PSE&G Public Service Electric and Gas Company

80 Park Plaza, Newark, NJ 07101 / 201 430-8217 MAILING ADDRESS / P.O. Box 570, Newark, NJ 07101

Robert L. Mittl General Manager
Nuclear Assurance and Regulation

May 16, 1984

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

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

Gentlemen:

HOPE CREEK GENERATING STATION
DOCKET NO. 50-354
FSAR COMMITMENT STATUS THROUGH APRIL 1984

Public Service Electric and Gas Company presently does not plan to issue Amendment No. 6 to the Hope Creek Generating Station Final Safety Analysis Report before July 1984. Accordingly, this letter is provided to document the status of Hope Creek Generating Station responses to NRC requests for additional information which were forecasted to be responded to by April 1984.

Attachment I is a tabulation of the Hope Creek Generating Station Final Safety Analysis Report commitments for April 1984 and the corresponding resolution for each commitment. Attachments II through V provide the responses to the questions forecasted to be responded to in April 1984, which will be included in Amendment No. 6.

Public Service Electric and Gas Company submitted Amendment No. 5 to the Hope Creek Generating Station Final Safety Analysis Report on April 30, 1984 (R. L. Mittl, PSE&G to A. Schwencer, NRC). Amendment No. 5 incorporated the responses to prior NRC requests for additional information forecasted for closure in February and March 1984, as outlined in PSE&G letters of March 5, 1984, and April 2, 1984, (R. L. Mittl, PSE&G to A. Schwencer, NRC).

Boo1
1/1

Should you have any questions in this regard, please contact us.

Very truly yours,

R J [signature]

Attachment I - Hope Creek Generating Station - FSAR Commitment Status through April 1984.
Attachment II - Response to Question 251.1
Attachment III - Response to Question 220.16
Attachment IV - Response to Question 430.94
Attachment V - Response to Question 640.9

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

Mr. W. H. Bateman (w/attach)
USNRC Senior Resident Inspector

MO 16 01/02-D

ATTACHMENT I
HOPE CREEK GENERATING STATION
FSAR COMMITMENT STATUS THROUGH APRIL 1984

FSAR Commitment Location	Commitment Resolution
1. Question/Response Appendix: Question 251.1	This commitment concerns demonstration of compliance to 10CFR50.55a and Appendices G and H of 10CFR Part 50 for ferritic reactor coolant pressure boundary materials. The response to this question is provided in Attachment II and will be included in Amendment 6 to the HCGS FSAR.
2. Question/Response Appendix: Question 220.16	This commitment concerns a description of the spent fuel racks conformance with applicable provisions of subsection NF of the ASME Code. The response to this question is provided in Attachment III and will be included in Amendment 6 to the HCGS FSAR.
3. Question/Response Appendix: Question 220.21	This commitment concerns submittal of a comparison of response spectra results from the finite-element and half-space methods. This information will be provided in July 1984.
4. Question/Response Appendix: Question 430.94	This commitment concerns submittal of a description of diesel generator fuel oil delivery to the site during flood conditions and the procedures used for refilling the storage tanks during flood conditions and non-flood conditions. The response to this question is provided in Attachment IV and will be included in Amendment 6 to the HCGS FSAR.
5. Question/Response Appendix: Question 460.4	This commitment concerns submittal of information on laboratory tests conducted under the process control program (PCP) for the solidification of solid radwaste, and on the laboratory/field instruction record sheet to be used within the PCP. This information will be provided in January 1985.

FSAR Commitment Location	Commitment Resolution
6. Question/Response Appendix: Questions 630.7, 630.9, 630.10, 630.12	These commitments concern submittal of information on training programs for licensed and non-licensed operations personnel, schedules for examinations prior to fuel loading and after criticality, training programs for management personnel and technical support staff, and fire protection training. This information has been provided in Amendment 5 to the HCGS FSAR.
7. Question/Response Appendix: Question 640.9	This commitment concerns submittal of the results of our findings on adequacy of the drainage in affected areas to preclude flooding, upon automatic sprinkler actuation. The response to this question is provided in Attachment V and will be included in Amendment 6 to the HCGS FSAR.

DJD:vw

5/16/84

M016 03/04-D

Attachment II

HOPE CREEK

QUESTION 251.1

Appendices G and H, 10CFR50 were revised in the Federal Register on May 27, 1983 and became effective on July 26, 1983.

