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

June 29, 1983

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

Subject: Virgil C. Summer Nuclear Station
Docket No. 50/395
Operating License No. NPF-12
Cold Overpressure Protection

Dear Mr. Denton:

South Carolina Electric and Gas Company (SCE&G) currently utilizes a cold Overpressurization Mitigation System (OMS) based on the protection afforded by two (2) pressurizer power operated relief valves (PORVs) augmented by administrative procedures. A complete description of this system is available in Section 5.2 of the Virgil C. Summer Nuclear Station Final Safety Analysis Report (FSAR).

In Supplement Four of the Virgil C. Summer Nuclear Station Safety Evaluation Report (SER), NUREG 0717, the NRC Staff concluded that no credible low-temperature overpressurization transients would fail the reactor vessel during first cycle operation. However, License Condition 2.C.10 states: "Prior to startup after the first refueling outage, SCE&G shall install an NRC staff-approved low-temperature overpressurization protection system. The preliminary design shall be provided for NRC Staff review no later than June 30, 1983."

SCE&G, in conjunction with the work being completed to obtain NRC approval of PORVs for cold overpressure protection, initiated an engineering evaluation to determine the feasibility of utilizing the mechanical relief valves in the Residual Heat Removal System (RHRS) for mitigating overpressurization during low temperature operation. The results of the engineering evaluation were positive, lending substantial credibility toward making a decision to utilize RHR relief valves as the most reliable and efficient method of mitigating cold overpressure transients.

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The present OMS employing the PORVs to mitigate an overpressurization transient is operational. SCE&G plans to utilize RHR relief valves as the method of mitigating cold overpressure transients in the future, but will continue to operate the present system until NRC Staff approval of the OMS based on the RHR relief valves is received. A Safety Analysis and appropriate attachments are included and constitute a preliminary design for NRC Staff review as required by License Condition 2.C.10. The SCE&G engineering design verification is continuing and is expected to be completed in September 1983. Any additional information resulting from the design verification will be forwarded for NRC Staff review. The revision to the FSAR will be incorporated after NRC approval is received.

Please respond in writing with your evaluation. If you require any additional information or have any questions, please let us know.

Very truly yours,

O. W. Dixon, Jr.

GMF:WRM:RBC:OWD/fjc

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1.0 SYSTEM DESCRIPTION

The Reactor Coolant System (RCS) cold Overpressurization Mitigating System (OMS) utilizes the Residual Heat Removal System Relief Valves (RHRSRVs) to mitigate overpressurization transients. The RHRSRVs are manufactured by the Crosby Valve and Gage Company (Figure 4). These relief valves are spring-loaded, bellows-type valves, one (1) installed on each RHRS suction header. The RHRSRVs have a setpoint of 450 psig, and are fully open at 495 psig (design accumulation of 10%). These valves discharge to the pressurizer relief tank which provides a conservatively assumed maximum back pressure of 100 psig. There are two (2) motor operated RHRS isolation valves between each of the RHRSRVs and the RCS. The RHRS suction lines are automatically isolated when RCS pressure exceeds 700 psig which is approximately 200 psig greater than the peak pressure resulting from the postulated heat input transient (Figure 3). The general arrangement of the RHRS isolation valves and RHRSRVs is attached (Figure 1) and is shown in more detail in the Virgil C. Summer Nuclear Station Final Safety Analysis Report (FSAR), Figure 5.5-4. Additionally, a comprehensive description of the RHRS can be found in the FSAR, Section 5.5.7.

The RHRSRVs are independent of any electric power supply or control circuitry and as such require no IEEE-279, Criteria 1971 qualification. The design of the circuitry associated with the motor operated RHRS isolation valves meets the requirements of IEEE-279, Criteria 1971. An alarm in the control room will be provided to inform the operator of any overpressurization transient exceeding the 450 psig setpoint of the RHRSRVs.

The RHRS piping and valves are classified as ASME Boiler and Pressure Vessel Code Section III, Code Class 2, Seismic Category I. System components important to safety are classified in accordance with the August 1970 Draft of the American National Standard Institute (ANSI) N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." The RHRS piping and valves are ANS Safety Class 2a. Further discussion of compliance with Regulatory Guide 1.26 can be found in the FSAR, Appendix 3A.

2.0 OPERATING BASIS EARTHQUAKE

The OMS is designed to function during an Operating Basis Earthquake (OBE) in addition to experiencing no degradation as a result of the OBE. The RHRS piping and valves are designed and classified as Seismic Category I. Compliance with Regulatory Guide 1.29 is discussed in the Virgil C. Summer Nuclear Station FSAR, Appendix 3A.

