PSEG Public Service Electric and Gas Company

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Robert L. Mittl General Manager
Nuclear Assurance and Regulation

June 29, 1984

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:

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

Attachment 1 is a current list which provides a status of the open items identified in Section 1.7 of the Draft 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.

Attachment 2 is a current list which identifies Draft SER Sections not yet provided.

In addition, enclosed for your review and approval (see Attachment 4) are the resolutions to those Draft SER open items listed in Attachment 3.

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

Very truly yours,

R L Mittl

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Attachments

C D. H. Wagner
USNRC Licensing Project Manager

W. H. Bateman
USNRC Senior Resident Inspector

FM05 1/2

ATTACHMENT 1

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL A. SCHWENCER LETTER DATED
5a&d	2.4.5	Wave impact and runup on service water intake structure	Complete	6/1/84
7b	2.4.11.2	Thermal aspects of ultimate heat sink	Complete	6/1/84
9	2.5.4	Soil damping values	Complete	6/1/84
10	2.5.4	Foundation level response spectra	Complete	6/1/84
11	2.5.4	Soil shear moduli variation	Complete	6/1/84
12	2.5.4	Combination of soil layer properties	Complete	6/1/84
13	2.5.4	Lab test shear moduli values	Complete	6/1/84
14	2.5.4	Liquefaction analysis of river bottom sands	Complete	6/1/84
15	2.5.4	Tabulations of shear moduli	Complete	6/1/84
16	2.5.4	Drying and wetting effect on Vincentown	Complete	6/1/84
17	2.5.4	Power block settlement monitoring	Complete	6/1/84
18	2.5.4	Maximum earth at rest pressure coefficient	Complete	6/1/84
19	2.5.4	Liquefaction analysis for service water piping	Complete	6/1/84
20	2.5.4	Explanation of observed power block settlement	Complete	6/1/84
21	2.5.4	Service water pipe settlement records	Complete	6/1/84
22	2.5.4	Cofferdam stability	Complete	6/1/84
23	2.5.4	Clarification of FSAR Tables 2.5.13 and 2.5.14	Complete	6/1/84
24	2.5.4	Soil depth models for intake structure	Complete	6/1/84
27	2.5.5	Slope stability	Complete	6/1/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
30	3.5.1.2	Internally generated missiles (inside containment)	Closed (5/30/84-Aux.Sys.Mtg.)	6/1/84
35	3.6.2	ISI program for pipe welds in break exclusion zone	Complete	6/29/84
36	3.6.2	Postulated pipe ruptures	Complete	6/29/84
41	3.8.2	Steel containment buckling analysis	Complete	6/1/84
42	3.8.2	Steel containment ultimate capacity analysis	Complete	6/1/84
43	3.8.2	SRV/LOCA pool dynamic loads	Complete	6/1/84
44	3.8.3	ACI 349 deviations for internal structures	Complete	6/1/84
45	3.8.4	ACI 349 deviations for Category I structures	Complete	6/1/84
46	3.8.5	ACI 349 deviations for foundations	Complete	6/1/84
47	3.8.6	Base mat response spectra	Complete	6/1/84
48	3.8.6	Rocking time histories	Complete	6/1/84
49	3.8.6	Gross concrete section	Complete	6/1/84
50	3.8.6	Vertical floor flexibility response spectra	Complete	6/1/84
53	3.8.6	Design of seismic Category I tanks	Complete	6/1/84
54	3.8.6	Combination of vertical responses	Complete	6/1/84
55	3.8.6	Torsional stiffness calculation	Complete	6/1/84
56	3.8.6	Drywell stick model development	Complete	6/1/84
57	3.8.6	Rotational time history inputs	Complete	6/1/84
58	3.8.6	"O" reference point for auxiliary building model	Complete	6/1/84
59	3.8.6	Overturning moment of reactor building foundation mat	Complete	6/1/84
60	3.8.6	BSAP element size limitations	Complete	6/1/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
61	3.8.6	Seismic modeling of drywell shield wall	Complete	6/1/84
62	3.8.6	Drywell shield wall boundary conditions	Complete	6/1/84
63	3.8.6	Reactor building dome boundary conditions	Complete	6/1/84
64	3.8.6	SSI analysis 12 Hz cutoff frequency	Complete	6/1/84
65	3.8.6	Intake structure crane heavy load drop	Complete	6/1/84
67	3.8.6	Critical loads calculation for reactor building dome	Complete	6/1/84
68	3.8.6	Reactor building foundation mat contact pressures	Complete	6/1/84
69	3.8.6	Factors of safety against sliding and overturning of drywell shield wall	Complete	6/1/84
70	3.8.6	Seismic shear force distribution in cylinder wall	Complete	6/1/84
71	3.8.6	Overturning of cylinder wall	Complete	6/1/84
72	3.8.6	Deep beam design of fuel pool walls	Complete	6/1/84
73	3.8.6	ASHSD dome model load inputs	Complete	6/1/84
74	3.8.6	Tornado depressurization	Complete	6/1/84
75	3.8.6	Auxiliary building abnormal pressure	Complete	6/1/84
76	3.8.6	Tangential shear stresses in drywell shield wall and the cylinder wall	Complete	6/1/84
77	3.8.6	Factor of safety against overturning of intake structure	Complete	6/1/84
78	3.8.6	Dead load calculations	Complete	6/1/84
79	3.8.6	Post-modification seismic loads for the torus	Complete	6/1/84
80	3.8.6	Torus fluid-structure interactions	Complete	6/1/84

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
81	3.8.6	Seismic displacement of torus	Complete	6/1/84
82	3.8.6	Review of seismic Category I tank design	Complete	6/1/84
83	3.8.6	Factors of safety for drywell buckling evaluation	Complete	6/1/84
84	3.8.6	Ultimate capacity of containment (materials)	Complete	6/1/84
85	3.8.6	Load combination consistency	Complete	6/1/84
88	3.9.1	Stress analysis and elastic-plastic analysis	Complete	6/29/84
89	3.9.2.1	Vibration levels for NSSS piping systems	Complete	6/29/84
91	3.9.2.2	Piping supports and anchors	Complete	6/29/84
92	3.9.2.2	Triple flued-head containment penetrations	Complete	6/15/84
93	3.9.3.1	Load combinations and allowable stress limits	Complete	6/29/84
94	3.9.3.2	Design of SRVs and SRV discharge piping	Complete	6/29/84
95	3.9.3.2	Fatigue evaluation on SRV piping and LOCA downcomers	Complete	6/15/84
96	3.9.3.3	IE Information Notice 83-80	Complete	6/15/84
97	3.9.3.3	Buckling criteria used for component supports	Complete	6/29/84
98	3.9.3.3	Design of bolts	Complete	6/15/84
99	3.9.5	Stress categories and limits for core support structures	Complete	6/15/84
100a	3.9.6	10CFR50.55a paragraph (g)	Complete	6/29/84
102	3.9.6	Leak testing of pressure isolation valves	Complete	6/29/84
107	4.2	Minimal post-irradiation fuel surveillance program	Complete	6/29/84

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
108	4.2	Gadolina thermal conductivity equation	Complete	6/29/84
110b	4.6	Functional design of reactivity control systems	Complete	6/1/84
111a	5.2.4.3	Preservice inspection program (components within reactor pressure boundary)	Complete	6/29/84
111b	5.2.4.3	Preservice inspection program (components within reactor pressure boundary)	Complete	6/29/84
111c	5.2.4.3	Preservice inspection program (components within reactor pressure boundary)	Complete	6/29/84
119	6.2	TMI item II.E.4.1	Complete	6/29/84
123	6.2.1.4	Butterfly valve operation (post accident)	Complete	6/29/84
124	6.2.1.5.1	RPV shield annulus analysis	Complete	6/1/84
125	6.2.1.5.2	Design drywell head differential pressure	Complete	6/15/84
129	6.2.2	Insulation ingestion	Complete	6/1/84
130	6.2.3	Potential bypass leakage paths	Complete	6/29/84
132	6.2.4	Containment isolation review	Complete	6/15/84
134	6.2.6	Containment leakage testing	Complete	6/15/84
138	6.6	Preservice inspection program for Class 2 and 3 components	Complete	6/29/84
139	6.7	MSIV leakage control system	Complete	6/29/84
141C	9.1.3	Spent fuel pool cooling and cleanup system	Complete	6/29/84
141g	9.1.3	Spent fuel pool cooling and cleanup system	Complete	6/15/84
142a	9.1.4	Light load handling system (related to refueling)	Closed (5/30/84-Aux.Sys.Mtg.)	6/29/84