QUESTION 251.1a

Identify ferritic reactor coolant pressure boundary materials that do not comply with the fracture toughness requirements of Section 50.55a and Appendices G and H of 10CFR Part 50.

RESPONSE

A major condition necessary for full compliance to Appendix G of 10 CFR Part 50 is satisfaction of the requirements of the Summer 1972 addenda to Section III of the ASME B&PV Code. This is not possible with the HCGS reactor pressure vessel (RPV) and the main steam isolation valves (MSIVs), which were purchased to earlier B&PV Code requirements. The RPV was qualified to the applicable General Electric RPV purchase specifications and by toughness testing in accordance with the 1968 edition of Section III of the ASME B&PV Code as well as the addenda through Winter 1969. The MSIVs were built to the draft addenda for Class 1 pumps and valves of the 1968 ASME B&PV Code.

QUESTION 251.1b

For materials which cannot meet the fracture toughness requirements of Section 50.55a and Appendices G and H of 10 CFR Part 50, provide alternative fracture toughness data and analyses to demonstrate their equivalence to the requirements of 10 CFR Part 50.

RESPONSE

For plants that received a construction permit prior to August 15, 1973, the NRC Branch Technical Position MTEB No. 5-2 provides guidance for making conservative estimates and assumptions that may be used to show compliance with the intent of the latest requirements. Paragraph 1.3 of MTEB 5-2 permits the use of other methods that can be shown to be conservative. Based on an evaluation of actual toughness data for plants of this period and on applicable data from Welding Research Council Bulletin 217, General Electric procedure Ø1006A006 was derived to estimate compliance with the intent of Appendix G and to establish values for the initial reference transition temperature (RT_{NDT}) of older RPV materials. For other materials, including the MSIVs, methods for evaluation of compliance with the intent of current Appendix G requirements are explained in Appendix 5A. Y

QUESTION 251.1c

To demonstrate conformance to Appendix G and H, 10 CFR Part 50:

QUESTION 251.1c(1)

Provide pressure-temperature limit curves for hydrostatic pressure and leak test, heat-up, cooldown and core operation.

RESPONSE

See Figure 5.3-1. When Appendix 5A was submitted in Amendment 1, this figure was revised to conform to the July 26, 1983 revision of Appendix G.

QUESTION 251.1c(2)

Identify the withdrawal schedule, lead factor, test samples and materials in the Reactor Vessel Materials Surveillance Program.

RESPONSE

The May 27, 1983 revision of Appendix H of 10 CFR 50 requires the withdrawal schedule for the surveillance program capsules to meet the requirements of ASTM standard E 185-82. The lead factors for the HCGS surveillance capsules are 0.86 at the inside surface of the vessel and 1.20 at one-quarter of the way through the vessel wall measured from the inside surface. These lead factors were calculated assuming that the vessel is symmetrical. This assumption was made because the vessel qualification program did not provide for measurements of vessel radii to identify any angular locations where the inside diameter of the vessel is larger than nominal. Hence, it is possible that a surveillance capsule could be located at an extended radius position. This would provide surveillance sample test results lower than calculated and nonconservative values for the peak fluence when it is estimated from the capsule data using the aforementioned lead factors. Details of the materials and test samples contained in the HCGS surveillance program are provided in Section 5A.4.

QUESTION 251.1c(3)

Indicate the reference temperature, RT_{NDT}, for materials in the reactor vessel closure flange region and the beltline regions.

RESPONSE

Reference temperature RT_{NDT} valves for materials in the RPV closure flange regions are given in Section 5A.2. Similar values for materials in the beltline region of the HCGS RPV are provided in Tables 5A-4 and 5A-5.

QUESTION 251.1c(4)

Indicate the chemical composition (copper, nickel and phosphorus), unirradiated upper-shelf energy, and projected end-of-life RT_{NDT} and upper-shelf energy for all beltline materials. RT_{NDT} projects are to be estimated using the "Guthrie Formula" in Commission Report SECY-82--465. Upper-shelf energy projects are to be estimated using Regulatory Guide 1.99, Rev. 1. These projects are to be for the end-of-life neutron fluence at the 1/4T and ID reactor vessel locations.

RESPONSE

Details of where beltline materials are located in the RPV are provided in Tables 5A-4 and 5A-5. Also contained in these tables are the appropriate fluence values used for the shift calculations.

