Commonwealth Edison
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Address Reply to: Post Office Box 767
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April 27, 1983

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

Subject: Quad Cities Station Units 1 and 2
NUREG 0737 Item II.D.1
Additional Information
NRC Docket Nos. 50-254 and 50-265

Reference (a): D. B. Vassallo letter to L. O. DelGeorge
dated January 4, 1983.

Dear Mr. Denton:

Reference (a) requested that the Commonwealth Edison Company provide, within sixty (60) days of receipt, certain information concerning NUREG 0737 Item II.D.1 "Performance Testing of BWR Safety/Relief Valves" for our Quad Cities Station.

The Attachment to this letter provides the requested information. We have compared the test facility configuration and information against that of our Quad Cities Station in order to assess the applicability of the resultant test facility data. However, this was performed only for the load combinations which result from the actuation of the valve and subsequent water flow as anticipated during the alternate cooling mode. These are the conditions which are commensurate with those of the test, thereby providing a common basis for comparison. No other loads (i.e. seismic) were considered.

It is our judgement that for the most part this information adequately demonstrates the applicability of the results of the BWR Owners Group Generic Test Report (NEDE-24988-P) to our Quad Cities Units 1 and 2. However, as stated in the attached response an evaluation the adequacy of the spring hangers with respect to increased dead weight will be performed. The results of this evaluation will be submitted by August 1, 1982.

A001

To the best of my knowledge and belief, the statements contained in the Attachment are true and correct. In some respects these statements are not based on my personal knowledge but upon information furnished by other Commonwealth Edison employees and Consultants. Such information has been reviewed in accordance with Company practice and I believe it to be reliable.

Please address any further questions that you or your staff may have concerning this matter to this office.

One (1) signed original and forty (40) copies of this letter with Attachment are provided for your use.

Very truly yours,

B. Rybak

B. Rybak
Nuclear Licensing Administrator

lm

Attachment

cc: RIII Inspector - Quad Cities
R. Bevan - NRR

6226N

NRC QUESTION 1

The test program utilized a "ramshead" discharge pipe configuration. Quad Cities Station Units 1 and 2 utilizes a "tee" quencher configuration at the end of the discharge line. Describe the discharge pipe configuration used at Quad Cities Station Units 1 and 2 and compare the anticipated loads on valve internals in the Quad Cities Station Units 1 and 2 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 Quad Cities Station Units 1 and 2 utilizes a "tee"quencher at the discharge pipe exit. The average total length of the 5 SRV discharge lines (SRVDL) between the SRV and quencher is 95.2 ft and the submergence length in the suppression pool is approximately 17.5 ft. The SRV test program utilized a ramshead at the discharge pipe exit, a pipe length of 112 ft and a submergence length of approximately 13 ft. Loads on valve internals during the test program are larger than loads on valve internals in the Quad Cities Station Units 1 and 2 configuration for the following reasons:

1. No dynamic mechanical load originating at the "tee" quencher is transmitted to the valve in the Quad Cities Station Units 1 and 2 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 Quad Cities Station Units 1 and 2 piping, thereby resulting in a bounding dynamic mechanical load on the valve in the test program. The first segment length in the test facility is 12 ft whereas this length is an average of 1.2 ft in the plant configuration.

3. Dynamic hydraulic loads (backpressure) are experienced by the valve internals in the Quad Cities Station Units 1 and 2 configuration. The backpressure loads may be either (i) transient backpressure occurring during valve actuation, or (ii) 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 and greater line submergence, and decrease with greater SRVDL air volume. The maximum transient backpressure occurs with high pressure steam flow conditions - a condition that Quad Cities Station Units 1 and 2 have experienced on numerous occassions during operation. The transient backpressure for the alternate shutdown cooling mode of operation is always much less than that for the design for steam flow conditions because of the lower upstream pressure and the slower 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 Quad Cities Station Units 1 and 2 SRVDLs.

