TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

October 29, 1982

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

In the Matter of the	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
		50-296

Enclosed is our response for Browns Ferry to the June 30, 1982 letter from D. B. Vassallo to H. G. Parris requesting additional information on NUREG-0737, Item II.D.1, "Testing of Relief and Safety Valves." This response has been prepared in coordination with the BWR Owners' Group. An extension to the original response deadline of August 30, 1982 was discussed with R. J. Clark of your staff.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L M Mills

L. M. Mills, Manager
Nuclear Licensing

Subscribed and sworn to before
me this 29th day of Oct. 1982.

Paulette H. White
Notary Public

My Commission Expires 9-5-84

Enclosure
cc (Enclosure):
U.S. Nuclear Regulatory Commission
Region II
ATTN: James P. O'Reilly, Regional Administrator
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Mr. R. J. Clark
Browns Ferry Project Manager
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, Maryland 20814

A046

ENCLOSURE

BROWNS FERRY NUCLEAR PLANT
RESPONSE TO NRC CONCERNS REGARDING
NUREG-0737, ITEM.D.1 - TESTING
OF SAFETY RELIEF VALVES

NRC QUESTION 1

The test program utilized a "rams head" discharge pipe configuration. Browns Ferry utilizes a "tee" quencher configuration at the end of the discharge line. Describe the discharge pipe configuration used at Browns Ferry and compare the anticipated loads on valve internals in the plant configuration to the measured loads in the test program. Discuss the impact of any differences in loads on valve operability.

RESPONSE TO QUESTION 1

The safety/relief valve discharge piping configuration at Browns Ferry utilizes a "tee" quencher at the discharge pipe exit. The average length of the 13 SRV discharge lines (SRVDL) is about 118 feet, and the submergence length in the suppression pool is approximately 10 feet. The SRV test program utilized a ramshead at the discharge pipe exit, a pipe length of 112 feet, and a submergence length of approximately 13 feet. Loads on valve internals during the test program are larger than loads on valve internals in the Browns Ferry configuration for the following reasons:

1. No dynamic mechanical load originating at the "tee" quencher is transmitted to the valve in the Browns Ferry configuration because there is at least one anchor point between the valve and the tee quencher.

2. The first length of the segment of piping downstream of the SRV in the test facility was longer than the Browns Ferry piping thereby resulting in a bounding dynamic mechanical load on the valve in the test program due to the larger moment arm between the SRV and the first elbow. The first segment length in the test facility is 12 feet, whereas the maximum length in the plant configuration is approximately 5 feet.

3. Dynamic hydraulic loads (backpressure) are experienced by the valve internals in the Browns Ferry configuration. The backpressure loads may be either (1) transient backpressures occurring during valve actuation, or (2) steady-state backpressures occurring during steady-state flow following valve actuation.

   (a) The key parameters affecting the transient backpressures are the fluid pressure upstream of the valve, the valve opening time, the fluid inertia in the submerged SRVDL, and the SRVDL air volume. Transient backpressures increase with higher upstream pressure, shorter valve opening times, greater line submergence, and smaller SRVDL air volume. The transient backpressure in the test program was maximized by utilizing a submergence of 13 feet which is greater than Browns Ferry and a pipe length of 112 feet which is less than Browns Ferry. The maximum transient backpressure occurs with high pressure steam flow conditions. The transient backpressure for the alternate shutdown cooling mode of operation is always much less than the design for steam flow conditions because of the lower upstream pressure and the longer valve opening time.

(b) The steady-state backpressure in the test program was maximized by utilizing an orifice plate in the SRVDL above the water level and before the ramshead. The orifice was sized to produce a backpressure greater than that calculated for any of the Browns Ferry SRVDLs.

The differences in the line configuration between the Browns Ferry plant and the test program as discussed above result in the loads on the valve internals for the test facility which bound the actual Browns Ferry loads. As additional consideration in the selection of the ramshead for the test facility was to allow more direct measurement of the thrust load in the final pipe segment. Utilization of a "tee" quencher in the test program would have required quencher supports that would unnecessarily obscure accurate measurement of the pipe thrust loads. For the reasons stated above, differences between the SRVDL configurations in Browns Ferry and the test facility will not have any adverse effect on SRV operability at Browns Ferry relative to the test facility.