Unirradiated and end-of-life upper-shelf engines and RT_{NDT} values are given in Table 251.1-1 along values for the shifts in RT_{NDT} calculated by the "Guthrie formula" found in Figure E-1 of Commission Report SECY-82-465. However, the radiation shift values used for the pressure-temperature limit curves presented in Figure 5.3-1 are those derived from shifts calculated according to the formula given in revision 1 of Regulatory Guide 1.99.

DBJ:pes/112G
4/3/84

TABLE 251.1-1 Upper-Shelf Energies and "Guthrie" RT_{NDT} Values[1]

Heat# or Heat/Flux	CHEMISTRY (Wt.%) Cu	Ni	P	Initial Upper-Shelf Energies (Ft.-Lb.)	R.G. 1.99 Rev.1 Projected EOL Upper-Shelf Energies (Ft.-Lb.)	RT_{NDT} (°F) Unirradiated	"Guthrie" Shift	"Guthrie" EOL
5K2963-1-2	0.07	0.58	0.009	102	89	-10	+68	+58
5K2530-1-2	0.08	0.56	0.010	86	75	+19	+72	+91
5K3238-1-2	0.09	0.63	0.012	76	66	+7	+77	+84
5K3230-1-2	0.07	0.56	0.010	121	105	-10	+68	+58
6C35-1-2	0.09	0.54	0.010	107	93	-11	+75	+64
6C45-1-2	0.08	0.57	0.008	97	84	+1	+72	+73
5K3025-1	0.15	0.71	0.012	75	69	+19	+84	+103
5K2608-1	0.09	0.58	0.012	75	69	+19	+66	+85
5K2698-1	0.10	0.58	0.010	75	69	+19	+69	+88
510-01205	0.09	0.59	0.010	>92.5	>80	-40	+76	+36
53040/1125-02205	0.08	0.63	0.010	135	117	-30	+73	+43
519-01205	0.010	0.53	0.010	>109	>100	-49	+47	-2
504-01205	0.010	0.51	0.011	>125	>115	-31	+47	+16
55733/1810-02205	0.10	0.68	0.013	>68	>62	-40	+70	+30
53040/1810-02205	0.10	0.68	0.012	>95	>87	-49	+70	+21
001-01205	0.02	0.51	0.012	>109	>100	-40	+49	+9
19468-1-4,5	0.12	0.81	0.011	>79	>73	-20	+74	+54
10024-1-2,3	0.14	0.84	0.010	>70	>64	-20	+80	+60

(1) This information is for a depth of 1/4 of the wall thickness from the inside surface of the vessel.

1. SCOPE OF APPLICATIONS AND OBJECTIVES

1.1 This procedure describes the method to be used for establishing the initial reference temperature (RT_{NDT}) for ferritic vessel steels for older plants where fracture toughness data may be incomplete. These methods represent a General Electric alternate position to the NRC Regulation 10CFR50 Appendix G for these plants.

2. METHODS

2.1 Vessel Plate (SA-533 Gr. B Cl. 1):

Predicted limiting property - either NDT (Nil-Ductility Transition Temperature) or transverse CVN (Charpy V-Notch) 50 ft-lb T.T. (Transition Temperature)

Usual data available - NDT and/or longitudinal CVN at +10 or +40°F

RT_{NDT} prediction method -

Operate on lowest longitudinal CVN ft-lb to get at least 50 ft-lb T.T. by adding 2°F per ft-lb or by plotting a curve (ft-lb versus temperature), where possible. Add additional 30°F to convert from longitudinal to transverse 50 ft-lb T.T.

NOTE: Where transverse CVN impact data are available, but the 50 ft-lb T.T. is not met, operate on the lowest CVN ft-lb to get at least 50 ft-lb T.T. by adding 3°F per ft-lb or by plotting a curve (ft-lb vs temperature), where possible. This extrapolation is valid for CVN test temperatures only in the range (-25° to +50°F).

Derive NDT, where missing, as equal to longitudinal CVN 35 ft-lb T.T.

RT_{NDT} is higher of NDT or transverse CVN 50 ft-lb T.T. -60°F

2.2 Forings (SA-508 Cl. 2):

Predicted limiting property - NDT or transverse CVN 50 ft-lb T.T.

Usual data available - NDT and/or CVN at single temperature

RT_{NDT} prediction method -

Derive CVN 50 ft-lb T.T. as for plate.

When only CVN values are available, estimate NDT as the lower of +70°F or the CVN test temperature where at least 100 ft-lb or 50 percent shear is achieved.

RT_{NDT} is higher of NDT or transverse CVN 50 ft-lb T.T. -60°F.

