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AUTH.NAME AUTHOR AFFILIATION
SORENSEN,G.C. Washington Public Power Supply System
RECIP.NAME RECIPIENT AFFILIATION
SCHWENCER,A. Licensing Branch 2

SUBJECT: Forwards response to 840914 request for addl info re BWR Owners Group response to NUREG-0737,Item II.D.1,addressed in NEDE-24988-P re analysis of safety/relief valve operability.

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TITLE: OR Submittal: TMI Action Plan Rqmt NUREG-0737 & NUREG-0660

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Washington Public Power Supply System
P.O. Box 968 3000 George Washington Way Richland, Washington 99352 (509) 372-5000

January 18, 1985
G02-85-024

Docket No. 50-397

Director of Nuclear Reactor Regulation
Attention: Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PLANT NO. 2
OPERATING LICENSE NPF-21, NRC REQUEST FOR
ADDITIONAL INFORMATION REGARDING BWROG'S
RESPONSE TO NUREG-0737, ITEM II.D.1

Reference:
1) Letter, A. Schwencer (NRC) to G. C. Sorensen (SS), same subject, dated September 14, 1984
2) Letter, G02-84-655, G. C. Sorensen (SS) to A. Schwencer (NRC), same subject, dated December 20, 1984

Reference 1 requested that the Supply System provide additional information to assist the Staff in their review of the BWR Owners' Group response to NUREG-0737, Item II.D.1 addressed in GE report NEDE-24988-P, "Analysis of Generic BWR Safety/Relief Valve Operability Test Results". Reference 2 established a submittal date of January 18, 1985. Our response is attached hereto.

Should you have additional questions in this matter, please contact Mr. P. L. Powell, Manager, WNP-2 Licensing.

Very truly yours,

GC Sorensen

G. C. Sorensen, Manager
Regulatory Programs

HLA/tmh
Attachment

cc: R Auluck - NRC
WS Chin - BPA
JB Martin - NRC RV
AD Toth - NRC Site

A044
1/1

NRC QUESTION 1

The test program utilized a "rams head" discharge pipe configuration. WNP-2 utilizes an "X" quencher configuration at the end of the discharge line. Describe the discharge pipe configuration used at WNP-2 and compare the anticipated loads on valve internals in the WNP-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 WNP-2 utilizes an "X" quencher at the discharge pipe exit. The average length of the WNP-2 SRV discharge lines (SRVDL) is 135 feet and the submergence length in the suppression pool is approximately 17.4 feet. The SRV test program utilized a ramshead at the discharge pipe exit, a pipe length of 112' and a submergence length of approximately 13'. Loads on valve internals during the test program are larger than loads on valve internals in the WNP-2 configuration for the following reasons:

1. No dynamic mechanical load originating at the "X" quencher is transmitted to the valve in the WNP-2 configuration because there is at least one anchor point between the valve and the quencher. In addition, all quenchers are anchored.
2. The first length of the segment of piping downstream of the SRV in the test facility was longer than the WNP-2 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 ft whereas this length is 8 ft maximum in the plant configuration.
3. Dynamic hydraulic loads (backpressure) are experienced by the valve internals in the WNP-2 configuration. The backpressure loads may be either (i) transient backpressures occuring during valve actuation, or (ii) steady-state backpressures occurring during steady-state flow following valve actuation.

   (a) The transient backpressure experienced by the valve internals in the WNP-2 configuration are bounded by the high pressure steam flow mode of the WNP-2 relief valves. Little or no transient backpressure loads occur during the alternate shutdown cooling mode since the SRV's are already open at the initiation of this mode. As described in detail in the response to question 5, the initiation of the alternate shutdown cooling mode occurs as a smooth transition from steam to water flow through the SRV's. This transition is accomplished by opening two SRV's following normal vessel depressurization and pumping water from the suppression pool via the RHR system into the vessel, filling the vessel and main steam lines and returning it through the open SRV's to the suppression pool.

(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 WNP-2 SRVDL's.

The differences in the line configuration between the WNP-2 plant and the test program as discussed above result in the loads on the valve internals for the test facility which bound the WNP-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 an "X" 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 WNP-2 and the test facility will not have any adverse effect on SRV operability at WNP-2 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 WNP-2 and compare the anticipated loads on valve internals for the WNP-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 WNP-2 safety-relief valve discharge lines (SRVDL's) are supported by a combination of snubbers, rigid supports, and spring hangers. The locations of snubbers and rigid supports at WNP-2 are such that the location of such supports in the BWR generic test facility is prototypical, i.e., in each case (WNP-2 and the test facility) there are supports near each change of direction in the pipe routing. Additionally, each SRVDL at WNP-2 has between 3 and 5 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 of 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 BWR's since the test facility was designed to be prototypical of the features pertinent to this issue. Furthermore, analysis of a typical WNP-2 SRVDL configuration has confirmed the applicability of this conclusion to WNP-2.

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 assured as they are designed for the larger steam discharge transient loads and valve operability is maintained.

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 concluded that sufficient margin exists in the WNP-2 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 SRV's 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 testing at Wyle Laboratories for demonstrating 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 Laboratories test log sheet for the CROSBY 6R10 valve tests is attached. This valve is used in the WNP-2 Nuclear Power Station.

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 WNP-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 WNP-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 WNP-2.

RESPONSE TO NRC QUESTION 4

The purpose of the S/RV test program was to demonstrate that the Safety Relief Valve (S/RV) will open and reclose under all expected flow conditions. The expected valve operating conditions were determined through the use of analysis of the accidents and anticipated operational occurences 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 relief valve. These 13 events were the ones identified from the evaluation of the events described in Regulatory Guide 1.70, Revision 2, with and without 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 WNP-2 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, were 15° to 50° subcooled liquid at 20 psig to 250 psig. These fluid conditions envelope the conditions expected to occur at WNP-2 in the alternate shutdown cooling mode of operation.