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
142b	9.1.4	Light load handling system (related to refueling)	Closed (5/30/84-Aux.Sys.Mtg.)	6/29/84
145	9.2.2	ISI program and functional testing of safety and turbine auxiliaries coolng systems	Closed (5/30/84-Aux.Sys.Mtg.)	6/15/84
146	9.2.6	Switches and wiring associated with HPCI/RCIC torus suction	Closed (5/30/84-Aux.Sys.Mtg.)	6/15/84
152	9.4.4	Radioactivity monitoring elements	Closed (5/30/84-Aux.Sys.Mtg.)	6/1/84
154	9.5.1.4.a	Metal roof deck construction classificiation	Complete	6/1/84
158	9.5.1.5.a	Class B fire detection system	Complete	6/15/84
159	9.5.1.5.a	Primary and secondary power supplies for fire detection system	Complete	6/1/84
161	9.5.1.5.b	Fire water valve supervision	Complete	6/1/84
162	9.5.1.5.c	Deluge valves	Complete	6/1/84
163	9.5.1.5.c	Manual hose station pipe sizing	Complete	6/1/84
164	9.5.1.6.e	Remote shutdown panel ventilation	Complete	6/1/84
165	9.5.1.6.g	Emergency diesel generator day tank protecton	Complete	6/1/84
168	12.5.2	Equipment, training, and procedures for inplant iodine instrumentation	Complete	6/29/84
170	13.5.2	Procedures generation package submittal	Complete	6/29/84
171	13.5.2	TMI Item I.C.1	Complete	6/29/84
172	13.5.2	PGP Commitment	Complete	6/29/84
173	13.5.2	Procedures covering abnormal releases of radioactivity	Complete	6/29/84

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
174	13.5.2	Resolution explanation in FSAR of TMI Items I.C.7 and I.C.8	Complete	6/15/84
181	15.9.5	TMI-2 Item II.K.3.3	Complete	6/29/84
182	15.9.10	TMI-2 Item II.K.3.18	Complete	6/1/84
185	7.2.2.2	Trip system sensors and cabling in turbine building	Complete	6/1/84
190	7.2.2.7	Regulatory Guide 1.75	Complete	6/1/84
191	7.2.2.8	Scram discharge volume	Complete	6/29/84
193	7.2.2.9	Reactor mode switch	Complete	6/1/84
194	7.3.2.2	Standard review plan deviations	Complete	6/1/84
197	7.3.2.5	Microprocessor, multiplexer and computer systems	Complete	6/1/84
200	7.4.2.2	Remote shutdown system	Complete	6/1/84
205	7.5.2.4	Plant process computer system	Complete	6/1/84
209	7.7.2.3	Credit for non-safety related systems in Chapter 15 of the FSAR	Complete	6/1/84
210	7.7.2.4	Transient analysis recording system	Complete	6/1/84
218	9.5.1.1	Fire hazards analysis	Complete	6/1/84
TS-3	4.4.5	Core flow monitoring for crud effects	Complete	6/1/84
LC-1	4.2	Fuel rod internal pressure criteria	Complete	6/1/84

ATTACHMENT 2 DATE: 6/29/84

DRAFT SER SECTIONS AND DATES PROVIDED

SECTION	DATE	SECTION	DATE
3.1			
3.2.1		11.4.1	
3.2.2		11.4.2	
5.1		11.5.1	
5.2.1		11.5.2	
6.5.1		13.1.1	
8.1		13.1.2	
8.2.1		13.2.1	
8.2.2		13.2.2	
8.2.3		13.3.1	
8.2.4		13.3.2	
8.3.1		13.3.3	
8.3.2		13.3.4	
8.4.1		13.4	
8.4.2		13.5.1	
8.4.3		15.2.3	
8.4.5		15.2.4	
8.4.6		15.2.5	
8.4.7		15.2.6	
8.4.8		15.2.7	
9.5.2		15.2.8	
9.5.3		15.7.3	
9.5.7		17.1	
9.5.8		17.2	
10.1		17.3	
10.2		17.4	
10.2.3			
10.3.2			
10.4.1			
10.4.2			
10.4.3			
10.4.4			
11.1.1			
11.1.2			
11.2.1			
11.2.2			
11.3.1			
11.3.2			

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ATTACHMENT 3

OPEN ITEM	DSER SECTION NUMBER	SUBJECT
35	3.6.2	ISI program for pipe welds in break exclusion zone
36	3.6.2	Postulated pipe ruptures
88	3.9.1	Stress analysis and elastic-plastic analysis
89	3.9.2.1	Vibration levels for NSSS piping systems
91	3.9.2.2	Piping supports and anchors
93	3.9.3.1	Load combinations and allowable stress limits
94	3.9.3.2	Design of SRV's and SRV discharge piping
97	3.9.3.3	Buckling criteria used for component supports
100a	3.9.6	10CFR50.55a paragraph (g)
102	3.9.6	Leak testing of pressure isolation valves
107	4.2	Minimal post-irradiation fuel surveillance program
108	4.2	Gadolina thermal conductivity equation
111A	5.2.4.3	Preservice inspection program (components within reactor pressure boundary)
111B	5.2.4.3	Preservice inspection program (components within reactor pressure boundary)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT
111C	5.2.4.3	Preservice inspection program (components within reactor pressure boundary)
119	6.2	TMI item II.E.4.1
123	6.2.1.4	Butterfly valve operation (post accident)
130	6.2.3	Potential bypass leakage paths
138	6.6	Preservice inspection program for class 2 and 3 components
139	6.7	MSIV leakage control system
141C	9.1.3	Spent fuel pool cooling and cleanup system
142a	9.1.4	Light load handling system (related to refueling)
142b	9.1.4	Light load handling system (related to refueling)
168	12.5.2	Equipment, training, and procedures for inplant iodine instrumentation
170	13.5.2	Procedures generation package submittal
171	13.5.2	TMI item I.C.1
172	13.5.2	PGP Commitment
173	13.5.2	Procedures covering abnormal releases of radioactivity
181	15.9.5	TMI-2 item II.K.3.3
191	7.2.2.8	Scram discharge volume

ATTACHMENT 4

DSER Open Item No. 35 (Section 3.6.2)

ISI PROGRAM FOR PIPE WELDS IN BREAK EXCLUSION ZONE

Assurance is needed regarding the augmented inservice inspection program for pipe welds in the break exclusion zone.

RESPONSE

For the information requested above, see the response to Question 210.14.

DSER Open Item No. 36 (Section 3.6.2)

POSTULATED PIPE RUPTURES

Additional information is required in several areas dealing with postulated pipe ruptures. Several tables and figures dealing with postulated rupture locations and their associated effects are incomplete. More information on the details of jet impingement and pipe whip analyses is required.

RESPONSE

For the information requested above, see the response to Question 210.21.

DSER Open Item No. 88 (Section 3.9.1)

STRESS ANALYSIS AND ELASTIC-PLASTIC ANALYSIS

Clarification is needed on experimental stress analysis and elastic-plastic analyses.

RESPONSE

For the information above, see the response to Questions 210.26 and 210.27.

DSER Open Item No. 89 (Section 3.9.2.1)

VIBRATION LEVELS FOR NSSS PIPING SYSTEMS

Additional information of the criteria to be used for determining acceptability of observed or measured vibration levels for NSSS piping systems needs to be included in the FSAR.

RESPONSE

For the information above, see the responses to Questions 210.29 and 210.30.

DSER Open Item No. 91 (Section 3.9.2.2)

PIPING SUPPORTS AND ANCHORS

Additional information is required on the design of piping supports and anchors which separate seismically designed piping and nonseismic Category I piping.

RESPONSE

For the information requested above, see the response to Question 210.34.

DSER Open Item No. 93 (Section 3.9.3.1)

LOAD COMBINATIONS AND ALLOWABLE STRESS LIMITS

More information is required on loading combinations, system operating transients and stress limits for each of the following for all classes of construction: vessels, pumps, valves, piping, and supports. Assurance must be provided for the functional capability of ASME Class 1, 2, and 3 piping systems. This is an open item.

RESPONSE

Non-NSSS:

The loading combinations for non-NSSS piping and supports are given in Tables 3.9-8 and 3.9-21. Table 3.9-21 has been revised to include allowable stresses under various plant conditions. In addition, the primary stress limits have been added to Tables 3.9-9 and 3.9-13.

Tables 3.9-10 and 3.9-15 have been revised to include the loading combinations and allowable stress limits for non-NSSS Class 1 valves and Class 2 and 3 valves respectively. Information on loading combinations, system operating transients, and stress limits for pumps were provided in response to Question 210.52. Information on safety-related vessels designed to the ASME Code are attached in Tables 93-1 and 93-2.

Functional capability of ASME Class 1, 2, and 3 piping system has been addressed in Question 210.39.

NSSS:

Information on loading combinations, system operating transients, and stress limits for pumps and vessels were provided in response to Question 210.52.

TABLE 93-1

SAFETY-RELATED VESSELS

The safety-related hydropneumatic accumulators (STACS) are demonstrated capable of withstanding the following loading conditions and associated loading combinations while stresses remain below the allowable stresses. The design, manufacturer, examination, testing, and inspection of the accumulators is in accordance with the ASME Boiler and Pressure Vessel Code, Section III Nuclear Power Plant Components, Division I for Class 3 components.

