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O. W. DIXON, JR.
VICE PRESIDENT
NUCLEAR OPERATIONS

July 30, 1982

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

Subject: Virgil C. Summer Nuclear Station
Docket No. 50/395
Safety and Relief Valve Report;
NUREG 0737 Item II.D.1

Dear Mr. Denton:

In response to NUREG 0737, Item II.D.1, South Carolina Electric & Gas Company (SCE&G) has participated in the Electric Power Research Institute (EPRI) Safety and Relief Valve test program to demonstrate the operability of the pressurizer Safety Valves, Power Operated Relief Valves (PORV's) and the PORV Block Valves and the adequacy of the piping and supports associated with these components.

The included attachments will address each area as follows:

ATTACHMENT 1 - Safety Valve Performance Evaluation
ATTACHMENT 2 - Power Operated Relief Valve Performance Evaluation
ATTACHMENT 3 - PORV Block Valve Performance Evaluation
ATTACHMENT 4 - Pressurizer Relief System Piping and Support Evaluation

The basic conclusion is that the valves have been demonstrated to perform their intended function as described in the subject NUREG and the piping and supports are adequate for design loads during valve operation. Any variations in valve performance or test conditions are discussed in the identified attachments.

The final design verification of the analysis contained in Attachment 4 is in progress and is expected to be complete by mid-August. We do not anticipate any change in the results of the analysis or the conclusions based on this analysis.

This submittal in conjunction with the April 1, 1982 submittal constitute a final report on NUREG 0737, Item II.D.1.

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If you have any questions or require additional information, please advise.

Very truly yours,

O. W. Dixon, Jr.

MDQ:OWD:glb

Attachments

cc: V. C. Summer w/o atts.
G. H. Fischer w/o atts.
H. N. Cyrus
T. C. Nichols, Jr. w/o atts.
O. W. Dixon, Jr.
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J. P. O'Reilly
H. T. Babb
D. A. Nauman
C. L. Ligon (NSRC)
W. A. Williams, Jr.
R. B. Clary
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G. J. Braddick
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NPCF
File

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ATTACHMENT 1

Safety Valve Performance Evaluation

July, 1982

References

References: a) Letter from T. C. Nichols, Jr., South Carolina Electric and Gas Company, to H. R. Denton, dated April 1, 1982, with attachments.

b) WCAP 10105, "Review of Pressurizer Safety Valve Performance As Observed In The EPRI Safety and Relief Valve Test Program," June, 1982.

Reference (a) and the attendent EPRI Reports documented the Safety Valves installed at the V. C. Summer Nuclear Station (VCS) and that the Safety Valves tested during the EPRI PWR Safety and Relief Valve Test Program represent the safety valves installed at VCS. The conditions for which the valves were tested envelope the range of expected operating and accident conditions for VCS. Furthermore, the testing demonstrated the functionability of the safety valves while identifying some anomalies in valve performance.

Westinghouse, through the Westinghouse Owners Group, has evaluated the observed safety valve performance during full scale testing and submitted reference (b). The discussion regarding upstream safety valve piping pressure oscillations is not applicable to VCS.

VCS utilizes a hot loop seal whose temperature exceeds that addressed in reference (b) with regard to pressure oscillations in the upstream piping. These pressure oscillations are addressed in Attachment 4 of this submittal and have been found to be acceptable.

In addition to reduced loads on the downstream piping, the hot loop seal also improves valve performance as shown in the 350°F loop seal tests.

Crosby Valve and Gage Co., the safety valve vendor, has evaluated the safety valve performance against VCS plant specific piping and valve information and determined that valve performance at VCS should be as good or better than observed valve performance during the EPRI test program.

Valve accelerations and nozzle loads are discussed in Attachment 4 of this submittal.

Based on the test program output, the evaluations of reference (b), the valve vendor evaluation and the plant specific piping and support evaluation, the operability of the VCS Safety Valves has been demonstrated in accordance with the requirements of NUREG 0737.

Letter From O. W. Dixon, Jr., SCE&G, To H. R. Denton, Dated July 30, 1982

ATTACHMENT 2

Power Operated Relief Valve Performance Evaluation

July, 1982

References: a) Letter from T. C. Nichols, Jr., South Carolina Electric and Gas Company, to H. R. Denton, dated April 1, 1982, with attachments.

Reference (a) and the attendent EPRI Reports documented the Power Operated Relief Valves (PORV's) installed at the V. C. Summer Nuclear Station (VCS) and that the PORV's tested during the EPRI PWR Safety and Relief Valve Test Program represent the PORV's installed at VCS. The conditions for which the valves were tested envelope the range of expected operating and accident conditions for VCS. Furthermore, the testing demonstrated the operability of the PORV's.