3.0 PRESSURE TRANSIENT ANALYSIS

The OMS design utilizing the RHRSRVs is capable of mitigating the consequences of cold overpressurization transients, thus maintaining RCS pressures within 10CFR50, Appendix G, limits. The OMS design is based on the theoretical "worst case" mass input and heat input event.

The mass input transient analysis was performed assuming the most severe event involves a single centrifugal charging pump. Specifically, a loss of instrument air is postulated, whereby the flow control valve on the charging line fails open and simultaneously the flow control valve on the RCS letdown line fails closed. Assuming zero RCS pressure, a maximum delivery of 650 gpm for the centrifugal charging pump is expected. The charging flow rate at an RCS pressure of 450 psig (RHRSRVs setpoint) is approximately 595 gpm. For the severe event, one (1) RHRSRV will allow RCS pressure to peak at approximately 478 psig (Figure 2). This pressure is within 10CFR50, Appendix G, limits.

The heat input transient mechanism considered for analysis involves a Reactor Coolant Pump (RCP) startup in one (1) loop with a water solid condition and temperature asymmetry in the RCS. The two most influential factors in the heat input transient are the magnitude of the secondary-to-primary temperature differential, and the initial temperature. With the start of a RCP, a rapid acceleration in pressure will result. This marked increase in pressure is due primarily to fluid expansion accompanying the RCS temperature increase. For the most severe event, a 50°F secondary-to-primary temperature differential and 300°F RCS temperature is assumed. One (1) RHRSRV is capable of mitigating the consequences of the overpressurization transient resulting from the aforementioned pressure increase. The analysis results provided (Figure 3) demonstrate that the RCS pressure is within 10CFR50, Appendix G, limits.

The OMS is designed to prevent RCS pressures in excess of 10CFR50, Appendix G, limits in the event of a single failure in addition to the occurrence initiating the transient. Power supplies for the motor operated RHRS isolation valves are designed to ensure that no single failure of the electrical system or loss of offsite power would isolate both of the RHRSRVs.

4.0 ADMINISTRATIVE PROCEDURES

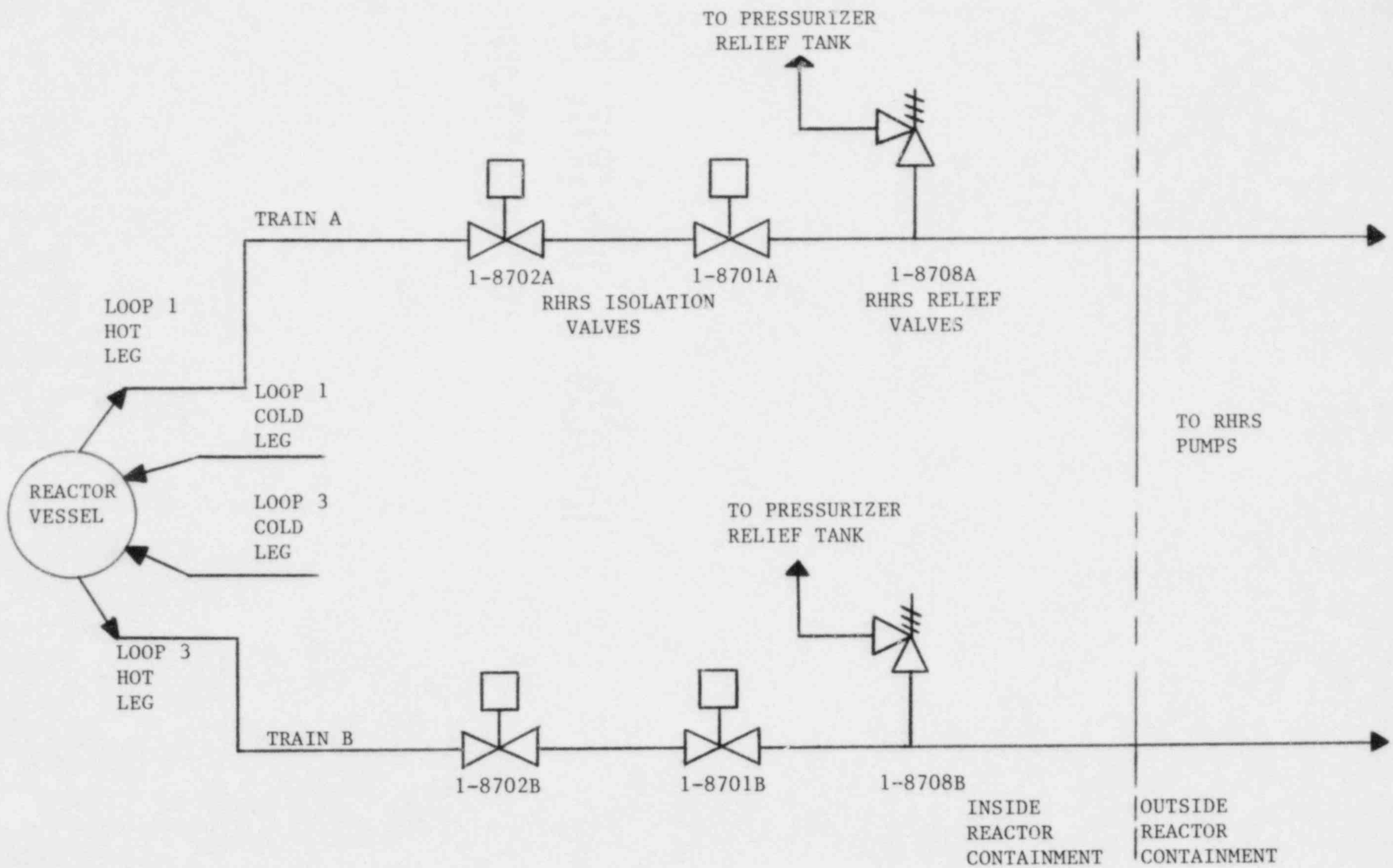
Although the system described previously mitigates the consequences of cold overpressurization transients, administrative procedures are recommended for minimizing the potential for any transient to challenge the OMS.

