TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

December 10, 1981

RECEIVED
DEC 14 1981
U.S. NUCLEAR REGULATORY COMMISSION

Director of Nuclear Reactor Regulation
Attention: Ms. E. Adensam, Chief
Licensing Branch No. 4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Ms. Adensam:

In the Matter of the Application of	)	Docket Nos. 50-390
Tennessee Valley Authority	)	50-391

Enclosed for NRC review are revised responses to NRC Reactor Systems Branch questions concerning Watts Bar Nuclear Plant. These revisions to questions 212.93, 212.97, 212.106, 212.107, 212.110, and 212.116 address NRC concerns specified during a telephone conference call on November 5, 1981.

If you have any questions concerning this matter, please get in touch with D. P. Ormsby at FTS 858-2682.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills, Manager
Nuclear Regulation and Safety

Sworn to and subscribed before me
this 10th day of Dec. 1981
Bryant M. Lowery
Notary Public
My Commission Expires 4/4/82

B001
s
1/1

Enclosure

8112160023 811210
PDR ADOCK 05000390
A PDR

212.93 Question

The Regulatory Requirement Review Committee, in a memorandum from E. Case, Committee Chairman, to L. Gossick, Executive Director for Operations (dated February 16, 1978), has approved a new staff position (BTP RSB 5-1) for the Residual Heat Removal System (RHR). The technical requirements for your plant are described below. Please respond to these requirements in sufficient detail to enable the staff to review your compliance in an expeditious fashion.

1. Provide safety-grade steam generator dump valves, operators, air and power supplies which meet the single failure criterion.

2. Provide the capability to cooldown to cold shutdown in less than 36 hours assuming the most limiting single failure and loss of offsite power or show that manual actions inside or outside containment or return to hot standby until the manual actions or maintenance can be performed to correct the failure provides an acceptable alternative.

3. Provide the capability to depressurize the reactor coolant system with only safety-grade systems assuming a single failure and loss of offsite power or show that manual actions inside or outside containment or remaining at hot standby until manual actions or repairs are complete provides an acceptable alternative.

4. Provide the capability for borating with only safety-grade systems assuming a single failure and loss of offsite power or show that manual actions inside or outside containment or remaining at hot standby until manual actions or repairs are completed provides an acceptable alternative.

5. Provide the system and component design features necessary for the prototype testing of both the mixing of the added borated water and the cooldown under natural circulation conditions with and without a single failure of a steam generator atmospheric dump valve. These tests and analyses will be used to obtain information on cooldown times and the corresponding AFW requirements.

6. Commit to providing specific procedures for cooling down using natural circulation and submit a summary of these procedures.

7. Provide or require a seismic Category I AFW supply for at least 4 hours at How Shutdown plus cooldown to the DHR system cut-in based on the longest time (for only onsite or offsite power and assuming the worst single failure), or show that an adequate alternate seismic Category I source will be available.

Response

1. The Watts Bar Nuclear Plant (WBN) steam generator powered atmospheric dump valves (one per generator) are seismically qualified. The supplies to these valves are from the plant safety grade auxiliary control air system. The power and air supplies to these valves are trainized (two valves per train), receiving necessary electrical power from the 125-volt vital battery system.

   The most limiting single failure would be the loss of one train of the safety grade air system, or one train of vital power. This would prevent control room initiated steam dumping via two of the fourpower operated relief valves. If such an event were to occur, operating personnel could enter the valve rooms (outside of containment) and throttle steam release from any affected valves by manual operation.

   The second most limiting single failure would be a mechanical breakdown within one of these S.G. relief valves so that the valve would be 'frozen' shut. In this case the operating personnel could circumvent the single active active failure as follows:

   (a) Bring the plant down to RHR cut-in conditions via natural convection in the remaining three active loops. If this cannot be done, WBN will proceed as in (b).

   (b) Alternately, a 6-inch hand valve upstream of the failed relief valve would be closed and the relief valve repaired or replaced. This would be done under emergency conditions.

2. The RCS is capable of being cooled via natural circulation. Sequoyah (SQN) is prototypical of WBN, and tests conducted at SQN have demonstrated the ability to cooldown to RHR initiation on natural convection. The time limit of 36 to 48 hours to the RHR cut in point currently suggested seems reasonable, barring unforeseen difficulties, and SQN prototype testing demonstrated a cooldown rate of 50° P hour to 450° F.

WBN has one RHR suction line containing two safety-grade valves in series for tie in of RCS to the RHR. However, to allow for the unlikely event of either or both of these valves failing to open during this final tie in operation, bypass valves have been added to provide an emergency flow path around either or both of the normal let down valves. These bypass valves have been outfitted with the same pressure and temperature interlocks as the normal valves to avoid overpressurization of or flooding in the RHR system.

3. The reactor coolant system is normally depressurized by means of the pressurizer spray, fed through a single valve FCV62-84, from the discharge of the centrifugal charging pump(s). Should this 2-inch valve fail to open every effort would be made to open it via portable compressed gas cylinder, or by maintenance, and if these attempts fail, system pressure could be reduced by blowing the pressurizer down through one of the two parallel power-operated relief valves provided for this purpose. (PCV68-334 or PCV68-340A).

