March 29, 2004

Mr. J. A. Scalice
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 1 — ISSUANCE OF AN AMENDMENT TO REVISE THE UPDATED FINAL SAFETY ANALYSIS REPORT FAILURE MODES AND EFFECTS ANALYSIS - USE OF OPERATOR ACTION (TAC NO. MB8102)

Dear Mr. Scalice:

The Commission has issued the enclosed Amendment No. 51 to Facility Operating License No. NPF-90 for Watts Bar Nuclear Plant, Unit 1. The amendment approves changes to the Updated Final Safety Analysis Report to revise the design and licensing basis failure modes and effects analysis for specific valves in the essential raw cooling water system, component cooling water system, and control air system to include the use of operator actions in certain postulated events. This amendment is in response to your application dated March 24, 2003, as supplemented on December 4, 2003, and February 12, 2004.

A copy of the safety evaluation is also enclosed. Notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/
Margaret H. Chernoff, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-390

Enclosures: 1. Amendment No. 51 to NPF-90
2. Safety Evaluation

cc w/enclosures: See next page

March 29, 2004

Mr. J. A. Scalice
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 1 — ISSUANCE OF AN AMENDMENT TO REVISE THE UPDATED FINAL SAFETY ANALYSIS REPORT FAILURE MODES AND EFFECTS ANALYSIS - USE OF OPERATOR ACTION (TAC NO. MB8102)

Dear Mr. Scalice:

The Commission has issued the enclosed Amendment No. 51 to Facility Operating License No. NPF-90 for Watts Bar Nuclear Plant, Unit 1. The amendment approves changes to the Updated Final Safety Analysis Report to revise the design and licensing basis failure modes and effects analysis for specific valves in the essential raw cooling water system, component cooling water system, and control air system to include the use of operator actions in certain postulated events. This amendment is in response to your application dated March 24, 2003, as supplemented on December 4, 2003, and February 12, 2004.

A copy of the safety evaluation is also enclosed. Notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Margaret H. Chernoff, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-390

Enclosures: 1. Amendment No. 51 to NPF-90
2. Safety Evaluation

cc w/enclosures: See next page

Distribution:

PUBLIC	PDII-2 R/F	EHackett	WBurton	MChernoff
BClayton (Hard Copy)	OGC	ACRS	GHill (2 copies)	SDinsmore
MRubin	JBongarra	DTrimble	AStubbs	DSolorio
SCahill, RII				

ADAMS Accession No. ML040900172

OFFICE	PDII-2/PM	PDII-2/LA	SPLB	SPSB	IROB	SRXB	OGC	PDII-2/SC(A)
NAME	MChernoff	BClayton	SE Input	SE Input	SE Input	JUhle	HMcGurren	WBurton
DATE	3/17/04	3/17/04	2/13/04	3/5/2004	2/17/04	3/18/04	3/25/04	03/29/04

OFFICIAL RECORD COPY

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-390

WATTS BAR NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 51
License No. NPF-90

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (TVA or the licensee) dated March 24, 2003, as supplemented by letters dated December 4, 2003, and February 12, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by Amendment No. 51, Facility Operating License No. NPF-90 is hereby amended to approve changes to the Watts Bar Unit 1 Updated Final Safety Analysis Report (UFSAR) Sections 6.2.4 and Tables 9.2-2, 9.2-9, and 9.3-7. These changes reflect the revised failure modes and effects analysis for certain valves in the essential raw cooling water system, component cooling water system, and control air system, as set forth in the application for amendment by TVA dated March 24, 2003, as supplemented by letters dated December 4, 2003 and February 12, 2004.