The 13 events and the plant-specific features that mitigate these events, are summarized in Table 1. Of these 13 events, only 11 are applicable to the WNP-2 plant because of its design and specific plant configuration. Two events, namely 3 & 11 are not applicable to the WNP-2 plant for the reasons listed below:

(a) Events 3 and 11 are not applicable, because WNP-2 does not have a HPCI System.

For the 11 remaining events, the WNP-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 demonstrated that the analysis is applicable to WNP-2 because the base case analysis does not

include any plant features with the exception of RCIC initiation on high drywell pressure which are not present in the WNP-2 design. For these events, Table 1 demonstrates that the WNP-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 WNP-2 plant. Furthermore, the time available for operator action is expected to be longer in the WNP-2 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 two-phase fluid at the S/RV inlet. Consequently, this event was simulated in the BWR S/RV test program. In WNP-2, this event involves flow of subcooled water (10 to 110°F subcooled) at a pressure of 75 to 172 psig. The test conditions of 15 to 50°F subcooled and 20 to 250 psig 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 WNP-2 plant-specific fluid conditions expected for the alternate shutdown cooling mode of operation.

PLANT FEATURES	#1 FW Cont. Failure, FW L8 Trip Failure	#2 Press. Reg. Failure	#3 Transient HPCI, HPCI L8 Trip Failure (Not Applicable)	#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 (Not Applicable)	#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 / S	X / NA	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

TABLE 1 - EVENTS EVALUATED

PLANT FEATURES	#1 FW Cont. Failure, FW LB Trip Failure	#2 Press. Reg. Failure	#3 Transient HPCI, HPCI L8 Trip Failure (Not Applicable)	#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 (Not Applicable)	#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 / S
FW Pumps Trip on Low Suction Pressure	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					

TABLE 1 - EVENTS EVALUATED

PLANT FEATURES	#1 FW Cont. Failure, FW L8 Trip Failure	#2 Press. Reg. Failure	#3 Transient HPCI, HPCI L8 Trip Failure (Not Applicable)	#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 (Not Applicable)	#12 SBA, Depress. & ECCS Over., Operator Error	#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
HPCS Initiation on Low Water Level	X / S	X / S	X / NA	X / S	X / S	X / S		X / S	X / S				X / S
HPCS Initiation on High Drywell Pressure			X / NA	X / S					X / S	X / S	X / NA	X / S	X / S
RCIC Initiation on High Drywell Pressure													X / NA

TABLE 1 - EVENTS EVALUATED

NRC QUESTION 5

The valves are likely to be extensively cycled in a controlled depresurization 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 WNP-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 to 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.

For alternate shutdown cooling, the operator opens two SRV's 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 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 control the flow rate into the vessel. 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 WNP-2 SRV to be extensively cycled for steam discharge conditions has been confirmed during steam discharge qualification testing of the valve by the valve vendor. Based on the qualification testing of the SRV's, the cycling of the valves in a controlled depressurization mode for steam discharge conditions will not adversely affect valve performance and 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 WNP-2. Show that the methodology used in the test program to determine the valve C_v will be consistent with the application at WNP-2.

RESPONSE TO NRC QUESTION 6

The flow coefficient, C_v, for the CROSBY 6R10 safety relief valve (SRV) utilized in WNP-2 was determined in the generic SRV test program (NEDE-24988-PO). The average flow coefficient calculated from the test results for the CROSBY 6R10 is reported in Table 5.2-1 of NEDE-24988-P. This test value has been used by Washington Public Power Supply System to confirm that the liquid discharge flow capacity of the WNP-2 SRV's 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 WNP-2 SRV's are capable of returning the flow injected by the RHR or CS pump to the suppression pool.

The flow coefficient for the CROSBY 6R10 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' downstream of the valve and the corresponding measured flowrate. Furthermore, the test conditions and test configuration were representative of WNP-2 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 the WNP-2 plant.

REFERENCES

1. F. L. Leverenz, "Probabilistic Evaluation of High Pressure Liquid Challenge of Safety/Relief Valve Piping," Science Applications, Inc. Palo Alto, California, April 1981.

2. Letter to D. G. Eisenhut (USNRC) from T. D. Keenan (BWR Owners' Group), November 14, 1979.

3. Letter to R. H. Vollmer (USNRC) from D. B. Waters (BWR Owners' Group), September 17, 1980.

OPERABILITY TEST REPORT

FOR

CROSBY 6R10 SRV

FOR

LOW PRESSURE WATER TESTS

FOR

GENERAL ELECTRIC COMPANY

GENERAL ELECTRIC

NUCLEAR ENERGY BUSINESS GROUP

J. J. Moss 2-10-82

APPROVED DATE

5682-78-1

VPF NO.

D810271

TRANSMITTAL NO.

175 Curtner Avenue
San Jose, California

TABLE I

OPERABILITY TEST LOG, SRV CR-1

TEST NO.	TEST MEDIA	LOAD LINE CONFIGURATION	TEST DATE	REMARKS
401	Steam	1	3/24/81	Backpressure low, changed orifice.
402	Steam	1	3/24/81	Test Acceptable
403	Water	1	3/24/81	Test Acceptable
404	Steam	1	3/24/81	Test Acceptable
405	Water	1	3/25/81	Test Acceptable
406	Steam	1	3/25/81	Test Acceptable
407	Water	1	3/25/81	Test Acceptable

[illegible]