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SCHWENCER,A.	Licensing Branch 2

SUBJECT: Forwards response to NRC 830329 ltr re SER Open Item 54 concerning performance testing of relief & safety valves. Encl info will be included in next FSAR amend.

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NIAGARA MOHAWK

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

October 3, 1984
(NMP2L 0185)

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Schwencer:

Re: Nine Mile Point Unit 2
Docket No. 50-410

Enclosed is information relating to Enclosure 12 to the Nuclear Regulatory Commission letter dated March 29, 1983. This is Safety Evaluation Report open item 54. This information was requested by your staff for use in evaluating the performance testing of relief and safety valves.

The enclosed information will be included in the next Final Safety Analysis Report Amendment.

Very truly yours,

T E Lempges

T. E. Lempges
Vice President
Nuclear Generation

TEL/DS:ja
Enclosure

cc: R. Gramm, NRC Resident Inspector

Project File (2)

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of Niagara Mohawk Power Corporation (Nine Mile Point Unit 2)	)))))	Docket No. 50-410

AFFIDAVIT

T. E. Lempges, being duly sworn, states that he is Vice President of Niagara Mohawk Power Corporation; that he is authorized on the part of said Corporation to sign and file with the Nuclear Regulatory Commission the documents attached hereto; and that all such documents are true and correct to the best of his knowledge, information and belief.

T E Lempges

Subscribed and sworn to before me, a Notary Public in and for the State of New York and County of Onondaga, this 3 day of October, 1984.

Janis M Macro
Notary Public in and for
Onondaga County, New York

My Commission expires:
JANIS M. MACRO
Notary Public in the State of New York
Qualified in Onondaga County No. 4784555
My Commission Expires March 30, 1985

SER#54

RESPONSE TO NRC ENCLOSURE 12

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 configuration 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

Unit 2 utilizes a tee quencher at the end of the main steam SRV discharge line (SRVDL). The test program described in NEDE-24988-P used a rams head discharge device with test conditions simulating the shutdown cooling mode. The impact of the difference on valve operability is accounted for as follows:

Valve operability is affected by dynamic loads on valve internals. The dynamic loads are governed by (a) back pressure of the SRV and (b) flow through the SRV. Higher back pressures and flow will produce higher dynamic loads.

(a) In the test program, the SRV inlet pressure was equal to 250 psig. The Unit 2 reactor pressure during shutdown cooling mode is approximately 135 psig. The maximum back pressure of the SRV is approximately 35 percent of the SRV inlet pressure; thus, the test program has qualified the SRV to work with back pressure of about two times that of Unit 2. This provides adequate margin to offset the difference in using a tee quencher.

(b) The test program has qualified the SRV with a rams head discharge device. The tee quencher allowed less flow (257 lbm/sec) than the rams head (263 lbm/sec) because it has higher flow resistance. Thus, operability of the SRV for Unit 2 SRVDL with a tee quencher will also be qualified.

RESPONSE TO NRC ENCLOSURE 12

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 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

Fourteen of 18 SRV discharge lines use no spring hangers between the SRV and first full anchor. The remaining four lines use spring hangers solely as deadweight supports and to limit deadweight stresses to less than 1,500 psi. Due to their low stiffness, spring hangers have an insignificant impact on the dynamic properties of the piping system and do not affect the operability of the valves. An adequate number of snubbers and rigid supports are provided to support piping for dynamic loads.

Measured stresses in the SRV discharge piping near the SRV outlet, from NEDE-24988-P, are given in Table 1. Computed stresses in Unit 2 SRV discharge piping near the SRV discharge outlet are given in Table 2. The computed stresses are consistently lower than the measured stresses, which are low. Therefore, it is expected that loads on Unit 2 SRVs will be lower than test loads. Hence, there is no impact on valve operability.

RESPONSE TO NRC ENCLOSURE 12

TABLE 1

MEASURED STRESSES FOR STEAM BLOWDOWN IN
SRV DISCHARGE PIPING NEAR SRV OUTLET
DIKKERS 8R10 SRV

Strain Gage*	Stress (psi)**
SG21	200
SG22	1,400
SG23	200
SG24	1,400

NOTES: *Strain gages SG21 through SG24 are at the same location, but differ in orientation.
**Based on 10-in Sch. 80 (0.593 in-thick) pipe.

RESPONSE TO NRC ENCLOSURE 12

TABLE 2

COMPUTED STRESSES FOR STEAM BLOWDOWN IN SRV DISCHARGE PIPING NEAR SRV OUTLET

Valve No. 2MSS*PSV-	Calc. No. AX-	Node	Computed Stress (psi)**
120	2A	205	564
121	2A	310	410
122	2A	410	397
123	2A	510	355
124	2B	204	340
125	2B	304	332
126	2B	404	386
127	2B	504	302
128	2B	604	303
129	2C	204	420
130	2C	304	491
131	2C	404	550
132	2C	504	308
133	2C	604	284
134	2D	212	699
135	2D	330	904
136	2D	577	952***
137	2D	721	903

NOTES: *SRSS (SRV discharge fluid transient, SRV inertia (max)).
**Based on 10-in Sch. 60 (0.5 in-thick) pipe.
***Maximum computed stress of 952 psi is much lower than the measured stress of 1,400 psi shown in Table 1.

