CP&L

Carolina Power & Light Company

JAN 14 1983

Office of Nuclear Reactor Regulation
ATTN: Mr. D. B. Vassallo, Chief
Operating Reactors Branch No. 2
United States Nuclear Regulatory Commission
Washington, DC 20555

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324
LICENSE NOS. DPR-71 AND DPR-62
NUREG-0737 ITEM II.D.1
RELIEF AND SAFETY VALVE TESTING

Dear Mr. Vassallo:

In response to your letter of July 12, 1982 concerning NUREG-0737 Item II.D.1, Relief and Safety Valve (S/RV) Testing, Carolina Power & Light Company (CP&L) is providing responses to your questions concerning the final results on S/RV testing for the Brunswick Steam Electric Plant, Unit Nos. 1 and 2. The final test results are contained in NEDE-24988-P, "Analysis of Generic BWR Safety/Relief Valve Operability Test Results," which were submitted to the NRC by the BWR Owners' Group on September 25, 1981. The applicability of the generic test results to the Brunswick S/RV's was demonstrated in our June 30, 1981 and October 5, 1981 submittals.

If you have any questions concerning this subject, please contact our staff.

Yours very truly,

S. R. Zimmerman

S. R. Zimmerman
Manager
Licensing & Permits

WRM/kjr (5876C10T2)
Enclosure

cc: Mr. S. D. MacKay (NRC)
Mr. D. O. Myers (NRC-BSEP)
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NRC QUESTION NO. 1:

The test program utilized a "rams head" discharge pipe configuration. Brunswick utilizes a "tee" quencher configuration at the end of the discharge line. Describe the discharge pipe configuration used at Brunswick 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.

CP&L Response: The safety/relief valve (S/RV) discharge piping configuration at Brunswick utilizes a "tee" quencher at the discharge pipe exit. The average length of the eleven S/RV discharge lines (SRVDL) is 75 feet and the submergence length in the suppression pool is approximately 7 feet. The S/RV test program utilized a ramshead at the discharge pipe exit, a pipe length of 112 feet and a submergence length of approximately 13 feet. Loads on valve internals during the test program are larger than loads on valve internals in the Brunswick configuration for the following reasons:

No dynamic mechanical load originating at the "tee" quencher is transmitted to the valve in the Brunswick 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 S/RV in the test facility was longer than the Brunswick piping, thereby resulting in a bounding dynamic mechanical load on the valve in the test program due to the larger moment arm between the S/RV and the first elbow. The first segment length in the test facility is 12 feet whereas these lengths are 1.5 to 12 feet in the plant configuration.

3. Dynamic hydraulic loads (backpressure) are experienced by the valve internals in the Brunswick 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 SRVDL and the SRVDL air volume. Transient backpressures increase with higher upstream pressure, shorter valve opening times, greater line submergence, and smaller SRVDL air volume. The transient backpressure in the test program was maximized by utilizing a submergence of 13 feet, which is greater than Brunswick, and a pipe length of 112 feet, which is more than Brunswick.

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 a 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 Brunswick SRVDL's.

The differences in the line configuration between the Brunswick plant and the test program as discussed above result in the loads on the valve internals of the test facility which bound the actual Brunswick 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, differences between the SRDVL configurations in Brunswick and the test facility will not have any adverse effects on S/RV operability at Brunswick relative to the test facility.

NRC QUESTION NO. 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 Brunswick 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.

CP&L Response: The Brunswick 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 Brunswick are such that the location of such supports in the BWR generic test facility is prototypical, i.e., in each case (Brunswick and the test facility) there are supports near each change of direction in the pipe routing. Additionally, each SRVDL at Brunswick has only 2,3, or 4 spring hangers, all of which are located in the drywell. The spring hangers, snubbers, and rigid supports were designed to accommodate combinations of loads resulting from piping dead weight, thermal conditions, seismic and suppression pool hydrodynamic events, and a high pressure steam discharge transient.