4. The normal method of heavily borate the WBN RCS is to take suction for the charging pump(s) from the 12 percent boric acid solution of the boric acid storage tank(s) via the boric acid transfer pumps and the normal charging line. This is a safety grade route. An alternate method of boration is to align the discharge of the boric acid pumps, as above, and align the discharge of one centrigual or the positive displacement charging pump with the RCS through the Boron Injection Tank (normal safety injection piping to the RCS).

   Should a single failure of a common valve in this normal charging line occur, every effort would be made to open it via handwheel, etc., and if these fail an alternate boration path would be used, admitting the 12 percent boric acid solution to the RCS via the Boron Injection Tank (Safety Injection Route) or via the RCP seal injection lines. These are also safety-grade flow paths.

5. Boration is performed through the normal Chemical and Volume Control System (CVCS) charging line(s) via the positive displacement or centrifugal charging pump(s), and sampling can be done continuously or intermittently from several sampling connections in the normal or alternate letdown lines, or from two separate hot legs of the RCS loops. In a 'worst

case' situation the amount of boron injected into the RCS can be determined by reduction of inventory and analysis of the boric acid tanks. Since this plant uses 12 wgt percent boric acid solution, the RCS can be adequately borated even if no RCS letdown is possible.

A natural circulation was performed on unit 1 at Sequoyah, with core power at 2-3 percent which in addition to providing that the RCS could be cooled at a rate of approximately 50o F per hour via steam dump from the steam generators, also proved that the RCS could be borated satisfactorily while natural circulation was taking place. The core was successfully poisoned with boron from this level of power within five minutes. Sampling showed that the boron was mixed satisfactorily in the RCS. (Reference Special Test No. 1 - Natural Circulation Tests 1-8, 9B). Cooldown was demonstrated using 1, 2, 3, or 4 loops (one loop at 1 percent power).

6. TVA has provided specific procedures for cooldown using natural circulation, based on these answers as a summary. A natural circulation test will be conducted as part of the startup test program.

7. The normal AFW is supplied from the condensate storage tank, which is a nonseismic tank. However, the design is such that low AFW pump suction line pressure admits ERCW to the AFW pumps suction via automatic multiple, separate safety-grade valves. The ERCW therefore provides the seismic Category I supply and can furnish emergency auxiliary feedwater indefinitely, assuming the worst single active failure.

The following is a more detailed scenario for the cold shutdown problem. (For WBN unit 1 or 2 may be mentioned, but the units are exact duplicates; therefore the analysis is valid.)

<u>COLD SHUTDOWN SCENARIO</u> <u>(Assuming loss of all non-seismic Category I equipment)</u>

The safe shutdown design basis of WBN is hot standby. The plant can be maintained in a safe hot standby condition while manual actions are taken to permit achievement of cold shutdown conditions following a safe shutdown earthquake with loss of offsite power. Under such conditions the plant is capable of achieving RHR initiation conditions (approximately 350o F, 400 psig) in approximately 36 to 48 hours, including the time required for any manual actions. To achieve and maintain cold shutdown, four key functions must be performed. These are: (I)

circulation of the reactor coolant, (II) removal of residual heat, (III) boration and makeup, and (IV) depressurization.

I. Circulation of Reactor Coolant

Circulation of the reactor coolant has two stages in a cooldown from hot standby to cold shutdown. The first stage is from hot standby to 350° F. During this stage, circulation of the reactor coolant is provided by natural circulation with the reactor core as the heat source and steam generators as the heat sink. Steam release from the steam generators is initially via the steam generator safety valves and occurs automatically as a result of turbine and reactor trip. Steam release for cooldown is via the steam generator power operated relief valves which can be operated manually with their handwheels. The steam generator power operated relief valves are accessible for local operation. The status of ech steam generator can be monitored using Class 1E instrumentation located in the Control Room. Three separate channels of indictions for both steam generator pressure and water level are available.

Feedwater to the steam generators is provided from the Auxiliary Feedwater System which as a minimum reserve of 190,000 gallons in the non-seismic condensate storage tank as the primary source and two separate Seismic Category 1 piping sub-systems. The first sub-system is composed of two motor-driven pumps each powered from a different emergency power train, and the second sub-system incorporates a turbine drive pump which can receive motive steam from either of two steam generators. Additional backup is from the fully qualified Seismic Category I Essential Raw Cooling Water System via fully qualified multiple automatic admission valves. The operation of the auxiliary feedwater system can be monitored using Class 1E instrumentation located in the Control Room. There is a single indication of the flows into each steam generator, and pump operating status lights for the motor driven pumps. There is one indication in the main control room for the level in each condensate storage tank. There is local indication for suction and discharge pressure for the turbine drive AFW pump.

The second stage of Reactor Coolant circulation is from 350° F to cold shutdown. During this stage, circulation of the reactor coolant is provided by the Residual Heat Removal Pumps and a fully qualified, redundant system.

II. Removal of Residual Heat

Removal of residual heat also has two stages in a cooldown

from hot standby to cold shutdown. The first stage is from hot standby to 350° F.

