PSE&G Public Service Electric and Gas Company

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Robert L. Mittl General Manager
Nuclear Assurance and Regulation

February 15, 1985

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, MD 20814

Attention: Mr. Albert Schwencer, Chief
Licensing Branch 2
Division of Licensing

Gentlemen:

SAFETY EVALUATION REPORT
OPEN AND CONFIRMATORY ITEM STATUS
HOPE CREEK GENERATING STATION
DOCKET NO. 50-354

Attachment 1 is a current list which provides a status of the open and confirmatory items identified in Sections 1.7 and 1.8 of the Safety Evaluation Report (SER). Items identified as "complete" are those for which PSE&G has provided responses and no confirmation of status has been received from the staff. We will consider these items closed unless notified otherwise. In order to permit timely resolution of items identified as "complete" which may not be resolved to the staff's satisfaction, please provide a specific description of the issue which remains to be resolved.

Enclosed for your review and approval (see Attachment 3) are the resolutions to the SER items listed in Attachment 2.

Should you have any questions or require any additional information on these items, please contact us.

Very truly yours,

R L Mittl

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Attachments

C D. H. Wagner
USNRC Licensing Project Manager (w/attach.)

A. R. Blough
USNRC Senior Resident Inspector (w/attach.)

ATTACHMENT 1

Item No.	Subject	Status	R. L. Mittl to A. Schwencer ltr. dated
OI-1	Riverborne Missiles	Partial Response	1/31/85
OI-2	Equipment Qualification	Partial Response	2/1/85
OI-3	Preservice Inspection Program	Partial Response	2/14/85
OI-4	GDC 51 Compliance	Open	
OI-5	Solid-State Logic Modules	NRC Action	
OI-6	Postaccident Monitoring Instrumentation	NRC Action	
OI-7	Minimum Separation Between Non-Class IE Conduit and Class IE Cable Trays	Open	
OI-8	Control of Heavy Loads	Completed	1/18/85
OI-9	Alternate and Safe Shutdown	NRC Action	
OI-10	Delivery of Diesel Generator Fuel Oil and Lube Oil	Closed	Amendment 8
OI-11	Filling of Key Management Positions	Open	
OI-12	Training Program Items	Completed	1/7/85
	(a) Initial Training Program	Completed	12/28/84
	(b) Requalification Training Program	Completed	1/7/85
	(c) Replacement Training Program	Completed	1/7/85
	(d) TMI Issues I.A.2.1, I.A.3.1, and II.B.4	Completed	1/7/85
	(e) Nonlicensed Training Program	Completed	1/7/85
OI-13	Emergency Dose Assessment Computer Model	Closed	1/7/85
OI-14	Procedures Generation Package	Closed	1/28/85
OI-15	Human Factors Engineering	Open	

Item No.	Subject	Status	R. L. Mittl to A. Schwencer ltr. dated
C-1	Feedwater Isolation Check Valve Analysis	Open	
C-2	Plant-unique Analysis Report	Completed	1/8/85, 1/11/85, & 1/31/85
C-3	Inservice Testing of Pumps and Valves	Open	
C-4	Fuel Assembly Accelerations	Completed	Amendment 8
C-5	Fuel Assembly Liftoff	Completed	Amendment 8
C-6	Review of Stress Report	Open	
C-7	Use of Code Cases	Completed	12/17/84
C-8	Reactor Vessel Studs and Fastners	Completed	2/15/85
C-9	Containment Depressurization Analysis	NRC Review	
C-10	Reactor Pressure Vessel Shield Annulus Analysis	NRC Review	
C-11	Drywell Head Region Pressure Response Analysis	NRC Review	
C-12	Drywell-to-Wetwell Vacuum Breaker Loads	NRC Review	
C-13	Short-Term Feedwater System Analysis	Open	
C-14	Loss-of-Coolant-Accident Analysis	Open	
C-15	Balance-of-Plant Testability Analysis	Completed	Amendment 8
C-16	Instrumentation Setpoints	Completed	2/15/85
C-17	Isolation Devices	Open	
C-18	Regulatory Guide 1.75	NRC Review	
C-19	Reactor Mode Switch	Open	
C-20	Engineered Safety Features Reset Controls	Open	