Because of the differences in the line configuration between the Quad Cities Station Units 1 and 2 and the test program, as discussed above, the resultant steady-state loads on the valve internals for the test facility bound the actual Quad Cities Station Units 1 and 2 loads. An 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, difference between the SRVDL configurations in Quad Cities Station Units 1 and 2 the test facility result in more severe loads during the tests for the alternate shutdown cooling mode of operation; therefore, SRV operability at Quad Cities Station Units 1 and 2 is assured by the tests.

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 Quad Cities Station Units 1 and 2 and compare the anticipated loads on valve internals for the Quad Cities Station Units 1 and 2 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 Quad Cities Station Units 1 and 2 safety/relief valve discharge lines (SRVDLs) are supported by a combination of snubbers, rigid supports, and spring hangers. The locations of snubbers and rigid supports at Quad Cities Station Units 1 and 2 are such that the location of such supports in the BWR generic test facility is prototypical, i.e., in each case (Quad Cities Station Units 1 and 2 and the test facility) there are supports near each change of direction in the pipe routing. Additionally, each SRVDL at Quad Cities Station Units 1 and 2 has only 1 or 2 spring hangers, all of which are located in the drywell. The spring hangers, snubbers, and rigid supports were designed to accomodate combinations of loads resulting from piping dead weight, thermal conditions, seismic and suppression pool hydrodynamic events, and a high pressure steam discharge transient during a steam discharge event.

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 resulting from the valve operation and subsequent water flow acting on the snubbers and rigid supports than during the steam discharge transient. This will more than offset the small increase in the deadweight 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 assured because 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 deadweight load due to the weight of 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, sufficient margin should exist in the Quad Cities Station Units 1 and 2 piping system design to adequately offset the increased deadweight load on the spring hangers in an unpinned condition due to a water filled condition. Nevertheless, the design margin existing in the SRVDL used for the alternate shutdown cooling mode of operation will be quantitatively evaluated. Furthermore, from a safe shutdown viewpoint, the effect of the water deadweight load does not affect the ability of SRVs to open and to establish the alternate shutdown cooling path since the loads occur in the SRVDL only after valve opening. Consequently, it is concluded that safe shutdown can be achieved using the alternate shutdown cooling mode of operation because valve operability has been demonstrated for water flow conditions.

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 testing at Wyle Laboratories for compliance with the alternate shutdown cooling mode requirement. All 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 failure 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 Laboratories test log sheet for the Dresser 6X8 and Target Rock Three State valve tests are attached. These valves are used in the Quad Cities Station Units 1 and 2.

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°F subcooled water test data,

(c) Presenting the data on the only test run performed for the 50°F subcooled water test condition.

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 quad Cities Station Units 1 and 2 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 Quad Cities Station Units 1 and 2. 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 Quad Cities Station Units 1 and 2.

RESPONSE TO NRC QUESTION 4

The purpose of the S/RV test program was to demonstrate that the Safety/Relief Valves (S/RVs) 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 their enclosure to the September 17, 1980 letter from D. B. Waters to R. H. Vollmer, identified 13 events which may result in liquid or two-phase S/RV inlet flow that would maximize the dynamic forces on the safety and relief valves. 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 or two-phase fluid at the valve inlet. Consequently, this was the event simulated in the S/RV test program. This conclusion and the test results applicable to Quad Cities Station Units 1 and 2 are discussed below. The alternate shutdown cooling mode of operation is 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°F to 50°F subcooled liquid at 20 psid to 250 psid. These fluid conditions envelope the conditions expected to occur at Quad Cities Station Units 1 and 2 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 events sequence. These events and the plant-specific features that mitigate these events, are summarized in Table 1. Of these 13 events, only 8 are applicable to the Quad Cities Station Units 1 and 2 plant because of its design and specific plant configuration. Five events, namely, 2, 5, 8, 10, and 13 are not applicable to the Quad Cities Station Units 1 and 2 plant for the reasons listed below:

a. Event 2 - Results in steam flow only because the S/RVs are located higher than the MSIVs.

