UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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September 7, 2005

EA-05-169

Tennessee Valley Authority
ATTN: Mr. K. W. Singer
Chief Nuclear Officer and
Executive Vice President
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: NRC INSPECTION REPORT NO. 05000390/2005013; PRELIMINARY GREATER THAN GREEN FINDING; WATTS BAR NUCLEAR POWER PLANT

Dear Mr. Singer:

This letter and the enclosed supporting documentation discuss a finding that appears to have greater than very low safety significance. As described in Section 1R20.2 of NRC Inspection Report 05000390, 391/2005002, issued on April 29, 2005, a finding was identified with respect to procedural non-compliances at Watts Bar. The finding involved a challenge to reactor coolant system (RCS) integrity by pressurizer power-operated relief valve (PORV) actuations and a challenge to RCS inventory control by the loss of RCS coolant via the open PORVs. On February 22, 2005, we determined that your staff made inappropriate operational decisions during the transition to solid plant operations to return a charging control valve to service following a design change and before all post-maintenance testing (PMT) was complete. As a result of the erratic control provided by the valve, operators failed to adequately implement procedures for solid plant operations, as required by the Watts Bar Technical Specifications, which resulted in multiple actuations of the pressurizer PORVs in low temperature over pressure (LTOP) mode.

On February 23, 2005, Problem Evaluation Report (PER) 77176 was initiated in the Watts Bar Corrective Action Program for the cycling of the pressurizer PORV the previous day. The operator log entry of the PER initiation was the first log entry that made any mention of the charging problems and PORV lifts from the previous day. The PER also implied that only a single lift of the PORV had occurred. The inspectors' review of the reactor coolant and charging system parameters for the period in question determined that the Cold Over-Pressure Mitigating System was challenged by the actuation of both PORVs multiple times during a two-hour period. The block valve for one PORV, 1-RFV-63-340A, had been closed to reduce containment gas problems via leakage from the valve packing and as such this PORV did not relieve actual pressure during a total of seven actuations. However, the other PORV, 1-RFV-63-334D, actuated a total of five times to reduce pressure in parallel with a group of five actuations by 1-RFV-63-340A. The inspectors determined that the first single actuation and the group of five/five actuations of 1-RFV-63-340A/1-RFV-63-334D were due to a failure to comply with procedural requirements contained in General Operating Instruction (GO)-6, Unit Shutdown from Hot Standby to Cold Shutdown. To transition to solid water operations, Section 5.5, Step [1] [e] states, “Slowly RAISE charging to fill Pzr at less than 30 gpm.” Contrary to this, the 30-gpm requirement was exceeded resulting in the PORV actuations. The requirement was exceeded when the normal charging flow control valve 1-FCV-62-93 exhibited erratic operation following activities to swap from bypass to normal charging and when operators swapped back to bypass charging.

Previous erratic control problems with 1-FCV-62-93 had resulted in a precaution and limits statement in GO-6 stating that it may cycle with RCS pressure below 500 psig when manually attempting to control low charging flow rates. During the transition to solid plant operations, the RCS pressure was less than 400 psig. Additionally, the Watts Bar RCS system description states that when the RCS is operated in the water-solid mode, the charging flow to the RCS is to be set at a constant value. This was not consistent with the operational decision to place 1-FCV-62-93 in service, before all post-modification testing was done, during the transition to solid plant operations.

Work Order (WO) 04-825584-000, which implemented the design change stated that the equipment cannot be declared operable until the modification turnover package was complete. However, the WO also allowed the valve to be returned to operation with testing to be done later when plant conditions allow. This WO allowance was implemented on February 22, 2005 with remaining tests, including a valve stroke under high differential pressure, not yet complete.

A last PORV actuation was due to RCS heat up and resultant pressure increase from the closure of the 1A Residual Heat Removal (RHR) heat exchanger outlet valve per System Operating Instruction (SOI) 74.01, Residual Heat Removal, Section 8.11, Flush of A Train RHR Heat Exchanger Bypass during Shutdown Cooling. This aspect was not described in the original PER 77176 problem description. The following action was contained in a procedure note, “The effect on RCS heat up/cool down should be evaluated.” This action was not appropriately implemented in that the performance of Section 8.11 during solid plant operation allowed sufficient RCS heat up to result in the actuation of the Pressurizer PORV.