Item No.	Subject	Status	R. L. Mittl to A. Schwencer ltr. dated
C-21	High Pressure Coolant Injection Initiation	Open	
C-22	IE Bulletin 79-27	Completed	Amendment 8
C-23	Bypassed and Inoperable Status Indication	NRC Review	
C-24	Logic for Low Pressure Coolant Injection Interlock Circuitry	Open	
C-25	End-of-Cycle Recirculation Pump Trip	Open	
C-26	Multiple Control System Failures	NRC Review	
C-27	Relief Function of Safety/Relief Valves	Completed	2/15/85
C-28	Main Steam Tunnel Flooding Analysis	Open	
C-29	Cable Tray Separation Testing	Open	
C-30	Use of Inverter as Isolation Device	Open	
C-31	Core Damage Estimate Procedure	Open	
C-32	Continuous Airborne Particulate Monitors	Open	
C-33	Qualifications of Senior Radiation Protection Engineer	Open	
C-34	Onsite Instrument Information	Open	
C-35	Airborne Iodine Concentration Instruments	Open	
C-36	Emergency Plan Items	Partial Response	11/9/84, 1/16/85, & 2/7/85
C-37	TMI Item II.K.3.18	Open	

ATTACHMENT 2

ITEM NO.	SER SECTION	SUBJECT
C-8	5.3.1.5	Reactor vessel studs and fasteners
C-16	7.2.2.5	Instrumentation Setpoint
C-27	7.7.2.2	Relief Function of Safety/Relief Valves

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ATTACHMENT 3

SER ITEM NO. C-8

REACTOR VESSEL STUDS AND FASTENERS

The reactor vessel studs and fasteners satisfy most of the recommendations of RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs." The FSAR does not discuss the nondestructive examinations of the bolts and nuts, and the applicant needs to confirm that the Code-specified inspections were performed. This is a confirmatory issue.

RESPONSE

FSAR Section 5.3.1.7 has been revised to provide the information requested above.

5.3.1.7 Reactor Vessel Fasteners

The reactor vessel closure head (flange) is fastened to the reactor vessel shell flange by multiple sets of threaded studs and nuts. The lower end of each stud is installed in a threaded hole in the vessel shell flange. A nut and washer are installed on the upper end of each stud. The proper amount of preload can be applied to the studs by a sequential tensioning using hydraulic tensioners. The design and analysis of this area of the vessel is in full compliance with all ASME B&PV Code, Section III, Class I requirements. The material for studs, nuts, and washers is SA-540 Grade B24. The maximum reported ultimate tensile strength for the bolting material is less than the 170,000 psi limitation in Regulatory Guide 1.65. Also the Charpy impact test recommendations in Paragraph IV.A.4 of Appendix G to 10 CFR 50 were not specified in the vessel order since the order was placed prior to issuance of Appendix G to 10 CFR 50. However, impact data from the certified materials report shows that all bolting materials have met the Appendix G impact properties. The nondestructive examinations prescribed by the revision of the ASME B&PV Code in effect at the time the fasteners were ordered were conducted by the fabricator. All fasteners were found to be acceptable. A phosphate coating was applied to threaded areas of studs, nuts and bearing areas of nuts, and washers to act as a rust inhibitor and to assist in retaining lubricant on these surfaces.

5.3.1.8 SRP Rule Review

5.3.1.8.1 Acceptance Criterion II.2

SRP 5.3.1 acceptance criterion II.2 requires that the reactor vessel and its appurtenances be fabricated and installed in accordance with ASME B&PV Code, Section III, Paragraph NB-4100. The manufacturer or installer of such components is required to certify, by application of the appropriate Code symbol and completion of an appropriate data report in accordance with ASME B&PV Code, Section III, Paragraph NA-8000, that the materials used comply with the requirements of NB-2000, and that the fabrication or installation comply with the requirements of NB-4000.

The HCGS RPV and appurtenances were manufactured in accordance with the 1968 edition of the ASME B&PV Code, Section III, which does not have NB-designated subarticles. In light of HCGS's compliance with 1968 ASME B&PV Code, Section III, and information

SER ITEM NO. C-16

INSTRUMENTATION SETPOINT

The staff will confirm that the resolutions of the generic issues concerning the setpoint methodology are appropriate and successfully applied to the Hope Creek Technical Specifications.

RESPONSE

FSAR Question Response 421.18 has been revised to reflect the NRC acceptance of the setpoint methodology program. The HCGS Technical Specifications will be revised by 12/85 as required.

QUESTION 421.18 (SECTIONS 7.2 AND 7.3)

Provide a detailed discussion on the methodology used to establish the technical specification trip setpoints and allowable values for the Reactor Protection System (including Reactor Trip and Engineered Safety Feature channels) assumed to operate in the FSAR accident and and transient analyses. Include the following information:

1. The trip setpoint and allowable value for the technical specifications.

2. The safety limits necessary to protect the integrity of the physical barriers which guard against uncontrolled release of radioactivity. The safety limits should be the limits established for licensing purposes, for example the technical specification safety limits on minimum critical power ratio (1.06), and reactor coolant system pressure (1325 psig).

3. The values assigned to each component of the combined channel error allowance (e.g., modeling uncertainties, analytical uncertainties, transient overshoot, response time, trip unit setting accuracy, test equipment accuracy, primary element accuracy, sensor drift, nominal and harsh environmental allowances, trip unit drift), the basis for these values, and the method used to sum the individual errors. Where zero is assumed for an error a justification that the error is negligible should be provided.

