TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401

5N 157B Lookout Place

January 13, 1986

Mr. Hugh L. Thompson, Jr., Director
Division of PWR Licensing-A
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Thompson:

In the Matter of)	Docket Nos.	50-259	50-390
Tennessee Valley Authority)		50-260	50-391
		50-296	50-438
		50-327	50-439
		50-328	

Please refer to your Generic Letter 85-12 dated June 28, 1985 subject, "Generic Letter 85-12 - Implementation of TMI Action Item II.K.3.5, Automatic Trip of Reactor Coolant Pumps (RCPs)."

Both Sequoyah and Watts Bar Nuclear Plants have chosen the Westinghouse Owners' Group (WOG) alternate criteria of tripping the pumps based on reactor coolant system (RCS) pressure. Enclosure 1 is our specific Sequoyah response and enclosure 2 is our Watts Bar response. The schedule for submitting the responses has been previously discussed with your project managers for Sequoyah (Carl Stahle) and Watts Bar (Tom Kenyon). If you have any questions, please telephone Fisher Campbell at FTS 858-4892, the licensing engineer for this item.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

J W Hufham

J. W. Hufham
Manager of Licensing

Sworn to and subscribed before me
this 13th day of Jan. 1986.

Paulette H. White
Notary Public

My Commission Expires 8-24-88

Enclosures

A046
/1

Add:

AD - J. KNIGHT (ltr only)
EB (BALLARD)
EICSB (ROSA)
PSB (GAMMILL)
PSB (BERLINGER)
FOB (BENAROYA)
AD - G. CRUTCHFIELD (ltr only)
EB (W. JOHNSTON)
RSB (THOMAS)
EICSB (PARR)
FOB (W. REGAN)
EB (LIAW)
PSB (L. HULMAN)
EICSB (SRINIVASAN)
RSB (ACTING)
FOB (VASSALLO)
AD - G. LAINAS (ltr only)

ENCLOSURE 1

A. Determination of RCP Trip Criteria

1. SQN uses Reactor Coolant System (RCS) pressure transmitters PT-68-66 and PT-68-69 for determination of need to trip RCPs. Upon reaching 1,250 psig during an uncontrolled depressurization, the reactor operator is required to initiate manual reactor coolant pump trip by means of main control room bench board handswitches. Pressure transmitter PT-68-66 provides its signal to an RCS pressure indicator on main control room panel M-6. Pressure transmitter PT-68-69 provides its signal to RCS pressure recorder PR-68-69 on main control room panel M-5. Both transmitters are qualified Post Accident Monitors and are powered by separate trains. This provides redundant signals to the operator for RCS wide range pressure indication for determination of when the pumps should be tripped.

2. The minimum accuracy requirements for PT-68-66 and PT-68-69 are listed in FSAR chapter 7, Table 7.5.1-2. This table identifies an accuracy requirement of ±2.5 percent of total span (span is 0 to 3,000 psig) or 75 psig for Condition IV events.

   Sequoyah defines adverse containment conditions as receipt of a Phase B containment isolation signal. Phase B isolation is received by means of receipt of 2.81 psig from 2 of 4 containment pressure transmitters. Upon receipt of this signal, the operator will immediately trip the RCPs, and these RCS pressure signals will no longer be used as parameters for RCP operation or RCP trip criteria.

3. The LOFTRAN computer code was used to perform the alternate RCP trip criteria analyses. Both Steam Generator Tube Rupture (SGTR) and non-LOCA event were simulated in these analyses. Results from the SGTR analyses were used to obtain all but three of the trip paramet rs. LOFTRAN is a Westinghouse-licensed code used for FSAR SGTR and non-LOCA analyses. The code has been validated against the January 1982 SGTR event at the Ginna plant. The results of this validation show that LOFTRAN can accurately predict RCS pressure, RCS temperatures, and secondary pressures, especially in the first 10 minutes of the transient. This is the critical time period when minimum pressure and subcooling is determined.

   The major causes of uncertainties and conservatism in the computer program results, assuming no changes in the initial plant conditions (i.e., full power, pressurizer lever, all SI and AFW pumps run), are because of either models or inputs to LOFTRAN. The following are considered to have the most impact on the determination of the RCP trip criteria:

   1. Break flow
   2. SI flow
   3. Decay heat
   4. Auxiliary feedwater flow

   The following sections provide an evaluation of the uncertainties associated with each of these items.