PLANT/SYSTEM OPERATING CONDITION	DESIGN AND SERVICE LIMITS	LOADING COMBINATION	ALLOWABLE SERVICE STRESS LIMIT
Design	-	PD+OBE	ND-3300
Normal	A	PO+DW+EL	ND-3300
Upset	B	PO+DW+EL+OBE	ND-3300
Faulted	B	PO+DW+EL+SSE	ND-3300

Where:

PD = Design Pressure

PO = Operating Pressure

DW = Dead Weight of vessel and contents

EL = External Loads due to connected piping

OBE = Operating Basis Earthquake

SSE = Safe Shutdown Earthquake

DSER OPEN ITEM 93

TABLE 93-2

SAFETY RELATED VESSELS

The Safety Related Expansion Tanks and Air Accumulators are demonstrated capable of withstanding the following loading conditions and associated loading combinations while stresses remain below the allowable stresses. These vessels include the SACS expansion tanks, control area chilled water system head tanks.

LOADING CONDITION	LOADING COMBINATION	ALLOWABLE STRESSES
Design	PD	ND-3300 for > 15 psig ND-3800 for ATM ND-3900 for 0-15 psig
Normal	PO + DW + EL	ND-3300 for > 15 psig ND-3800 for ATM ND-3900 for 0-15 psig
Upset	PO + DW + EL + OBE	Code Case 1607-1 for > 15 psig Code case 1657 for ATM, 0-15 psig
Faulted	PO + DW + EL + SSE	Code Case 1607-1 for > 15 psig Code Case 1657 for ATM, 0-15 psig

where:

PD = Design Pressure

PO = Operating Pressure

DW = Dead Weight of vessel and contents

EL = External Loads due to connected piping

OBE = Operating Basis Earthquake

SSE = Safe Shutdown Earthquake

TABLE 3.9-21

DESIGN LOADING COMBINATIONS FOR SUPPORTS
FOR ASME B&PV CODE CLASS 1, 2 AND 3 NON-NSSS COMPONENTS

Condition		Design Loading Combinations[1][2][3]	Allowable stress
Hydrostatic test	(a)	HTDW	0.8 Sy
Normal and upset	(a)	$DW + TH + (OBE^2 + RVC^2)^{1/2}$	ASME Section III Appendix XVII
	(b)	DW + TH + OBE + RVO	
	(c)	DW + TH + FV	
Emergency	(a)	$DW + TH + (OBE^2 + FV^2)^{1/2}$	ASME SECTION III, Appendix XVII
Faulted	(a)	DW + TH + SSE + RVO	ASME Section III, Appendix F[4]
	(b)	$DW + TH + (SSE^2 + RVC^2)^{1/2}$	
	(c)	$DW + TH + (SSE^2 + DBA^2)^{1/2}$	

(1) Loads due to OBE, SSE, and DBA include both inertia portion and anchor movement portion when spectra method is used. The loads from the inertia portion and anchor movement portion are combined by the SRSS method.

(2) For torus-attached piping, the loading combinations used in evaluating the pipe support loads are those given in the Plant Unique Analysis Application Guide (PUAAG) (NEDO-24583-1, October 1979 (Table 5-2)).

(3) Definition of symbols used:

HTDW - piping dead weight due to hydrostatic test

TH - reaction at the support due to thermal expansion of the pipe

DW - dead weight

OBE - operating basis earthquake[1]

RVC - transient response of the piping system associated with relief valve opening in a closed system

RVO - transient response of the piping system associated with relief valve opening in an open system

DSER OPEN ITEM 93

TABLE 3.9-21 (cont)

FV - transient response of the piping system associated with fast valve closure time less than 5 seconds

SSE - safe shutdown earthquake(1)

DBA - design basis accident(1)

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(4) For essential safety-related (ESR) systems allowable stress not to exceed S_y.

TABLE 3.9-9

DESIGN CRITERIA FOR ASME B&PV CODE CLASS 1 NON-NSSS PIPING

Condition	Applicable Code Paragraph ~~Stress Limits~~ (1)(2)	Primary stress limits
Design	NB-3221 and NB-3652	1.5 Sm
Normal	NB-3222 and NB-3653	1.5 Sm
Upset	NB-3223 and NB-3654	1.8Sm but not greate than 1.5 Sy
Emergency	NB-3224 and NB-3655	2.25Sm but not greate than 1.8 Sy
Faulted	NB-3225 and NB-3656	3.0 Sm

(1) As specified by the ASME B&PV Code, Section III, 1974 through Winter 1974 Addenda, except for the following:

a. Class 1, 1-inch and smaller piping are designed to ASME Section III, 1975 Summer Addenda, Paragraph NB-3630(d)(1).

b. Class 1 flanges are designed to ASME Section III, 1979 Summer Addenda, Paragraph NB-3658.

c. Class 1 branch connections are designed to ASME Section III, 1979 Summer Addenda, Paragraph NB-3653.1.

(2) Functional capability of essential piping is ensured per NEDO-21985, September 1978.

TABLE 3.9-13

DESIGN CRITERIA FOR ASME B&PV CODE CLASS 2 AND 3 NON-NSSS PIPING

Condition	Stress Limits[1][2]
Design, normal, upset and emergency	The piping conforms to the requirements of Section III, paragraphs NC-3600
Faulted[2]	The piping conforms to the requirements of ASME Code Case 1606

INSERT

delete

(1) As specified by the ASME B&PV Code, Section III, 1974 through Winter 1974 Addenda, except for Class 2 and 3 flanges, which are designed to 1979 Winter Addenda, Paragraph NC and ND-3652.

(2) Functional capability of essential piping is ensured per NEDO-21985, September 1978.

Insert

TABLE 3.9-13

CONDITION	APPLICABLE CODE PARAGRAPH (1) (2)	PRIMARY STRESS LIMITS
Design:		
Sustained Loads	NC, ND-3652.1	1.0Sh
Occasional Loads	NC, ND-3652.2	1.2Sh
Normal and Upset	NC, ND-3652.2 & 3611	1.2Sh
Emergency	NC, ND-3611	1.8Sh
Faulted	Code Case 1606	2.4Sh

DSER OPEN ITEM 93

TABLE 3.9-10

DESIGN CRITERIA FOR ASME B&PV CODE CLASS 1 NON-NSSS VALVES

INSERT A

Condition	Stress Limits
Design	NB-3521[1]
Normal and upset	NB-3200 or NB-3500[1] (Standard Design Rules)
Emergency[2]	NB-3526
Faulted[2]	NB-3527

delete

[1] As specified by the ASME B&PV Code, Section III, 1974 through Winter 1974 Addenda.

[2] Where valve function must be ensured (active valves) during emergency or faulted conditions, the specified emergency or faulted conditions for the plant ~~is~~ are considered the normal condition for the valve.

3) As required by subsection NB of ASME Section III other loads such as thermal transient and thermal gradients may require additional consideration in addition to those primary stress-producing loads listed.

4) Definition of symbols used:

PD - Design pressure

PO - Operating pressure at noted plant condition

OBE - Operating Basis Earthquake loads (inertia portion) excluding loads from attached piping

SSE - Safe Shutdown Earthquake loads (inertia portion) excluding loads from attached piping

B - Piping end loads at noted plant condition

DSER OPEN ITEM 93

INSERT A T3.9-10

<table>
<tr><th>Plant Condition</th><th>Design Loading Combinations (4)</th><th>Stress Limits (1)</th></tr>
<tr><td>Design</td><td>PD</td><td rowspan="2">The valve shall conform to the requirement of Paragraph NB-3500 (Standard Design Rules)</td></tr>
<tr><td>Normal</td><td>P0n + Bn</td></tr>
<tr><td>Upset (3)</td><td>P0u + OBE + Bu</td><td>NB 3525</td></tr>
<tr><td>Emergency (2)</td><td>P0e + Be</td><td>NB 3526</td></tr>
<tr><td>Faulted (2)</td><td>P0f + SSE + Bf</td><td>NB 3527</td></tr>
</table>

MP 84 112/05 1-mr

DSER OPEN ITEM 93

TABLE 3.9-15

DESIGN CRITERIA FOR ASME B&PV CODE CLASS 2 AND 3 NON-NSSS VALVES

Condition	Stress Limits[1]
Design and normal	The valve conforms to the requirements of Section III, Paragraphs NC-3500 and ND-3500
Upset, emergency, and faulted[2]	The valve conforms to the requirements of ASME Code Case 1635-1

(1) As specified by the ASME B&PV Code, Section III, 1974 through Winter 1974 Addenda.