Of primary importance is the basic method of operation of the plant. Normal plant operating procedures will maximize the use of a pressurizer cushion (steam/nitrogen bubble) during periods of low pressure, low temperature operation. This cushion will dampen the plant's response to potential transient generating inputs, providing easier pressure control with slower response rates.

An adequate cushion reduces the severity of some potential transients such as RCP-induced heat input and slows the rate of pressure rise for others. For modes of operation where water solid plant operation is possible, procedures further highlight precautions that minimize the potential for developing overpressurization transients. For example, Technical Specification 3.1.2.4 provides a clear indication that a maximum of one (1) centrifugal charging pump shall be operable whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F. These procedures provide reasonable assurance that most potential transients can be terminated by operator action before the OMS is challenged.

Figure 1

GENERAL ARRANGEMENT OF THE RHRS ISOLATION AND RELIEF VALVES



VIRGIL C. SUMMER NUCLEAR STATION MASS INPUT TRANSIENT ANALYSIS RESULTS					
REACTOR COOLANT SYSTEM TEMPERATURE (°F)	RHRS RELIEF VALVE SETPOINT PRESSURE (PSIG)	RHRS RELIEF VALVE BACK PRESSURE (PSIG)	*DESIGN MASS INPUT (GPM)	REACTOR COOLANT SYSTEM PEAK PRESSURE (PSIG)	APPENDIX G REACTOR COOLANT SYSTEM PRESSURE LIMIT (PSIG)
350	450	100	595	475.7	NONE
300	450	100	595	476.2	NONE
250	450	100	595	476.6	1812
200	450	100	595	476.9	1088
150	450	100	595	477.2	700
100	450	100	595	477.4	575

*One Charging Pump, Delivery at 450 PSIG
Reactor Coolant System Pressure

Figure 2

VIRGIL C. SUMMER NUCLEAR STATION HEAT INPUT TRANSIENT ANALYSIS RESULTS							
STEAM GENERATOR SECONDARY TEMPERATURE (°F)	REACTOR COOLANT SYSTEM TEMPERATURE (°F)	SECONDARY TO PRIMARY TEMPERATURE DIFFERENTIAL (°F)	RHRS RELIEF VALVE SETPOINT PRESSURE (PSIG)	RHRS RELIEF VALVE BACKPRESSURE (PSIG)	EXPANSION EXCESS VOLUME RELIEF (GPM)	*REACTOR COOLANT SYSTEM PEAK PRESSURE (PSIG)	APPENDIX G REACTOR COOLANT SYSTEM PRESSURE LIMIT (PSIG)
350	300	50	450	100	967	501.5	NONE
300	250	50	450	100	768	488	1810
250	200	50	450	100	572	475	1088
200	150	50	450	100	347	462.3	700
150	100	50	450	100	256	458.8	575

*Relief Valve Flow Equal to Expansion Excess Volume
Based on Steam Generator Tube Surface Area of 48,000 Ft.^{2}

Figure 3

RESIDUAL HEAT REMOVAL RELIEF VALVES
MANUFACTURER'S DATA

Manufacturer	Crosby Valve and Gage
Valve Size & Type	3" Inlet x 4" Outlet Type B
Valve Style	JB-35-TD-WR
Pressure Setpoint	450 PSIG
Design Temperature	400 °F
Pressure Class	600 PSIG x 150 PSIG
Orifice Size/Area	Size L/2.853 $in.^2$
Design Accumulation	10%
Design Features	Spring Loaded Bellows Type
Tag Numbers	XVR-8708A XVR-8708B

Figure 4