Since that time, Westinghouse has completed the VCS plant specific cold overpressure protection analysis. Test conditions envelope these results. In order to provide complete information, Appendix A of this attachment contains the revised pages of the "Valve Inlet Fluid Conditions for Pressurizer Safety and Relief Valves in Westinghouse Designed Plants."

Valve nozzle loads and accelerations are discussed in Attachment 4 of this submittal.

Based on the test program output and the results of the plant specific piping and support evaluation, the operability of the VCS PORV's has been demonstrated in accordance with the requirements of NUREG 0737.

Letter From O. W. Dixon, Jr., SCE&G, To H. R. Denton, Dated July 30, 1982

APPENDIX A

"Valve Inlet Fluid Conditions for Pressurizer Safety and Relief Valves in Westinghouse Designed Plants"

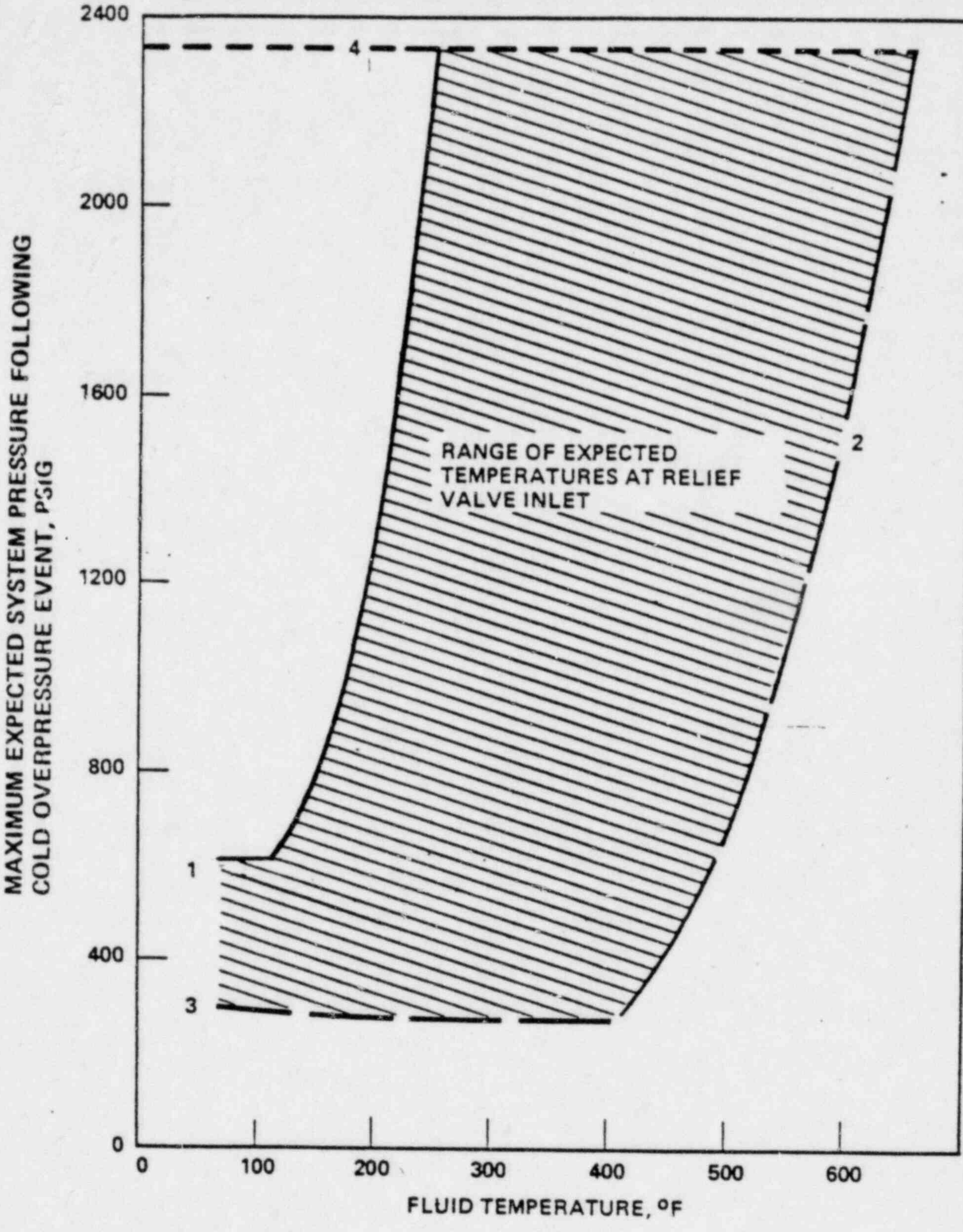

Figure 5-1 (Revised) Potential Cold Overpressure Fluid Conditions at the Relief Valve Inlet