(2) Where valve function must be ensured (active valves) during emergency or faulted conditions, the specified emergency or faulted plant conditions are considered as the normal condition for the valve.

delete

See attached new Table 3.9-15 pages 1 and 2

DSER OPEN ITEM 93

TABLE 3.9-15

DESIGN CRITERIA FOR ASME B&PV CODE CLASS 2&3 NON-NSSS VALVES

Plant Condition	Design Loading Combination (1) (4)	Stress Limits (1) (2) (3)
Design	PD	The valve shall conform to the requirements of Section III, 1974 Paragraphs NC-3500 or ND-3500,as applicable
Normal	POn + Bn	$S_m \leq 1.0S$ $(S_m \text{ or } S_L) + S_b \leq 1.50S$
Upset	POu + OBE + Bu	$S_m \leq 1.1S$ $(S_m \text{ or } S_L) + S_b \leq 1.65S$
Emergency	POe + Be	$S_m < 1.5S$ $(S_m \text{ or } S_L) + S_b \leq 1.8S$
Faulted	POf + SSE + Bf	$S_m \leq 2.0S$ $(S_m \text{ or } S_L) + S_b \leq 2.4S$

(1) Definition of symbols:

Sm = General membrane stress

SL = Local membrane stress

Sb = Bending stress

S = Allowable stress

DSER OPEN ITEM 93

TABLE 3.9-15

(1) Definition of symbols (cont'd):

PD = Design pressure

PO = Operating pressure at noted plant condition

OBE = Operating basis earthquake loads (inertia portion) excluding loads from attached piping

SSE = Safe shutdown earthquake loads (inertia portion) excluding loads from attached piping

B = Piping end loads at noted plant condition

(2) As specified by the ASME B&PV Code, Section III, 1974, through Winter 1974 Addenda.

(3) Where valve function must be ensured (active valves) during emergency or faulted conditions, the specified emergency or faulted plant conditions are considered as the normal condition for the valve.

(4) As required by subsection NC, ND of ASME Section III, other loads such as thermal transient and thermal gradients may require additional consideration in addition to those primary stress producing loads listed.

DSER OPEN ITEM 93

DSER Open Item No. 94 (Section 3.9.3.2)

DESIGN OF SRVs AND SRV DISCHARGE PIPING

The staff has reviewed Section 3.9.3.3 of the applicant's FSAR with respect to the design and installation, and testing criteria applicable to the mounting of pressure relief devices used for the overpressure protection of ASME Class 1, 2, and 3 components. This review, conducted in accordance with SRP Section 3.9.3 (NUREG-0800), includes evaluation of the applicable loading combinations and stress criteria. The design review extends to consideration of the means provided to accommodate the rapidly applied reaction force when a safety valve or relief valve opens, and the transient fluid-induced loads applied to the piping downstream of a safety or relief valve in a closed discharge piping system.

The staff requires additional information on the design of safety and relief valves (SRVs) and the main steam SRV discharge piping.

RESPONSE

Information on the main steam SRV discharge piping is provided in response to Question 210.45.

DSER Open Item No. 97 (Section 3.9.3.3)

BUCKLING CRITERIA USED FOR COMPONENT SUPPORTS

The staff's review of Section 3.9.3.4 of the applicant's FSAR relates to the methodology used by the applicant in the design of ASME Class 1, 2, and 3 component supports. The review includes assessment of design and structural integrity of the supports. The review addresses three types of supports: plate and shell, linear, and component standard types. More information regarding the design and construction of ASME Class 1, 2, and 3 component supports is required.

The applicant should provide the buckling criteria used for component supports.

RESPONSE

For the information requested above see response to Question 210.49.

DSER Open Item No. 100a (Section 3.9.6)

10CFR50.55a PARAGRAPH (g)

The applicant must provide a commitment that the inservice testing of ASME Class 1, 2, and 3 components will be in accordance with the rules of 10CFR50.55a, Paragraph (g).

RESPONSE

For the information requested above see FSAR Sections 5.2.4 and 6.6.

DSER Open Item No. 102 (Section 3.9.6)

LEAK TESTING OF PRESSURE ISOLATION VALVES

The applicant has not yet responded to the staff's concern regarding the leak testing of pressure isolation valves.

RESPONSE

For the information requested above, see the response to Question 210.56.

DSER OPEN ITEM 107 (Section 4.2)

MINIMAL POST-IRRADIATION FUEL SURVEILLANCE PROGRAM

The applicant must provide a minimal post-irradiation fuel surveillance program consistent with SRP Section 4.2.II.D.3.

RESPONSE

General Electric and the NRC have negotiated a post-irradiation fuel surveillance program that meets the requirements of SRP 4.2. The NRC has been requiring that certain applicants (Perry, Hanford and Limerick) commit to perform visual inspections of a prescribed percentage of discharged bundles each cycle. In a letter dated November 23, 1983, General Electric proposed an alternative program that would transfer much of the burden for these inspections from utilities to General Electric. In a letter dated January 18, 1984, the NRC staff described what would be an acceptable program and requested additional detail from General Electric. In letters dated January 27, 1984, and February 29, 1984, General Electric addressed the NRC questions, and the NRC has verbally agreed to the General Electric program. Public Service Electric and Gas endorses this program for the Hope Creek plant.

MFN-218-83
JSC-072-83

GENERAL ELECTRIC

NUCLEAR POWER SYSTEMS DIVISION
GENERAL ELECTRIC COMPANY • 175 CURTNER AVENUE • SAN JOSE, CALIFORNIA 95125

MC 682, (408) 925-3697

November 23, 1983

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Attention: C. H. Berlinger, Chief
Core Performance Branch

Gentlemen:

SUBJECT: POST-IRRADIATION FUEL SURVEILLANCE PROGRAM

Reference: Letter, J. S. Charnley (GE) to F. J. Miraglia (NRC), "Proposed Revision to GE Licensing Topical Report NEDE-24011-P-A", February 25, 1983

The NRC has recently required that newly licensed plants adopt a post-irradiation fuel surveillance program that consists essentially of routine visual inspection of discharged fuel at each refueling outage. The purpose of this letter is to propose the use of the fuel surveillance program described in the attachment, in place of the program required by the NRC at newly licensed plants. General Electric believes that its program meets the intent of Section II, Part D, of Standard Review Plan (SRP) 4.2 (NUREG-0800), regarding fuel surveillance. Because of the number of plants coming on-line in the near future that will be affected by this issue, GE requests that the NRC expedite consideration of this matter.

General Electric Fuel Performance Verification Program

The General Electric fuel performance verification program is described in the proprietary attachment to this letter. The attachment is considered proprietary because it contains information which GE customarily maintains in confidence and withholds from public disclosure. This information has been handled and classified as proprietary as indicated in the affidavit provided in the reference letter. We hereby request that this information be withheld from public disclosure in accordance with the provisions of 10CFR2.790.

DSER OPEN ITEM 107

GE Program and SRP 4.2

Regarding post-irradiation fuel surveillance, SRP 4.2 states that a program "should be described for each plant to detect anomalies or confirm expected fuel performance...For a fuel design like that in other operating plants, a minimum acceptable program should include a qualitative visual examination of some discharged fuel assemblies from each refueling."

GE defines expected fuel performance as "the fuel will not fail". Failure criteria used in the design process contain conservatisms that adequately bound conditions that may exist at any plant, and provide margin to actual fuel failure limits. Additionally, operating limits are established such that sufficient margins are maintained to the design limits during normal operation and transients (in accident analyses, all fuel is conservatively assumed to fail).

Expected fuel performance as defined above is confirmed on a generic basis for a fuel design through the inspection of LTA's, and on a plant-specific basis through offgas surveillance. The LTA program detects anomalies that may arise, with the added advantage of accomplishing this prior to the time that the anomaly might appear in production fuel. As discussed earlier, a visual examination of some of the discharged fuel from two early applications of a new fuel design will also be performed, in order to confirm the expected performance of that fuel design.

Discussion of GE Program

GE believes that the program it proposes meets or exceeds the intent of SRP 4.2 and is also more cost effective. The numerous benefits of the GE program are presented below.

Inspection of LTA's of new designs provides timely, detailed, and useful information that can be fed back into fuel design, analysis, and manufacture. LTA's of new designs are usually placed in operation at least a year before in-reactor introduction of production fuel. Prior to irradiation, these LTA's may undergo detailed visual, nondestructive, and dimensional characterization. Key measurements may be taken of specific bundle features and additional detailed examinations may be performed on specific fuel rods. Interim examinations may be performed at the end of each operating cycle. Upon discharge, a final inspection is performed on the previously characterized fuel rods and final measurements may be taken of key bundle features. As required, more extensive evaluations may be performed, including destructive testing. This detailed surveillance of LTA's for new designs provides: (1) early identification of potential fuel performance concerns; (2) continuous knowledge of overall fuel performance; and (3) systematic acquisition of detailed behavioral data allowing a comparison of predicted versus observed performance characteristics, thus providing feedback into the design process from fuel operated in a commercial reactor.

As discussed previously, the detection of fuel failures results in an investigation into the cause, and corrective actions where appropriate.

A general visual inspection of the exterior surfaces of a statistically significant number of fuel bundles (24 total - twelve at each of two plants) to confirm the absence of any anomalous behavior at end-of-life discharge for a new fuel design represents ample additional confirmation of the design.

Because fewer bundles are examined in greater depth (LTAs) than in the program required by the NRC of newly licensed plants, and because the visual inspections are limited to 24 bundles at end-of-life for a new design rather than at the end of every cycle in perpetuity, the GE program leads to a significant reduction in the total costs to utilities, while simultaneously providing more valuable data. If a utility were to contract for the type of visual examination the NRC is proposing the cost to the utility would be on the order of $60,000 per reload (assuming 12 bundles are inspected at each outage), in addition to personnel and dechanneling costs. If the utility were to perform the visual inspection itself, the cost in terms of training personnel, procuring proper equipment, performing the inspection, and exposing workers to radiation, would also be substantial.