3. This license amendment is effective as of the date of its issuance. Implementation of the amendment is the incorporation into the next UFSAR update made in accordance with 10 CFR 50.71(e), of the changes to the description of the facility as described in TVA's application dated March 24, 2003, as supplemented by letters dated December 4, 2003, and February 12, 2004, and evaluated in the staff's Safety Evaluation attached to this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

William F. Burton, Acting Chief, Section 2
Project Directorate II
Division of Project Licensing Management
Office of Nuclear Reactor Regulation

Date of Issuance: March 29, 2004

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 51 TO FACILITY OPERATING LICENSE NO. NPF-90

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-390

1.0 INTRODUCTION

By letter dated March 24, 2003, as supplemented on December 4, 2003, and February 12, 2004, (ADAMS Accession Nos. ML030860487, ML033460393, and ML040500646, respectively), the Tennessee Valley Authority (the licensee) submitted a request for approval of changes to the Updated Final Safety Analysis Report (UFSAR) for Watts Bar (WBN), Unit 1. The changes would revise the design and licensing basis failure modes and effects analysis for specific valves in the essential raw cooling water system, component cooling water system, and control air system to include the use of operator actions in certain postulated events.

The supplemental letters provided clarifying information that did not expand the scope of the original amendment request and did not change the initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

The WBN UFSAR currently indicates that containment integrity is maintained automatically subsequent to an accident with a single failure of an outboard containment isolation valve by the containment design and by the containment isolation system. In the March 24, 2003, letter, the licensee stated that a condition had been identified in which containment integrity, accident flood levels, and containment sump boron concentrations subsequent to a high energy line break (HELB) could not be assured automatically as stated in the UFSAR. In certain postulated events, manual actions may be required using equipment not currently evaluated in the UFSAR. The licensee proposed changes to the UFSAR to revise the discussion of containment isolation and to update the failure modes and effects analysis for specific valves affected by this condition.

The regulatory requirements directly applicable to the containment design are Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, General Design Criterion (GDC) 16, “Containment design,” GDC 50, “Containment design bases,” and GDC 54, “Piping systems penetrating containment.”

GDC 16 requires that reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

GDC 50 requires, in part, that the reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss of coolant accident.

GDC 54 requires, in part, that piping systems penetrating primary reactor containment shall be provided with leak detection, isolation and containment having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems.

The regulatory guidance applicable to the acceptability of accrediting manual actions is contained in the following documents:

1. NUREG-800, “Standard Review Plan” (draft for comment, 2003);

2. NUREG-0711, Rev.1, “Human Factors Engineering Program Review Model” (2002);

3. NUREG-0700, Rev.2, “Human-System Interface Design Review Guideline” (2002);

4. Information Notice (IN) 97-78, “Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times” (1997);

5. NUREG-1764, “Guidance for the Review of Human Actions, Draft Report for Comment” (2002);

6. Regulatory Guide (RG) 1.174, “An Approach To Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes To The Licensing Basis” (1998);

7. RG 1.177, “An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications” (1998);

8. Information Notice (IN) 91-18, “Information to Licensees Regarding Two Inspection Manual Sections On Resolution of Degraded and Non-Conforming Conditions and on Operability” (1991); and

9. ANSI/ANS Standard 58.8 (1994), “Time Response Design Criteria for Safety-Related Operator Actions.”

The licensee stated that this amendment request was not considered a risk-informed initiative per RG 1.174; however quantitative risk insights were provided by the licensee. The staff used NUREG-0800, “Standard Review Plan,” Chapter 19, Appendix D, for guidance in the review of the risk information provided by the licensee.

3.0 TECHNICAL EVALUATION

The licensee determined that manual action may be required to ensure containment integrity and isolation of one nonsafety-related air line, and three safety-related water lines penetrating containment, in the event of an HELB concurrent with a single failure of an outboard containment isolation valve. These lines, which include a compressed air system (CAS) supply line, an essential raw cooling water (ERCW) supply line, and component cooling water system (CCS) supply and return lines, all have a common configuration for piping penetrating containment. The containment penetration consists of an automatic motor-operated isolation valve outside containment and a check valve inside containment. The licensee determined that these specific lines are not adequately protected inside containment from pipe whip or jet impingement. As a result, an HELB inside containment, concurrent with a single failure of the outboard containment isolation valve to close, could result in continued flow through the affected line into the containment. If unmitigated, inflow of water from the water systems could result in containment flood levels being exceeded or dilution of post-accident sump boron concentrations; and the unmitigated inflow of compressed air could result in over-pressurization of the containment. The licensee proposes to isolate these lines and, thus, mitigate the accident through the use of operator action to achieve manual isolation of the lines.

