REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8303110288 DOC.DATE: 83/03/07 NOTARIZED: NO DOCKET #
FACIL:50-220 Nine Mile Point Nuclear Station, Unit 1, Niagara Powe 05000220
AUTH.NAME AUTHOR AFFILIATION
MANGAN,C.V. Niagara Mohawk Power Corp.
RECIP.NAME RECIPIENT AFFILIATION
VASSALLO,D.B. Operating Reactors Branch 2

SUBJECT: Forwards responses to NRC 830104 plant-specific questions re BWR Owners Group NEDE-24988-P, "Analysis of Generic BWR Safety/Relief Valve Operability Test Results," per TMI Action Plan II.D.1.

DISTRIBUTION CODE: A046S COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 16
TITLE: OR Submittal: TMI Action Plan Rgmt NUREG-0737 & NUREG-0660

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	NRR ORB2 BC 01	7	7			
INTERNAL:	ELD/HDS3	1	0	IE/DEP DIR 33	1	1
	IE/DEP EPDS	1	1	IE/DEP/EPLB	3	3
	NRR PAWLSON,W.	1	1	NRR/DHFS/DEPY29	1	1
	NRR/DL DIR 14	1	1	NRR/DL/ADL 16	1	1
	NRR/DL/ADSA 17	1	1	NRR/DL/ORAB 18	3	3
	NRR/DSI/ADRS 27	1	1	NRR/DSI/AEB	1	1
	NRR/DSI/ASB	1	1	NRR/DSI/RAB	1	1
	NRR/DST DIR 30	1	1	REG FILE 04	1	1
	RGN1	1	1			
EXTERNAL:	ACRS 34	10	10	INPO,J.STARNES	1	1
	LPDR 03	1	1	NRC PDR 02	1	1
	NSIC 05	1	1	NTIS	1	1

TOTAL NUMBER OF COPIES REQUIRED: LTTR 43 ENCL 42

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]

NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

March 7, 1983

Mr. Domenic B. Vassallo, Chief
Operating Reactors Branch No. 2
Division of Licensing
United States Nuclear Regulatory Commission
Washington, D.C. 20555

Re: Nine Mile Point Unit 1
Docket No. 50-220
DPR-63

Dear Mr. Vassallo:

Enclosure 1 of your January 4, 1983 letter identified plant specific questions regarding the Boiling Water Reactor's Owners Group submittal entitled "Analysis of Generic BWR Safety/Relief Valve Operability Test Results" (NEDE-24988-P). The attachment to this letter provides our responses to your questions.

Sincerely,

CVMangan

C. V. Mangan
Vice President
Nuclear Engineering & Licensing

CVM/BDW:bd

Attach.

A046

Nine Mile Point Unit 1

Response to Nuclear Regulatory Commission Request for Additional Information: TMI Action Plan II.D.1, Relief and Safety Valve Test Requirements

QUESTION 1: The test program utilized a "rams head" discharge pipe configuration. Most plants utilize a "tee" quencher configuration at the end of the discharge line. Describe the discharge pipe configuraton used at your plant 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: The safety/relief valve discharge piping configuration at Nine Mile Point Unit 1 utilizes a "y" quencher at the discharge pipe exit. The lengths of the 14 inch diameter relief valve discharge lines range from approximately 94 feet to 102 feet. The submergence in the suppression pool is approximately 6 feet. The test program used a rams head at the discharge pipe exit, a pipe length of 112 feet and a submergence of approximately 13 feet. Loads on valve internals tested in the program were greater than loads expected on valve internals in the Nine Mile Point Unit 1 configuration for the following reasons:

1. No dynamic mechanical load originating at the "y" quencher is transmitted to the valve in the Nine Mile Point Unit 1 configuration because there is an anchor point and a bellows between the valve and the "y" quencher.

2. The first segment of piping from the relief valve in the test facility was longer than the Nine Mile Point Unit 1 piping. This resulted in a bounding dynamic mechanical load on the valve in the test program due to the larger moment arm between the safety relief valve and the first elbow. The first segment length in the test facility is 12 feet whereas this length is less than 2 feet in the Nine Mile Pont Unit 1 configuration.

3. Dynamic hydraulic loads (backpressure) are experienced by the valve internals in the Nine Mile Point Unit 1 configuration. The backpressure loads may be either (i) transient backpressures 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 relief valve discharge line and the relief valve discharge line air volume. Transient backpressures increase with higher upstream pressure, shorter valve opening times, greater line submergence, and smaller relief valve discharge line air volume. The transient backpressure in the test program was maximized by using a submergence of 13 feet, which is greater than Nine Mile Point Unit 1 and a discharge line air volume which is less than that at Nine Mile Point Unit 1. 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 relief valve discharge line above the water level and before the rams head.