The dynamic load effects on the piping and supports of the test facility due to the water discharge event (the alternate shutdown cooling mode) were found to be significantly lower than corresponding loads resulting from the high pressure steam discharge event. As stated in NEDE-24988-P, this finding is considered generic to all BWR's since the test facility was designed to be prototypical of the features pertinent to this issue. Futhermore, analysis of a typical Brunswick SRVDL configuration has confirmed the applicability of this conclusion to Brunswick.

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.

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

NRC QUESTION NO. 3:

Report NEDE-24988-P did not identify 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.

CP&L Response: No functional defic.encies or anomalies of the safety relief or relief valves w.re 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. The Wyle Laboratories test log sheet for the Target Rock two-stage valve tests is attached. This valve is used in the Brunswick Plant.

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 Table 4.2-1 (Attachment 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.

ATTACHMENT 1 (Response No. 3)

NEDE-24988-P

ANALYSIS OF GENERIC BWR SAFETY/RELIEF VALVE OPERABILITY TEST RESULTS

TABLE 4.2-1
SUMMARY OF REDUCED DATA
TARGET ROCK 6X10-2 STAGE S/RV WITH LOADS I SUPPORTS

Test Parameter		Test Data		
Description	Units	Steam, Saturated Run 301	Water, 15°F Subcooling Run 303	Water, 50°F Subcooling Run 307
Fluid inlet temperature	°F	555	249	215
Steam flow rate at ( ) psig	lbs/hr	838,900 (1128)	N/A	N/A
Average backpressure (Wyle Data)	%	25.5	N/A	N/A
Operability, open/closed upon command	yes/no	yes	yes	yes
Opening time, main valve disc	msec	24	*	*
Average water pressure	psig	N/A	270	264
Average water flow rate	gpm	N/A	6784	6619
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	N/A	N/A	yes

TABLE 4.2-1
SUMMARY OF REDUCED DATA
TARGET ROCK 6X10-2 STAGE S/RV WITH LOADS I SUPPORTS

Test Parameter		Test Data, Maximum Dynamic Values		
Description	Units	Steam, Saturated Run 301	Water, 15°F Subcooling Run 303	Water, 50°F Subcooling Run 307
SRVDL Acceleration (A2) - 2nd Section	"g's"	2	<1	<1
SRVDL Acceleration (A3) - 3rd Section	"g's"	<1	<1	<1
SRVDL Acceleration (A4) - 4th Section	"g's"	6	<1	<1
Load Cell (L1), 1st Horiz. Support	LBS	2,000	1,000	1,000
Load Cell (L3), 2nd Horiz. Support	LBS	13,000	3,000	2,000
Load Cell (L2), 3rd Horiz. Support	LBS	12,000	5,000	4,000
Load Cell (L4), 4th Horiz. Support	LBS	26,000	2,000	2,000
Stress (SG9), 5th Horiz. Support	psi	3,500	100	200
Stress (SG10), Steam Chest	psi	700	500	200
Stress (SG11), Steam Chest - Middle	psi	1,000	500	200
Stress (SG12), Steam Chest	psi	700	500	200
Stress (SG13), Sweepolet	psi	800	600	500
Stress (SG14), Sweepolet	psi	2,000	600	500
Stress (SG15), Sweepolet	psi	700	500	500
Stress (SG16), Sweepolet	psi	2,000	700	1,300

TABLE 4.2-1
SUMMARY OF REDUCED DATA
TARGET ROCK 6X10-2 STAGE S/RV WITH LOADS I SUPPORTS

Test Parameter		Test Data, Maximum Dynamic Values		
Description	Units	Steam, Saturated Run 301	Water, 15°F Subcooling Run 303	Water, 50°F Subcooling Run 307
SRVDL Stress (SG21), 1st Section	psi	200	200	200
SRVDL Stress (SG22), 1st Section	psi	1,500	500	500
SRVDL Stress (SG23), 1st Section	psi	200	100	200
SRVDL Stress (SG24), 1st Section	psi	1,500	400	500
SRVDL Stress (SG25), 2nd Section	psi	700	200	200
SRVDL Stress (SG26), 2nd Section	psi	800	200	200
SRVDL Stress (SG27), 3rd Section	psi	700	50	200
SRVDL Stress (SG28), 3rd Section	psi	700	100	200
SRVDL Stress (SG29), 3rd Section	psi	700	100	200
SRVDL Stress (SG30), 3rd Section	psi	700	50	200
SRVDL Stress (SG5), 4th Section	psi	1,000	100	200
SRVDL Stress (SG6), 4th Section	psi	1,000	100	200
SRVDL Stress (SG7), 4th Section	psi	1,500	100	200
SRVDL Stress (SG8), 4th Section	psi	3,200	100	200

NRC QUESTION NO. 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 in the plants. Describe the events and anticipated conditions at Brunswick 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 Brunswick. 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 Brunswick.