The proposed GE program will allow the NRC to maximize the utilization of its resources by eliminating routine, repetitive review. Legitimate concerns will be easily recognized under the program proposed by GE.

<u>Summary</u>

GE proposes a fuel performance verification program consisting of inspection of LTA's, offgas surveillance and visual examination of a limited but statistically significant number of fuel bundles of two early commercial applications of new fuel designs. GE believes that this program meets or exceeds the intent of SRP 4.2 regarding fuel surveillance, and in addition is cost-effective for GE and the utilities as well as the NRC, while providing timely, detailed, and useful information that will be of benefit in enhancing fuel performance.

Please contact W. A. Zarbis (408-925-5070) or myself if you have any questions.

Very truly yours,

J. S. Charnley

J. S. Charnley
Fuel Licensing Manager
Nuclear Safety and Licensing Operation

JSC:csc/I09091*

cc: L. S. Gifford (GE-Beth)
L. S. Rubenstein (NRC)
G. G. Sherwood (GE)

ATTACHMENT

General Electric Fuel Performance Verification Program

The General Electric fuel performance verification program consists of inspection of lead test assemblies (LTA's) for new designs, and offgas monitoring of all designs throughout their lifetime.

For new fuel designs, GE will, in addition, agree to conduct a general visual examination of the exterior surfaces of a statistically significant number of fuel bundles (12 bundles) upon discharge from each of two early commercial applications of the new product. The visual examination will be made using binoculars, borescope, periscope, or TV and will be sufficient to meet the objectives presented in SRP 4.2 for visual inspections. The schedule and scope of LTA inspections is contingent on both the availability of the fuel as influenced by plant operation and the expected value of the information to be obtained.

General Electric's LTA's are selectively inspected using one or more of the following techniques:

1) Leak detection tests, such as sipping.
2) Visual inspection with various aids such as binoculars, borescope, or periscope, with a photographic record of observations as appropriate.
3) Nondestructive testing of selected fuel rods by ultrasonic and eddy current test techniques.
4) Dimensional measurements of selected fuel rods.

Unexpected conditions or abnormalities which may arise are analyzed, and examination of selected fuel rods in hot cell facilities may be undertaken when the expected value of the information to be obtained warrants this type of examination. Results of this surveillance program will be updated annually by a GE proprietary letter report.

The use of LTA's provides early verification of performance targets as well as early indication of potential performance anomalies.

Specific plant fuel failures are accurately detected by offgas surveillance. Offgas surveillance is performed for all operating plants, and leak detection tests such as sipping are performed by the utilities at the end of each cycle, if warranted (based on analysis of the offgas surveillance results). Offgas surveillance is a very sensitive measure of fuel performance, and General Electric fuel failure statistics include fuel failures estimated as a result of offgas measurements. These fuel failure statistics will be updated in the annual letter report.

DSER OPEN ITEM 107

If many fuel failures are detected, an analysis or investigation is initiated to determine the cause of the failures. In addition to review of operational parameters such as power history and water chemistry and of GE's current overall fuel experience base, the investigation may include site examinations, and when appropriate, searches of manufacturing records, tests of manufactured spare rods if available, and hot cell examination of selected irradiated fuel rods.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

RECEIVED
JAN 3 0 1984
R. L. GRIDLEY

JAN 1 8 1984

Mr. R. L. Gridley, Fuel and Services
Licensing Manager
Nuclear Safety & Licensing Operation
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Mr. Gridley:

Subject: Post-Irradiation Fuel Surveillance

Reference: Letter from J. S. Charnley (GE) to C. H. Berlinger (NRC), "Post-Irradiation Fuel Surveillance Program," November 23, 1983.

Your letter of November 23, 1983 proposes that generic vendor surveillance on lead test assemblies (LTAs) be substituted for routine licensee surveillance to satisfy Section II, Part D of Standard Review Plan 4.2. Section II, Part D contains two subparts that are relevant to your request. Subpart 2 describes on-line monitoring and Subpart 3 describes post-irradiation surveillance.

In our view, the licensee offgas surveillance that is mentioned in your letter clearly satisfies Subpart 2 mentioned above. This offgas surveillance program has been proposed by all recent BWR operating license applicants and in all cases we have approved it.

Your letter of November 23, 1983 also proposes that generic vendor surveillance on LTAs supported by a visual examination of some discharged fuel from two early applications of the new fuel design be substituted for routine licensee surveillance to satisfy Section II, Part D of Standard Review Plan Section 4.2.

While we agree with the goal of your letter, that is, to provide a better balance between the reduction of the regulatory burden on individual licensees and the NRC's interests in maintaining the present high level of fuel performance and in identifying potential new anomalies at an early stage, we find the General Electric Company proposal as described in your letter to be inadequate for several reasons.

We believe that past experience has shown that a surveillance program which looks only at LTAs and the first two core loadings is not sufficient. Fuel problems have occurred with standard fuel designs that have been in service for many years. Some of these problems were due to specific one-of-a-kind problems but other problems have been more generic in nature. We consider a small amount of visual surveillance to be important because we are concerned with all types of fuel damage

DSER OPEN ITEM 107

including that which could affect control rod insertability and accident doses, as well as those mechanisms that would lead to detectable (i.e., by offgas) cladding leaks during normal operation. Many examples of excessive wear, tearing of metal parts, fuel rod deformation and excessive crud buildup have been observed visually in fuel that showed no evidence of cladding failure under normal operating conditions.

We consider these differences in fuel surveillance programs to be reconcilable. General Electric topical report NEDE 24343-P "Experience With Fuel Through January 1981" describes the GE fuel surveillance program. One aspect of this program is stated to be an overall post-irradiation visual examination of selected fuel bundles. We would consider this present GE program to be equivalent to that described in the Standard Review Plan if General Electric would: (1) verify that this program includes post-irradiation visual inspection of standard design fuel bundles which have not been identified as leakers by sipping or other methods and (2) that the current GE fuel surveillance program for standard fuel designs will continue at its present level of effort.

In addition, we have some specific questions on details of the surveillance program proposed in your November 23, 1983 letter. The program which is described makes almost all the commitments to the type of surveillance conditional. It is not clear in the letter what these conditions are. For example, the letter states that "prior to irradiation, these LTAs <u>may</u> undergo detailed visual, nondestructive and dimensional characterization."

We believe it is important to discuss and clarify under what circumstances the conditions would be met so that these inspections would be done. Also, it is not clear in your letter what threshold of offgas activity would result in a non-routine inspection of the standard fuel designs. This should be discussed and clarified.

It continues to be our position that operating reactor licensees have the final responsibility for the performance of the fuel in their reactors. Although we agree in principle with the GE proposal to lessen the burden on these licensees, if a problem is discovered it is still the responsibility of the licensees to assure that adequate steps are taken to assure safe operation of the fuel at their facilities. We will also attempt to assure that, should a licensee who is presently a GE customer choose not to continue that relationship, the licensee will subsequently adopt an acceptable fuel surveillance program.

The results of these fuel inspections performed at a licensee's facility or performed on fuel irradiated at a licensee's facility will be covered by the reporting requirements of Paragraph 50.73(a)(2)(ii) concerning the degradation of "principal safety barriers," such as the fuel.

DSER OPEN ITEM 107

While [illegible] discussion is necessary to determine if your proposal constitutes an acc[illegible]able alternative to the SRP, we believe your proposal is a positi[illegible] approach.

Sincerely,

L S Rubenstein

L. S. Rubenstein, Assistant Director
for Core and Plant Systems
Division of Systems Integration, NRR

cc: J [illegible]harnley

DSER OPEN ITEM 107

GENERAL ELECTRIC

NUCLEAR POWER SYSTEMS DIVISION
GENERAL ELECTRIC COMPANY • 175 CURTNER AVENUE • SAN JOSE, CALIFORNIA 95125
MC 682, (408) 925-3697

February 29, 1984

MFN-024-84
JSC-10-84

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Attention: L. S. Rubenstein, Assistant Director
Core and Plant Systems

Gentlemen:

SUBJECT: FUEL SURVEILLANCE PROGRAM

References:
1) L. S. Rubenstein (NPC) to R. L. Gridley (GE), "Post-Irradiation Fuel Surveillance," January 18, 1984
2) NEDE-24343-P, "Experience with BWR Fuel Through January 1981," May 1981
3) J. S. Charnley (GE) to C. H. Berlinger (NRC), "Post-Irradiation Fuel Surveillance Program," November 23, 1984

This letter provides additional details requested by the NRC on GE's fuel surveillance program, and replaces our letter of January 27 on this subject.

The fuel surveillance program presented in your letter of January 18 (Reference 1) assures adequate verification of safe fuel performance while still maintaining efficient use of industry resources, and is acceptable to General Electric. We would like to take this opportunity to provide additional information in order to address the points raised in your letter.

Reference 1 states that the fuel surveillance program described in NEDE-24343 (Reference 2) could be considered equivalent to that described in the Standard Review Plan if GE would: "(1) verify that this program includes post-irradiation visual inspection of standard design fuel bundles which have not been identified as leakers by sipping or other methods, and (2) that the current GE fuel surveillance program for standard fuel designs will continue at its present level of effort."