The NRC staff review focused on three major aspects of the proposed changes. These are: (1) impact on post-accident conditions, including containment isolation, containment pressure, containment flood levels, and containment sump boron concentration, (2) acceptability of the proposed manual actions, and (3) an evaluation of the risk impact and potential risk implications. The staff's review of each of these areas is summarized in the following paragraphs.

3.1 Post-Accident Conditions

3.1.1 Containment Isolation

The licensee, in its letter dated March 24, 2003, proposes that operators isolate the CAS, CCS, and ERCW lines penetrating containment, in the event of the failure of an outboard containment isolation valve, by closing the following valves, as applicable: nonsafety related CAS valve 0-ISV-32-1013, or other up stream manual valves; CCS safety-related valves 1-ISV-70-516 (supply) and 1-ISV-70-700 (return); and ERCW safety-related valve 1-ISV-67-523B. The staff has reviewed the configuration of the CAS, CCS, and ERCW piping penetrating containment provided by the licensee. Since the valves that are to be relied upon for alternative isolation are in series with the isolation valves that are assumed failed, and are within scope of the licensee's maintenance rule program, the staff agrees with the licensee that these valves may be used for isolation of the piping systems in the event of a failure of the outboard containment valves. Therefore, GDC 54 continues to be satisfied.

3.1.2 Containment Pressure

In the event of an HELB inside containment with a consequential break of a CAS, ERCW, or CCS line concurrent with a single failure of an associated outboard containment isolation valve, the line break that will have the greatest impact on containment pressure is the CAS line break. The CAS line break results in the potential release of 1446 standard cubic feet per minute (scfm) of control air to the containment, as opposed to 36 gallons per minute (gpm) of water for the

CCS system, and 38 gpm of water for the ERCW. The control air consists of primarily noncondensible gases and, therefore, will act to increase the pressure inside the containment. Since the ice condenser and containment spray reduce pressure by condensing water vapor, they will have little affect on reducing the pressure increase caused by the CAS line break. The inflow of water into the containment through ERCW and CCS line breaks will not add sufficient energy to the containment to result in a significant increase in containment pressure. The licensee is proposing to use manual actions in lieu of automatic actions to isolate the CAS line in the event of a CAS line HELB concurrent with a failure of the associated outboard containment isolation valve. The licensee has determined that the CAS line can be isolated within 90 minutes of the accident, and has performed a new containment pressurization analysis based on the large break loss-of-coolant accident (LOCA) analysis in UFSAR Section 6.2, concurrent with CAS line break isolated at 90 minutes into the accident. The licensee has indicated that Westinghouse has performed a study to evaluate the effect on the containment of 1446 scfm air leakage for 90 minutes. To perform the evaluation, Westinghouse used the current UFSAR containment integrity analysis with the following modifications:

- Added an air leak into the containment corresponding to a guillotine break of the main CAS line inside containment; and,
- Credited an additional train of containment spray.

The licensee states that in the accident scenario being evaluated, since the single failure is the failure of the control air outboard isolation valve, both trains of the containment spray system are available. With the exception of the two modifications identified above the analysis is consistent with the UFASR analysis. The containment pressure response to line breaks inside the containment is given in UFSAR Section 6.2, “Containment Systems.” The peak containment pressure is 10.64 psig and is the result of a large break LOCA. The peak pressure for the new analysis discussed above is 10.24 psig. The lower peak pressure is a result of credit being taken for both trains of the containment spray system. The staff has reviewed the containment pressure evaluation provided by the licensee. Based on the fact that calculated peak pressure is less than the design pressure, the staff finds the proposed 90-minute isolation of the CAS line and resulting peak pressure acceptable. Therefore, the staff has concluded that GDC 16 and 50 continue to be satisfied.