NRC QUESTION 2

The test configuration utilized no spring hangers as pipe supports. Plant specific configurations do use spring hangers in conjunction with snubber and rigid supports. Describe the safety-relief valve pipe supports used at Browns Ferry and compare the anticipated loads on valve internals for the plant pipe supports to the measured loads in the test program. Describe the impact of any differences in loads on valve operability.

RESPONSE TO QUESTION 2

The Browns Ferry SRVDLs are supported by a combination of snubbers, rigid supports, and spring hangers. The locations of snubbers and rigid supports at Browns Ferry are such that the locations of such supports in the BWR generic test facility are prototypical, i.e., in each case (Browns Ferry and the test facility) there are supports near each change of direction in the pipe routing. Additionally, several SRVDLs at Browns Ferry have spring hangers all of which are located in the drywell. The spring hangers, snubbers, and rigid supports were designed to accommodate combinations of loads resulting from piping dead weight, thermal conditions, seismic and suppression pool hydrodynamic events, and a high pressure steam discharge transient.

The dynamic load effects on the piping and supports of the test facility due to the water discharge event (the alternate shutdown cooling mode) were found to be significantly lower than corresponding loads resulting from the high pressure steam discharge event. As stated in NEDE-24988-P, this finding is considered generic to all BWRs since the test facility was designed to be prototypical of the features pertinent to this issue.

During the water discharge transient there will be significantly lower dynamic loads acting on the snubbers and rigid supports than during the steam discharge transient. This will more than offset the small increase in the dead load on these supports due to the weight of the water during the alternate shutdown cooling mode of operation. Therefore, design adequacy of the snubbers and rigid supports is ensured as they are designed for the larger steam discharge transient loads.

This question addresses the design adequacy of the spring hangers with respect to the increased dead load due to the weight of the water during the liquid discharge transient. As was discussed with respect to snubbers and rigid supports, the dynamic loads resulting from liquid discharge during the alternate shutdown cooling mode of operation are significantly lower than those from the high pressure steam discharge. Therefore, it is believed that sufficient margin exists in the Browns Ferry piping system design to adequately offset the increased dead load on the spring hangers in an unpinned condition due to a water filled condition. Furthermore, the effect of the water dead weight load does not affect the ability of SRVs to open to establish the alternate shutdown cooling path since the loads occur in the SRVDL only after valve opening.

NRC QUESTION 3

Report NEDE-24988-P did not identify any valve functional deficiencies or anomalies encountered during the test program. Describe the impact on valve safety function of any valve functional deficiencies or anomalies encountered during the program.

RESPONSE TO QUESTION 3

No functional deficiencies or anomalies of the safety-relief or relief valves were experienced during the esting at Wyle Laboratories for compliance with the alternate shutdown cooling mode requirement. All of the valves subjected to test runs, valid and invalid, opened and closed without loss of pressure integrity or damage. Anomalies encountered during the test program were all due to failures of test facility instrumentation, equipment, data acquisition equipment, or deviation from the approved test procedure.

The test specification for each valve required six runs. Under the test procedure, any anomaly caused the test run to be judged invalid. All anomalies were reported in the test report. The Wyle test log sheet for the Target Rock 2-stage valve tests is attached, which is the valve used at Browns Ferry.

Each Wyle test report for the respective valves identifies each test run performed and documents whether or not the test run is valid or invalid and states the reason for considering the run invalid. No anomaly encountered during the required test program affects any valve safety or operability function.

All valid test runs are identified in table 2.2-1 of NEDE-24988-P. The data presented in table 4.2-1 for each valve were obtained from the table 2.2-1 test runs and were based upon the selection criteria of:

(a) Presenting the maximum representative loading information obtained from the steam run data,

(b) Presenting the maximum representative water loading information obtained from the $15^{\circ}F$ subcooled water test data,

(c) Presenting the data on the only test run performed for the $50^{\circ}F$ subcooled water test condition.