To conservatively simulate a double-ended tube rupture in safety analyses, the break flow model used in LOFTRAN includes a substantial amount of conservatism (i.e., predicts higher break flow than actually expected). Westinghouse has performed analyses and developed a more realistic break flow model that has been validated against the Ginna SGTR tube rupture data. The break flow model used in the WOG analyses has been shown to be approximately 30 percent conservative when the effect of the higher predicted break flow is compared to the more realistic model. The consequence of the higher predicted break flow is a lower than expected predicted minimum pressure.

The SI flow inputs used were derived from best estimate calculations, assuming all SI trains operating. An evaluation of the calculational methodology shows that these inputs have a maximum uncertainty of ±10 percent.

The decay heat model used in the WOG analyses was based on the 1971 ANS 5.1 standard. When compared with more recent 1979 ANS 5.1 decay heat inputs, the values used in the WOG analyses are higher by about 5 percent. To determine the effect of the uncertainty because of the decay heat model, a sensitivity study was conducted for SGTR. The results of this study show that a 20 percent decrease in decay heat resulted in only a 1 percent decrease in RCS pressure for the first 10 minutes of the transient. Since RCS temperature is controlled by the steam dump, it is not affected by the decay heat model uncertainty.

The AFW flow rate input used in the WOG analyses are best-estimate values, assuming that all auxiliary feed pumps are running, minimum pump start delay, and no throttling. To evaluate the uncertainties with AFW flow rate, a sensitivity study was performed. Results from the 2-loop plant study show that a 64 percent increase in AFW flow resulted in only an 8 percent decrease in minimum RCS pressure, a 3 percent decrease in minimum RCS subcooling, and an 8 percent decrease in minimum pressure differential. Results from the 3-loop plant study show that a 27 percent increase in AFW flow resulted in only a 3 percent decrease in minimum RCS pressure, a 2 percent decrease in minimum RCS subcooling, and a 2 percent decrease in pressure differential.

The effects of all these uncertainties with the models and input parameters were evaluated, and it was concluded that the contributions from the break flow conservatism and the SI uncertainty dominate. The calculated overall uncertainty in the WOG analyses, as a result of these considerations for SON units 1 and 2, is +30 to +200 psig for the minimum RCS pressure RCP trip setpoint. Because of the minimal effects from the decay heat model and AFW input, these break flow results include only the effects of the uncertainties due to the model and SI flow inputs.

B. Potential Reactor Coolant Pump Problems

1. SQN has two "levels" of containment isolation, Phase A and Phase B. Phase B provides a higher level of isolation since it responds to a more severe transient.

Phase A containment isolation is initiated at containment pressure of +1.54 psid with respect to annulus pressure. SQN's reactor coolant pumps are designed to operate following Phase A isolation. All necessary equipment and water services remain in operation.

Each RCP is equipped with a thermal barrier heat exchanger in the event seal injection water is interrupted for any reason. The thermal barrier heat exchangers are cooled by component cooling water. Upon loss of seal injection, RCS water will reverse flow up the shaft, through the thermal barrier for cooling, to the seals to prevent pump damage. This arrangement may be utilized, by procedure, to allow attempts to restore the interrupted seal injection flow. While this arrangement is in use, lower bearing temperature should not exceed 210°F. If bearing temperature reaches 210°F, the operator is instructed to stop the pump.

For Phase B isolation, which occurs at containment pressure of +2.81 psid with respect to the annulus, water services to the RCP are interrupted. In the event of a Phase B isolation, the operators are instructed to trip the RCPs in order to prevent damage to the pumps. A pressure transient of this magnitude would be indicative of a LOCA or MSLB of sufficient magnitude such that RCP operation would not significantly impact the transient behavior.

2. Tripping an RCP is accomplished by the operator placing the RCP handswitch on the main control room benchboard in the stop position. This energizes trip coil 52T on the feeder breaker to the pump. The control power for this circuit is supplied by a 250-volt DC power supply, which is supplied by batteries and chargers. The chargers are normally supplied by the 480-volt auxiliary common board and are alternately powered by 480-volt shutdown board. The trip coil is energized to actuate, however, if power were lost, the normal feeder breaker could be manually tripped locally at the switchgear. These relays and breakers are located in an area outside containment that would not be subject to adverse containment conditions.