3.1.3 Containment Flood Level

In the event of an HELB inside containment with a consequential break of a CAS, ERCW, or CCS line concurrent with a single failure of an associated outboard containment isolation valve, the line breaks that will have the greatest impact on containment flood level are the ERCW and CCS line breaks. The ERCW and CCS line breaks both result in the discharge of water into the containment. The ERCW line break discharges water into the containment at a rate of 36 gpm, and the CCS line break discharges water into the containment at a rate of 38 gpm. The licensee has revised its maximum containment water level calculation to include an evaluation of the effect of potential LOCA-induced ERCW and CCS line breaks on the containment flooding analysis. Based on the new analysis it was determined that the minimum timeframe for exceeding the maximum flood level is 16 hours. The staff has reviewed the maximum containment water level calculation provided by the licensee. Based on our review the staff concurs that the water level in the containment will not exceed flood levels before 16 hours postaccident.

Since flow will be allowed to exist for up to 16 hours after the onset of the accident, the staff requested that the licensee verify that failure to isolate these lines will not result in loss of water supply to structures, systems, and components that require ERCW or CCS supply to operate. In its December 4, 2003, letter, the licensee states that the ERCW is supplied from the Tennessee River, and that the supply header for the ERCW system can supply up to 7,000 gpm. The licensee concluded, and the staff concurs that the loss of approximately 40 gpm through the ERCW line break would have a negligible impact on the overall flow distribution through the ERCW system piping. The licensee also states that the required flow through the CCS piping is approximately 5100 gpm, and a loss of 40 gpm through this line will not have a significant impact on the flow distribution in the header. Because the CCS system has two trains, should a loss of inventory occur in the train with the break, the redundant train will be available to supply the structures, systems and components required to be in service during the postaccident period. Based on the above review, the staff finds that the proposed operator isolation of the CCS and ERCW line within 16 hours of the event acceptable.

3.1.4 Containment Sump Boron Concentration

In the event of an HELB inside containment with a consequential break of a CAS, ERCW, or CCS line concurrent with a single failure of an associated outboard containment isolation valve, the line breaks that will have the greatest impact on containment sump boron concentration are the ERCW and CCS line breaks. The licensee revised the post-LOCA sump boron concentration calculations to include the effects of an additional 40 gpm inflow. The NRC staff's review and conclusions regarding the post LOCA sump boron concentration is documented in the staff's Safety Evaluation dated October 8, 2003 (Watts Bar License Amendment 48, ADAMS Accession No. ML032890316).

3.2 Acceptability of Manual Actions

In its technical analysis the licensee indicated that crediting manual actions is necessary only if specific outboard containment isolation valves fail to close concurrent with an HELB inside containment. The outboard containment isolation valves for the affected lines are: 1-FCV-32-110 (CAS); 1-FCV-67-107 (ERCW); 1-FCV-70-92 (CCS return line); and 1-FCV-70-140 (CCS supply line).

In the case of a CAS line break, the licensee's analysis concluded that the accident peak containment pressure is not exceeded if the control air line break is manually isolated within the first 90 minutes of the accident (a large break LOCA, analyzed in the licensee's current UFSAR, bounds the CAS line break inside containment). For the ERCW and CCS line breaks, the licensee's analysis concluded that the ERCW or CCS line must be isolated within 16 hours subsequent to an HELB inside containment and a concurrent consequential rupture of the lines and a single failure of the associated outboard containment isolation valve.

In evaluating the acceptability of crediting proposed manual actions to mitigate the postulated accidents, the licensee indicated that it used guidance contained in NRC Information Notice 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times," and ANSI/ANS 58.8, "Time Response Design Criteria for Safety-Related Operator Actions." The licensee indicated that it performed Mission Dose Calculations for each of the four valves that required manual operation to determine if the operator actions could be accomplished within the time required to maintain containment integrity. These missions were walked down to validate the assumed response times. Four separate missions were initiated from the Main Control Room and the same four were also initiated from the Operations Support Center. Each mission assumed the operator to be dressed in anticontamination clothing and self-contained breathing apparatus (SCBA) and traveling at a slow pace (1.5 feet/second). Each mission required one operator and one radiological control (RADCON) technician (this staffing is within the licensee's Technical Specification requirements).