The differences in the line configuration between the Nine Mile Point Unit 1 plant and the test program result in the loads on the valve internals for the test facility which bound the actual Nine Mile Point Unit 1 valve loads. An additional consideration in the selection of the rams head for the test facility was to allow more direct measurement of the thrust load in the final pipe segment. Utilization of a "tee" or "y" 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 relief valve discharge line configurations in Nine Mile Point Unit 1 and the test facility will not have any adverse effect on relief valve operability at Nine Mile Point Unit 1.

QUESTION 2: The test configuration uses 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 your plant 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: The Nine Mile Point Unit 1 relief valve discharge lines are provided with an anchor point located less than 2 feet from the valve discharge. The proximity of the anchor to the valve precludes the transmittal of any mechanical loads from the discharge line to the valve internals. Additionally, an expansion bellows located between the valve and the anchor prevents loads from affecting valve operability. The remainder of the pipe supports at Nine Mile Point Unit 1 are such that the location of the supports for the test facility is prototypical, i.e. there are supports near each change in the pipe routing. The relief valve discharge line 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 Boiling Water Reactor's 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 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 assured as they are designed for the larger steam discharge transient loads.

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 likely that sufficient margin exists in the Nine Mile Point Unit 1 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 safety/relief valves to open since the loads occur in the safety/relief valve discharge line only after valve opening. Therefore, the alternate shutdown cooling path can be established.

QUESTION 3: Report NEDE-24988-P did not report any valve functional deficiencies or anomalies encountered during the test program. Describe the impact of valve safety function of any valve functional deficiencies or anomalies encountered during the program that were not reported.

RESPONSE: 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 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. Attachment 1 is the Wyle Laboratories test log sheet for selective Dresser Electromatic relief valve tests. This valve is used in the Nine Mile Point Unit 1 Nuclear Power Station.

Each Wyle test report for the respective valves identifies each test run performed, 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 Attachment 1 for each valve were obtained from the Table 2.2-1 test runs and were based upon the following selection criteria:

(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.

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 the 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 the features at your plant. 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 your plant.

RESPONSE: The purpose of the test program was to demonstrate that the safety/relief valve will open and close under expected flow conditions. The expected valve operating conditions were determined through analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. Single failures were applied to these analyses so the dynamic forces on the safety and relief valves would be maximized. Test pressures were the highest predicted by conventional safety analysis procedures. The Boiling Water Reactor 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 relief valve inlet flow that would maximize the dynamic forces 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 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 test program. This conclusion and the test results applicable to Nine Mile Point Unit 1 are discussed below. The alternate shutdown cooling mode of operation is described in the response to Nuclear Regulatory Commission Question 5.

The valve inlet fluid conditions tested in the Boiling Water Reactor Owners Group program, as documented in NEDE-24988-P, are 15^{o} to 50^{o} subcooled liquid at 20 psig to 250 psig. These fluid conditions envelope the conditions expected to occur in the alternate shutdown cooling mode of operation at Nine Mile Point Unit 1.

The Boiling Water Reactor 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 can be considered applicable to the Nine Mile Point Unit 1 plant because of its design and specific plant configuration. Five events, namely 4,5,6,9, and 10 are not applicable to the Nine Mile Point Unit 1 plant for the reasons listed below:

- events 4, 6 and 9 are not applicable because Nine Mile Point Unit 1 does not have a RCIC system and
- events 5 and 10 are not applicable because Nine Mile Point Unit 1 does not have a HPCS system.

For the eight remaining events, the Nine Mile Point Unit 1 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 Boiling Water Reactor Owners Group submittal of September 17, 1980. The comparison has demonstrated that in general, the base case analysis is applicable to Nine Mile Point Unit 1. The base case analysis does not include any mitigating plant features with the exception of a high water level motor driven feedwater pump trip, which are not already present in the Nine Mile Point Unit 1 design. As discussed in the notes for Table 1, a high level pump trip will be installed during the 1984 refueling outage. Table 1 demonstrates that plant mitigating features assumed in the event evaluations are also existing features in the Nine Mile Point Unit 1 plant. Features included in this base case analysis are similar to plant features in the Nine Mile Point Unit 1 design.

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 relief valve inlet. Consequently, this event was simulated in the Boiling Water Reactor test program. Calculations for Nine Mile Point Unit 1 indicate that for the conservative assumptions of a flow rate of approximately 3000 gpm, torus water temperature of 70°F, and decay heat at 10 minutes following shutdown, the relief valve liquid discharge would be approximately 48°F subcooled. The test conditions envelope these plant conditions.