The NRC has determined that the procedural noncompliances identified above represent a performance deficiency that had an impact on safety by affecting the cold over-pressure mitigation or low temperature over-pressure system required by the Watts Bar Technical Specifications (TS). Specifically, TS 5.7.1.1 states that written procedures shall be implemented and maintained covering the activities in the applicable procedures recommended by RG 1.33, Revision 2, Appendix A, February 1978, of which Part 2.j requires a procedure for hot standby to cold shutdown and Part 3.c requires a procedure for shutdown cooling system. GO-6, Unit Shutdown from Hot Standby to Cold Shutdown, Section 5.5, Step [1] [e] states, “Slowly RAISE charging to fill Pressurizer at less than 30 gpm.” SOI-74.01, Residual Heat Removal, Section 8.11, implemented a flush of the A train RHR heat exchanger bypass during shutdown cooling and contained a note which stated, “The effect on RCS heatup/cool down should be evaluated.” Each procedure was not adequately implemented approaching and during solid plant operations on February 22, 2005. This performance deficiency constitutes an apparent violation of TS 5.7.1.1, in that, TVA failed to follow approved procedures, which resulted in a challenge to RCS integrity by pressurizer PORV actuations and a challenge to RCS inventory control by the loss of RCS coolant via the open PORVs. Accordingly, this finding is identified as an Apparent Violation (AV) 05000390/2005013-01, Failure to Implement and Maintain Shutdown Procedures which Resulted in Pressurizer PORV Actuations. The finding is being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedures for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600. The current Enforcement Policy is included on the NRC's Web site at www.nrc.gov; select **What We Do, Enforcement,** then **Enforcement Policy**.

This finding was assessed using the applicable Significance Determination Process (SDP) and was preliminarily determined to be a Greater-Than-Green finding (i.e., a finding with some increased importance to safety, which may require additional NRC inspection). The finding appears to have greater than very low safety significance, primarily because the dominant SDP scenario for this event results from a failure of the PORV to open following a demand on the overpressure protection system (OPS) and the residual heat removal (RHR) suction relief valve failing to close after being challenged. Subsequent failure of both RHR isolation valves *or* failure of the operator to open a PORV to establish feed and bleed results in core damage. The results of the NRC's Phase 3 SDP are attached to this letter. We will consider any additional information you may have that could assist the NRC in making a final significance determination.

Before we make a final decision on this matter, we are providing you an opportunity to: (1) present to the NRC your perspectives on the apparent violation and the facts and assumptions used by the NRC to arrive at the finding and its significance at a Regulatory Conference or (2) submit your position on the finding to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of your receipt of this letter and we encourage you to submit supporting documentation at least one week prior to the conference in an effort to make the conference more efficient and effective. If a Regulatory Conference is held, it will be open for public observation. The NRC will also issue a press release to announce the conference. If you decide to submit only a written response, such a submittal should be sent to the NRC within 30 days of the receipt of this letter.

Please contact Stephen Cahill at (404) 562-4520 within ten business days of the date of your receipt of this letter to notify the NRC of your intentions. If we have not heard from you within ten days, we will continue with our significance determination decision and you will be advised by separate correspondence of the results of our deliberations on this matter.

Since the NRC has not made a final determination in this matter, no Notice of Violation is being issued for these inspection findings at this time. In addition, please be advised that the number and characterization of the apparent violation described in the enclosed inspection report may change as a result of further NRC review.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Charles Casto, Director
Division of Reactor Projects

Docket No.: 50-390
License No.: NPF-90

Enclosure: SDP Phase III Summary

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RISK FROM LTOP AND SHUTDOWN LOCA'S
DURING WATER SOLID MODE AT WATTS BAR

The Probabilistic Safety Assessment Branch (SPSB) evaluated the risk significance of repeated challenges to the power operated relief valves (PORVs) at Watts Bar on February 23, 2005, when the licensee was in a water solid condition. This assessment estimated the delta increase in core damage frequency (CDF) resulting from the (1) increased likelihood of a low-temperature over pressure event, and (2) the increased likelihood of a shutdown loss-of-coolant accident (LOCA) resulting from relief valves sticking open.