Two-Loop Plants	Name	Owner
NSP	Prairie Island #1	Northern States Power
NRP	Prairie Island #2	Northern States Power
WPS	Kewaunee	Wisconsin Public Service

Three-Loop Plants	Name	Owner
SCE	San Onofre #1	Southern California Edison
CPL	H. B. Robinson #2	Carolina Power & Light Co.
FPL	Turkey Point #3	Florida Power & Light Co.
FLA	Turkey Point #4	Florida Power & Light Co.
VPA	Surry #1	Virginia Electric & Power Co.
VIR	Surry #2	Virginia Electric & Power Co.
DLW	Beaver Valley #1	Duquesne Light Co.
VRA	North Anna #1	Virginia Electric & Power Co.
ALA	Joseph M. Farley #1	Alabama Power Co.
VGB	North Anna #2	Virginia Electric & Power Co.
APR	Joseph M. Farley #2	Alabama Power Co.
CGE	* Virgil C. Summer #1	South Carolina Electric & Gas
DMW	Beaver Valley #2	Duquesne Light Company
CQL	Shearon Harris #1	Carolina Power & Light Co.
CRL	Shearon Harris #2	Carolina Power & Light Co.
CSL	Shearon Harris #3	Carolina Power & Light Co.
CTL	Shearon Harris #4	Carolina Power & Light Co.

Four-Loop Plants	Name	Owner
IPP	Indian Point #2	Consolidated Edison Co. of New York
INT	Indian Point #3	Power Authority, State of New York
CWE	Zion #1	Commonwealth Edison
COM	Zion #2	Commonwealth Edison
AEP	Donald C. Cook #1	American Electric Power Co.
AMP	Donald C. Cook #2	American Electric Power Co.
PGE	Diablo Canyon #1	Pacific Gas & Electric Power

2.3 COLD OVERPRESSURE TRANSIENTS

2.3.1 Mass Input Events

Based on probability of occurrence and in-plant operating experience, the most credible mass input events producing a net injection of mass into the reactor coolant system (RCS) involve failure in the air supply system, which causes the charging flow control valve to open, and/or isolation of letdown. Mass injection based on single charging pump operation is the most likely mass input mechanism, producing typical charging rates up to 120 gpm following isolation of letdown, and higher rates for air supply system failure.

Although precluded at low temperature by administrative procedure, two-charging pump operation was considered in all plants except Virgil C. Summer to develop maximum input capability and thus provide additional flexibility in the operation of the cold overpressure mitigation system. For V.C. Summer, consistent with administrative procedure, single charging pump operation was considered, together with the higher mass input rates associated with air supply system failure. Maximum input capability associated with these mechanisms as applied to all plants analyzed to date is shown in Figure 2-1. The PORV inlet conditions presented in Figure 5-1 also include these mechanisms.

Operation of the PORV at a predetermined setpoint pressure is employed by Westinghouse in the Cold Overpressure Mitigation System (OMS) to arrest the pressure transient caused by the above mechanisms. Mitigation of the transient on valve opening results in the RCS pressure turning over. This produces a transient peak overpressure. The PORV continues to open until valve capacity matches the net mass injection rate, after which the reset pressure is reached and the valve begins to close. PORV closure arrests the decreasing RCS pressure and reinitiates the pressure increase to complete the pressure transient cycle. This mimimum pressure is termed the transient pressure undershoot and is determined by the blowdown setting of the PORVs (nominally 20 psi). Pressure cycling continues until action is taken to remove the mass input mechanism.

Selection of PORV setpoints for pressure control of mass input-induced transients are based on a water-solid reactor coolant system, which produces pressure excursions significantly higher than for a RCS with