C. Operator Training and Procedures (RCP Trip)

1. During the process of certification as a licensed reactor operator or Senior Reactor Operator, the candidate will encounter the RCP trip criteria during several phases of his training. In each of these phases, the candidate will encounter either classroom or simulator training which will address RCP trip criteria. SQN's licensed operators and Shift Technical Advisors received training (classroon and simulator) on the background and basis for the RCP trip criteria and its use in EOPs during weeks 3 and 4 requalification 1983.

The general philosophy regarding the need for RCP trip versus the desire to keep the pumps running is summarized below.

Immediately upon receipt of Phase B containment isolation, the reactor operator is to initiate manual reactor coolant pump trip. This is required since Emergency Raw Cooling Water and Component Cooling Water supplies to containment are interrupted by containment isolation valve closure. Tripping the pump immediately upon receipt of Phase B precludes pump damage due to overheating.

During an uncontrolled Reactor Coolant System depressurization upon reaching 1250 psig, the operator is to initate manual RCP trip. This setpoint was calculated using WOG guidelines which consider RCS pressure, RCS subcooling, and secondary pressure dependent RCS pressure. Operators have been trained on this setpoint, and it has been stressed that the depressurization must be an uncontrolled depressurization. This must be accompanied by at least one high head injection (centrifugal charging pump) or intermediate head injection (safety injection pump) pump running and delivering flow to the RCS. If these conditions are not met, the RCPs should not be tripped. The RCPs should remain in operation in order to facilitate core residual heat removal.

It has been stressed in operator training that if the RCP trip criteria is not fully met, it is more advantageous to continue RCP operation than to trip the RCPs. In virtually all non-LOCA accidents, RCP operation is desired for its heat removal and pressure control assistance.

2. Procedures which include RCP trip-related operations are as follows:

Instruction Title	Instruction Number	*RCP Trip Operations
Reactor Trip or Safety Injection	E-0	a, c, e
Reactor Trip Response	ES-0.1	b, c, e
SI Termination	ES-0.2	b, c, e
Natural Circulation Cooldown	ES-0.3	b, c, e
Loss of Reactor or Secondary Coolant	E-1	a, c, e
Post-LOCA Cooldown	ES-1.1	b, f
Transfer to RHR Containment Sump	ES-1.2	-
Transfer to Hot Leg Recirculation	ES-1.3	-

Instruction Title	Instruction Number	RCP Trip Operations
Faulted Steam Generator Isolation	E-2	-
Steam Generator Tube Rupture	E-3	e, f
SI Termination Following SGTR	ES-3.1	b, c, e, f
Post-SGTR Cooldown Using Backfill	ES-3.2	e, f
Post-SGTR Cooldown by Ruptured Steam Generator Depressurizaiton	ES-3.3	e, f
Foldout Page	E-FOP	a

Instruction	Title	RCP Trip Operations
FR-C.1	Response to Inadequate Core Cooling	f, g
FR-P.1	Response to Pressurized Thermal Shock	f
FR-Z.1	Response to High Containment Pressure	a
FR-I.1	Response to Voids in Reactor Vessel	b, c, d

*RCP Trip Operations

(a) RCP trip using WOG alternate criteria
(b) RCP restart
(c) Decay heat removal by natural circulation
(d) Primary system void removal
(e) Use of steam generators with and without RCPs operating
(f) RCP trip for other reasons
(g) Special restart without support conditions

Enclosure 2

A. Determination Of RCP Trip Criteria

1. Watts Bar uses pressure transmitters (PTs) 68-63 and 68-64 to determine the RCS pressure. When 1400 psig is reached as indicated by these pressure transmitters and pressure is decreasing uncontrollably, the reactor coolant pumps are to be tripped. These transmitters are made by different manufacturers and are powered by separate trains of power. This provides redundant signals to the operator for RCS wide-range pressure to let me know when the pumps should be tripped.

2. The minimum accuracy for PTs 68-63 and 68-64 is $\pm$ 116 psi per Table 7.5-1 of the Final Safety Analysis Report (FSAR). The transmitters are located outside containment and should not be affected by adverse containment conditions.

3. The LOFTRAN computer code was used to perform the alternate RCP trip criteria analyses. Both steam generator tube rupture (SGTR) and non-LOCA (loss of coolant accident) event were simulated in these analyses. Results from the SGTR analyses were used to obtain all but three of the trip parameters. LOFTRAN is a Westinghouse licensed code used for FSAR, SGTR, and non-LOCA analyses. The code has been validated against the January 1982 SGTR event at the Ginna plant. The results of this validation show the LOFTRAN can accurately predict RCS pressure and RCS temperatures of the transient. This is the ciritical time period when minimum pressure and subcooling is determined.