During this stage, the steam generators act as the means of heat removal from the reactor coolant system. Initially, steam is released from the steam generators via the steam generator safety valves to maintain hot standby conditions. When the operators are ready to begin the cooldown, the steam generator power operated relief valves are slightly opened by local operation with their handwheels. As the cooldown proceeds, the operator will occasionally adjust these valves to increase the amount they are open. This allows a reasonable cooldown rate to be maintained. Feedwater makeup to the steam generators is provided from the Auxiliary Feedwater System. The Auxiliary Feedwater System has the ability to remove decay heat by providing feedwater to all four steam generators for extended periods of operation. Communications for these actions will be by use of hand held two-way radios.

The second stage is from 350° F to cold shutdown. During this stage the Residual Heat Removal (RHR) System is brought into operation. The Residual Heat Removal Exchangers in the RHR system act as the means of heat removal from the Reactor Coolant System. In the RHR Heat Exchanger, the residual heat is transferred to the Component Cooling System which ultimately transfers the heat to the Essential Raw Cooling Water System (ERCW). The Component Cooling and the ERCW systems are both designed to Seismic Category 1. The RHR system includes two Residual Heat Removal Pumps and two Residual Heat Removal Exchangers. Each RHR pumps is powered from different emergency power trains and each RHR Heat Exchanger is cooled by a different Component Cooling loop. If any component in one RHR loop becomes inoperable, cooldown of the plant is not compromised, however, the time for cooldown would be extended. An audio visual alarm in the main control warns the operator of low flow in the RHR system.

The operation of the RHR system can be monitored using Class 1E instrumentation in the Control Room. There is indication of the pump discharge flow, the pump operating status and the Component Cooling flow from the discharge of the RHR heat exchangers.

III. Boration and Makeup

Boration is accomplished using portions of the Chemical and Volume Control System (CVCS). Boric acid 12 wt. % from the Boric Acid Tanks, (each of which has two redundant emergency power heaters) is supplied to the suction of the Centrifugal Charging Pumps by the Boric Acid Transfer Pumps. The Centrifugal Charging Pumps inject the borated

water into the Reactor Coolant System via the normal charging and reactor coolant pump seal injection flow paths. The three Boric Acid Tanks, for Boric Acid Transfer Pumps, and the associated piping are of Seismic Category 1 design. There is sufficient boric acid capacity in one tank to provide for a cold shutdown for one unit with the most reactive rod withdrawn. The Boric Acid Transfer Pumps are each powered from different emergency power trains. The Boric Acid Tank level can be monitored to verify the operability of the boration portion of the CVCS. For this, credit is taken for operator action in using a portable differential pressure indicator which can be connected to the level signal lines from the Boric Acid Tanks.

Makeup, in excess of that provided as 12 wt. % boric acid is provided form the Refueling Water Storage Tank (RWST) using Centrifugal Charging Pumps and the same injection flow paths as described for boration. Two motor operated valves, each powered from different emergency power trains and connected in parallel, will transfer the suction of the charging pumps to the RSWT. Makeup from the RWST can be monitored using Class 1E instrumentation in the Control Room. Separate redundant channels of RWST level indication exist.

Depressurization

Depressurization is accomplished using portions of the Chemical and Volume Control System (CVCS). Either 12 wt. % boric acid or refueling water can be used as desired for depressurization with the flow path being from the Centrifugal Charging Pumps to the auxiliary spary valve in the Pressurizer. The two Centrifugal Charging Pumps of the CVCS are of Seismic Category I, and are powered from different emergency power trains. The pumps can be operated from and their operating status monitored in the Control Room. The depressurization of the reactor coolant system can be monitored using Class 1E instrumentation in the Control Room. Available to the operator are four channels of Pressurizer pressure, three channels of Pressurizer level and two channels of reactor coolant pressure.

Maintaining RCS Temperature and Pressure Without Letdown

In performing the cooldown, the operator will integrate the functions of heat removal, boration and makeup, and depressurization so that these functions can be accomplished without letdown from the reactor coolant system. Boration, cooldown, and depressurization will be accomplished in a series of short steps arranged to keep Reactor Coolant System temperature and pressure and

Pressurizer level in the desired relationships. However, to demonstrate that boration and depressurization can be done without letdown, a simpler scenario can be used. First, the operators borate the RCS to the cold shutdown conditions, taking advantage of the steam space available in the pressurizer. Second, the operators use the cooldown contraction to low their pressurizer water level. Finally, the operators use auxiliary spary from the CVCS to depressurize the plant to 425 psia.

The assumed initial conditions following plant trip are:

RCS Temperature = 547° F RCS Pressure = 2250 psia
Pressurizer Water Volume = 500 ft%; Pressurizer Steam Volume = 1300 ft^3

To calculate if boration can be accomplished without letting down and without taking the plant water solid, worst case conditions of end of life and maximum peak Xenon were assumed. These result in a requirement for 600 cubic feet of 12 wt. % boric acid at 165° F to reach cold shutdown conditions. When added to the RCS, the boric acid would be heated to 547° F and would expand to 800 cubic feet. Since this volume is less than the 1300 cubic feet available in the pressurizer steam space, boration to cold shutdown concentrations can be accomplished without letdown, without taking the plant water solid, and without cooling down.

The cooldown from 547° F to 350° F decreases the volume of water in the RCS by approximately 1700 cubic feet. Some of this contraction is used to reduce the pressurizer water level to the no-load water level (following the increase caused by the boration) and the remainder is compensated for by makeup from the refueling water storage tank.