The first item is specifically considered in the GE program. However, inspection of non-leakers is not performed on a routine basis but only in cases when information of special interest can be obtained. In these cases, a total visual examination is performed. For instance, if GE desired technical information on a particular subject such as end plug

DSER OPEN ITEM 107

wear or model verification data, then the inspections described in the first item would be performed. These inspections are performed at a variety of plants and include plants in which no fuel problems are expected.

Regarding the second item above, the GE fuel surveillance program is currently planned to continue at approximately its present level of effort. The number and type of inspections will vary from year to year, of course, depending on offgas measurements and the degree of technical interest as explained in the previous paragraph.

The next point raised in Reference 1 concerns the conditional aspect of GE's lead test assembly (LTA) program described in Reference 3. Detailed measurements of LTA's are not performed prior to irradiation in all cases. When the LTA's represent significant design changes, though, such as the advanced LTA's in Browns Ferry 3 and Peach Bottom 3, detailed measurements are performed prior to irradiation. In addition, detailed examinations are performed at the end of each operating cycle on specific LTA's and upon discharge of most LTA's, depending on the subsequent interest in implementing the design change demonstrated in the LTA.

The final point raised in Reference 1 addresses the threshold of offgas activity that would result in non-routine inspection of standard fuel designs. The offgas activity threshold would (a) vary from plant to plant, (b) be contingent on the amount of fuel failures predicted from the increase in offgas, and (c) depend on whether the cause of the failures could be identified without performing an examination. Inspections would generally be performed if the number of failures predicted is on the order of ten bundles, but this number could be more or less depending on the surrounding circumstances. For example, if offgas activity approaches technical specification limits and a cause cannot be assessed, fuel inspections could be performed even if the number of fuel bundles with failures is judged to be fewer than ten. On the other hand, if the cause is assessed - for instance, control blades were withdrawn at power - an inspection would not be performed even if the number of fuel bundle failures were greater than ten.

We hope that this response provides the clarification required to arrive at a mutually acceptable surveillance program.

Very truly yours,

J. S. Charnley

J. S. Charnley, Fuel Licensing Manager
Nuclear Safety and Licensing Operation

JSC:jg/b01231

cc: L. S. Gifford
G. G. Sherwood

DSER OPEN ITEM 107

DSER OPEN ITEM 108 (SECTION 4.2)

GADOLINA THERMAL CONDUCTIVITY EQUATION

The gadolinia thermal conductivity equation used in the GESSAR-II fuel centerline melting analysis described in NEDE-24011 was not the same equation submitted and approved in Appendix B to NEDE-23785-1-P. GESSAR-II references NEDE-20943-P (which was withdrawn), which provided a different gadolinia thermal conductivity equation. This raises a concern about the adequacy of GE's gadolinia fuel incipient melting calculations for Hope Creek (in particular, Table 2-4 of NEDE-24011-P). The applicant should confirm the adequacy of Table 2-4 in NEDE-24011-P or submit updated results for review.

RESPONSE

Discussions with the staff of the NRC Core Performance Branch led to agreement with General Electric that this issue is generic. At the NRC staff's request, General Electric recounted this agreement in a February 2, 1984, letter to L. S. Rubenstein. This issue has been resolved for the Perry and Hanford SERs based on prior information identical to that contained in the February 2 letter. The fuel design evaluation and the results described in this letter are also applicable to the Hope Creek fuel.

GENERAL ELECTRIC

NUCLEAR POWER SYSTEMS DIVISION
GENERAL ELECTRIC COMPANY • 175 CURTNER AVENUE • SAN JOSE, CALIFORNIA 95125

MC 682 (408)925-3697

JSC-04-84
MFN-015-84

February 2, 1984

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, DC 20555

Attention: L. S. Rubenstein, Assistant Director
Core and Plant Systems

Gentlemen:

SUBJECT: OVERHEATING OF GADOLINIA FUEL PELLETS

This letter addresses the confirmatory issue pertaining to the overheating of gadolinia fuel pellets that has appeared in the Safety Evaluation Reports of individual plants seeking operating licenses. The NRC has treated this issue as plant-specific and has resolved the issue on at least two plant dockets based on the information presented in the second paragraph of this letter. However, this issue is in fact generic, and GE hereby requests closure of this issue on a generic basis.

An improved fuel rod thermal-mechanical design code has recently been developed and qualified that utilizes the revised gadolinia fuel thermal conductivity relations. This code has been reviewed and approved by the NRC; application of this code is currently being reviewed by the NRC on the NEDE-24011 docket. This approved code was used to evaluate all GE fuel to be used for new plants. The results of the evaluation for fuel centermelting indicate that gadolinia fuel melting is not expected to occur during normal steady-state operation or during the largest whole core anticipated operational transient.

Please call me if I can be of any assistance on this matter.

Very truly yours,

J S Charnley

J. S. Charnley
Fuel Licensing Manager
Nuclear Safety and Licensing Operation

cc: C. H. Berlinger (NRC)
L. S. Gifford (GE)
R. Lobel (NRC)
G. G. Sherwood (GE)

DSER OPEN ITEM 108

Dupe PDR
~~8402080201~~

DSER Open Item No. 111a and b (Section 5.2.4.3)

PRESERVICE INSPECTION PROGRAM (COMPONENTS WITHIN REACTOR PRESSURE BOUNDARY)

The SER input will be completed after the applicant (a) dockets a complete and acceptable PSI program (b) submits all relief requests with a supporting technical justification.

RESPONSE

The preservice inspection program has been submitted under separate cover (April 13, 1984, letter from R. L. Mittl - PSE&G to A. Schwencer - NRC). In addition, for the information requested in Item b, see response to Question 250.3.

DSER Open Item No. 111c (Section 5.2.4.3)

PRESERVICE INSPECTION PROGRAM (COMPONENTS WITHIN REACTOR PRESSURE BOUNDARY)

The initial ISI program has not been submitted by the Applicant. This program will be evaluated after the applicable ASME Code Edition and Addenda can be determined based on Section 50.55a(b) of 10CFR Part 50, but before inservice inspection commences during the first refueling outage.

RESPONSE

For the information requested above see FSAR Sections 5.2.4 and 6.6.

DSER Open Item No. 119 (Section 6.2)

TMI ITEM II.E.4.1

The HCGS has two redundant hydrogen recombiner packages used for post-accident combustible gas control. The containment penetration associated with the hydrogen recombiner system are a combined design. The hydrogen recombiners are isolated by two isolation valves on suction inlet and are located downstream from the purge system isolation valves. In order to properly evaluate this design, we require information from the applicant outlined in Section 6.2.4. We will report on the resolution of this matter in a supplement to this report.

RESPONSE

For the information requested above see the response to DSER Open Item 132.

DSER Open Item No. 123 (Section 6.2.1.4)

BUTTERFLY VALVE OPERATION (POST ACCIDENT)

Vacuum in the suppression chamber is relieved by a 24-inch vacuum breaker assembly located in each of the two lines between the reactor building and the suppression chamber free space. Each assembly consists of a check-type vacuum relief valve and a pneumatically operated butterfly valve in series. The butterfly valve is located between the containment and the check-type valve. The check-type valves are self-activating and can be remote manually operated from the main control room for testing purposes. The butterfly valves which are normally closed for containment isolation purposes, are activated by differential pressure between the reactor building and the suppression chamber free space. The butterfly valves can also be remote manually operated from the main control room for testing purposes. The power failure position of the butterfly valves is the closed position. The air supply for these valves is a non ESF supply. We will require the applicant to comment on how these valves can be operated post accident, if this air supply is unavailable.

RESPONSE

The butterfly valves are provided with an accumulator which is designed to ASME Code, Section III, Class 3 requirements. The accumulators are provided with a make-up source from the safety-related primary containment instrument gas supply system. (See Figure 6.2-29).

In the event of an air failure, the accumulator will supply the air to activate the valves. Therefore, the valves can be operated post-accident. See revised Section 6.2.1.1.4.1.

2. Flow through the vents is adiabatic.

3. The temperature of the suppression chamber atmosphere is equal to the temperature of the suppression pool.

4. No credit is taken for heat losses to the drywell wall, suppression chamber walls, and internal structures.

6.2.1.1.4 Negative Pressure Design Evaluation

6.2.1.1.4.1 Containment Vacuum Relief Valves

The containment is designed to withstand an external-to-internal differential pressure of 3 psi. To ensure that this design limit is not exceeded, vacuum relief valves are provided to limit the inward pressure loading on the drywell and suppression chamber walls to no more than 2.5 psi.

Vacuum in the drywell is relieved by eight 24-inch vacuum relief valves located on the vent header of the drywell-to-suppression chamber vent system. These valves are self-actuating, check-type, that can also be remote-manually operated from the main control room for testing purposes. The vacuum relief valves between the drywell and the suppression chamber are sized to provide a total flow area of no less than approximately one-sixteenth of the net vent system cross-sectional area.