CP&L Response: The purpose of the safety/relief valve (S/RV) test program was to demonstrate that the S/RV will open and reclose under all expected flow conditions. The expected valve operating conditions were determined through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. Single failures were applied to these analyses so that the dynamic forces on the safety and relief valves would be maximized. Test pressures were the highest predicted by conventional safety analysis procedures. The BWR Owners' Group, in 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 valve. These events were identified by evaluating the initial events described in Regulatory Guide 1.70, Revision 2, with and without the additional conservatism of a single active component failure or operator error postulated in the event sequence. It was concluded from this evaluation that the alternate shutdown cooling mode is the only expected event which will result in liquid at the valve inlet. Consequently, this was the event simulated in the S/RV test program. This conclusion and the test results applicable to Brunswick are discussed below. The alternate shutdown cooling mode of operation has been described in the response to NRC Question No. 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, to 50, subcooled liquid at 20 psig to 250 psig. These fluid conditions envelope the conditions expected to occur at Brunswick 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 10 are applicable to the Brunswick plant because of its design and specific plant configuration. Three events, namely 5 (Transient HPCS, L8 Trip Failure), 6 (Transient RCIC Hd. Spr.), and 10 (SBA, HPCS, HPCS L8 Trip Failure) are not applicable to the Brunswick plant for the reasons listed in Attachment 2.

For the 10 remaining events, the Brunswick specific features, such as trip logic, power supplies, instrument line configuration, alarms and operator actions, have been compared to the base case analysis presented in the BWR Owners' Group submittal of September 17, 1980. The comparison has demonstrated that in each case, the base case analysis is applicable to Brunswick because the base case analysis does not include any plant features which are not already present in the Brunswick design or affect the transient. For these events, Table 1 demonstrates that the Brunswick specific features are included in the base case analyses presented in the BWR Owners' Group submittal of September 17, 1980. It is seen from Table 1 (Attachment 2), that all plant features assumed in the event evaluation are also existing features in the Brunswick plant. All features included in this base case analysis are similar to plant features in the Brunswick design. Furthermore, the time available for operator action is expected to be longer in the Brunswick 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. The test conditions envelope the plant conditions and will be bounded by the Emergency Procedures.

As discussed above, the BWR Owners' Group evaluated transients including single active failures that would maximize the dynamic forces on the S/RVs. 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 Brunswick plant-specific fluid conditions expected for the alternate shutdown cooling mode of operation.

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 (Note 3*)	#6 Transient RCIC Hd. Spr. (Note 4*)	#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 (Note 3*)	#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 (Note 1*)	X/S(1*)	X/S(1*)											
Reactor Scram on Neutron Flux Monitor		X/S											
Reactor Scram on MSIVs Closure (Note 2*)		X/S(2*)											
Reactor Scram on High Radiation								X/S					
Reactor Scram on High Drywell Pressure									X/S	X/N/A (3*)	X/S	X/S	X/S
Reactor Scram on Low Water Level													X/S
Reactor Isolation on Low Water Level													X/S