For the CAS air line, manual isolation must occur within 90 minutes of an accident concurrent with the single failure of the outboard containment isolation valve to close. A safety-related containment pressure monitor will indicate to the operator if the containment pressure is still rising, thus, identifying if the closure of the valve (0-ISV-32-1013) was successful. The valve is located in the general area of the Auxiliary Building on elevation 713. The CAS valve chain operator is located in an area where dose rates are not a large factor according to the licensee. Consequently, a time delay to access the area is not an issue for isolating the CAS valve. The total dose received during the mission to isolate the CAS valve, including ingress and egress, is approximately 1 Rem.

The licensee indicated that if the CAS is not adequately isolated by closing one of the manual isolation valves, the operators may consider stopping the air compressors by removing breakers 0-BKR-32-25-A, 0-BKR-32-26-B, 0-BKR-32-27, and 0-BKR-32-4900A within 90 minutes of the accident. The breakers are located in the Control Building and Turbine Building. Transit time from the Main Control Room to the CAS valve is 4.9 minutes, as stated by the licensee, and 5 minutes is used in the licensee's analysis to close the valve. In its December 4, 2003, submittal, the licensee explains that if the CAS valve is not isolated until 30 minutes after the accident, approximately 60 minutes are available to dispatch a team to remove the air compressor breakers. Because of the location of the breakers, maximum calculated dose rate, and the allowable dose per mission, two teams of operators may be dispatched if needed. The mission dose calculation indicated that 55.9 minutes would be needed to remove the air compressor breakers using one team. Two teams could be dispatched simultaneously, with one team removing 0-BKR-32-25-A, 0-BKR-32-26-B, 0-BKR-32-27 in 39.4 minutes and the second team removing breaker 0-BKR-32-4900A in 20.7 minutes. Using two teams would also allow more time to monitor the containment pressure.

The licensee also stated that if the CAS line is not isolated within the 90 minute timeframe, containment will not actually overpressurize as the 90 minute closure time for the air valve was selected based on a reasonable time for operators to leave the control room and close the valve and was not based on the time for over-pressurization of the containment. The licensee explains the basis for the overpressurization time limits in its December 4, 2003, submittal.

Manual isolation of the 6-inch ERCW supply to the lower containment cooler Group D from the main supply Header 1B must be completed within 16 hours after the postulated accident and can be accomplished by closing the safety-related valve 1-ISV-67-523B, which is located in the Auxiliary Building at elevation 692. The containment sump level monitor can be used to determine if the valve closure was successful. In its March 24, 2003, submittal, the licensee indicated that personnel may not be able to enter the room until the dose rate is less than or equal to 40.4 Rem/hr assuming maximum possible dose rates in all areas traveled during the mission. Consequently, the room may not be accessible for 10 hours to permit manual operation of the valve. The licensee indicated, however, that these limitations are conservative and based on maximum calculated dose rates in all areas using design basis assumptions. Actual mission staffing, protective equipment, and times are based on actual plant dose rates as determined by site RADCON personnel. According to the licensee, the valve is routinely manipulated for the leak-rate testing of the containment isolation valve. The licensee also indicated that scaffolding has been erected to allow easy access to the valve and that the scaffolding is under administrative control for as long as the condition for manual closure of the valve exists, thus, reducing the time and dose to manually close the valve. The licensee further stated that task accomplishment, including ingress and egress in the penetration room is approximately 6 minutes and that the mission can be accomplished within the allowable 16 hours and 5 Rem dose limit.