OPERABILITY TEST REPORT

FOR

TARGET ROCK 6X10 SRV

FOR

LOW PRESSURE WATER TESTS

FOR

GENERAL ELECTRIC COMPANY

175 Curtner Avenue
San Jose, California

TABLE I

TEST LOG FOR SRV TR-1

Test No.	Test Media	Load Line Configuration	Test Date	Remarks
301	Steam	I	3/17/81	Acceptable
302	Water	I	3/17/81	GN_2 Regulator failed. Data not acceptable.
303	Water	I	3/17/81	Acceptable
304	Steam	I	3/17/81	Acceptable
305	Water	I	3/18/81	Acceptable
306	Steam	I	3/18/81	Acceptable
307	Water	I	3/18/81	Acceptable
308	Water	I	3/18/81	Special test at elevated temperature and low pressure requested by G.E.

on the safety and relief valve. These events were identified by evaluating the initial events described in Regulatory Guide 1.70, revision 2 with and without the additional conservatism of a single active component failure or operator error postulated in the event sequence. It was concluded from this evaluation that the alternate shutdown cooling mode is the only expected event which will result in liquid at the valve inlet. Consequently, this was the event simulated in the S/RV test program. This conclusion and the test results applicable to Browns Ferry are discussed below. The alternate shutdown cooling mode of operation has been described in the response to NRC question 5.

The S/RV inlet fluid conditions tested in the BWR Owners' Group S/RV test program, as documented in NEDE-24988-P, are 15-50°F subcooled liquid at 20-250 psig. These fluid conditions envelope the conditions expected to occur at Browns Ferry in the alternate shutdown cooling mode of operation.

The BWR Owners' Group identified 13 events by evaluating the initiating events described in Regulatory Guide 1.70, revision 2 with the additional conservatism of a single active component failure or operator error postulated in the event sequence. These events and the plant-specific features that mitigate these events are summarized in table 1. Of these 13 events, only 11 are applicable to Browns Ferry because of its design and specific plant configuration. Two events, namely numbers 5 and 10, are not applicable to Browns Ferry because the design does not include a HPCS.

NRC QUESTION 4

The purpose of the test program was to determine valve performance under conditions anticipated to be encountered in the plants. Describe the events and anticipated conditions at Browns Ferry Nuclear Plant for which the valves are required to operate and compare these plant conditions to the conditions in the test program. Describe the plant features assumed in the event evaluations used to scope the test program and compare them to plant features at Browns Ferry. For example, describe high level trips to prevent water from entering the steam lines under high pressure operating conditions as assumed in the test event and compare them to trips used at Browns Ferry.

RESPONSE TO NRC QUESTION 4

The purpose of the safety relief valve (S/RV) test program was to demonstrate that the S/RV will open and reclose under all expected flow conditions. The expected valve operating conditions were determined through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, revision 2. Single failures were applied to these analyses so that the dynamic forces on the safety and relief valves would be maximized. Test pressures were the highest predicted by conventional safety analysis procedures. The BWR Owners' Group, in its enclosure to the September 17, 1980 letter from D. B. Waters to R. H. Vollmer, identified 13 events which may result in liquid or 2-phase S/RV inlet flow that would maximize the dynamic forces

As discussed above, the BWR Owners' Group evaluated transients including single failures that would maximize the dynamic forces on the safety relief valves. As a result of this evaluation, the alternate shutdown cooling mode is the only expected event involving liquid or 2-phase flow. Consequently, this event was simulated in the BWR S/RV test program. The fluid conditions and flow conditions tested in the BWR Owners' Group test program conservatively envelope the Browns Ferry plant-specific fluid conditions expected for the alternate shutdown cooling mode of operation.

For the 11 remaining events, Browns Ferry specific features, such as trip logic, power supplies, instrument line configuration, alarms and operator actions, have been compared to the base case analysis presented in the BWR Owners' Group submittal of September 17, 1980. The comparison has demonstrated that in each case, the base case analysis is applicable to Browns Ferry because the base case analysis does not include any plant features which are not already present in the Browns Ferry design except for RCIC initiation on high drywell pressure. This assumption makes the vessel overfill analysis conservative for Browns Ferry. For these events, table 1 lists the Browns Ferry specific features that are included in the base case analyses presented in the BWR Owners' Group submittal of September 17, 1980. All features included in this base case analysis are similar to plant features in the Browns Ferry design. Furthermore, the time available for operator action is expected to be longer in the Browns Ferry plant than in the base case analysis for each case where operator action is required.