2.3 Weld Metal (Used to Join SA-533 Gr. B CL. 1 Plates and SA-508 CL. 2 Forgings):

Predicted limiting property - CVN 50 ft-lb T.T.

Usual data available - CVN values at single or at several test temperatures.

RT_{NDT} prediction method -

Operate on lowest CVN ft-lb to get at least 50 ft-lb T.T. by adding 2°F per ft-lb or by plotting a curve (ft-lb versus temperature), where possible

RT_{NDT} is the CVN 50 ft-lb T.T. - 60°F. If NDT is available, it will be considered also. In absence of NDT data, RT_{NDT} shall not be lower than -50°F.

2.4 Vessel Plate (SA-533 Gr. B Cl. 1) and Forging (SA-508 Cl. 2) Weld HAZ:

RT_{NDT} assumed same as for base material. Weld procedure qualification test requirements indicate this assumption is valid.

2.5 Bolting Material (SA-540 Gr. B24):

CVN 45 ft-lb and 25 MLE (Mils Lateral Expansion) are required at no higher than preload temperature or Lowest Service Temperature (LST)

Usual data available - CVN ft-lb and MLE at +10°F

LST prediction method -

If preceding CVN requirements are met at test temperature, then it is LST.

If at least 30 ft-lb, but less than 45 ft-lb and 25 MLE, are met at test temperature, then add 60°F to the test temperature for LST.

QUESTION 220.16 (SECTION 3.8.4)

Indicate whether materials, fabrication, welding, and quality control of the spent fuel racks are in conformance with applicable provisions of subsection NF of the ASME code. If not, identify and justify the deviations.

RESPONSE

~~The spent fuel rack specification requires conformance with the Subsection NF requirements for Class 3 component supports. The purchase order is scheduled to be awarded in December 1983. This response will be verified and deviations, if any, will be provided after award of the purchase order. Table 3.2-1 has been revised to show Subsection NF as the principal code for the racks.~~

The design, fabrication, and welding of the racks conforms with the applicable provisions of Subsection NF. Table 3.2-1 has been revised to show Subsection NF as the principal construction code for the racks.

The materials used for the racks are ASTM A-240-80b, Type 304L and ASTM A-564-79, Type 630 stainless steels. The ASTM A-240-80b material specification is identical to the ASME SA-240, 80S'82, material specification for Type 304L steel, and the ASTM A-564-79

FEB 28 '84 0259775

2/3

QUESTION 220.16 RESPONSE

material specification is identical to the ASME SA-564, 80 S'82, material specification for Type 630 steel.

Certified material test reports are provided for all spent fuel rack steel.

The spent fuel rack materials are not ~~procured from a supplier with a Q.A. Program that meets the requirements of NCA-3800, as required by Subsection NF. Instead,~~ ~~they are~~ procured under a Q.A. Program that is intended to comply with

- 10CFR50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants".
- ANSI/ASME N45.2, "Quality Assurance Program Requirements for Nuclear Facilities", and
- ANSI/ASME NQA-1, "Quality Assurance Program Requirements for Nuclear Power Plants."

QUESTION 220.16 RESPONSE

Table 1.11-1 and Section 3.8.4.8 have been revised to address this exception to the requirements of Standard Review Plan 3.8.4.

II.4.f
Spent fuel rack material should conform to Section III, Subsection NF, of the ASME Code.

ASTM steel procured under an ANSI N45.2 Q.A. Program, instead of steel procured under an ASME Code Q.A. Program in accordance with Subsection NF, is used.

3.8.4.8

TABLE 1.11-1 (cont)

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Differences	FSAR Section(s) Where Discussed
3.8.4 (Rev 1)	II.2		3.8.4.8
	Category I structures shall be designed in accordance with Specification ACI 349 as augmented by Regulatory Guide 1.142.	Category I structures are designed in accordance with Specification ACI 318-71.	
	II.2		8.8.1
	Conformance to Regulatory Guides 1.10, 1.55 and 1.94.	Nonconformance, in part, with Regulatory Guides 1.10, 1.55, and 1.94.	
	II.4.d		3.8.2.8
	Design reports are acceptable if it contains the information specified in Appendix C.	Sufficient information is available in forms other than those outlined in Appendix C.	
3.8.5 (Rev 1)	II.4.e		3.8.2.8
	Design report is considered acceptable if it satisfies the guidelines of Appendix C to SRP 3.8.4.	Sufficient information is available in forms other than those outlined in Appendix C.	
3.9.3 (Rev 1)	II.1		3.9.3.5
	Acceptability of the combination of design and service loadings applicable to the design of Class 1, 2, and 3 components should be judged by comparison with positions stated in Appendix A of SRP 3.9.3.	Design and service loadings applicable to the design of Class 1, 2, and 3 components do not conform, in part, to Appendix A or SRP 3.9.3.	
3.9.5 (Rev 2)	II.b		3.9.5.4
	Design and construction of the core support structures is to conform to the requirement of Subsection NG of Section III of the ASME Code.	Design and construction of the core support structures do not specifically conform to Subsection NG of Section III of the ASME Code.	