Isolation of the 6-inch CCS supply and return lines for the reactor coolant pump oil cooler penetrating containment must be accomplished within 16 hours of the accident, concurrent with the single failure of the outboard containment isolation valve to close according to the licensee's analysis. Isolation of the piping lines entering containment is accomplished by locally closing safety-related valves 1-ISV-70-516 (supply) and 1-ISV-70-700 (return), as applicable. The CCS return line valve (1-ISV-70-700), which is located in the Auxiliary Building at elevation 692, Penetration Room, must be isolated since back flow into the containment can occur with a line break inside containment concurrent with a single failure of the outboard containment isolation valve to close. The maximum calculated accident dose rate in the room may not permit manual operation of the valve until 12 hours after the accident or until the general area dose in the room is less than or equal to 35.6 Rem/hr. Manual operation of the valve can be accomplished in 7 minutes, according to the licensee's analysis, including ingress and egress. Monitoring the CCS surge line tank level and makeup can ensure that the valve was successfully closed. In its December 4, 2003, submittal, the licensee indicated that it did not specify time cautions for closing the CCS supply valve (1-ISV-70-516) because the accident dose rate at the location of this valve is not as high as that for the return valve.

For each of the proposed manual actions, the licensee indicated that the dose calculations included dose incurred in transit to and from the Main Control Room or the Operations Support Center, dose incurred in performing the manual actions, and dose resulting from ingress and egress to and from the valve or breaker locations. Dose calculations were performed under worst-case conditions which required operators to wear SCBA with a protection factor of 400. The licensee stated that the required manual actions can be successfully performed within 10 CFR Part 50, Appendix A, GDC 19 limits. Indications for ERCW, CCS, and CAS lines which fail to isolate flow into containment and specific guidance for isolating the lines have been included in the licensee's emergency operating procedures. In its December 4, 2003, submittal, the licensee stated that walk-downs have been independently performed by both the preparer of the mission dose calculations and a member of the Operations organization, knowledgeable in the use of emergency operating procedures, to determine that the actions can be accomplished within the required times. In its March 24, 2003, submittal, the licensee indicated that the applicable emergency operating procedure for Reactor Trip and Safety Injection has been revised to include the changes needed to address the required manual actions. The training lesson plan for these revisions was revised and included in the licensee last Operator Re-qualification Training for licensed operators. The licensee further indicated that operators are provided with status light indications, located in the Main Control Room which identify that the outboard containment isolation valves have failed to close. The instrumentation used, and detailed in the licensee's March 24, 2003, submittal, is safety-related. In addition, for each of the manual actions proposed, the licensee stated that the operator assigned to close the valve in an unisolated line will be dedicated to that task.

3.3 Evaluation of Risk Impacts

In its March 24, 2003, submittal, the licensee briefly describes the HELB initiating events, the initiating event frequencies, and the general scenario leading from the HELB to the failure to isolate containment. The licensee estimated that the frequency of occurrence of the HELB, consequential failure of the ERCW, CSS, or CAS piping, and subsequent failure to isolate containment as 4E-7/year. Details of the particular model and the failure probability estimates for the events modeled in the evaluation were not provided.

NUREG/CR-5750, "Rates of Initiating Events at U. S. Nuclear Power Plants: 1987 - 1995," provides generic estimates of the following pressurized water reactor pipe failure events. The estimates provided by the licensee are similar but slightly lower than the following estimates.

Large Pipe Break LOCA	5E-6/year
Medium Pipe Break LOCA	4E-5/year
Small Pipe Break LOCA	5E-4/year
Steam Line Break Inside Containment	1E-3/year
Feedwater Line Break	3E-3/year

NUREG-1715, Volume 4, "Component Performance Study - Motor-Operated Valves, 1987 - 1998, Commercial Power Reactors," provides a generic estimate for a motor operated valve failing to close of 5E-3 per demand. NUREG-1715, Volume 4," Component Performance Study - Air-Operated Valves, 1987 - 1998," provides a generic estimate for an air-operated valve failing to close of 5E-3 per demand.

An HELB does not inevitably cause a failure of the ERCW, CSS, or CAS piping. The design basis evaluation leading to the amendment request determined that, following an HELB in some specified locations, a consequential failure of this piping could not be excluded. A reasonable estimate of the likelihood of consequential failure would require a substantial effort and include walking down the potentially affected piping. The licensee provided no evaluation of the potential for consequential failure.