The dominant core damage scenario for this event results from a failure of the PORV to open following a demand on the overpressure protection system (OPS) and the residual heat removal (RHR) suction relief valve failing to close after being challenged. Subsequent failure of both RHR isolation valves *or* failure of the operator to open a PORV to establish feed and bleed results in core damage. This event was evaluated as having a likelihood of 2E-5.

Additionally, the likelihood of having an interfacing systems LOCA (ISLOCA) in the RHR system from failure of the OPS to work and failure of the RHR suction relief valve was evaluated. Credit was given to the operators by using the RHR isolation valves to isolate a postulated rupture in the RHR system. This action reduced the CDF from ISLOCAs to 8E-7.

<u>Description of the Event as Relayed from the Resident Inspector (RI)</u>

On February 23, 2005, the inspectors identified a control room log entry which described the initiation of problem event report (PER) 77176 for cycling of the pressurizer PORVs. The repeated cycling occurred as a result of problems associated with (1) charging flow control valve (1-FCV-62-93) erratic control, and (2) implementation of a design change notice (DCN) to raise control air pressure on the actuator for 1-FCV-62-93. The DCN was implemented to eliminate the erratic control of the valve. The inspectors performed a review of the reactor coolant and charging system parameters for the period in question. The inspectors determined that the pressurizer PORV (setpoints adjusted for cold over-pressure conditions as required by Technical Specification 3.4.12) had actuated a total of seven times (2 single actuations and a group of five actuations) in an approximate 2-hour period. The inspectors determined that first single and group of five actuations were due to a failure to follow procedures regarding general operating instruction (GOI) -6, Unit Shutdown from Hot Standby to Cold Shutdown. To transition to solid water operations section 5.5, step [1] [e] states, 'Slowly RAISE charging to fill pressurizer (PZR) at less than 30 gpm.' Contrary to this, the licensee exceeded the 30 gpm requirement and experienced the first PORV actuation. The actuation occurred when 1-FCV-62-93 exhibited erratic operation following activities to swap from bypass to normal charging. The inspectors noted that while the DCN had been previously implemented while the plant was on bypass charging, all of the post maintenance testing had not yet been completed. Since 1-FCV-62-93 operation was still erratic, the licensee swapped back to bypass charging resulting in the group of five PORV actuations.

The inspectors also determined that, contrary to the original PER problem description, the last pressurizer PORV actuation was due to RCS heatup and resultant pressure increase from the closure of the 1A RHR heat exchanger outlet valve per system operating instruction (SOI) 74.01, section 8.11, 'Flush of a Train RHR Heat Exchanger Bypass during Shutdown Cooling.' The inspectors determined that this procedure was not adequately maintained in that the following action was contained in a procedure note, 'The effect on reactor coolant system (RCS) heatup/cooldown should be evaluated.' The performance of section 8.11 during solid plant operation resulted in sufficient RCS heatup to result in another actuation of the PORV.

Plant Mitigation Capability and Event Details

- Final safety evaluation report (FSAR) states (page 5.2-36) that one PORV is sufficient for pressure relief considering one charging pump is charging water into a water solid reactor coolant system at approximately 485 gpm with the letdown path isolated. Both PORVs were available; however, one block valve was shut, so only one PORV was credited with providing automatic over pressure protection.

- As specified in the FSAR (page 5.2-38), the power was locked out to all but one charging pump when RCS cold leg temperatures are below 350 degrees F. Based on information from the RI, the handswitch for the disabled charging pump and both safety injection (SI) pumps were taken to pull-to-lock. The disabled charging pump's discharge isolation valve and discharge bypass valve were locked and shut. Each SI pump discharge isolation valve was locked shut and tagged. Operations reported that it would take approximately 10 minutes to restore the disabled charging pump and both SI pumps if needed.

- The unit was shutdown at 0001 on February 22, 2005, and the first PORV actuation occurred on 1315 on February 22, 2005. The last PORV actuation occurred on 1508 on February 22, 2005.