FEB 28 '84 0259775

facility. They are not shear walls and are designed to the working stress method of UBC, as listed in Table 3.8-7.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

Materials, quality control, and special construction techniques are discussed in Section 3.8.6.

3.8.4.7 Testing and Inservice Inspection Requirements

Testing and inservice inspection are not required for Seismic Category I structures other than the primary containment and its internals.

3.8.4.8 SRP Rule Review

3.8.4.8.1 Concrete Design

Acceptance Criteria II.2 of SRP 3.8.3 and 3.8.4 requires that Category I structures be designed in accordance with Specification ACI 349 as augmented by Regulatory Guide 1.142. The HCGS design was based on the requirements of Specification ACI 318-71.

The Category I structures concrete design for HCGS began prior to the issue of Specification ACI 349 (1976). As a result, all concrete design is based on using Specification ACI 318-71 with the following clarifications:

A review of the design of the HCGS Seismic Category I structures indicates that there is no impact due to differences in the structural acceptance criteria between ACI 318-71 and ACI 349-76 as augmented by Regulatory Guide 1.142.

The load combinations used are in conformance with the following SRP sections except that the 0.9 load factor on dead load as required by ACI 349-76 was not used:

Structures	SRP Section
Primary Containment Internal Concrete Structures	3.8.3.II.3.b.
Other Seismic Category I Concrete Structures	3.8.4.II.3.b.

Based on parametric analyses, an adequate design margin exists to compensate for the effects of the reduced dead load factor.

Table 3.8-18 provides a comparison of the allowable ductility ratios used for design of the concrete structural components subjected to impactive and impulsive loadings and the criteria outlined in Appendix C of ACI 349 as modified by Regulatory Guide 1.142.

Except for flexural beams and slabs subjected to impactive loads, the allowable ductility ratios used in the design are less than or equal to those in the Regulatory Guide. The allowable ductility ratios for beams and slabs used in design are based on the evaluation of test data reported in References 3.8-5 and 3.8-6 and tests performed by the Architect-Engineer.

The test results consistently demonstrate that actual ductility ratios in excess of 50 are reached prior to failure. Therefore, by limiting the values to 10 for beams and 30 for slabs, the design is conservative. Furthermore, the flexural members are designed to meet additional reinforcing requirements (See Table 3.8-18) to ensure ductile behavior.

3.8.4.8.2 Structural Steel Design

Table 3.8-19 provides a comparison of the allowable ductility ratios used for design of structural steel subjected to impactive and impulsive loading, and the criteria outlined in Appendix A of NUREG-0800, SRP Section 3.5.3. Except for flexure in beams subjected to impactive loads (other than the tornado missiles) and axial tension members subject to impulsive loads, the ductility ratios are essentially identical. Based on the recommendations provided in References 3.8-5 and 3.8-6 and tests performed by the Architect-Engineer, it has been demonstrated that steel members under flexural loads can sustain higher ductility ratios (on the order of 30) without collapse. Therefore, a limiting value of 20 used in the design is conservative. Furthermore, additional design and fabrication features (such as box sections, lateral bracings, NDE, etc.) are incorporated in the flexural members to preclude buckling and to ensure material quality.

Regarding the ductility ratio for axial tension members subject to compression loads, the HCGS limit of 3 is always conservative for the types of steel used.

ADD INSERT 1 HERE

X

INSERT 1

3.8.4.8.3 Spent Fuel Rack Design

Acceptance Criterion II.4.f requires that the spent fuel racks be designed in compliance with Appendix D of SRP 3.8.4, which requires that construction materials should conform to Section III, Subsection NF of the ASME Code.

The spent fuel racks are constructed of ASTM A-240 and ASTM A-564 stainless steel. The A-240 and A-564 material specifications are identical to the ASME SA-240 and SA-564 material specifications.

All rack steel is supplied with certified material test reports.