As discussed above, the Boiling Water Reactor 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 Boiling Water Reactor safety/relief valve test program. The fluid conditions and flow conditions tested in the Boiling Water Reactor Owners Group test program conservatively envelope the Nine Mile Point Unit 1 plant-specific fluid conditions expected for the alternate shutdown cooling mode of operation.

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 close?

RESPONSE: The Boiling Water Reactor safety/relief valve operability test program was designed to simulate the alternate shutdown cooling mode, which is the only expected liquid discharge event for Nine Mile Point Unit 1. 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 either the emegency condensers or the relief valves to discharge steam to the suppression pool. If relief valve operation is required, the operator cycles the valves 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 the normal shutdown cooling system. If that system is unavailable because the valve on the shutdown cooling suction line fails to open, the operator could initiate the alternate shutdown cooling mode.

For alternate shutdown cooling, the operator opens one relief valve and initiates a core spray pump utilizing the suppression pool as the suction source. The reactor vessel is filled allowing water to flow into the main steam lines, out of the relief valve and back to the suppression pool. Cooling of the system is provided by use of a containment spray heat exchanger. As a result, an alternate cooling mode is maintained.

In order to assure continuous long term heat removal, the relief valve 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 control the flow rate into the vessel. Consequently, no cycling of the relief valve is required for the alternate shutdown cooling mode. No cycling of the relief valve was performed for the generic Boiling Water Reactor relief valve operability test program.

The ability of the Nine Mile Point Unit 1 relief valve to be extensively cycled for steam discharge conditions has been observed during valve tests performed by the valve vendor. Our discussions with the valve vendor indicate these test results provide assurance that cycling of the valve in a controlled depressurization mode for steam discharge conditions will not adversely affect valve performance. Documentation from the vendor to verify this is being pursued. The probability of the valve failing to open or close is extremely low.

QUESTION 6: Describe how the values of valve C_V in report NEDE-24988-P will be used at your plant. Show that the methodology used in the test program to determine the valve C_V will be consistent with the application at your plant.

RESPONSE: The flow coefficient, C_V, for the Dresser Electromatic relief valve utilized in Nine Mile Point Unit 1 was determined in the generic test program (NEDE-24988-P). The average flow coefficient calculated from the test results is reported in Table 5.2-1 of NEDE-24988-P. This test value has been used to confirm that the liquid discharge flow capacity of the relief valves will be sufficient to remove core decay heat when injecting into the reactor pressure vessel in the alternate shutdown cooling mode. The C_V value determined in the relief valve test demonstrates that the Nine Mile Point Unit 1 relief valves are capable of returning the flow injected by the core spray pump to the suppression pool.

If it were necessary for the operator to place the plant in the alternate shutdown cooling mode, he would assure that adequate core cooling was being provided by monitoring the core spray flow rate, reactor vessel pressure and reactor vessel temperature.

The flow coefficient was determined from the 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), 3 feet downstream of the valve and the corresponding measured flowrate. Furthermore, the test conditions and test configuration were representative of Nine Mile Point Unit 1 plant conditions for the alternate shutdown cooling mode, e.g. pressure upstream of the valve, fluid temperature, friction losses and liquid flow rate. Therefore, the reported C_V values are appropriate for application to the Nine Mile Point Unit 1 plant.

SUMMARY OF REDUCED DATA
DRESSER ELECTROMATIC 6X8 S/RV WITH LOADS I SUPPORTS

Test Parameter Description	Units	Test Data: Steam, Saturated Run 609	Water, 15°F Subcooling Run 606	Water, 50°F Subcooling Run 608
Fluid inlet temperature	°F	555	217	161
Steam flow rate at (.1146) PSIG	lbs/hr	533.300	N/A	N/A
Average backpressure	%	36.3.	N/A	N/A
Operability, open/closed upon command	yes/no	yes	yes	yes
Opening time, main valve disc	msec	17	*	*
Average water pressure	psig	N/A	276	270
Average water flow rate	gpm	N/A	4482	4431
S/RV test facility integrity, after run	yes/no	yes	yes	yes
S/RV integrity, post test hydro	yes/no	yes	yes	yes
S/RV internal parts integrity post test disassembly/inspection	yes/no	yes	N/A	N/A

* The main disc opening time could not be estimated, because the S/RVDL pressure and accelerometer values were too low to cause a perceptible change in the traces.