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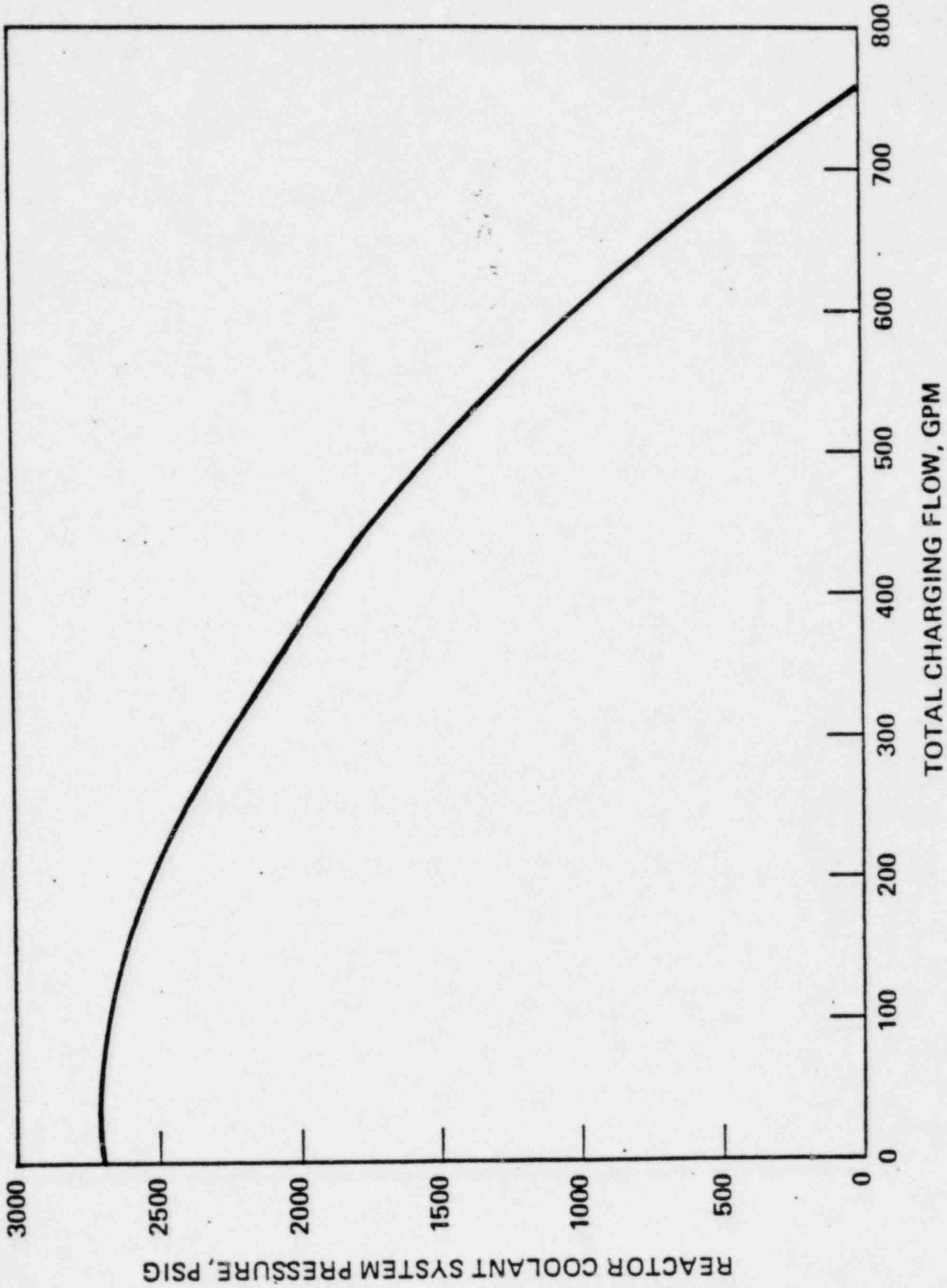

Figure 2-1. Maximum Charging Capacity versus Reactor Coolant System Pressure

a pressurizer steam cushion. Setpoint selection is also based on an algorithm which considers the reactor vessel NDT pressure limit and the integrity of the reactor coolant pump No. 1 Seal.

Except for the V.C. Summer Plant*, valve opening and closure times of 2 seconds are assumed, and valve setpoints are staggered such that operation of the first valve will mitigate the event so that the other valve will not be challenged.

2.3.2 Heat Input Events

The heat input case which has the potential for the most severe pressure transient is that in which the steam generators exhibit a higher temperature than the remainder of the reactor coolant system. The magnitude of the difference in temperature is dependent on the means by which the temperature asymmetry was achieved, but a typical difference is considered to be about $50^{o}F$.

For the heat input transient with the initial reactor coolant temperature $50^{o}F$ less than the temperature in the steam generators and with all reactor coolant pumps off, one of the two reactor coolant pumps is started to circulate the reactor coolant through the warmer steam generators. As the coolant flow begins, the warm water in the tubes of the steam generator in the active loop is forced out and into the reactor coolant pump where it is pumped into and mixed with the colder reactor coolant. In the inactive loops, the warmer water from the tubes of the steam generator is forced out in a reverse direction due to the backflow in the inactive loops, and also mixed with the cooler reactor coolant. This initial mixing of the warm water with the larger volume of cooler water causes an initial shrinkage effect which tends to decrease the initial coolant pressure.