The rack materials are procured under a Q.A. Program that is intended to comply with

- 10CFR50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."
- ANSI/ASME N45.2, "Quality Assurance Program Requirements for Nuclear Facilities", and
- ANSI/ASME NQA-1, "Quality Assurance Program Requirements for Nuclear Power Plants."

QUESTION 430.94 (SECTION 9.5.4)

In Section 9.5.4.2.6 of the FSAR you state that the emergency flood protected truck fill connection for the fuel oil storage tanks is located inside the auxiliary building at Elevation 102 feet (plant grade level). In Section 3.4, Table 3.4-1 you state that the design flood elevation for the D/G building is 120.4 feet with the still water height at 113.8 feet. Provide the following:

a. Describe or provide adequate drawings to show the location of the emergency fuel oil storage tank fill connection.

b. Assuming the emergency fill connection must be used to refill the fuel oil storage tanks. Describe how fuel oil will be delivered to the site during flood conditions and describe the procedures that will be used in refilling the storage tanks during flood conditions and non-flood conditions. The procedures should include fuel hose routing and fire watcher.

c. Describe how flood water is prevented from entering the building during refueling operations. (SRP 9.5.4, Parts I, II & III).

RESPONSE

The diesel fuel oil emergency fill line is located in the auxiliary building at floor elevation 102 feet-0 inches and a center line elevation of 106 feet-6 inches. The emergency diesel fuel oil fill connection is located in an area which is flood protected by the auxiliary building main service entry doors. The location of the diesel fuel oil connection is shown on Figure 430.94-1, reference Figure 1.2-35 for location relative to watertight door.

Response to Item (b) will be provided in April, 1984.

Leakage through the door seals is removed by drainage systems in the building. The flood doors are capable of withstanding the flood height as described in Section 3.4 and Table 3.4-1.

The diesel fuel oil tanks are designed for a seven day supply with the diesel generators operating at full capacity, reference Section 9.5.4.3. It is not anticipated that localized flooding will prevent refueling from occuring at some time during this (SOP) period. System Operating Procedures shall address the refilling of the storage tanks from any fill connection and shall also include proper fuel hose routing and the establishment of a fire watch, when necessary. Abnormal Operating Procedure, [redacted], Acts of Nature, shall provide direction to the operator as to which SOP is to be used, dependent upon environmental conditions. ~~[illegible]~~ These procedures shall be in place by 1/85.

QUESTION 640.9 (SECTION 14.2.12)

Modify FSAR Subsection 14.2.12.1.29 (KC-Fire Protection - Deluge) to provide assurance that:

1. Upon automatic sprinkler actuation, adequate drainage in the affected spaces is provided to preclude flooding (including expected hand-held hose volume).

2. A walk-down of plant equipment is conducted to identify potential incidences where the actuation of fire suppression systems could cause damage to or inoperability of systems important to safety.

See IE Information Notice 83-41: Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment, June 22, 1983.

RESPONSE

The results of our findings on adequacy of the drainage to preclude flooding, upon automatic sprinkler actuation, will be available March 1984.

Section 14.2.12.1.29.b has been revised to include a prerequisite walkdown of the fire protection system to identify potential areas where the fire protection system could cause damage.

Section 14.2.12.1.29.b has been revised to address the provision to drain areas where automatic sprinkler actuation might affect Safe Shutdown equipment.

2. The system responds to simulated fire signals.

3. The refrigeration system operates to maintain pressure and temperature as specified by the manufacturer's technical instruction manual.

14.2.12.1.29 KC-Fire Protection - Deluge

a. Objective

The test objective is to verify the capability of the fire protection system to deliver water to the sprinkler system, pre-action and deluge systems, hose stations, and hydrants at rated pressure and flow.

b. Prerequisites

1. Component tests have been completed and approved.

2. System instrumentation has been calibrated and approved.

3. AC and dc power are available.

4. The diesel fire pump local fuel oil storage tank is in service.

5. Adequate fire protection water supply is available.

6. A walkdown has been performed to identify components or areas that may be susceptible to damage due to actuation of the deluge system.

INSERT A HERE

c. Test Method

1. All valves, controls, alarms, interlocks, and logic are checked for proper operation.

2. Normal system flow paths are verified.

INSERT
A

7. Floor drains have been provided to remove the expected fire fighting water flow from automatic sprinkler systems, hand hose lines, etc. Temporary build up of water in the affected spaces will not flood Safe Shutdown equipment