The major causes of uncertainties and conservatism in the computer program results, assuming no changes in the initial plant conditions (i.e., full power, pressurizer level, all safety injection (SI) and auxiliary feedwater (AFW) pumps running) are due to either models or inputs to LOFTRAN. The following are considered to have the most impact on the determination of the RCP trip criteria.

1. Break flow

2. SI flow

3. Decay heat

4. Auxiliary feedwater flow

The following sections provide an elevation of the uncertainties associated with each of these items.

To conservatively simulate a double ended tube rupture in safety analyses, the break flow model used in LOFTRAN includes substantial amount of conservatism, (i.e., predicts higher break flow than actually expected). Westinghouse has performed analyses and developed a more realistic break flow model that has been validated against the Ginna tube rupture data. The break flow model used in the WOG analyses has been shown to be approximately 30 percent conservative when the effect of the higher predicted break flow is compared to the more realistic model. The consequence of the higher predicted break flow is a lower than expected predicted minimum pressure.

The SI flow input used was derived from best estimate calculations, assuming all SI trains operating. An evaluation of the calculational methodology shows that these inputs have a maximum uncertainty of ±10 percent.

The decay heat model used in the WOG analyses was based on the 1971 American Nuclear Society (ANS) 5.1 standard. When compared with the more recent 1979 ANS 5.1 decay heat inputs, the values used in the WOG analyses are higher by about 5 percent. To determine the effect of the uncertainty due to the decay heat model, a sensitivity study was conducted for SGTR. The results of this study show that a 20 percent decrease in decay heat resulted in only a 1 percent decrease in RCS pressure for the first 10 minutes of the transient. Since RCS temperature is controlled by the steam dump, it is not affected by the decay heat model uncertainty.

The AFW flow rate input used in the WOG analyses are best estimate values, assuming that all auxiliary feed pumps are running, minimum pump start delay, and no throttling. To evaluate the uncertainties with AFW flow rate, a sensitivity study was performed. Results from the 2-loop plant study show that a 64 percent increase in AFW flow resulted in only an 8 percent decrease in minimum RCS pressure, a 3 percent decrease in minimum RCS subcooling, and an 8 percent decrease in minimum pressure differential. Results from the 3-loop plant study show that, a 27 percent increase in AFW flow resulted in only a 3 percent decrease in minimum RCS pressure, and 2 percent decrease in minimum RCS subcooling, and a 2 percent decrease in pressure differential.

The effects of all these uncertainties with the models and input parameters were evaluated and it was concluded that the contributions from the break flow conservatism and the SI uncertainty dominate. The calculated overall uncertainty in the WOG analyses as a result of these considerations for the Watts Bar units is -150 to 150 psig for the RCS pressure RCP setpoint. Due to the minimal effects from the decay heat model and AFW input, these results include only the effects of the uncertainties due to the break flow model and SI flow inputs. The minimum RCS pressure setpoint calculated by the WOG for the bounding accidents of a SGTR, steamline or feedline break was 1698 psia. The Watts Bar criteria of 1400 psig provides approximately 130 psi margin even when the analysis uncertainty is considered. This margin adequately covers any plant specific uncertainties. This margin also ensures that the RCPs will not be tripped when it would be detrimental.

B. Potential reactor coolant pump problems

1. Watts Bar defines a containment isolation as a Phase A isolation which occurs at 1.54 psid in containment with respect to the annulus. Our reactor coolant pumps are designed to operate following a Phase A isolation and all the necessary equipment and water supplies required to keep them operable do not receive a Phase A isolation. If seal water is interrupted due to loss of a charging pump, a second pump can be started in a matter of seconds and the thermal barrier which is cooled by component cooling water will keep the shaft cool during this period.

   The only time that water services are interrupted to the RCPs is when a Phase B isolation occurs at 2.81 psid in containment with respect to the annulus. The pumps would be shut off due to the loss of component cooling water to the RCP motor oil coolers. However, this high of a containment pressure would indicate a relatively large LOCA or steamline break in which case the RCPs would not significantly contribute to the cooldown.