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

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~~ (Note 3*)	#6 ~~Transient RCIC Hd. Spr.~~ (Note 4*)	#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 (Note 3*)	#11 SBA, HPCI, HPCI L8 Trip Failure	#12 SBA, Depress. & ECCS Over., Operator Error	#13 LBA, ECCS Overf Brk Isol
Low Pressure ECCS Initiation on High Drywell Pressure												X/S	X/S
Low Pressure Initiation on Low Water Level (Note 5*)													X/S(6*)
FW Pumps Trip on Low Suction Pressure	X/S												
HPCI Trip on High Backpressure			X/S								X/S		
RCIC Trip on High Backpressure				X/S					X/S				
Turbine Trip on Vessel High Level	X/S	X/S											
MSIVs Closure on Low Turbine Inlet Pressure (Note 6*)	X/S(6*)	X/S(6*)						X/S(6*)					
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. 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 (Note 3*)	#6 Transient RCIC Hd. Spr. (Note 4*)	#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 (Note 3*)	#11 SBA, HPCI, HPCI L8 Trip Failure	#12 SBA, Depress. & ECCS Over., Operator Error	#13 LBA, ECCS Overf Brk Isol
High Water Level 7 Alarm	X / S		X / S	X / S	X / N/A (3*)				X / S	X / N/A (3*)	X / S	X / S	X / S
High Drywell Pressure Alarm													
FW Level 8 Trip	X / S	X / S											
RCIC Level 8 Trip			X / S	X / S	X / N/A (3*)				X / S	X / N/A (3*)	X / S		X / S
HPCS Level 8 Trip (Note 3*)				X / N/A (3*)	X / N/A (3*)				X / N/A (3*)	X / N/A (3*)			X / N/A (3*)
HPCI Level 8 Trip			X / S	X / S					X / S		X / S		X / S
HPCI/S and RCIC Initiation on Low Water Level	X / S	X / S	X / S	X / S	X / N/A (3*)	X / N/A (4*)		X / S	X / S				X / S
HPCI/S initiation on High Drywell Pressure			X / S	X / S					X / S	X / N/A (3*)	X / S	X / S	X / S
RCIC Initiation on High Drywell Pressure													X / N/A ([illegible])

TABLE 1 - EVENTS EVALUATED

NOTES

1. Reactor will scram if at greater than 30% power. Below 30% power, reactor scram would occur due to reactor low water level after reactor feed pumps are tripped or by manual scram.

2. Reactor scram from MSIV closure only occurs in the Run mode. Reactor scram would occur from reactor low water level after reactor feed pumps are tripped in the Startup mode.

3. No HPCS System at Brunswick Plant.

4. No head spray to RCIC Systm at Brunswick Plant.

5. Only if reactor pressure is less than 410 psi. Low pressure would exist for a LBA.

6. MSIV closure on low turbine inlet pressure occurs only in the Run mode. In the Startup mode, MSIV closure would occur after the reactor feed pumps trip due to reactor low water level or other existing signal.

7. While RCIC would not automatically initiate on high drywell pressure, it would initiate on the reactor low water level that would exist during a LBA.

NRC QUESTION NO. 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?

CP&L Response: The BWR safety/relief valve (S/RV) operability test program was designed to simulate the alternate shutdown cooling mode, which is the only expected liquid discharge event for Brunswick.

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 S/RV's to discharge steam to the suppression pool. If S/RV 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 one S/RV 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 S/RV 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 S/RV 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 S/RV is required for the alternate shutdown cooling mode, and no cycling of the S/RV was performed for the generic BWR S/RV operability test program.

The ability of the Brunswick S/RV 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 S/RV'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 NO. 6:

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

CP&L Response: The flow coefficient, C_V, for the Target Rock two-stage safety/relief valve (S/RV) utilized in Brunswick was determined in the generic S/RV test program (NEDE-24988-P). The average flow coefficient calculated from the test results for the Target Rock two-stage S/RV is reported in Table 5.2-1 of NEDE-24988-P. This test value has been used by Carolina Power & Light to confirm that the liquid discharge flow capacity of the Brunswick S/RV'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 S/RV test demonstrates that the Brunswick S/RV's are capable of returning the flow injected by the RHR or CS pump to the suppression pool.

If it was necessary for the operator to place the Brunswick plant 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 vessel temperature.

The flow coefficient for the Target Rock two-stage valve reported in NEDE-24988-P was determined from the S/RV 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 feet downstream of the valve and the corresponding measured flow rate. Furthermore, the test conditions and test configuration were representative of Brunswick 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 Brunswick Plant.