Vacuum in the suppression chamber is relieved by a 24-inch vacuum breaker assembly located in each of two lines between the reactor building and the suppression chamber free space. Each assembly consists of a check-type vacuum relief valve and a pneumatically operated butterfly valve mounted in series, with the butterfly valve located between the containment and the check-type valve. The check-type valves are self-actuating and can be remote-manually operated from the main control room for testing purposes. The butterfly valves, which are normally closed for containment isolation purposes, are actuated by differential pressure between the reactor building and the suppression chamber free space. The butterfly valves can also be remote-manually operated from the main control room for testing purposes. The controls and instrumentation for each butterfly valve are powered from different Class 1E electrical channels to ensure that

Insert

failure of a single electrical channel does not disable more than one vacuum breaker assembly. Each vacuum breaker assembly is sized on the basis of the flow of air from the reactor building required to limit the containment collapse pressure to within 2.5 psi. The maximum containment depressurization rate is a function of the containment spray flow rate and temperature and the assumed initial conditions of the containment atmosphere. Low spray temperatures and containment atmospheric conditions that yield the minimum numbers of contained noncondensable moles of gas are assumed for conservatism.

The containment vacuum relief valves are qualified to Seismic Category I criteria and are designed and manufactured in accordance with the requirements of the ASME B&PV Code, Section III, Class 2. The valves and appurtenances are designed to operate at a maximum pressure and temperature of 62 psig and 340°F, respectively, concurrent with a maximum relative humidity of 100%. During such environmental conditions, the valves open fully within 1 second, with a 0.25 psi differential pressure existing across the valve. Each valve is equipped with redundant valve-position limit switches, which are suitably sensitive to provide main control room indication of valve closure to a tolerance of 0.01 inch.

6.2.1.1.4.2 Containment Depressurization Evaluation

Negative pressure differentials (negative corresponding to an inward loading) across the drywell walls are caused by the rapid depressurization of the drywell. Events that cause depressurization in the drywell are:

a. Cooling cycles

b. Inadvertent containment spray actuation during normal operation

c. Steam condensation following RCS pipe ruptures with inadvertent containment spray actuation.

Cooling cycles result in minor pressure transients in the drywell, which occur slowly and are controlled by heating and ventilating equipment. Inadvertent spray actuation during normal operation results in a more significant pressure transient and becomes important in sizing the suppression chamber-to-reactor building vacuum breaker assemblies. Steam condensation following RCS pipe ruptures with inadvertent containment spray actuation within the drywell results in the most severe pressure

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The normal air supply for these valve actuators is from the instrument air system. To assure these butterfly valves can operate post-accident, they are provided with an accumulator which is designed to ASME Code, Section III, Class 3 requirements. The accumulators are provided with a make-up source from the safety-related primary containment instrument gas supply system. (See Figure 6.2-29).

DSER OPEN ITEM 123

DSER Open Item No. 130 (Section 6.2.3)

POTENTIAL BYPASS LEAKAGE PATHS

Although the primary containment is enclosed by the secondary containment, there are systems that penetrate both the primary and secondary containment boundaries, creating potential paths through which radioactive material in the primary containment could bypass the filtration, recirculation, ventilation system. The criteria by which potential bypass leakage paths are determined are the BTP CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants." These criteria include specific requirements for barriers - such as water sealing systems, leakage control systems, and closed systems employed to process or preclude bypass leakage. Utilizing these criteria the applicant has identified in FSAR Table 6.2-15 those lines penetrating the primary containment that are potential reactor building bypass leakage paths, and the bypass leakage barrier(s) that will prevent bypass leakage. Since the applicant has not fully responded to our concerns regarding the Containment Isolation System (Section 6.2.4), we are unable to complete our review of the potential bypass leakage paths. We will report on this matter in a supplement to this SER.

RESPONSE

For the information requested above see the response to DSER Open Item No. 132.

DSER Open Item No. 138 (Section 6.6)

PRESERVICE INSPECTION PROGRAM FOR CLASS 2 AND 3 COMPONENTS

The complete evaluation of the PSI program will be presented in a supplement to the SER after the applicant submits the required examination information and identifies all plant specific areas where ASME Code Section XI requirements cannot be met and provides a supporting technical justification.

RESPONSE

For the information requested above, see the response to DSER Open Item 111a and b.

DSER Open Item No. 139 (Section 6.7)

MSIV LEAKAGE CONTROL SYSTEM

The MSIVLCS is protected from the dynamic effects associated with the LOCA, the only pipe break and event where this system is required to operate. However, insufficient information has been provided in the FSAR to allow us to conclude that the components of each subsystem are protected by separation and barriers against internally and externally generated missiles such that their function will not be impaired under postulated LOCA conditions. Thus, we cannot conclude that the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," and the guidelines of Regulatory Guide 1.96, Positions C.2 and C.4, are satisfied.

RESPONSE

Section 6.7.3.1 has been revised to discuss the effects of internally and externally generated missiles on the MSIV sealing system.

The effects of single active failures (including one MSIV failure to close) are provided in Section 6.7.3.2.

6.7.2.4 Equipment Required

The following equipment/components are provided:

a. Piping - Process piping is carbon steel pipe throughout and it is designed and constructed to ASME B&PV Code, as discussed in Section 3.2.

b. Valves - Motor-operated, air-operated, relief, and check valves

c. Instrumentation - The requirements and criteria for the MSIV sealing system instrumentation are discussed in Chapter 7.

The remainder of the piping and components are discussed in Section 9.3.6.

6.7.3 SYSTEM EVALUATION

An evaluation of the capability of the main steam isolation valve (MSIV) sealing system to control the release of radioactivity from the MSIVs following a loss-of-coolant accident (LOCA) has been conducted. The results of this evaluation are presented in the following sections.

6.7.3.1 Functional Protection Features

The equipment in the two independent subsystems (inboard and outboard) are physically separated. The equipment is designed to operate under the expected environmental conditions appropriate to the equipment location.

internal

protected by separation and barrier from

caused by equipment failure (see Section 3.5.1

The MSIV sealing system equipment is ~~arranged so as to minimize the~~ exposure of the system components to missiles, pipe breaks, and jet forces caused by the LOCA event. Equipment is located in the reactor building, hence the effects of the design basis recirculation line break would not impact the system ability to function. Furthermore, the primary containment instrument gas system equipment that supplies gas to the MSIV sealing system is located in the reactor building outside the steam tunnel.

and postulated external missiles

DSER Open Item No. 141C (Section 9.1.3)

SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

The applicant has not committed to include the portions of the cooling and cleanup systems which are not normally operating in the inservice inspection and periodic functional testing programs as decribed in Sections 6.6 and 3.6.6 of the SRP. The Applicant has not specified the frequency of the testing. Thus, the requirements c General Design Criterion 45, "Inspection of Cooling Water Systems," and 46, "Testing of Cooling Water Systems," are not satisfied.

RESPONSE

The spent fuel cooling system does not perform a specific function in shutting down the reactor or in mitigating the consequences of an accident; therefore, does not meet the criteria for being included in ASME B&PV Code Section XI testing requirements.

DSER OPEN ITEM 142a AND b (SECTION 9.1.4)

LIGHT LOAD HANDLING SYSTEM (Related To Refueling)

Redundant interlocks and limit switches have not been provided to prevent accidental collision with pool walls. The applicant must provide these redundant interlock and limit switches or provide the results of an analysis which shows that the effects of a fuel bundle colliding with the pool wall is bounded by the fuel handling accident analysis in Chapter 15 of the FSAR.

Based on the above, we cannot conclude that the requirements of General Design Criteria 61, "Fuel Storage and Handling and Radioactivity Control" and 62, "Prevention of Criticality in Fuel Storage and Handling" and the guidelines of Regulatory Guide 1.13, Position C.3 with respect to prevention of unacceptable radioactivity releases and criticality accidents are satisfied.

RESPONSE

Strict administrative and procedural controls will assure that a collision of a fuel bundle with the pool wall will not occur. An analysis has shown that a postulated, accidental fuel-bundle collision with the pool wall cannot result in more mechanical damage to the bundle hardware or the fuel or result in more fission product release than could result from the postulated drop of a fuel bundle over the reactor core. The analysis of the fuel drop accident, described in Section 15.7.4, shows that the resulting fission product release would be within the guidelines of 10CFR100. Since the consequences of the wall collision accident would be bounded by those of the fuel drop accident, redundant interlocks and limit switches are not required.

DSER OPEN ITEM 168 (SECTION 12.5.2)

EQUIPMENT, TRAINING, AND PROCEDURES FOR INPLANT IODINE INSTRUMENTATION.

The applicant will utilize portable ventilation systems equipped with HEPA filters, or HEPA and sorbant filters, to minimize airborne contamination in highly contaminated areas. Continuous air monitors will be used to monitor airborne concentrations at specific work locations. Section III.D.3.3 of NUREG-0737 states that each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident. The applicant will provide a description of the equipment, training, and procedures to comply with Section III.D.3.3 by June 1, 1985. This is an open item.

Response

A description of the equipment, training, and procedures for accurately determining the airborne iodine concentration in areas within the HCGS plant where personnel may be present during an accident will be provided by June 1, 1985. These procedures and associated training will meet the intent of Section III.D.3.3 of NUREG-0737.