- At 1449, on February 22, 2005, a 12-inch diameter containment penetration, a maintenance port, was opened. The penetration has a blind flange which is located outside containment. The penetration had emergency closure capability from outside containment (remove/cut all hose/cables passing through the penetration and install the outboard blind flange) and was required to be closed within 15 minutes of loss of shutdown cooling. Since (1) the penetration could be closed from outside containment, (2) operators would not have to be concerned with a degraded containment environment immediately following an extended loss of core cooling, and (3) the penetration was required to be closed within 15 minutes, SPSB assumed containment closure could be established. Therefore, risk from LERF was not evaluated except for ISLOCA scenarios.

- Refueling water storage tank had 368,000 gal of inventory at the start of the event.

- The RHR Suction Relief valve lifts at 450 psig and has a capacity of 900 gal per minute (gpm). The required flow rate is 480 gpm at 350 degrees F and 690 gpm at 200 degrees F which is the combined flow capacity of both charging pumps. Watts Bar RHR system is arranged with a single loop RHR suction (dual loop discharge), and therefore has only one relief valve on a 3-inch line connected to the 14-inch suction piping
- RHR Discharge Relief Valves: Setpoints are 600 psig and both valves are on a 2-inch line connected to 8-inch piping. Downstream of the relief valves are the pressure boundary check valves. The relief capacity of each valve is 20 gpm.
- The design pressure of the RHR piping is 600 psig.

- The operators had pressurizer level indication, pressurizer level low alarms, and the core exit thermocouples.

- Based on a time of 13-hours post shutdown (decay heat estimated as 24MW), the amount of inventory necessary to maintain boiloff was estimated as 174 gals per minute.

PROBABILISTIC ASSESSMENT

A cold overpressure (COP) event tree was developed for this event. The endstates are:

- OK - The RHR function not interrupted and reactor vessel integrity preserved.

- RCS BLOWDOWN-RHR-OK - This scenario results when the PORVs lift but fail to reseat. According to discussions with Reactor Systems, failure of a PORV to reseat will result in loss of RCS inventory until RCS pressure reaches atmospheric conditions. However, the inventory loss is not expected to result in loss of inventory from the RCS hot legs and a loss of RHR pump suction. The pressurizer is expected to remain above fifty percent full.

  Since the RCS pressurizer is expected to remain above fifty percent full and the core exit thermocouples will indicate that core cooling has been maintained, the operators are not expected to operate the charging pump in the safety injection mode, thereby eliminating the potential for increasing RCS pressure.

  Since this event does not lead to a loss or interruption of the RHR function, this scenario does not lead to a shutdown initiating event and was not analyzed further.

- BLOW-DOWN-LOCA - This scenario results from the RHR suction relief valve failing to reseat after a challenge following the PORV failing to open after a challenge. Leak path termination requires closure of one of two RHR isolation valves which causes a loss of the RHR function. Failure of leak path termination results in a loss of RCS inventory that leads to a loss of RHR pump suction. This scenario is evaluated using a Watts Bar LOCA event tree to obtain a conditional core damage frequency (delta CDF).

‘ ISLOCA - Failure of the PORV and the RHR suction relief valve is assumed to result in failure of the RHR system (starting with failure of the RHR pump seals) once RCS pressure exceeds 1500 -1800 psig. Isolation of the break requires closure of the RHR isolation valves.

Quantification of Cold Over Pressure Event Tree

Initiating Event - Cold Over Pressure Challenge - The initiating event, RCS pressure increase that challenges the OPS, was quantified as a frequency. As reported by the RI, there were seven challenges to the OPS system.

OPS@372 PSIG - Both PORVs were available for cold over pressure protection. However, one PORV block valve was closed, so only one PORV was credited for automatic cold over pressure protection. Failure for a PORV to open on demand was estimated as 6E-3 based on the SPAR model for Watts Bar, Revision 3i.

OPS RESEATS - Failure of the PORV to reseat following a demand after passing water as .1 based on the SPAR model for Watts Bar, Revision 3i.

