectrum report only addresses the RMOV Board B scenario. The prior GEH LOCA analysis (which is provided in Reference 3 as Attachments 18 and 19) includes the CLTP based evaluations of the battery failure impacting RMOV Board A.

The limiting Appendix K break for GE14 fuel at current licensed power was shown to be a 0.24 ft^2 break in the recirculation discharge piping with single failure of the battery supplying RMOV Board B. This break resulted in an Appendix K PCT of 1828°F. In the GEH method, additional uncertainties are applied to the Appendix K results to derive a Licensing Basis PCT. For the GE14 fuel type, the new Licensing Basis PCT value was determined to be 1920°F.

The report in Attachment 20 of Reference 3 provides a thermal limit set down for the GE13 fuel type, which allows the Licensing Basis PCT value of 1810°F reported in Attachment 18 of Reference 3 to remain applicable.

With the submittal of the above mentioned reports in TS-473 (Reference 3), the LOCA analysis of record for BFN, Unit 1, will transition from an Extended Power Uprate basis to a CLTP basis.

In evaluating the single failure of the ADS, it becomes necessary to credit manual operator action as part of the event mitigation. The analysis of all LOCA events involving the failure of the battery which powers RMOV Board B assumes that the operators open 4 ADS valves manually 10 minutes after the event begins. The crediting of manual operator action is being used as a temporary measure until such time as the ADS can be modified at BFN to ensure automatic functioning for single failures. As a result, the guidance provided in Section C.5, "Use of Temporary Manual Action in Place of Automatic Action in Support of Operability," of NRC Inspection Manual Part 9900: "Technical Guidance, Operability Determination and Functionality Assessments for Resolution of Degraded or Non-conforming Conditions Adverse to Quality or Safety," has been utilized. Enclosure 1 provides the background and justification for making this assumption in the LOCA analyses. Enclosure 2 includes the TVA commitment to perform a modification at BFN Unit 1 to ensure automatic functioning of 4 ADS valves for single failures.

The new licensing PCT values noted above are all well below the 2200°F limit. The other 10 CFR 50.46 requirements (e.g., maximum cladding oxidation, maximum hydrogen generation) are maintained within limits.

There is a new regulatory commitment contained in this letter as reflected in Enclosure 2. Should you have any questions concerning this submittal, please contact Terry Cribbe at (423) 751-3850.

Respectfully,

R. M. Krich

Enclosures: 1. Operator Actions for Manual Automatic Depressurization System Actuation
2. Regulatory Commitment

cc (Enclosures):
NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Browns Ferry Nuclear Plant

Enclosure 1

Browns Ferry Nuclear Plant, Unit 1

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Operator Actions for Manual Automatic Depressurization System Actuation

Introduction

During a meeting between the Tennessee Valley Authority (TVA) and the NRC on February 17, 2010, NRC staff requested TVA provide additional information regarding Automatic Depressurization System ADS manual operation credit in certain portions of the Loss-Of-Coolant Accident (LOCA) reanalysis done for Brown Ferry Nuclear Plant (BFN). The additional information provided here, with respect to ADS manual operation credit, is consistent with the information provided by Detroit Edison Company for Fermi 2 in a letter to the NRC dated June 10, 2009 (Reference 1). NRC approval of Fermi 2 credit for ADS manual operation is documented in the NRC letter and associated Safety Evaluation dated June 30, 2009 (Reference 2). The LOCA reanalysis for BFN was performed to address a single failure disabling the automatic functioning of ADS. The single failure involves a loss of the 250VDC battery supplying power to Reactor Motor Operated Valve (RMOV) Board B. Both logic trains of automatic ADS initiation instrumentation are powered from RMOV Board B; consequently, loss of power to the board results in the loss of automatic ADS function. For this single failure scenario, operators would still be able to manually open 4 ADS valves.