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DSER Open Item No. 170 (Section 13.5.2)

PROCEDURES GENERATION PACKAGE SUBMITTAL

In Attachment D to a letter from R. L. Mittl to the Director, NRR, dated April 15, 1983, and in the FSAR Amendment 4 clarification to the letter, the applicant has committed to implement Supplement 1 to NUREG-0737 (Generic Letter 82-33) and has committed to submit its procedures generation package (PGP) in January 1985. The PGP will be based on the BWR Emergency Procedure Guidelines prepared by General Electric and the BWR Owners Group which have been reviewed by the staff and approved by Generic Letter 83-05, dated February 8, 1983.

RESPONSE

As noted in the Draft SER it is the intent of PSE&G to submit the HCGS procedures generation package in January 1985.

DSER Open Item No. 171 (Section 13.5.2)

TMI-2 ITEM I.C.1

The staff will review the PGP for compliance with Supplement 1 to NUREG-0737. Our review must be completed prior to issuance of the operating license and will be addressed in a supplement to this Safety Evaluation Report. Until the review is completed, Task Action Plan Item I.C.1 is considered open.

RESPONSE

The procedures generation package will be submitted to the NRC for review in January 1985.

DSER Open Item No. 172 (Section 13.5.2)

PGP COMMITMENT

The applicant should commit to the following: 1) the PGP will be submitted to NRC three months prior to start of operator training, 2) all proposed operating and maintenance procedures will be completed at least three months prior to fuel loading, and 3) procedures will be available for review in advanced draft form at least six months prior to fuel loading. It is the staff position that procedures must be completed in sufficient time to ensure operator and appropriate plant staff familiarization. The FSAR should be modified to describe how adequate operator and plant staff familiarization will be assured.

RESPONSE

FSAR Sections 1.10, 13.5.2 and 13.5.2.1 have been revised to provide the requested information.

activities will be listed with a brief overview of their scope. This procedure will be deleted at the start of the first refueling.

In addition to these station administrative procedures, operationally oriented administrative procedures provide guidelines for the operations senior shift supervisors and their shift crews, as well as procedures for night order book usage and control. Operations administrative procedures meet the requirements of 10 CFR 50.54(i), (j), (l), and (m). Figure 13.5-1 indicates the main control room area designated as "at the controls," the area restricted to licensed personnel and the limitations of the reactor operator while manipulating the controls.

delete

~~13.5.2 OPERATING AND MAINTENANCE PROCEDURES~~

~~The operating and maintenance procedures meet the relevant requirements as discussed in Section 1.8.~~

~~13.5.2.1 Main Control Room Operating Procedures~~

~~The following categories delineate those procedures that are performed primarily within the main control room.~~

13.5.2.1.1 Operating Instructions

Operating instructions are provided for startup, normal, manual, and automatic modes of operation of each system or subsystem related to plant safety. Detailed checkoff lists are included, where appropriate, within each procedure. These lists prescribe the proper valve lineup or switch position for the addressed mode of operation.

13.5.2.1.2 Overall Plant Operating Procedures

Overall plant operating procedures provide instructions for integrated plant operations. Checkoff lists are used for confirming completion of major steps in the proper sequence.

DSER OPEN ITEM 172

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13.5.2 OPERATING AND MAINTENANCE PROCEDURE

The operating and maintenance procedures meet the relevant requirements as discussed in Section 1.8.

It is planned that most operating and maintenance procedures will be completed at least three months prior to fuel load and will be available for review in advance draft form at least six months prior to fuel load. This will provide sufficient lead time to ensure that plant personnel can become familiar with them. Where practical the preoperational testing phase will be used to demonstrate the adequacy of the operating procedures.

13.5.2.1 MAIN CONTROL ROOM OPERATING PROEDURES

The following categories delineate those procedures that are performed primarily within the main control room. Operator familiarization with these procedures is acquired though initial, requalification and replacement training programs. Furthermore, these procedures will be utilized in simulator training.

DSER Open Item 172

Response

See Section 13.1 for discussion of the PSE&G and HCGS organizations.

The safety review group reports directly to the general manager - nuclear support as discussed in Section 13.4.4 and shown on Figure 13.1-8.

* I.C.1 SHORT-TERM ACCIDENT ANALYSIS AND PROCEDURE REVIEW

Position

In our letters of September 13 and 27, October 10 and 30, and November 9, 1979, we required licensees of operating plants, applicants for operating licenses, and licensees of plants under construction to perform analyses of transients and accidents, prepare emergency procedure guidelines, upgrade emergency procedures, and to conduct operator retraining (see also Item I.A.2.1 of this report). Emergency procedures are required to be consistent with the actions necessary to cope with the transients and accidents analyzed. Analyses of transients and accidents were to be completed in early 1980, and implementation of procedures and retraining were to be completed 3 months after emergency procedure guidelines were established; however, some difficulty in completing these requirements has been experienced. Clarification of the scope of the task and appropriate schedule revisions were included in NUREG-0737, Item I.C.1.

Pending staff approval of the revised analysis and guidelines, the staff will continue the pilot monitoring of emergency procedures described in Item I.C.8 (NUREG-0660). The adequacy of the boiling water reactor vendor's guidelines will be identified to each near-term operating licensee during the emergency procedure review.

Response

All emergency procedures will be written following the guidelines of INPO 82-017, Emergency Operating Procedure Writing Guideline, and the guidelines of the BWR Owners Group-Emergency Procedures Committee, as long as the guidelines do not contradict existing NRC directives. These procedures will be available March 1, 1985. The PGP will be submitted in January 1985. This will provide a minimum of 3 months prior to the start of operator training on the emergency procedures.

DSER OPEN ITEM 172

DSER Open Item No. 173 (Section 13.5.2)

PROCEDURES COVERING ABNORMAL RELEASES OF RADIOACTIVITY

A procedure or procedures covering abnormal releases of radioactivity should be included among the available procedures.

RESPONSE

The following procedures will be developed to address abnormal releases of radioactivity:

1. Abnormal operating procedure titled "Abnormal Release of Radioactivity."

2. Emergency operating procedure titled "Radioactivity Release Control" (this procedure will be developed from BWROG emergency procedure guidelines).

DSER OPEN ITEM 181 (Section 15.9.5)

TMI-2 ITEM II.K.3.3

Response given in Section 1.10 of the FSAR is not acceptable. The staff requires a detailed response explaining how PSE&G is planning to comply with the requirements of Item II.K.3.3.

Response

HCGS will report any failure of a safety relief valve to open or close when called upon, within 24 hours by phone, confirmed the first working day following the event by telegraph (or similar transmission) and followed up with a written report in two weeks. This written report will be in the form of a Licensee Event Report.

The PSE&G HCGS annual report to the NRC will list each safety relief valve which is challenged during the year and will include the number of times each is challenged.

These reporting requirements will be included in the HCGS Technical Specification.

c. Increase in drywell sump level.

- II.K.2 COMMISSION ORDERS ON BABCOCK & WILCOX PLANTS

Response

These requirements are not applicable to HCGS.

- II.K.3 FINAL RECOMMENDATIONS OF B&O TASK FORCE

- II.K.3.1 INSTALLATION AND TESTING OF AUTOMATIC PORV ISOLATION SYSTEM

Response

This requirement is not applicable to HCGS.

- II.K.3.2 REPORT ON OVERALL SAFETY EFFECT OF PORV ISOLATION SYSTEM

Response

This requirement is not applicable to HCGS.

- II.K.3.3 FAILURE OF PORV OR SAFETY VALVE TO CLOSE

Position

Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report. This requirement is to be met before fuel load.

Response

See Attached →

~~PSE&G will comply with the requirements of this item prior to fuel load.~~

- II.K.3.5 AUTOMATIC TRIP OF REACTOR COOLANT PUMPS DURING LOCA

Response

This requirement is not applicable to HCGS.

- II.K.3.7 EVALUATION OF PORV OPENING PROBABILITY DURING OVERPRESSURE TRANSIENT

Response

This requirement is not applicable to HCGS.

- II.K.3.9 PROPORTIONAL INTEGRAL DERIVATIVE (PID) CONTROLLER MODIFICATION

Response

This requirement is not applicable to HCGS.

- II.K.3.10 PROPOSED ANTICIPATORY TRIP MODIFICATION

Response

This requirement is not applicable to HCGS.

Response

HCGS will report any failure of a safety relief valve to open or close when called upon, within 24 hours by phone, confirmed the first working day following the event by telegraph (or similar transmission) and followed up with a written report in two weeks. This written report will be in the form of a Licensee Event Report.

The PSE&G HCGS annual report to the NRC will list each safety relief valve which is challenged during the year and will include the number of times each is challenged.

These reporting requirements will be included in the HCGS Technical Specification.

DSER Open Item No. 191 (Section 7.2.2.8)

SCRAM DISCHARGE VOLUME

The applicant is required to revise FSAR Figure 7.2-1 to show the correct SDV level instrumentation design.

RESPONSE

For the information requested above, see the response to Question 421.14.