Simultaneously, the cooler reactor coolant that enters the steam generator begins to be heated as it moves through the tube bundle. As heat is added to the coolant due to heat transfer from the secondary water in the steam generator, the coolant attempts to expand and cause a resultant pressure increase. The net effect of the expansion due to the heat transferred to the coolant and the shrinkage effect due to the mixing of the warm water with the cooler coolant is a relatively

*Measured valve opening/closure times for V.C. Summer were 1.5 sec/1.0 sec

The fluid conditions at the inlet to safety valves range from 667°F to 672°F at 2507 psia with a maximum discharge rate of 628.3 gpm. No liquid discharge from the safety valves of the 2- and 3-loop reference plants was observed during the analysis. The fluid conditions at the inlet to PORVs range from 498°F to 569°F at 2353 psia with a maximum discharge rate of 1104.1 gpm. In this case no liquid discharge from the PORVs of the 2-loop reference plant is observed during the interval that the transient was analyzed.

In general valves open on steam and no liquid discharge is observed until the pressurizer becomes water solid. This is plant dependent and can vary anywhere from 20 minutes to more than six hours.

5.4 PLANT-SPECIFIC VALVE INLET CONDITIONS RESULTING FROM COLD OVERPRESSURIZATION EVENTS

Setpoints for the cold overpressurization mitigation system are conservatively determined to accommodate the rapid pressurization rates (up to 100 psi/sec) produced by cold overpressure transients (Section 2.3) during water-solid, low temperature operation of the reactor coolant system. In practice, however, fluid conditions at the relief valve inlet are not restricted to low temperature, subcooled water. A variable fluid condition (steam or water) and temperature (saturated to subcooled) at the valve inlet is possible due to administrative requirements for maintaining a pressurizer steam bubble during low temperature operations when pressure excursions due to cold overpressurization events are a possibility (Section 4.3).

The maximum range of potential cold overpressure fluid conditions at the relief valve inlet, covering all Westinghouse plants analyzed to date, may be inferred from Figure 5-1. These plants include: Comanche Peak Units 1 and 2, SNUPPS, Sequoyah Units 1 and 2, Watts Bar Units 1 and 2, South Texas Units 1 and 2, Byron/Braidwood Units 1 and 2, and Virgil C. Summer. A description of the indexed curves used to define the range of potential fluid conditions is presented below.

<u>Legend Applicable To Figure 5-1</u>

Index	Description
1	Locus of maximum primary system pressures developed following PORV #2 operation (limiting condition/ water-solid RCS)

Letter From O. W. Dixon, Jr., SCE&G, To H. R. Denton, Dated July 30, 1982

ATTACHMENT 3

PORV Block Valve Performance Evaluation

July, 1982

References: a) Letter from T. C. Nichols, Jr., South Carolina Electric and Gas Company, to H. R. Denton, dated July 29, 1981.

b) Letter from R. C. Youngdahl, Chairman, EPRI Research Advisory Committee, Consumers Power Company, to H. R. Denton, dated June 1, 1982, with attachments.

As stated in reference (a) and documented in reference (b), the PORV Block Valves installed at the V. C. Summer Nuclear Station (VCS) are the same as the Westinghouse 3GM88 valves with the Limitorque type SB-00-15 operator that were tested at the Marshall Steam Station in conjunction with the EPRI Safety and Relief Valve test program. These tests demonstrated the operability of the PORV Block Valves during full flow steam conditions.

Additionally, during the start-up test program, a less rigorous but equally demonstrative test was performed on each PORV Block Valve at VCS by successfully stroking them closed and then open with the PORV open at normal system operating temperature and pressure.

Included in reference (b) is a report prepared by Westinghouse that details tests and analyses performed on Westinghouse Gate Valves to evaluate valve performance. Of significance in this report is that the Limitorque SB-00-15 develops adequate stem thrust to close the valve and that the saturated steam fluid condition poses the greatest challenge to gate valve closure.

The successful testing of the VCS plant specific valves at Marshall, the in-plant tests during Hot Functional Test 2, and the tests, analyses and conclusions by Westinghouse conclusively demonstrates the PORV Block Valve operability for fluid conditions described in NUREG 0737, Item II.D.1. Furthermore, the operability demonstrated by the PORV's for all expected inlet fluid conditions greatly enhances the expected reliability of the PORV Block Valves.