Event 7, the alternate shutdown cooling mode of operation, is the only expected event which will result in liquid or 2-phase fluid at the S/RV inlet. Consequently, this event was simulated in the BWR S/RV test program. In Browns Ferry, this event involves flow of water which is expected to be subcooled at a pressure less than 250 psig. The test conditions clearly envelope these plant conditions.

TABLE 1 - EVENTS EVALUATED

PLANT FEATURES	#1 FW Cont. Fail., FW L8 Trip Failure	#2 Press. Reg. Fail.	#3 Transient HPCI, HPCI L8 Trip Failure	#4 Transient RCIC, RCIC L8 Trip Failure	#5 Transient HPCS, HPCS L8 Trip Failure	#6 Transient RCIC Hd. Spr.	#7 Alt. Shutdown Cooling, Shutdown Suction Unavailable	#8 MSL Brk OSC	#9 SBA, RCIC, RCIC L8 Trip Failure	#10 SBA, HPCS, HPCS L8 Trip Failure	#11 SBA, HPCI, HPCI L8 Trip Failure	#12 SBA, Depress. & ECCS Over., Operator Error	#13 LBA, ECCS Overf Brk Isol
High Water Level 7 Alarm	X / S		X / S	X / S	X / NA				X / S	X / NA	X / S	X / S	X / S
High Drywell Pressure Alarm													
FW Level 8 Trip	X / S	X / S											
RCIC Level 8 Trip			X / S	X / S	X / NA				X / S	X / NA	X / S		X / S
HPCS Level 8 Trip				X / NA	X / NA				X / NA	X / NA	NA		X / NA
HPCI Level 8 Trip			X / S	X / S					X / S		X / S		X / S
HPCI/S and RCIC Initiation on Low Water Level	X / S	X / S	X / S	X / S	X / NA	X / NA		X / S	X / S				X / S
HPCI/S initiation on High Drywell Pressure			X / S	X / S					X / S	X / NA	X / S	X / S	X / S
RCIC Initiation on High Drywell Pressure													X / NA

TABLE 1 - EVENTS EVALUATED

PLANT FEATURES	#1 FW Cont. Fail., FW L8 Trip Failure	#2 Press. Reg. Fail.	#3 Transient HPCI, HPCI L8 Trip Failure	#4 Transient RCIC, RCIC L8 Trip Failure	#5 Transient HPCS, HPCS L8 Trip Failure	#6 Transient RCIC Hd. Spr.	#7 Alt. Shutdown Cooling, Shutdown Suction Unavailable	#8 MSL Brk OSC	#9 SBA, RCIC, RCIC L8 Trip Failure	#10 SBA, HPCS, HPCS L8 Trip Failure	#11 SBA, HPCI, HPCI L8 Trip Failure	#12 SBA, Depress. & ECCS Over., Operator Error	#13 LBA, ECCS Overf Brk Isol
MSIV Closure on High Radiation								X / S					
Reactor Scram on Turbine Trip	X / S	X / S											
Reactor Scram on Neutron Flux Monitor		X / S											
Reactor Scram on MSIVs Closure		X / S											
Reactor Scram on High Radiation								X / S					
Reactor Scram on High Drywell Pressure									X / S	X / NA	X / S	X / S	X / S
Reactor Scram on Low Water Level													X / S
Reactor Isolation on Low Water Level													X / S

KEY: X - Feature considered in Base Case Analysis
S - Feature in Plant Specific Design
NA - Not Applicable