RHR-SUCTION-RV-LIFTS - Failure of the RHR suction relief valve to lift was estimated using the same failure rate of a single RCS SRV failing to open on demand (1E-3/demand) based on the SPAR model for Watts Bar, Revision 3i.
.
RHR-SUCTION-RV-CLOSES - Failure of the RHR suction relief valve to close following a demand was estimated using the failure rate of a PORV failing to reseat after a challenge since the RHR suction relief valve is designed to pass water. This failure of 3E-2 was based on the SPAR model for Watts Bar, Revision 3i.

OP STOPS PUMPs or OPENS PORV - Failure of the operator to terminate the pressure excursion by: (1) stopping charging flow, or (2) opening the alternate PORV and associated block valve following failure of the RHR suction relief valve to lift was estimated as 1.0. Based on discussions with the Division of Engineering and the Reactor Systems Branch, once a pressure excursion has been initiated, the operator will not have enough time to respond before the RHR system failure is expected to occur (around 1500 to 1800 psig - 2.5 times the design pressure). Failure of the RHR system was believed to occur before failure of the reactor vessel based on preliminary materials information from the Division of Engineering.

Using the top event values discussed earlier, the likelihood of scenarios 4 and 6 given the event are:

Scenario 4: The likelihood of having a LOCA through the open RHR suction relief valve was estimated as 1E-3. This scenario is further evaluated in the shutdown SDP phase 2 event tree.

Scenario 6: The likelihood of having an interfacing systems LOCA resulting from rupture of the RHR piping was assessed as 4E-5.

Scenarios 4 and 6 were then analyzed using the Watts Bar LOCA event tree. Each scenario was analyzed separately because the event tree top event probabilities were found to be different for each scenario.

Quantification of COP Scenario 4 Using the Loss of Inventory PWR Event Trees POS 2

LOI - The likelihood of a loss of inventory from a stuck open RHR suction relief valve, COP scenario 4, was estimated as 1E-3.

RCS injection before core damage-based on the licensee's mitigation capability, both charging pumps and both safety injection pumps were credited as being able to keep the reactor core covered. Assuming a multi-train failure rate of 1E-3 for charging and safety injection, the failure probability of RCS injection is driven by operator error rather than equipment failure. The probability of operators failing to inject via available sources following a loss of the operating train of residual heat removal system and prior to core damage is on the order of 1×10^{-4}. (Inspection Manual Chapter Appendix G Phase 2 Worksheet 2, "SDP for a Westinghouse 4-Loop Plant - Loss of Level Control in POS 2"). Thus, the failure probability of RCS injection before core damage is estimated as 1E-4.

Isolate RHR and Open PORV - If the RHR suction relief valve were to stick open, it was assumed that the operators would attempt to (1) close the RHR isolation valves, and (2) open the alternate PORV to initiate feed and bleed. Conservatively assuming the design flow rate of 900 gpm, the operators have more than 1 hour to close the valve and initiate RCS injection and bleed through a PORV. (The remaining RWST inventory would last over 17 hours assuming a charging flow rate of 300 gpm.)

To complete the recovery, the operators must close one of two RHR isolation valves *and* open the alternate PORV and its block valve. (It was assumed that the PORV that responded to the initial challenge is failed). Thus, failure for this top event is both RHR isolation valves failing to close *or* the alternate PORV *or* its block valve failing to open. This failure likelihood was estimated as;

> (3E-3 failure for RHR isolation valve to close - Watts Bar SPAR model, revision 3i)*(.1beta factor for both RHR isolation valves failing to close due to common cause) + (PORV block valve failing top open (3E-3 Watts BAR SPAR model, revision 3i) + (PORV failing to open on demand 6E-3) = 9 E-3

The likelihood of the operator failing to close the RHR isolation valves and open a PORV was estimated using HRA Worksheets for LP&SD contained in the SPAR-H methodology page B-3. To simplify this analysis, the diagnosis probability defines the operator recovery. The inferred definition of diagnosis is any cognitive decision making that is necessary to perform a task.