SUMMARY OF REDUCED DATA
DRESSER ELECTROMATIC 6X8 S/RV WITH LOADS I SUPPORTS

Test Parameter		Test Data, Maximum Dynamic Values		
Description	Units	Steam, Saturated Run 609	Water, 15°F Subcooling Run 606	Water, 50°F Subcooling Run 608
SRVDL Acceleration (A2) - 2nd Section	"g's"	4	<1	<1
SRVDL Acceleration (A3) - 3rd Section	"g's"	3	<1	<1
SRVDL Acceleration (A4) - 4th Section	"g's"	4	<1	<1
Load Cell (L1), Horiz. Support	lbs	3,000	1,000	1,000
Load Cell (L3), Horiz. Support	lbs	15,000	1,000	1,000
Load Cell (L2), Horiz. Support	lbs	14,000	3,000	3,000
Load Cell (L4), Horiz. Support	lbs	29,000	2,000	1,000
Stress (SG9), Horiz. Support	psi	2,000	500	200
Stress (SG10,) Steam Chest	psi	500	200	300
Stress (SG11), Steam Chest - Middle	psi	700	200	300
Stress (SG12), Steam Chest	psi	*	200	300
Stress (SG13), Sweepolet	psi	2,100	1,000	1,200
Stress (SG14), Sweepolet	psi	4,600	900	1,200
Stress (SG15), Sweepolet	psi	2,000	1,000	1,000
Stress (SG16), Sweepolet	psi	4,100	1,000	700

* strain gage failed - no data available

SUMMARY OF REDUCED DATA
DRESSER ELECTROMATIC 6X8 S/RV WITH LOADS I SUPPORTS

Test Parameter Description	Units	Test Data, Maximum Dynamic Values: Steam, Saturated Run 609	Water, 15°F Subcooling Run 606	Water, 50°F Subcooling Run 608
SRVDL Stress (SG21), 1st Section	psi	<100	200	300
SRVDL Stress (SG22), 1st Section	psi	700	200	300
SRVDL Stress (SG23), 1st Section	psi	<100	200	300
SRVDL Stress (SG24), 1st Section	psi	1,000	200	300
SRVDL Stress (SG25), 2nd Section	psi	300	200	200
SRVDL Stress (SG26), 2nd Section	psi	700	200	700
SRVDL Stress (SG27), 3rd Section	psi	600	600	200
SRVDL Stress (SG28), 3rd Section	psi	<100	400	<100
SRVDL Stress (SG29), 3rd Section	psi	500	600	100
SRVDL Stress (SG30), 3rd Section	psi	1,000	400	200
SRVDL Stress (SG5), 4th Section	psi	900	300	200
SRVDL Stress (SG6), 4th Section	psi	1,000	200	200
SRVDL Stress (SG7), 4th Section	psi	1,600	300	200
SRVDL Stress (SG8), 4th Section	psi	2,100	200	<100

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 / NA	X / NA	X / S	X / S	X / S
Reactor Scram on Low Water Level													X / S.
Reactor Isolation on Low Water Level													X / 1

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

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 / NA	X / NA				X / NA	X / NA	X / S	X / S	X / S
High Drywell Pressure Alarm													
FW Level 8 Trip*	X / 2	X / 2											
RCIC Level 8 Trip			X / NA	X / NA	X / NA				X / NA	X / NA	X / NA		X / NA
HPCS Level 8 Trip				X / NA	X / NA				X / NA	X / NA			X / NA
HPCI Level 8 Trip *			X / 2	X / NA					X / NA		X / 2		X / 2
HPCI/S and RCIC Initiation on Low Water Level	X / 3	X / 3	X / 3	X / NA	X / NA	X / NA		X / 3	X / NA				X / 3
HPCI/S initiation on High Drywell Pressure			X / 4	X / NA					X / NA	X / NA	X / 4	X / 4	X / 4
RCIC Initiation on High Drywell Pressure													X / NA

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

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 ECCS Initiation on Low Water Level													X/5
FW Pumps Trip on Low Suction Pressure	X/S												
HPCS Trip on High Backpressure			X/NA								X/NA		
RCIC Trip on High Backpressure				X/NA					X/NA				
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					

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

Notes for Table 1

1. Reactor Isolation occurs on low low water level.

2. A high water level trip of the Feedwater/HPCI pumps (see Note 3) does not currently exist. This trip will be installed during the 1984 refueling outage. The precise trip level is undetermined but will be below the main steam nozzles.

3. The Nine Mile Point Unit 1 high pressure coolant injection system utilizes the plants condensate, feedwater booster pumps and electric motor driven feedwater pumps. On a low water level signal the normal feedwater control switches over to the HPCI mode of control. A detailed description of the HPCI system is provided in the First Supplement to the Final Safety Analysis Report, dated May 1968.

4. The HPCI system at Nine Mile Point Unit 1 initiates only on low water level or a turbine trip.

5. Low pressure ECCS initiates on low low water level at Nine Mile Point Unit 1.

* The high water level alarm and the high level trip feature discussed in item 2 above may not correspond exactly to the Level 7 and Level 8 features of the base case.

[illegible]