To calculate if depressurization can be accomplished without letting down and without taking the plant water solid, it was assumed that the Pressurizer was at saturated conditions with 500 cubic feet of water, 1300 cubic feet of steam, and the Pressurizer metal, all at 653° F (2250 psia). It was further assumed that no additional water would be removed from the pressurizer by the cooldown contraction. With these assumptions, and including the effect of heat input from the pressurizer metal, it was determined that spraying in approximately 820 cubic feet of 165° F water would produce saturated conditions at 425 psia (450° F) with a water volume of 1550 cubic feet and a steam volume of 250 cubic feet.

The results of the calculations described above demonstrate that boration and depressurization can be accomplished

without letdown, without taking the plant water solid, and without taking full credit for the available volume credited by the cooldown contraction.

A more detailed single failure evaluation is presented below:

I. Circulation of the Reactor Coolant

From Hot Standby to 350o F - Four reactor coolant loops and steam generators are provided, any one of which can provide natural circulation flow for adequate core cooling. Even with the single mechanical failure (of a steam generator power operated relief valve), three of the reactor coolant loops and steam generators remain available. As pointed out in 1b all four loops and steam generators would be available, if necessary, by doing valve repair.

II. Removal of Residual Heat

A. From Hot Standby to 350o F

1. Steam generator power operated relief valves - Four valves are provided (one per generator), any one of which is sufficient for residual heat removal. In the event of a single failure, three power operated relief valves remain available, with the fourth available by valve repair as stated in 1b.

2. Auxiliary feedwater pumps - Two motor driven and one steam driven auxiliary feedwater pumps are provided. In the event of a single failure adequate pumps remain available, to provide sufficient feedwater.

3. Flow control valves - Air operated, (the normal flow control valves for the motor driven AFW pumps fail open; those for the steam driven pump flow fail closed but have air accumulators to supply them for adequate control times, plus handwheels if needed to enable manual opening). In the event of a single failure of one flow control valve (which affects flow to one steam generator from either a motor driven pump or the steam driven pump) auxiliary feed flow can still be provided to all four steam generators.

4. Backup source - A backup source for auxiliary feedwater is provided via automatic, multiple, qualified valves which admit ERCW water (a fully

qualified system) to the suctions of the AFW pumps when required to supply adequate NPSH to these pumps. An alarm for low pressure in the suction lines of the three AFW will annunciate in the main control room.

III. Boration and Makeup

A. Boric Acid Tanks A, B, and C

Three boric acid tanks are provided for the two reactor units of the plant. Each tank contains sufficient 12% boric acid to borate one unit for cold shutdown with the most reactive rod fully withdrawn. Piping and the four safety grade acid transfer pumps are arranged so that the two units can share the tanks and, if necessary, the pumps.

B. Boric Acid Transfer Pumps

2A-A and 2B-B (Unit 2) - Two pumps are normally aligned to each reactor unit. Each of these pumps is powered from a different emergency diesel power train. In the event of a single failure, either pump will provide sufficient boric acid flow.

C. If Isolation Valve 1-FCV62-140, normally closed, cannot be opened, an alternate path can be established by opening 1-FCV62-138B, or by using the Boron Injection Tank (normal safety injection) routing.

D. Isolation Valves LCV 62-135 and LCV62-136 - Powered from different emergency power trains; only one of these normally closed valves needs to be opened to provide a makeup flow path from the RWST to the centrifugal charging pumps.

E. Centrigual Charging Pumps (two per unit) - Each pump is powered from a different emergency power train. In the event of a single failure, either pump provides sufficient boration or makeup flow.

F. Flow Control Valve FCV62-93 - This normally open valve fails open on loss of air or power. If FCV62-93 were to close spuriously the centrifugal charging pumps would safely operate on their miniflow circuits. Efforts would be made to open it, however, boration could be accomplished by starting the postive displacement charging pump, or using the safety injection - boron injection tank path, or by using the seal injection path.

G. Flow Control Valve FCV62-89 - This normally open valve fails open on loss of air or power. If FCV62-89 was stuck nearly closed as a result of a single failure, manual bypass valve 62-538 could be opened locally.

H. Isolation Valves FCV62-90 and FCV62-91 - If either of these normally open, motor operated valves, which are powered from different emergency power trains, should close spuriously, operator action could be used to deenergize the valve operator and reopen the valve with its handwheel.

I. Isolation Valve FCV62-85 - If this normally open, fail open valve should close spuriously, alternate charging valve FCV62-86, which fails open, could be used.

IV. Depressurization

Auxiliary Spray VAalve FCV 62-84 - This normally closed valve fails closed on loss of air or power. Use of a portable nitrogen bottle would allow FCV 62-84 to be opened. If FCV 62-84 was stuck closed as a result of a single failure, the redundant Seismic Category I overpressure protection valves (PCV 68-334 and PCV 68-340A) can be used to depressurize the RCS by venting the pressurizer to the PRT.

Environmental Qualification of the RHR Section Isolation Valves

The RHR suction isolation valves are qualified for the steam line break environment (280° F). Therefore they are qualified for the less severe environment which would result from venting the pressurizer to depressurize the RCS.

Comparison of WBN with Referenced SQN Test

WBN is essentially the same plant design as Sequoyah in-so-far as flow and cooldown characteristics are concerned.