RESPONSE TO NRC ENCLOSURE 12

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

No functional deficiencies or anomalies of the safety relief valves were experienced during the testing at Wyle Laboratories in compliance with the alternate shutdown cooling mode requirement. All of the valves subjected to test runs 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 system, 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 Dikkers valve test is in Table 3.

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

All valid test runs are identified in Table 2.2-1 or 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 on the following selection criteria:

1. Presenting the maximum representative loading information obtained from the steam run data.

2. Presenting the maximum representative water loading information obtained from the 15°F subcooled water test data.

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

RESPONSE TO NRC ENCLOSURE 12

TABLE 3

OPERABILITY TEST LOG - SRV DK-1

Test No.	Media	Load Line Configuration	Date	Test Remarks
101	Steam	1	03/03/81	Acceptable
102	Water	1	03/03/81	Acceptable
103	Steam	1	03/03/81	Acceptable
104	Water	1	03/04/81	Acceptable
105	Steam	1	03/04/81	Acceptable
106	Water	1	03/04/81	Acceptable

RESPONSE TO NRC ENCLOSURE 12

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 plant 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 (SRV) will open and reclose under all expected flow conditions. The expected valve operating conditions were determined from accidents and anticipated transients referenced in Regulatory Guide 1.70, Revision 2. Additional single failures were considered so that dynamic forces on SRVs would be maximized. By this approach, the BWR Owners' Group, in the 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 SRV inlet flow that would maximize the dynamic forces on the SRV. Among the 13 events, the alternate shutdown cooling mode was found to be the only expected event which will result in liquid flow at the valve inlet. Consequently, this event was simulated in the SRV test program. The conclusion and test results applicable to Unit 2 are discussed subsequently.

The SRV inlet conditions in the test program, as documented in NEDE-24988-P, are 15°F to 50°F subcooled liquid at 20 psig to 250 psig. For Unit 2, the inlet condition during shutdown cooling (Event 7) is approximately 35°F subcooling at a pressure of approximately 135 psig.

The 13 events and the plant-specific features that mitigate these events are summarized in Table 4. Of these 13 events, only 11 are applicable to the Unit 2 design. Two events, namely 3 and 11, are not applicable because Unit 2 does not have an HPCI system. For the 11 remaining events, the Unit 2 specific features, such as trip logic, power supplies, instrument line configuration, alarms, and

operator actions, have been compared to the base plant analysis presented in the BWR Owners' Group submittal dated September 17, 1980. For these events, Table 4 demonstrates that the Unit 2 specific features are included in the base plant analyses. Furthermore, the time available for operator action is expected to be longer in the Unit 2 plant than in the base plant analysis for each event where operator action is required.

As discussed previously, the BWR Owners' Group evaluated transients, including single active failures, that would maximize the dynamic forces on the SRVs. 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 SRV test program. The fluid and flow conditions tested in the BWR Owners' Group test program conservatively envelope the Unit 2 plant-specific fluid conditions expected for the alternate shutdown cooling mode of operation.

RESPONSE TO NRC ENCLOSURE 12

TABLE 4

EVENTS EVALUATED

Plant Features	No. 1 - FW Cont. Fail, FW L8 Trip Failure	No. 2 - Press. Reg. Fail.	No. 3 - Transient HPCI, HPCI L8 Trip Failure	No. 4 - Transient RCIC, RCIC L8 Trip Failure	No. 5 - Transient HPCS, HPCS L8 Trip Failure	No. 6 - Transient RCIC Hd. Spr.	No. 7 - Alt. Shutdown Cooling, Shutdown Suction Unavailable	No. 8 - MSL Brk OSC	No. 9 - SBA, RCIC, RCIC L8 Trip Failure	No. 10 - SBA, HPCS, HPCS L8 Trip Failure	No. 11 - SBA, HPCI, HPCI L8 Trip Failure	No. 12- SBA, Depress. & ECCS Over., Operator Error	No. 13 - LBA, ECCS Overf Brk Isol
High water Level 7 alarm	X/S		X/NA	X/S	X/S				X/S	X/S	X/NA	X/S	X/S
High drywell pressure alarm													
FW Level 8 trip	X/S	X/S											
RCIC Level 8 trip			X/NA	X/S	X/S				X/S	X/S	X/NA		X/S
HPCS Level 8 trip				X/S	X/S				X/S	X/S			X/S
HPCI Level 8 trip			X/NA	X/NA					X/NA		X/NA		X/NA
HPCI/S and RCIC initiation on low water level	X/S	X/S	X/NA	X/S	X/S	X/S		X/S	X/S				X/S
HPCI/S initiation on high drywell pressure			X/NA	X/NA					X/S	X/S	X/NA	X/S	X/S
RCIC initiation on high drywell pressure													X/NA
Low-pressure ECCS initiation on high drywell pressure									(2)	(2)	(2)	X/S	X/S
Low-pressure initiation on low water level													X/S