b. Event 5 - There is no HPCS system at Quad Cities Station Units 1 and 2.

c. Event 8 - Results in steam flow only because the S/RVs are located higher than the MSIVs.

d. Event 10 - There is no HPCS system at Quad Cities Station Units 1 and 2.

e. Event 13 - There are no procedures requiring break isolation. The operator is trained to respond to high water level indication and alarms before the vessel is filled to the MSL level.

For these eight remaining events, the Quad Cities Station Units 1 and 2 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

has demonstrated that in each case, the base case analysis is applicable to Quad Cities Station Units 1 and 2 because the base case analysis does not include any plant features which are not already present in the Quad Cities Station Units 1 and 2 design. For events, 1, 3, 4, 6, 9, 11, and 12, Table 1 demonstrates that the Quad Cities Station Units 1 and 2 specific features are included in the base case analysis presented in the BWR Owners Group submittal of September 17, 1980. It is seen from Table 1, that all plant features assumed in the event evaluation are also existing features in the Quad Cities Station Units 1 and 2. All features included in this base case analysis are similar to plant features in the Quad Cities Station Units 1 and 2 design. Furthermore, the time available for operator action is expected to be longer at the Quad Cities Station Units 1 and 2 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 two-phase fluid at the S/RV inlet. Consequently, this event was simulated in the BWR S/RV test program. At Quad Cities Station Units 1 and 2, this event involves flow of subcooled water (approximately 15°F to 50°F subcooled) at a pressure of approximately 20 psig to 250 psig. The test conditions clearly envelope these plant conditions.

As discussed above, the BWR Owners Group evaluated transients including single active 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 two-phase flow. Consequently this event was tested in the BWR S/RV test program. The fluid conditions and flow conditions tested in the BWR Owners Group test program conservatively envelope the Quad Cities Station Units 1 and 2 plant-specific fluid conditions expected for the alternate shutdown cooling mode of operation.

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 or two-phase flow discharge event for Quad Cities Station Units 1 and 2. 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 SRV's to discharge steam into the suppression pool. If SRV operation is required, the operator cycles the valves in order to assure that the cooldown rate is maintained within the technical specification limit of 100°F per 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.

As discussed in the preceeding paragraph, if the normal equipment is postulated to be unavailable, then the operator will initiate the alternate shutdown cooling mode of operation. For alternate shutdown cooling, the operator opens one or more SRVs and initiates either an RHR or core spray pump utilizing the suppression pool as the suction source. The reactor vessel is filled such that water is allowed to flow into the main steam lines and out of the SRV(s) and back to the suppression pool. Cooling of the system is provided by use of an RHR heat exchanger. As a result, an alternate cooling mode is maintained.

In order to assure continuous long term heat removal, the SRV is kept open and no cycling of the valve is performed. In order to control the reactor vessel cooldown rate, the operator is instructed to limit flow into the vessel by throttling the injection valve. Consequently, no cycling of the SRV is required for the alternate shutdown cooling mode, and no cycling of the SRV was performed for the generic BWR SRV operability test program.

The ability of the Quad Cities Station Units 1 and 2 SRV to be extensively cycled for steam discharge conditions has been confirmed during steam discharge qualification testing of the valve by the valve operator. Based on the qualification testing of the SRVs, the cycling of the valves in a controlled depressurization mode for steam discharge conditions will not adversely affect valve performance and thus the probability of the valve to fail open or closed is extremely low.

NRC QUESTION 6

Describe how the values of valve C_v's in report NEDE-24988-P will be used at Quad Cities Station Units 1 and 2. Show that the methodology used in the test program to determine the valve C_v will be consistent with the application of Quad Cities Station Units 1 and 2.