2. Tripping an RCP is accomplished by the operator placing the RCP handswitch in the main control room on the stop position. This supplies power to 52 TC which is a trip coil on the feeder breaker to the pump. The control power for the circuit is 250V batteries and chargers. The charger is normally supplied by the 480V auxiliary common board and is alternately supplied by the 480V shutdown board. The trip coil is energized to actuate; but if the control power is lost, either the normal feeder breaker or the motor protection breaker could be tripped locally at the switchgear by manual operation. None of the relays or breakers required to trip the pump are located inside containment and therefore are not subject to adverse containment conditions.

C. 1. Watts Bar operator training on RCP trip criteria during a small break LOCA incorporates many facets. This training is presented in classroom, on a simulator, and on shift. The classroom portion consists of in-class lectures on the various emergency procedures. On the simulator, the procedures are reviewed and reinforced with actual hands on experience. On the job, each operator must review all emergency procedures on a yearly basis. Should a revision to a procedure be made, the operator is required to review the change in a timely manner.

With regard to the general philosophy concerning the need to trip the RCPs versus the desire to keep the pumps running, the following statements can best summarize Watts Bar's position.

When containment pressure reaches the Phase B pressure setpoint, this is an indication of adverse containment conditions and ensures the RCPs are tripped in a timeframe consistent with analysis for a small break LOCA. When Phase B conditions exist, cooling water to the RCPs is lost due to the Phase B isolation signal, therefore, the RCPs should be tripped.

Uncontrolled RCS system pressure decrease to a calculated pressure setpoint is also a criteria for tripping the RCPs. The pressure setpoint of 1400 psig was calculated using WOG guidelines considering RCS pressure, RCS subcooling, and secondary pressure dependent RCS pressure. Operators have been trained to the fact that RCS depressurization must be an uncontrolled depressurization. This ensures the RCPs are not tripped on controlled depressurizations such as an operator initiated depressurization for SGTR and other normal depressurization evaluations. Uncontrolled depressurization must be accompanied by at least one high head SI pump delivering flow to the RCS. One of the fundamental conditions that must be satisfied to trip the RCPs is at least one high pressure SI pump (centrifugal charging pump (CCP) or SI) must be delivering flow to the RCS. If this condition does not exist, the RCPs should not be tripped, regardless of RCS pressure. RCPs should be operated to provide core heat removal.

Plant operators are trained to recognize the RCP trip parameters, and to trip the RCPs, if appropriate. They are also trained that if these parameters are not met, it is more desirable to have the RCPs running. In virtually all non-LOCA accidents, it is more advantageous to have the RCPs in operation. This provides additional margin to safety criteria limits and compliments operator actions during recovery. RCP operation enhances core heat removal and makes RCS pressure control easier.

2. Procedures which include RCP trip related operations

INSTR	TITLE	RCP TRIP OPERATIONS*
E-0	Reactor Trip *or* Safety Injection	a,c,e
ES-0.1	Reactor Trip Response	b,c,e
ES-0.2	SI Termination	b,c,e
ES-0.3	Natural Circulation Cooldown	b,c,e
E-1	Loss of Reactor or Secondary Coolant	a
ES-1.1	Post-LOCA Cooldown	a,b,c,e,f
ES-1.2	Transfer to CNTMT Sump	
ES-1.3	Transfer to Hot Leg Recirc	
E-2	Faulted Steam Generator Isolation	
E-3	Steam Generator Tube Rupture	e
ES-3.1	SI Termination Following SGTR	b,c
ES-3.2	Post SGTR Cooldown Using Backfill	f
ES-3.3	Post SGTR Cooldown by Ruptured S/G Depressurization	F
E-FOP	Foldout Page	a (continuously monitored)

INSTR	TITLE	RCP TRIP OPERATIONS*
F.0.2	Core Cooling Status Tree	a
FR-C.1	Response to Inadequate Core Cooling (ICC)	e,f,g
FR-P.1	Response to Pressurized Thermal Shock	b,f
FR-Z.1	Response to Phase B CNTMT Pressure	a,f
FR-I.3	Responsible Voids in Reactor Vessel	b,c,d

*(a) RCP trip using WOG alternate criteria
(b) RCP restart
(c) Decay heat removal by natural circulation
(d) Primary system void removal
(e) Use of steam generators with and without RCPs operating
(f) RCP trip for other reasons
(g) Special restart without support condition