TABLE 4

EVENTS EVALUATED

Plant Features	No. 1 - FW Cont. Fail, FW L8 Trip Failure	No. 2 - Press. Reg. Fail.	No. 3 - Transient HPCI, HPCI L8 Trip Failure	No. 4 - Transient RCIC, RCIC L8 Trip Failure	No. 5 - Transient HPCS, HPCS L8 Trip Failure	No. 6 - Transient RCIC Hd. Spr.	No. 7 - Alt. Shutdown Cooling, Shutdown Suction Unavailable	No. 8 - MSL Brk OSC	No. 9 - SBA, RCIC, RCIC L8 Trip Failure	No. 10 - SBA, HPCS, HPCS L8 Trip Failure	No. 11 - SBA, HPCI, HPCI L8 Trip Failure	No. 12- SBA, Depress. & ECCS Over., Operator Error	No. 13 - LBA, ECCS Overf Brk Isol
FW pumps trip on low suction pressure or high vibration	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/S											
MSIV closure on low turbine inlet pressure	X/S	X/S						X/S					
MSIV closure on high steam flow		X/S						X/S					
MSIV closure on high steam tunnel temperature								X/S					
MSIV closure on high radiation								X/S					
Reactor scram on turbine trip	X/S	X/NA	(3)										
Reactor scram on neutron flux monitor		X/NA	(3)										
Reactor scram on MSIV closure		X/S											
Reactor scram on high radiation								X/S	(1)				
Reactor scram on high drywell pressure									X/S	X/S	X/NA	X/S	X/S

TABLE 4

EVENTS EVALUATED

Plant Features	No. 1 - FW Cont. Fail, FW L8 Trip Failure	No. 2 - Press. Reg. Fail.	No. 3 - Transient HPCI, HPCI L8 Trip Failure	No. 4 - Transient RCIC, RCIC L8 Trip Failure	No. 5 - Transient HPCS, HPCS L8 Trip Failure	No. 6 - Transient RCIC Hd. Spr.	No. 7 - Alt. Shutdown Cooling, Shutdown Suction Unavailable	No. 8 - MSL Brk OSC	No. 9 - SBA, RCIC, RCIC L8 Trip Failure	No. 10 - SBA, HPCS, HPCS L8 Trip Failure	No. 11 - SBA, HPCI, HPCI L8 Trip Failure	No. 12- SBA, Depress. & ECCS Over., Operator Error	No. 13 - LBA, ECCS Overf Brk Isol
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.

NOTES: (1) As a consequence of MSIV closure.
(2) High vessel pressure prevents initiation.
(3) Scram bypassed below 30 percent power.

RESPONSE TO NRC ENCLOSURE 12

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

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 Unit 2. The sequence of events leading to the alternate shutdown cooling mode follows.

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°F per hr. 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.

For alternate shutdown cooling, the operator opens the SRVs and initiates an RHR 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, out of the SRV, and back to the suppression pool. Cooling of the system is provided by use of an RHR heat exchanger.

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

The ability of the Unit 2 SRV to withstand extensive cycling for steam discharge conditions has been confirmed during steam discharge qualification testing of the valve by the valve vendor. Based on the SRV qualification testing, cycling of the valves in a controlled depressurization mode for steam discharge conditions will not affect valve performance adversely. Furthermore, the probability of the valve to fail open or closed is extremely low.

RESPONSE TO NRC ENCLOSURE 12

QUESTION 6.

Describe how the values of the valve C_V's 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 will be consistent with the application at your plant.

RESPONSE

The flow coefficient, C_V, for the Dikkers safety relief valve (SRV) was determined in the generic SRV test program (NEDE-24988-P). The average flow coefficient calculated from test results for the Dikkers valve is reported in Table 5.2-1 of NEDE-24988-P. This test value has been used by NMPC to confirm that the liquid discharge flow capacity of the SRVs will be sufficient to remove core decay heat in the alternate shutdown cooling mode. The C_V value determined in the SRV test demonstrates that the SRVs are capable of returning flow to the suppression pool.

If the alternate shutdown cooling mode is required, the operator is to ensure that adequate core cooling is provided by monitoring the following parameters: RHR flow rate and reactor vessel pressure and temperature.

The flow coefficient for the Dikkers 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 3 ft downstream of the valve and the corresponding measured flowrate. Furthermore, test conditions and configurations were representative of Unit 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 Unit 2.

[illegible]