TABLE 1 - EVENTS EVALUATED

PLANT FEATURES	#1 FW Cont. Fail., FW L8 Trip Failure	#2 Press. Reg. Fail.	#3 Transient HPCI, HPCI L8 Trip Failure	#4 Transient RCIC, RCIC L8 Trip Failure	#5 Transient HPCS, HPCS L8 Trip Failure	#6 Transient RCIC Hd. Spr.	#7 Alt. Shutdown Cooling, Shutdown Suction Unavailable	#8 MSL Brk OSC	#9 SBA, RCIC, RCIC L8 Trip Failure	#10 SBA, HPCS, HPCS L8 Trip Failure	#11 SBA, HPCI, HPCI L8 Trip Failure	#12 SBA, Depress. & ECCS Over., Operator Error	#13 LBA, ECCS Overf Brk Isol
Low Pressure ECCS Initiation on High Drywell Pressure												X / S	X / S
Low Pressure Initiation on Low Water Level													X / S
FW Pumps Trip on Low Suction Pressure	X / S												
HPCS Trip on High Backpressure			X / S								X / S		
RCIC Trip on High Backpressure				X / S					X / S				
Turbine Trip on Vessel High Level	X / S	X / S											
MSIVs Closure on Low Turbine Inlet Pressure	X / S	X / S						X / S					
MSIVs Closure on High Steam Flow		X / S						X / S					
MSIVs Closure on High Steam Tunnel Temperature								X / S					

NRC QUESTION 5

The valves are likely to be extensively cycled in a controlled depressurization mode in a plant-specific application. Was this mode simulated in the test program? What is the effect of this valve cycling on valve performance and probability of the valve to fail open or to fail closed?

RESPONSE TO NRC QUESTION 5

The BWR safety/relief valve (SRV) operability test program was designed to simulate the alternate shutdown cooling mode which is the only expected liquid discharge event for Browns Ferry. The sequence of events leading to the alternate shutdown cooling mode is given below.

Following normal reactor shutdown, the reactor operator depressurizes the reactor vessel by opening the turbine bypass valves and removing heat through the main condenser. If the main condenser is unavailable, the operator could depressurize the reactor vessel by using the SRVs to discharge steam to the suppression pool. If SRV operation is required, the operator cycles the valves in order to ensure that the cooldown rate is maintained within the technical specification limit of $100^{\circ}F$ each hour. When the vessel is depressurized, the operator initiates normal shutdown cooling by use of the RHR system. If that system is unavailable because the valve on the RHR shutdown cooling suction line fails to open, the operator initiates the alternate shutdown cooling mode.

NRC QUESTION 6

Describe how the values of valve C_v's in report NEDE-24988-P will be used at Browns Ferry. Show that the methodology used in the test program to determine the valve C_v will be consistent with the application at Browns Ferry.

RESPONSE TO NRC QUESTION 6

The flow coefficient C_v for the Target Rock 2-stage safety-relief valve (SRV) utilized in Browns Ferry was determined in the generic SRV test program (NEDE-24988-P). The average flow coefficient calculated from the test results for the Target Rock 2-stage valve is reported in table 5.2-1 of NEDE-24988-P. This test value has been used by TVA to confirm that the liquid discharge flow capacity of the Browns Ferry SRVs will be sufficient to remove core decay heat when injecting into the reactor pressure vessel (RPV) in the alternate shutdown cooling mode. The C_v value determined in the SRV test demonstrates that the Browns Ferry SRVs are capable of returning the flow injected by the RHR or CS pump to the suppression pool.

If it were necessary for the operator to place the Browns Ferry plant in the alternate shutdown cooling mode, he would ensure that adequate core cooling was being provided by monitoring the following parameters: RHR or CS flow rate, reactor vessel pressure, and reactor vessel temperature.

The flow coefficient for the Target Rock 2-stage valve reported in NEDE-24988-P was determined from the SRV flow rate when the valve inlet was pressurized to approximately 250 psig. The valve flow rate was measured with the supply line flow venturi upstream of the steam chest. The C_v for the valve was calculated using the nominal measured pressure differential between the valve inlet (steam chest) and three feet downstream of the valve and the corresponding measured flowrate. Furthermore, the test conditions and test configuration were representative of Browns Ferry plant conditions for the alternate shutdown cooling mode, e.g., pressure upstream of the valve, fluid temperature, friction losses, and liquid flowrate. Therefore, the reported C_v values are appropriate for application to Browns Ferry.