The RMOV Board B loss of DC power analysis assumes operators manually open 4 ADS valves 10 minutes after the break occurs. This is required because automatic ADS initiation logic is disabled in this scenario. The 10-minute assumption, although shorter than the 20-minute time period provided in the guidance in Section 6.3 of NUREG-0800 (Reference 3), is consistent with the earliest post accident delay after which credit for manual operator actions (such as initiation of suppression pool cooling) is considered for licensing and design basis analyses at BFN and has been demonstrated, as reflected below, to be conservative relative to the timing of the actual operator action.

The NRC requested TVA provide information to demonstrate BFN operators have appropriate procedures, training, control room indications, and time to ensure this manual action can be reliably performed as assumed in the LOCA analysis with RMOV Board B loss of DC power.

Procedures

In the event of a small break where the high pressure makeup cannot maintain water level, BFN Emergency Operating Instruction u-C-1 (Level Control) contains a step directing the operator to Emergency Operating Instruction u-C-2 (Emergency Depressurization). Note: Here and in the discussion of control room indications to follow, "u" represents the reactor unit number.

The BFN Emergency Operating Instructions (EOIs) are based on the NRC approved Boiling Water Reactor Owners Group Emergency Procedure Guidelines (EPGs), and Severe Accident Guidelines (SAGs). BFN EOIs require operators begin depressurization prior to reactor water level reaching a value of -180 inches on the narrow range instruments, which corresponds to the EPG Minimum Steam Cooling Reactor Water Level (MSCRWL). The MSCRWL is defined as the lowest reactor water level at which the covered portion of the core will generate sufficient steam to prevent any clad temperature in the uncovered portion of the core from exceeding 1500° Fahrenheit assuming the most limiting top peaked power shape prior to the event. For

BFN, the MSCRWL is 18 inches below the top of the active fuel. Water levels between the top of active fuel and the MSCRWL represent the range in which the operator must recognize manual depressurization is required. The procedure also requires depressurization to begin prior to water level reaching -200 inches. This level corresponds to the Minimum Zero Injection Water Level defined in the EPGs and ensures the clad temperature does not exceed 1800° Fahrenheit, with no injection.

In addition, consistent with the NRC approved Revision 4 of the EPGs, manual inhibition of the ADS is required to allow time for the high pressure Emergency Core Cooling Systems (ECCSs) to restore water level and avoid unnecessary core uncovery. This action effectively makes reactor vessel depressurization a manual action regardless of the particular LOCA event.

The key steps in BFN depressurization instruction EOI u-C-2 are:

- Attempt to manually open all 6 ADS valves (minimum of 4 valves are required for depressurization).
- If the minimum number of 4 cannot be achieved with ADS only, manually open other non-ADS safety relief valves with the goal of opening a total of 6 valves (minimum of 4 total is still required).
- Bypass and restore drywell pneumatics as necessary.
- Actions are provided once depressurization is complete to establish shutdown cooling or pursue alternate emergency depressurization systems if the minimum 4 relief valves cannot be opened.

Training

The BFN licensed operator training program utilizes a simulator facility in accordance with 10 CFR 55.46(c), "Plant-referenced simulators." Shift crews are typically evaluated in the simulator during each training cycle. Each simulator evaluation includes performance of Critical Tasks, which must be accomplished in order to pass the simulator evaluation. For simulator scenarios involving small breaks requiring depressurization, one Critical Task includes commencing emergency depressurization after level drops below top of active fuel, but before reaching -200 inches. Another Critical Task of interest is for the crew to restore and maintain water level above the top of active fuel after depressurization is complete. If an operator fails to meet a Critical Task, he is removed from shift and a remediation plan is determined. The remediation plan typically requires additional training time in the simulator and a retest involving a scenario similar to the one the operator failed.