Four operator actions have been identified and will be proceduralized, one each for the CCS and ERCW systems, and two alternative actions for the CAS system. Some of the actions are relatively complex, including some actions in high radiation areas, requiring a breathing apparatus. Such actions are only feasible because of the relatively long time intervals available to complete the action, ranging from 90 minutes in the worst-case (double ended guillotine LOCA with consequential rupture of the CAS line) to more than 16 hours.

Estimating the probability of the failure of the human actions required to isolate the lines is also a complex task requiring detailed information of the instrument indications available to guide the operators, the specific actions, and the times available to perform the task. The licensee used an operator action estimate already defined in the probabilistic risk assessment (PRA). The estimate defined in the PRA was developed as the probability of failure to manually close an automatic isolation valve following the failure of the automatic Engineered Safety Features

Actuation System. The licensee stated that the additional local, manual actions required in these scenarios are a continuation of the first action because the operators have recognized that the containment isolation has failed and will continue to try to isolate containment. This assumption is used in human reliability analysis, but is not applicable to all sequences of human actions and it is not clear that this assumption is applicable to all these actions based on the complexity of some of the required actions.

CCS and ERCW Systems

The diversion of water flow through the ruptured and unisolated ERCW and CCS system lines is not great enough to directly cause the loss of other structures, systems and components cooled by the system. Therefore, the only risk implications of performing the human actions in lieu of automatic actions are associated with failing to isolate the flow before the excess water leads to damage inside or out of the containment.

As discussed above, the expected frequency of all HELB breaks is about 5E-3/year. This frequency coupled with the probability of failure of the automatic valve to close (5E-3) yields a frequency estimate of about 2.5E-5/year. This frequency estimate does not include (1) the probability that the HELB ruptures one of the at-risk pipes and (2) the probability that the operator fails to isolate the line within 16 or more hours. Furthermore, the 40-gpm loss of water through the bypass lines does not affect the proper operation of other mitigating systems and all the equipment nominally available to mitigate these scenarios would be available in these scenarios. The staff concludes that the frequency of core damage and large early release would be much less than 1E-5/year and less than 1E-6/year, respectively.

CAS System

Unlike the diversion of water in the ERCW and CCS systems, the diversion of the air flow in the CAS system is great enough to directly cause the failure of structures, systems and components outside containment that require the pressurized air to operate. The CAS system is a nonsafety-related system that provides air to nonsafety-related structures, systems and components. Some of these structures, systems and components can, however, be used to mitigate accident scenarios. The licensee stated in its December 4, 2003, submittal that the risk achievement worth of the failure of the CAS system to function in the PRA is 4.3.

The licensee reported that the most time-limited human action is isolation of the CAS system following a nonisolable large break LOCA. In this scenario, the LOCA is contributing to the containment pressure and the pressurized air from the CAS system is contributing to the containment pressure. The operator must isolate the CAS system within 90 minutes to prevent reaching the containment design pressure. Without a detailed evaluation, the staff does not find it reasonable to assume a high probability (essentially 1.0 with the licensee's assumptions) that the human actions will be completed within the 90 minutes. A conservative frequency of requiring the human actions to be completed within 90 minutes can be estimated as the frequency of a large LOCA (4.5E-5/year) and the probability that the automatic valve fails to close (5E-3) or 2.25E-7/year. This frequency does not include the probability that the HELB ruptures the CAS lines or that the operators fail to close the manual valve which would further lower this frequency. Therefore, even with a high probability of the operators failing to isolate the line, this scenario represents a very low risk.

The expected frequency of all HELB breaks is about 5E-3/year. This frequency, coupled with the probability of failure of the automatic valve to close (5E-3), yields a frequency estimate of about 2.5E-5/year. This frequency estimate does not include (1) the probability that the HELB ruptures one of the at-risk pipes and (2) the probability that the operator fails to isolate the line within 90 minutes. Although some nonsafety-related equipment will be lost until the CAS line is isolated, safety-related structures, systems and components will remain available to mitigate the transient and, if there is no large break LOCA, containment pressure will rise more slowly so the operators will have more than 90 minutes to prevent the containment pressure from reaching the design pressure. Furthermore, only containment pressure is being increased and there will be no flooding nor boron dilution to complicate mitigation of these scenarios. Isolation of the air line will immediately stop the pressure rise and pressure may be relieved by providing a controlled release. The check valve in the CAS line preventing flow from inside to outside containment will also prevent (with a high likelihood) a core damage event from becoming a consequential large early release event if the air flow is stopped prior to overpressure failure of the containment.