At Sequoyah tests have been run to ascertain that the 6-inch steam generator relief valves could be operated manually via their handwheels, and it was demonstrated that all four such valves (one per steam generator) could be so operated. It was further demonstrated that the plant could be cooled down by this method at the rate of 50° F per hour. It should therefore not be necessary to perform similar tests at WBN.

REFERENCES

Function	FSAR Fig.	TVA Dwg No.	Title
I Circulation-			Main and Reheat

	Steam Pump	10.3-1	27W801-1	Steam
II	Removal of Residual Heat	10.4-19	47W803-2	Auxiliary Feedwater
III	Boration and Makeup	9.3-17	47W809-5	CVCS Chemical Control
		9.3-14	47W809-2	CVCS Chemical Control
		9.3-13	47W809-1	Chem & Vol Control System
		6.3-1	47W811-1	Safey Injection System
IV	Depressurization RHR Tie In	5.1-1	47W813-1	Reactor Coolant System
		5.5-6	47W810-1	Residual Heat Removal System

Diablo Canyon is similar to Watts Bar and tests conducted at Diablo Canyon would demonstrate the ability to cooldown to RHR initiation on natural circulation. TVA will address any unforeseen safety concerns identified during the Diablo Canyon test.

212.97 Question
(6.3)
(212.86)

Provide further information about preoperational tests to demonstrate UHI performance. Provide UHI accumulator values such as line resistance and isolation setpoint assumed in safety analysis (with tolerance bands) and acceptance criteria (including instrument uncertainty) for the preop tests. Describe how tests demonstrate that no nitrogen entraining vortices would degrade UHI performance.

Response

The Westinghouse UHIS pre-operational test program has consisted of both a low pressure blowdown (90 psig initial pressure) and a high pressure blowdown (1240 psig initial pressure). The low pressure blowdown portion of the test provides direct measurement of the relative resistance of the UHIS injection piping which with a specific UHIS hydraulic isolation valve (HIV) closure time establishes the water accumulator level instrument setpoint. The high pressure blowdown portion of the test was specified in order to: 1) verify the closing performance of the four hydraulic isolation valves (HIV's), and 2) verify that the water accumulator level instrumentation setpoint, as determined by the low pressure blowdown test, results in the delivery of the proper volume of water to the reactor vessel upper head. This high pressure blowdown test was utilized in order to provide fluid flow conditions that are more severe than the flow conditions expected during UHIS operation following the worst postulated loss of coolant accident. Thus, direct observations of the HIV operation and volume delivery could be made at conservatively high flow rates.

The high pressure UHIS blowdown tests performed at Sequoyah unit 1 and other plants have indeed provided direct verification that the UHIS HIV's perform as expected. These tests have also verified the correlation between the piping resistance of the UHIS injection piping (determined by the low pressure UHIS blowdown test), the closing time of the UHIS HIV and the water accumulator level setpoint required to obtain the desired delivered water volume.

The complete UHIS test program including both the low and high pressure blowdowns have been performed on the Sequoyah unit 1 UHIS. The low pressure test has determined the piping resistance of the unit 1 UHIS injection piping, the high pressure test has verified HIV

valve closure and resultant water volume delivered. Since the Watts Bar units are mechanically and hydraulically identical to the Sequoyah unit 1 UHIS, except that SQN units have a 43° elbow in the discharge legs that does not exist in WBN. An adjustment factor will be provided for slight change in relative resistance and UHI level setpoint to take into account the 43° elbow in SQN that is not in WBN.
To provide actual verification of this similarity between the units, Watts Bar will perform the low pressure blowdown portion of the UHIS pre-operational test program. The results of this test are to show that; 1) the resistance of the Watts Bar units 2 UHIS injection piping and 2) the HIV closing times are identical to Sequoyah unit 1's, except for the above stated adjustment for the 43 ° elbow. Westinghouse has provided test acceptance criteria for these parameters, see Table Q212.97-1. By assuring that these criteria have been met and by utilizing the unit 1 water level setpoint, TVA will be assured that the volume delivered by the Watts Bar units, UHIS will be the same as Sequoyah unit 1 and will be within the delivered water volume band contained in the Sequoyah Appendix K ECCS analysis.

The benefits of not performing a high pressure blowdown test on the Watts Bar units UHIS would be two-fold:

1. The time required for UHIS testing would be shortended and thus would reduce the amount of test personnel time required and also would make the reactor coolant system available for other necessary pre-operational work and testing.

2. The need for valve disk and seats repair or replacement after the test would be eliminated. Although all the UHIS isolation valves have been modified to minimize valve damage during the high pressure blowdown test, it is recognized that the UHIS isolation valves are subjected to very severe ΔP and flow conditions during the test. These test conditions far exceed that which the valves would see during the worst possible LOCA recovery situation. Thus, although the 'post-test' valve leakage has been acceptable for system isolation following a LOCA, valve repair was required to reduce valve leakage for system isolation when the RCS was depressurized for refueling, etc.

In the safety analysis the UHI accumulator discharge line resistance is assumed to be a fl/D of 23.0 and the accumulator isolation valve closure setpoint is 945 $ft^3 \pm 50\ ft^3$ of water discharged into the reactor vessel.