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During 2009, a total of sixteen BFN operator crews were presented with at least one simulator scenario requiring emergency depressurization based on water level indications. Operator lesson plan OPL177.078 was the principle scenario presented to the crews in 2009. In this scenario, the reactor is operating at 100% power, when an earthquake occurs. The operators then shutdown the reactor using plant procedure 0-AOI-100-5. Aftershocks then cause a feedwater line break to occur resulting in all available High Pressure Coolant Injection (HPCI) flow being diverted through the feedwater line break. Reactor Core Isolation Cooling (RCIC) is also lost due to a break in the RCIC steam supply line. This leaves the operators with control rod drive flow and standby liquid control as the only sources of high pressure injection. Reactor water level will decrease to the point where the crew is expected to manually depressurize the reactor (the automatic ADS would have already been inhibited per the EOIs) in accordance with EOI u-C-2. Following manual depressurization, low pressure ECCS subsystems then terminate the event. All sixteen crews successfully passed this scenario and initiated manual depressurization in accordance with the Critical Task requirements mentioned earlier.

This training scenario is comparable to the feedwater line break cases included in the LOCA reanalyses performed by GE Hitachi and AREVA. In those analyses, the feedwater line break is assumed to be large enough to result in all of the HPCI flow being diverted out of the break, just as in the training scenario. A manual depressurization is required both in the training scenario and in the LOCA analyses. The operators must manually initiate ADS in the training scenario due to the ADS being inhibited early in the event. In the LOCA analysis, automatic ADS would not occur due to the RMOV Board B loss of DC power and the operators are assumed to manually start depressurization at 10 minutes, rather than on indicated level. Low pressure ECCS subsystems are used to recover reactor water level back above the top of active fuel in both the training scenario and the LOCA analyses.

In the LOCA analyses, the most limiting case for the RMOV Board B loss of DC power is the recirculation discharge line break, rather than the feedwater line break. While this is a different break from the simulator scenario described above, the differences in terms of operator action are not significant. In the recirculation line break with RMOV Board B loss of DC power, HPCI is still available, but is unable to maintain water level for the worst break size. The LOCA analysis does not credit the Reactor Core Isolation Cooling (RCIC) system, so operators would still be presented with a situation where the high pressure systems (i.e., HPCI and RCIC systems) cannot maintain level above the top of active fuel, thereby requiring manual depressurization. Results of the training scenario evaluation indicate BFN operators would be capable of mitigating the limiting recirculation line break event presented in the revised LOCA analysis reports.

Control Room Indications

In the event manual depressurization is required, operators have several independent means to confirm a given ADS or safety relief valve is open. The hand switch for each valve has two lights next to it, one red and one green, indicating the position of the solenoid on the valve operator. A change from green to red provides indication the solenoid has repositioned and the valve has opened. The valve position indicator lights are powered from the same source as the valve itself. For the case of RMOV Board B loss of DC power, the 4 remaining ADS valves would be operable and have position indication.

Each valve also has an acoustic flow indicator located on panel u-FMT-1-4, above the valve hand switches. When a valve is opened, a set of Light Emitting Diode (LED) lights on this indicator will light up. Each valve has its own associated set of LED lights. The acoustic monitoring detects the sound made by the steam as it passes through the relief valve piping and provides the operator with a positive indication the valve is passing steam. The acoustic flow indicator is powered by 120VAC power and would be unaffected by failure of the battery powering RMOV Board B.

The operator also has the ability to look at the tailpipe temperatures for each ADS or relief valve discharge pipe. This temperature is available to be viewed as a display point on the control room computer monitors or by looking at chart recorder u-TR-1-1, which is also in the control room. The observed elevated temperature readings would provide further independent confirmation the valves of interest are in fact open. The tailpipe temperature indicators are also powered by 120VAC power and would be unaffected by failure of the battery powering RMOV Board B.

The operators would be able to determine depressurization is occurring by observing reactor pressure indicators in the control room. These instruments are safety-related divisional Class 1E equipment. The operators would use the narrow range reactor level instrumentation in order to determine the need for the manual depressurization. Like the pressure instrumentation, the narrow range level instrumentation is safety-related divisional Class 1E equipment.