The proposed change will amend the design and licensing basis to identify specific operator actions that will be necessary to ensure isolation of four lines penetrating containment and containment integrity. This change will be made in lieu of installing pipe jet and pipe whip protective devices for the at-risk piping, and in lieu of installing additional automatic closing isolation valves. Implementation of the proposed human actions results in the highest risk estimate among the three alternatives. Installing pipe whip and pipe jet protective devices for the at-risk piping would preclude the consequential rupture of the system piping. Installing additional automatic closing isolation valves results in a higher risk than installing protective devices because an active component (the valve) would need to successfully function instead of the passive protection devices. Authorizing the proposed human actions in lieu of automatic valve actions further increases the risk (compared to installing protective devices) because the probability of the operator failing to perform the relatively complex tasks is generally greater than the probability a valve would fail to close.

Based on the available information, the staff believes that any increase in core damage frequency and large early release frequency associated with authorizing the human actions instead of the other alternatives is less than the risk increase that may be authorized under Regulatory Guide 1.174 guidance. Therefore, the staff finds that the licensee’s proposed changes do not reveal an unforseen hazard or a substantially greater potential for a known hazardous event to occur. The staff did not identify “special circumstances” that, if reviewed on a risk-informed basis, might warrant attaching conditions to or denying the proposed changes.

4.0 SUMMARY

The NRC staff has reviewed the licensee’s proposed changes to the failure modes and effects analysis for the specific four lines penetrating containment, in which manual action is being relied upon to ensure containment integrity under certain conditions. As described in detail in the preceding sections, the staff reviewed the impacts of the proposed changes on containment integrity, postaccident flood levels, postaccident sump boron concentration, and containment pressure, and concluded that the licensee would remain in compliance with GDC 16, 50, and 54. The staff reviewed the licensee’s proposed manual actions, and concluded that the actions were consistent with the available guidance regarding the use of manual action. This review is described in detail in Section 3.2. The staff also reviewed the risk insights provided by the licensee, and concluded that no unforeseen hazard or substantially greater potential for a known hazard to occur was involved in this amendment, and, as described in Section 3.3 above, the proposed changes are acceptable from a risk perspective.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (68 FR 18286). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: James Bongarra
Stephen Dinsmore
Angelo Stubbs

Date: March 29, 2004

cc:
Mr. Karl W. Singer, Senior Vice President
Nuclear Operations
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. James E. Maddox, Vice President
Engineering & Technical
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. William R. Lagergren
Site Vice President
Watts Bar Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Spring City, TN 37381

General Counsel
Tennessee Valley Authority
ET 11A
400 West Summit Hill Drive
Knoxville, TN 37902

Mr. T. J. Niessen, Acting General Manager
Nuclear Assurance
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Mark J. Burzynski, Manager
Nuclear Licensing
Tennessee Valley Authority
4X Blue Ridge
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Paul L. Pace, Manager
Licensing and Industry Affairs
Watts Bar Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Spring City, TN 37381

Mr. Larry S. Bryant, Manager
Watts Bar Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Spring City, TN 37381

Senior Resident Inspector
Watts Bar Nuclear Plant
U.S. Nuclear Regulatory Commission
1260 Nuclear Plant Road
Spring City, TN 37381

Rhea County Executive
375 Church Street
Suite 215
Dayton, TN 37321

County Executive
Meigs County Courthouse
Decatur, TN 37322

Mr. Lawrence E. Nanney, Director
Division of Radiological Health
Dept. of Environment & Conservation
Third Floor, L and C Annex
401 Church Street
Nashville, TN 37243-1532

Ms. Ann P. Harris
341 Swing Loop Road
Rockwood, Tennessee 37854