The UHI water volume uncertainty band applied analysis is developed as a bound to the possible variation in the total UHI water delivered. Variation in total delivery is postulated to result from two causes: variation which results from measurement error and the uncertainty associated with system performance and variation which results from a single failure in the UHI system.

Each source of variation in total UHI delivery and its associated volume contribution is given in attached Table Q212.97-2. The volume contributions associated with single failure results from a postulated system malfunctions and are therefore not subject to statistical consideration. The single failure volumes are added directly to the nominal UHI volume setpoint. The remaining volume sources are the result of system uncertainties.

Nitrogen samples of the discharged fluid are taken just before and during the UHI hi-pressure blowdown. These samples have consistently demonstrated no significant amount of nitrogen is intrained in the discharged fluid.

TABLE Q212.97-1

WATTS BAR UNITS 1 AND 2 LOW PRESSURE BLOWDOWN
TEST ACCEPTANCE CRITERIA

a) Relative resistance - 3.02 ± 0.52

b) HIV static opening time 4.0 seconds ± 0.25

c) HIV static closing time - 3.5 seconds ± 0.05

d) Level setpoint for actuation of HIV - 103.4 inches ± 0.5 atmospheric pressure

TABLE Q212.97-2

UHI WATER VOLUME UNCERTAINTY

SOURCE	ASSOCIATED VOLUME FT^3
SINGLE FAILURE	
TRAIN FAILURE	+55
ONE VALVE CLOSES PREMATURELY	-51
TANK LEVEL INSTRUMENTATION ACCURACY	± 2.2
INSTRUMENT SETTING TOLERANCE	± 2.0
TANK VOLUME TOLERANCE	± 19
HYDRAULIC ISOLATION VALVE STROKING TIME	± 12.
TANK LEVEL READING ACCURACY	± 11.
TOTAL ERROR	198.4
TOTAL ERROR AT .95 PROBABILITY	156.

212.106 (9.2.7) (212.37) (212.87) Question:

An analysis of NPSH for RHR pumps in the recirculation mode of ECCS operation, dated April 1979 and the analysis summary presented in FSAR Section 9.2.7.1 (Amendment 35) are inconsistent in results and calculational assumptions. The 4/79 analysis employed techniques subsequently approved for Sequoyah; however, inputs for the Sequoyah analyses had to be corrected in order to make them acceptable and the resultant calculated NPSH margin reduced from 31.3 ft. to 2.8 ft. Explain the discrepancy between Watts Bar documentations (4/79 Analyses, and Amendment 35) and, in tabular form, list input assumptions (LPI flow, worst case, etc.), criteria (NPSH required at given flow, test rate 'as installed' LPI must meet, etc.), and summary of calculated NPSH available (specify term by term).

Response:

The basic discrepancy between the April 1979 analysis and FSAR Amendment 35 is the assumption of maximum water temperature. An earlier analysis used the 160°F maximum of the April 1979 analysis, but the FSAR assumption was provided extra conservatism by using 190°F. This change effectively reduced the RHR pump excess head by 11-1/2 feet from vapor pressure and atmospheric pressure effects (33.8 feet in April 1979 and 23 feet in FSAR).

Tabulation of Assumptions, Criteria, and Operating Data

	4/79 Analysis	FSAR Amend. 35	5500 GPM ANALYSIS
Maximum water temperature - assumed	160°	190°	190°
Containment atmosphere - assumed	14.7 lb/in²a	14.7 lb/in²a	14.7 lb/in^{2}a
Containment flood elevation - assumed, RWST and ice melt only	716 ft.	716 ft.	716 ft.
RHR pump rated flow - criteria, 2 pumps operating concurrent	4500 gpm ea.	4500 gpm ea.	5500 GPM ea.
Containment spray pumps rated flow - criteria, 2 pumps operating concurrent	8000 gpm total	8000 gpm total	8000 GPM ea.
Pipe configuration - data, identical	- maximum loss path -		------
RHR pump NPSHR - data, manufacturer's test curve	17.5 ft.	17.0 ft.	24.5 ft.

Summary of Calculated NPSH (+ Excess Head)

	4/79 Analysis	FSAR Amend. 35	5500 GPM Analysis
Static head 716' - 678' - 7-1/2'	+37'-4-1/2"	+37'-4-1/2"	+37' - 4½"
Containment atmosphere	+34'-8-1/4"	+35	+35
Pipe loss	-8'-4-5/8"	-8'-4"	-10' -3"
Sump and screen loss	-1'-1-7/8"	-11-1/4"	-11 3/4"
NPSHR	-17'-6"	-17'	-24' - 6"
Vapor pressure	-11'-2-1/2"	-22'-3-3/8"	-22' - 3 3/8
Excess head resultant	33'-9-3/4" 33.8' shown	23'-9-7/8" 23' shown	14' - 4 3/8'

212.107 (6.3) (9.2.7) (212.39)

Question:

Table 6.3-3a lists times, volumes, etc., associated with ECCS switchover from injection to recirculation and provides a bsis for RWST sizing. This table does not include instrument uncertainty (assume ± 3% of full instrument space unless documented justification of another value is provided and approved), nor does it consider single failures in switchover. Justify the RWST volume, setpoints, and alarms, addressing the above consideration. The justification should identify all functional requirements for the RWST (e.g., adequate injection volume to assure NPSH for recirculation, sufficient remaining volume.