RESPONSE TO NRC QUESTION 6

The flow coefficient, C_v, for the Dresser 6X8 and Target Rock Three Stage Safety relief valves (SRVs) utilized in Quad Cities Station Units 1 and 2 was determined in the generic SRV test program (NEDE-24988-P). The average flow coefficient calculated from the test results for the Dresser 6X8 and Target Rock Three Stage valves is reported in Table 5.2-1 of NEDE-24988-P. This test value has been used by Commonwealth Edison Company to confirm that the liquid discharge flow capacity of the Quad Cities Station Units 1 and 2 SRVs will be sufficient to remove core decay heat when injecting water into the reactor pressure vessel (RPV) in the alternate shutdown cooling mode. The C_v value determined in the SRV test demonstrates that the Quad Cities Station Units 1 and 2 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 Quad Cities Station Units 1 or 2 in the alternate shutdown cooling mode, he would assure that adequate core cooling was being provided by monitoring the following parameters: RHR or CS flow rate, reactor vessel pressure and reactor coolant temperature.

The flow coefficients for the Dresser 6X8 and Target Rock Three Stage valves reported in NEDE-24988-P were 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 3 ft downstream of the valve and the corresponding measured flowrate. Furthermore, the test conditions and test configuration were representative of Quad Cities Station Units 1 and 2 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 the Quad Cities Station Units 1 and 2 plant.

OPERABILITY TEST REPORT

FOR

DRESSER 6X8 SRV

FOR

LOW PRESSURE WATER TESTS

FOR

GENERAL ELECTRIC COMPANY

175 Curtner Avenue
San Jose, California

TABLE I
OPERABILITY TEST LOG, SRV DR-1

TEST NO.	MEDIA	LOAD LINE CONFIGURATION	TEST DATE	REMARKS
601	Steam	1	4/15/81	Back pressure too high.
602	Steam	1	4/15/81	Installed 5.75" orifice. Test acceptable.
603	Water	1	4/15/81	Steam chest pressure low.
604	Water	1	4/15/81	Test acceptable.
605	Steam	1	4/16/81	No data on tape.
606	Water	1	4/16/81	Test acceptable.
607	Steam	1	4/16/81	Test acceptable.
608	Water	1	4/16/81	Test acceptable.
609	Steam	1	4/16/81	Rerun of Test # 605. Test acceptable.

Replaced L1 snubber for 608 and 609.

OPERABILITY TEST REPORT

FOR

TARGET ROCK THREE STAGE SRV

FOR

LOW PRESSURE WATER TESTS

FOR

GENERAL ELECTRIC COMPANY

175 Curtner Avenue
San Jose, California

TABLE I

TEST LOG FOR SRV TR-2

TEST NO.	TEST MEDIA	LOAD LINE CONFIGURATION	TEST DATE	REMARKS
201	Steam	I	3/10/81	Back pressure low. Test Unacceptable.
202	Steam	I	3/10/81	Installed 6.8" orifice. Test Acceptable.
203	Water	I	3/10/81	Test Acceptable.
204	Steam	I	3/11/81	Test Acceptable.
205	Water	I	3/11/81	Pipe loads high. See NOA # 5.
206	Steam	I	3/11/81	Test Acceptable.
207	Water	I	3/11/81	Not Acceptable. Low steam chest pressure.
208	Water	I	3/11/81	Test Acceptable. Water temperature low.
209	Water	I	3/30/81	Test Acceptable.
210	Water	I	3/30/81	Test Acceptable.
211	Water	I	3/30/81	Test Acceptable.