4. The margin (i.e., the difference between the safety limit and the setpoint less the combined channel error allowance).

5. Identify any trip for which the setpoint and allowable value in the technical specifications will be assigned best estimate values and for which you do not have an analysis of errors and/or uncertainties to confirm that the trip function will occur before the actual value of the measured parameter exceeds that assumed in the plant safety analysis. Provide justification for this nonanalytical approach.

RESPONSE

Public Service Electric and Gas is currently participating with a number of other utilities and the General Electric Company in a review of ~~generic discussions with the NRC staff concerning~~ the methodology used to establish the technical specification trip setpoints and allowable values for the reactor protection system. All the issues raised by this question are being covered by ~~these generic discussions. After they are concluded, the resolutions of the~~ this program. Public Service Electric and Gas endorses the action

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~~generic issues will be applied to the HCGS, as appropriate, and HCGS-specific details the NRC may request will be supplied.~~

plan and schedule forwarded to the NRC Staff in Reference 1. NRC acceptance of this plan is provided in Reference 2. The action plan includes addressing all NRC open items, formally issuing the General Electric instrument setpoint methodology document, and documenting the plant-unique setpoint evaluations of each RPS and ESF trip function assumed to operate in the analyses described in Chapters 6 and 15. Modifications will be initiated for the technical specifications if necessary. The expected completion date is December, 1985.

Reference 1: John F. Carolan, Chairman - LRG, Instrument Setpoint Methodology Group, to T. M. Novak, Assistant Director for Licensing, NRC, "Action Plan to Answer the NRC Staff Concerns on Setpoint Methodology for GE Supplied Protection System Instrumentation," June 29, 1984.

Reference 2: B.J. Youngblood (NRC) to John F. Carolan, Chairman - LRG ISM Group, "Acceptance of Action Plan to Answer NRC Staff Concerns on Setpoint Methodology for GE Supplied Protection System Intrumentaion," July 23, 1984.

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SER ITEM NO. C-27

RELIEF FUNCTION OF SAFETY/RELIEF VALVES

However, in its review of FSAR Section 15, the staff assumed that the instrumentation and controls associated with the relief function of the SRVs were safety related. The applicant has indicated in response to this concern that the instrumentation and control equipment associated with the SRV function is nonsafety-related. This does not appear to be consistent with the design of previously reviewed BWRs. This would mean that the control circuits for the SRVs would contain both safety-related circuitry (relief function) without proper isolation.

The staff is currently reviewing the information provided regarding the relief function of the SRVs and will determine the adequacy of the design in a supplement to this report. This is a confirmatory item.

RESPONSE

The following information is provided to clarify the above confirmatory item.

The safety-related relief function of all 14 SRVs is provided by the entirely mechanical, self-actuating action inherent in each of the valves. Pressure relief can also be provided manually by the reactor operator via solenoids actuated by separate remote-manual switches. This manual relief function provides operational flexibility and is not considered a safety function. The Hope Creek design does, however, use safety-grade devices and Class 1E power to perform the manual relief function.

In addition to the relief function, 5 of the SRVs perform the safety-related automatic depressurization function utilizing solenoids "A" and "B", one for each of the logic trains that initiate the depressurization function (see FSAR Section 7.3.1). The 9 relief-function-only valves utilize only solenoid "A" for the manual relief function.

Table 7.1 of the SER was provided as part of the response to NRC Questions 440.33 and 421.54. This table lists nonsafety-grade systems and components assumed to contribute to mitigation in analyses described in Chapter 15 of the FSAR. However, the reference in the note to this table to ".. nonsafety-related instrumentation in the relief mode..." of the SRVs is a reference to the nonsafety function of the

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RESPONSE (CONT'D)

manual relief mode and not to the qualification of the electrical components. Since the components and power supplies of both the manual relief function and the automatic depressurization function are safety grade, no isolation problems exist between these functions.

FSAR Table 440.33-1 has been revised to clarify this item.

TABLE 440.33-1 (cont)

INFREQUENT EVENTS		
15.2.2	Load rejection without bypass	Relief valves
15.2.3	Turbine trip without bypass	Relief valves
LIMITING EVENTS		
15.3.3	Recirculation pump seizure	Level-8 turbine trip and feedwater pump trip, turbine bypass relief valves
15.3.4	Recirculation pump shaft break	Level-8 turbine trip and feedwater pump trip, turbine bypass, relief valves

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(1) "Relief ~~V~~alves" refers to the nonsafety-related manual ~~instrumentation & in the~~ relief mode of the SRVs. Although this mode does not serve a safety function, its electrical ~~and~~ components and power supply are safety grade.

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