Response:

The refueling water storage tank (RWST) is held within the limits of 370,000 to 375,000 gallons of borated (2,000 to 2,100 ppm of boron) water during normal plant operation (SQN technical specifications, assuming WBN will be the same). (Capacity at overflow is 378,000 gallons, FSAR Section 9.2.7). At 3 percent of full span instrument error would equate to -11,340 gallons or +3,000 gallons (limited by the overflow).

Table 6.3-3a assumes a starting supply of approximately 372,000 gallons in the RWST. The autoswitchover is automatically done when the RWST level reaches 120,000 gallons, coincident with an in-containment water level equal to or greater than 2-1/2 feet above the floor of the containment. A this point the operators start to verify the autoswitchover and perform manual initiation of valve operations, pump starts and stops, and verify overall core and containment cooling systems performance. In arriving at the 'Time Elaspsed (total in seconds)' figures, an operator was given 10 seconds to initiate each action, followed by an exact addition of the time for the equipment to respond (i.e., 10 seconds for a fast valve to open or close and up to two minutes for slow valves).

At the time the containment spray pumps are restarted (560 seconds after operator actions began or 23.4 minutes after the automatic initiation of the safety injection signal), the indicated water level in the RWST is 47,000 gallons. At three percent of full scale, instrument error would confine this to between 58,340

and 35,660 gallons, a range which would not affect NPSH of the pumps or afford the possibility of the RWST running dry. The level of water inside containment at this time will be at least 13 feet 2 inches above the top of the containment floor (and the top of the containment sump). Tests performed by TVA have shown that NPSH and vortexing are not problems at the time of autoswitchover (120,000 ± 11,340 gallons in RWST). Vortex suppressors have been added to both the RWST and the containment sump as further insurance against vortexing. The entire contents of the RWST are therefore 'useable' but the procedure does not empty the RWST (to avoid pump cavitation).

Single failures were considered in the redundant system design (Reference FSAR Section 6.3.2.11). A single failure (i.e., of one train of pump) would not affect the water levels in the RWST or the containment sump at autoswitchover since these are controlled by actual water levels and not be flow rates. Such failures would stretch out the times involved, giving the operators more time to complete each action.

The most limiting single failure would be the loss of an entire train of pumps, probably caused by a loss of power, not replaced by the diesels. IN this case the above loss of a train of pumps, plus one containment spray pump would be experienced. The safety injection and containment spray systems are designed to take such a failure; one train of safety injection pumps plus one containment spray pump will mitigate FSAR Chapter 15 accidents.

Other single failures considered include the failure of suction valves feeding water from the RWST to close (e.g. FCF 74-3 or FCV 74-21). In this case the operator during his procedural verification of the completion of the autoswitchover would note this from his valve position lights and RWST level indication in the MCR, stop the RHR pump involved, close the common suction valve, FCV 63-1, and then restart the RHR pump.

Should a sump suction valve fail to open (e.g. FCV 63-72, 73 to RHR pumps or FCV 72-44,45 to containment spray pumps), the associated pump would also be stopped by the operator to prevent cavitation. If the valve could be opened by any reasonable means, the pump would then be restarted; otherwise the affected equipment would have to remain idle, a situation still within the design basis of the plant.

Excessive water levels (above 13 feet 2 inches inside containment) and possible equipment flooding cannot be brought about by the water contained in the RWST, because the tank has an overflow capacity of only 378,000 gallons. The maximum range of water available from the RWST, with a three precent instrument error is 378,000 to 358,660 gallons. The maximum difference of 19,340 gallons equates to a level change inside containment of approximately 8 inches, and most of the error would result in a lower than calculated level inside containment (because of the overflow design limitation of the tank contents).

212.110 (6.3) **Question:**

Questions 212.96 and 212.26 address valve 8813 in the miniflow header common to the HPI pumps. The applicant has provided a power removal inhibit function in the control room to assure against mispositioning of this valve by an active single failure, but has not responded to question 212.96 which requires provision to verify the flow path through the valve during normal operation (assurance against passive failures). We require that this issue be resolved prior to operation of Watts Bar.

Response:

See response to 212.96 provided in Amendment 44.
Safety injection pump miniflow path is verified at lease once every 31 days or once every month through the use of the appropriate Surveillence Instructions that verify valve positions and flow pursuant to the Technical Specifications.

212.116 (6.3) Question:

With regard to the potential for debris impairing containment sump/ECCS performance, four considerations should be addressed;

1) containment design and potential sources of sump debris (e.g., insulation, etc.),

2) Maintenance of containment 'as licensed' cleanliness (free of loose debris),

3) instrumentation and alarms to alert the operator to a degradation in post-LOCA ECCS performance,

4) procedures to be taken in the event of post-LOCA ECCS performance degradation.

For item 1, a detailed survey of insulation and other potential debris sources is required; however, this item may be resolved by providing a general summary of the insulation used at Watts Bar, an assessment by the applicant of its potential to produce post-LOCA debris, and a commitment to provide a detailed survey prior to startup following the first refueling outage.

The following questions are provided as guidance into the nature and detail of the type of response expected to resolve the above concerns.