The performance shaping factors (PSFs) for this operator recovery were assumed to be: expansive time, extreme stress, moderately complex diagnoses, and nominal procedures. It was assumed that shutdown loss of inventory procedures would direct the operators to search for the source of a leak, and procedures exist for using RCS injection and RCS bleed through a PORV. All other PSFs were assumed to be nominal. Using the SPAR-H methodology, the operator failing to isolate the RHR suction valves and initiate RCS bleed was assumed to be 1E-3. Thus, failure of this top event is assumed to be driven by equipment error and was estimated as 9E-3 +1E-3 = 1E-2. Failure of this top event is assumed to lead to core damage since it is dominated by failure of alternate PORV and its block valve failing to open.

RHR Recovery before RWST Depletion - To isolate the stuck open RHR relief valve, the RHR system must be isolated. If the RHR system was isolated, the relief valve may reseat once pressure in the RHR system is reduced significantly below the lift setpoint of 450 psi. If the valve were to reseat, the RHR isolation valves could be opened, and restoration of a train of RHR could begin. Failure of the valve not to reseat after RHR system pressure was reduced was given a value of screening value of .5 due to lack of data. Failure of RHR recovery before RWST depletion is assumed to be driven by failure of the stuck open relief valve to reseat.

RWST Makeup before Core Damage - One train of RWST makeup was assumed to be available and low a RWST inventory alarm. It was assumed that the RWST makeup rate could keep up with the RCS boiloff rate 17 hours into the event after the RHR system was isolated. The failure of RWST makeup before core damage has been assigned the nominal value used in the Shutdown SDP worksheets of 1E-2.

Referring to the Watts Bar LOCA event tree, three core damage scenarios were quantified, sequences 3, 4, and 5. Using the top event values discussed above, the likelihood of core damage from COP scenario 4 is:

1E-3 ((.5)*(1E-2) + (1E-2) + (1E-4)) = 1.5E-5

Quantification of COP Scenario 6 Using the Loss of Inventory PWR Event Trees POS 2

LOI - The likelihood of a loss of inventory from a interfacing systems LOCA, COP scenario 6, was estimated as 4E-5.

RCS Injection before Core Damage - The same failure likelihood from COP Scenario 2 was except modified for extreme stress using the SPAR-H methodology 5*(1E-4) = 5E-4.

Leak Path Terminated before RWST Depletion - If the RHR system had a rupture, it was assumed that the operators would attempt to close the RHR isolation valves to try to isolate the leak and open a PORV to initiate feed and bleed. Based on quantification of this top event in COP scenario 4, this event is driven by equipment failure rather than operator error. This top event was quantified in scenario 4 as 1E-2. Failure of this top event by failing to isolate the RHR system or failing to establish a RCS bleed path is assumed to lead to core damage.

RHR Recovery before RWST Depletion - Since there is a rupture of the RHR system, no credit is given for RHR recovery.

RWST Makeup before Core Damage - One train of RWST makeup was assumed to be available and low a RWST inventory alarm. It was assumed that the RWST makeup rate could keep up with the RCS boiloff rate 18 hours into the event after the RHR system was isolated. The failure of RWST makeup before core damage has been assigned the nominal value used in the Shutdown SDP worksheets of 1E-2.

Referring to the Watts Bar LOCA event tree, three core damage scenarios were quantified, sequences 3, 4, and 5. Using the top event values discussed above, the likelihood of core damage from COP scenario 6 is:

4E-5 ((1.0)*(1E-2) + (1E-2) +(5E-4)) = 8E-7

<u>CONCLUSION</u>

The likelihood of having an interfacing systems LOCA (ISLOCA) in the RHR system from failure of the OPS to function and failure of the RHR suction relief valve was evaluated. Credit was given to the operators by using the RHR isolation valves to isolate a postulated rupture in the RHR system. This action reduced the CDF from ISLOCAs to 8E-7.

The dominant core damage scenario for this event results from a failure of the PORV to open following a demand on the OPS and the RHR suction relief valve failing to close after being challenged. Subsequent failure of both RHR isolation valves *or* failure of the operator to open a PORV to establish feed and bleed results in core damage.

The overall calculated CDF from repeated challenges to the PORVs during solid water operations was 2E-5/year.