As noted earlier, manual depressurization in the BFN LOCA analysis is only assumed for the single failure of the DC battery powering RMOV Board B. If this failure were to occur, the operator is provided with annunciators indicating the failure. Annunciator u-EA-57-100 illuminates when RMOV Board B loses power. The loss of RMOV Board B power disables the automatic ADS function, resulting in the annunciator in control room panel u-XA-55-3C window 32 becoming lit. This annunciator reads "ADS BLOWDOWN POWER FAILURE." This annunciator has an associated Annunciator Response Procedure 9-3C, highlighting automatic ADS logic has failed and manual valve operation will be required to depressurize. These annunciators are powered by 48VDC power and are unaffected by failure of the battery powering RMOV Board B.

In summary, operators are provided with adequate instrumentation and indicators to recognize the need for depressurization and to successfully carry out the action.

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Timing Aspects

The limiting LOCA event in both vendor analyses is a small break of the recirculation discharge line with single failure of the battery powering RMOV Board B. In this event the LOCA analysis from both vendors shows reactor water level reaches the top of active fuel approximately three minutes into the event. This represents the time at which operators would be required to start manual depressurization per the EOIs. This time is much less than the 10 minute time assumed for the manual operator action in the LOCA analysis. In addition, the LOCA analyses from both GE Hitachi and AREVA show that for this limiting break, the clad temperature reaches the peak value before the 10-minute point is reached. This means the issue of operator timing with regard to manual depressurization is irrelevant for the limiting break, since natural depressurization allows the low pressure ECCS to act prior to the 10 minute manual ADS initiation point being reached.

For feedwater line breaks, HPCI injection line breaks, and steam line breaks (with single failure of the battery powering RMOV Board B), the PCT is not reached prior to the 10 minute point and the results are dependent upon the manual ADS initiation. However, these events are non-limiting in terms of peak clad temperature compared to the recirculation line breaks. For these breaks, the PCT is approximately 1100° F or less for both vendors. Therefore, a brief delay in operator action (one to two minutes) beyond the 10-minute assumption would not challenge PCT limits for these breaks.

For core spray line breaks with the RMOV Board B failure, the HPCI would be available and the analyses by both vendors show that the RMOV Board A failure (which disables HPCI but leaves automatic ADS function) is much more limiting. In the GE analysis of RMOV Board B failure, the water level never reaches a point where manual ADS would be required, so the assumption of manual ADS at 10 minutes becomes an arbitrary assumption that does not affect the results. The AREVA analysis of the event with RMOV Board B failure shows the PCT remains below 1000° F. A delay in operator action beyond 10 minutes for this break is either non-consequential or would result in only a modest PCT increase that would not challenge the limiting PCT.

The action to manually depressurize would only require a few minutes or less to perform, including any restoration of drywell pneumatics. The 10-minute assumption used in the LOCA analyses is reasonable and conservative relative to the actual operator timing that would occur.

Summary

Based on the information presented above, the operators have the training, procedures, instrumentation, control room indicators, and adequate time to complete the critical task of performing emergency manual depressurization of the reactor in response to a small break LOCA.

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References

1. Letter from J. H. Plona (DTE Energy) to NRC, "Additional Information to Support Review of Plant Specific Emergency Core Cooling System Evaluation Model Reanalysis," dated June 10, 2009.

2. Letter from M. L. Chawla (NRC) to J. M. Davis (Detroit Edison Company), "Fermi 2 - Approval of Plant Specific Emergency Core Cooling System (ECCS) Evaluation Model Reanalysis (TAC No. MD9169)," dated June 30, 2009.

3. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 6.3, "Emergency Core Cooling System," Revision 3, March 2007.

Enclosure 2

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Regulatory Commitment

Automatic Depressurization System Modification

The Automatic Depressurization System (ADS) will be modified to provide a single failure proof automatic initiation capability of 4 ADS valves, regardless of which 250VDC battery fails. This modification is expected to be made to BFN, Unit 1, during the Unit 1 outage in the fall of 2012.