For item that have been previously resolved, you may respond by referring to the previous documentation. Items 1, 2, and 3 must be resolved prior to full power operation; item 4 must be resolved prior to startup following the first refueling.

1. In addition to insulation debris resulting from LOCA effects, debris can be generated within the containment from other sources, such as (1) degraded materials (paint chips), and (2) items which are taken into and left in the containment following maintenance and inspection activities.

   Describe how the housekeeping program for Watts Bar will control and limit debris accumulation from these sources. The objective is to assure that debris capable of defeating the post-LOCA core cooling functions are identified and removed from the containment. The response should include references to specific procedures or other means to assure that 'as licensed' cleanliness will be

attained prior to initial operation and prior to each resumption of operation.

2. Address the degree of compliance of Watts Bar with the following recommendation which is also set forth as item C.14 of Regulatory Guide 1.82:

   'Inservice inspection requirements for coolant pump components (trash racks, screens, and pump suction inlets) should include the following:

   a. Coolant sump components should be inspected during every refueling period downtime, and

   b. The inspection should be a visual examination of the components for evidence of structural distress or corrosion.'

3. The resolution of the concerns noted below plus the provisions of adequate NPSH under non-debris conditions, and adequate housekeeping practices are expected to reduce the likelihood of problems during recirculation. However, in the event that LPI recirculation system problems such as pump cavitation or air entrainment to occur, the operator should have the capability to recognize and contend with the problems.

   Both cavitation and air entrainment could be expected to cause pump vibration and oscillations in system flow rate and pressure. Show that the operator will be provided with sufficient instrumentation and appropriate indications to allow and enable detection of these problems. List the instrumentation available giving both the location of the sensor and the readout.

   The incidence of cavitation, air entrainment or vortex formation could be reduced by reducting the system flow rate. The operator should have the capability to perform indicated actions (e.g., throttling or terminating flow, resort to alternate cooling system, etc). Show that the emergency operating instructions and the operator training consider the need to monitor the long-term performance of the recirculation system and consider the need for corrective actions to alleviate problems.

4. With regard to the sump tests on Watts Bar, the responses to the following concerns pertaining to potential sump screen blockage are required:

a. Various types of insulation may be used in the containment. For each type provide the following information:

   (1) The manufacturer, brand name, volume and area covered.

   (2) A brief description of the material and an estimate of the tendency of this material either to form particles small enough to pass through the fine screen in the sump or to block the sump trash racks or sump screens.

   (3) Location of the material (metal mirrored, foam glass, foam rubber, foam concrete, fiberglass, etc.) with respect to whether a mechanism exists for the material to be transported to the sump.

b. Provide an estimate of the amount of debris that the sump inlet screens may be subjected to during a loss-of-coolant accident. Describe the origin of the debris and design features of the containment sump and equipment which would preclude the screens becoming blocked or the sump plugged by debris. Your discussion should include consideration of at least the following sources of possible debris: equipment insulation, sand plug materials, reactor cavity annulus sand tanks or sand bags for biological shielding, containment loose insulation, and debris which could be generated by failure of non-safety related equipment within the containment. Entry of sand plug materials into the containment sump and the possibility of sand covering the recirculation line inlets prior to the initiation of recirculation flow from the containment should be specifically addressed.

   Please provide this information along with your conclusion regarding the percentage of the screens which would be expected to be blocked by particles of all sizes, including those greater than 250 mils.

c. With respect to the conclusion that debris with a specific gravity greater than unity will settle before reaching the sump cover, consider the potential for flow paths which may direct significant quantities of debris laden coolant

into the lower containment in the vicinity of the sump and the availability or lack of sufficient horizontal surface areas or obstructions to promote settlings or holdup of debris prior to reaching the sump.

d. Does metal mirror insulation house other materials, fibrous or otherwise, which could become debris is the insulation were blown off as a result of a LOCA?

e. If the Watts Bar containment contains loose insulation, include examples of how the insulation will be precluded from reaching the sump.

(1) Provide a schematic drawing of the post-LOCA water level in containment during the recirculation mode relative to the elevation of the ECCS sump floor. Include on this drawing the location of the containment water level sensor and the elevations corresponds to readings of zero and 100 percent of range on the control room indicator.

(2) Provide several large scale drawings of the containment structures, systems and components at elevations.

(3) Does the Watts Bar utilize or similar materials in the containment during power operation for purposes such as reactor cavity annulus biological shielding (e.g., sand tanks or sand bags) or reactor cavity blow out sand plugs?

Response:

It is TVA's understanding that resolution of Generic Task A-43 'Containment Emergency Sump Reliability' is being evaluated by Burns and Roe as consultants.

TVA's interim position on task A-43 was submitted to NRC by letter dated September 11, 1981 from L. M. Mills to E. Adensam.

To support resolution of this issue the above requested information was provided to Mr. R. Reyer of Burns and Roe by letters dated October 20, 1980 and December 23, 1980.

A visual inspection is performed before establishing containment integrity, whenever containment is entered after containment integrity has been established, and at least once every 18 months. This inspection is performed through the use of the appropriate surveillence instructions for the containment area and for the RHR sump area.

The RCS is monitored at all times during an accident for indication of inadequate core cooling through the plant emergency operating instructions.