TEST REPORT NO. 17476-03
Revision A

NOTICE OF ANOMALY

NOTICE NO. 5 P. O. NUMBER: 205-XH212 WYLE JOB NO. 17476-03

CONTRACT NUMBER: N/A

CATEGORY: ☐ SPECIMEN ☐ PROCEDURE ☒ TEST EQUIPMENT DATE: 3/14/81

TO: General Electric Company ATTN: Mr. R. Miller

PART NAME: Target Rock 3-Stage SRV PART NO. N/A

TEST: Low Pressure Water I. D. NO. TR-2

SPECIFICATION: WTP 17450-01 PARA. NO. N/A

NOTIFICATION MADE TO: J. Mross/A. Sallman DATE: 3/14/81

NOTIFICATION MADE BY: L. Millsaps VIA: Verbal

REQUIREMENTS:

N/A

DESCRIPTION OF ANOMALY:

When the water control valve was opened to initiate the test, the entire system was
subjected to a shock wave similar to water hammer. As a result, loads of approximately
10,000 and 16,000 pounds were observed at Struts 1 and 2. Review of the recorded data
showed no abnormal pressure in the discharge line, but did show sharply varying pres-
sure in the steam chest and inlet water pipe.

DISPOSITION – COMMENTS – RECOMMENDATIONS:

The recorded data shows that the anomaly occurred in the inlet piping and/or steam chest
and, therefore, was not caused by the SRV. The probable cause was the forming of vapor
in the inlet pipe because of the higher water temperature (233°F) and the low pressure
(8 to 10 psig). The vapor then compressed when subjected to the higher pressure water
(300 psig), thus causing a shock wave in the water system. Since the discharge pipe
loads were caused by the shock wave rather than the SRV, the data must be considered
invalid.

The test was not repeated. However, three other water tests were conducted on this SRV,
and all data was consistent. In addition, water tests were performed on a two-stage
Target Rock SRV, and no anomalies occurred.

It is, therefore, recommended that the test not be repeated.

DISTRIBUTION:
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1 Copy: Operations Director

TEST WITNESS —
—
REPRESENTING —

ENGINEER L. Millsaps
QUALITY CONTROL
PROJECT MANAGER

WYLE LABORATORIES
Huntsville Facility

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
gh Water Level 7 Alarm	X/S		X/S	X/S	X/NA				X/S	X/NA	X/S	X/S	X/NA
gh Drywell Pressure Alarm													
W Level 8 Trip	X/S	X/NA											
IC Level 8 Trip			X/S	X/S	X/NA				X/S	X/NA	X/S		X/NA
PCS Level 8 Trip				X/NA	X/NA				X/NA	X/NA			X/NA
PCI Level 8 Trip			X/S	X/S					X/S		X/S		X/NA
PCI/S and RCIC Initiation on Low Water Level	X/S	X/NA	X/S	X/S	X/NA	X/S		X/NA	X/S				X/NA
PCI/S initiation on High Drywell Pressure			X/S	X/S					X/S	X/NA	X/S	X/S	X/NA
IC Initiation on High Drywell Pressure													X/NA

RII:rf/G09163-1*

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 / NA
Low Pressure Initiation on Low Water Level													X / NA
FW Pumps Trip on Low Suction Pressure	X / S												
HPCS Trip on High Backpressure			X / NA								X / NA		
RCIC Trip on High Backpressure				X / S					X / S				
Turbine Trip on Vessel High Level	X / S	X / NA											
MSIVs Closure on Low Turbine Inlet Pressure	X / S	X / NA						X / NA					
MSIVs Closure on High Steam Flow		X / NA						X / NA					
MSIVs Closure on High Steam Tunnel Temperature								X / NA					

FRII:rf/G09163-2*

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 / NA					
Reactor Scram on Turbine Trip	X / S	X / NA											
Reactor Scram on Neutron Flux Monitor		X / NA											
Reactor Scram on MSIVs Closure		X / NA											
Reactor Scram on High Radiation								X / NA					
Reactor Scram on High Drywell Pressure									X / S	X / NA	X / S	X / S	X / NA
Reactor Scram on Low Water Level													X / NA
Reactor Isolation on Low Water Level													X / NA

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

FRH:rf/G09163-3*