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Robert L. Mittl General Manager
Nuclear Assurance and Regulation

October 29, 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 SEPTEMBER 1984

Public Service Electric and Gas Company presently does not plan to issue Amendment No. 8 to the Hope Creek Generating Station Final Safety Analysis Report before November 1, 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 September 1984.

Attachment I is a tabulation of the Hope Creek Generating Station Final Safety Analysis Report commitments for September 1984, and the corresponding resolution for each commitment. Attachments II through VI provide responses to commitments forecasted to be responded to in September 1984, which will be included in Amendment No. 8 or 9.

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

Very truly yours,

RL Mittl/RP Douglas

Attachment I - Hope Creek Generating Station - FSAR Commitment Status through September 1984
Attachment II - Response to FSAR Section 3.11.2.6
Attachment III - Response to Question 220.15
Attachment IV - Response to Question 270.2
Attachment V - Response to Question 410.38
Attachment VI - Response to SRAI(5)

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

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

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HOPE CREEK GENERATING STATION
FSAR COMMITMENT STATUS THROUGH SEPTEMBER 1984

FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
1. NRC Generic Letter 83-28 Response: (PSE&G to NRC, 3/30/84)	This commitment concerns providing station operating procedures referenced in the response to NRC Generic Letter 83-28 of 3/30/84. This information will be provided in November 1984.
2. FSAR Section 1.14.1.37.2	This commitment concerns providing setpoints for undervoltage relays and system voltages. This information is provided in Amendment 6 to the HCGS FSAR.
3. FSAR Section 3.11.2.6	This commitment concerns providing a tabulation of all safety-related mechanical equipment located in a harsh environment. This information is provided in Attachment II and will be included in Amendment 9 to the HCGS FSAR.
4. FSAR Table 13.1-4	This commitment concerns providing the resume for the Technical Engineer. This information is provided in letter; R. L. Mittl (PSE&G) to A. Schwencer (NRC) dated August 15, 1984, and will be included in Amendment 8 to the HCGS FSAR.
5. Question/Response Appendix: Question 100.6	Re: TMI Item I.A.3.1: This commitment concerns Hope Creek simulator training.

FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
5. Question/Response Appendix: Question 100.6 (Cont'd)	Information in Section 13.2, which is provided in letter; R. L. Mittl (PSE&G) to A. Schwencer (NRC), dated October 3, 1984, and which will be included in Amendment 8 to the HCGS FSAR, indicates that simulator training is being conducted at the Susquehanna Training Center until the Hope Creek simulator is operational. The Hope Creek simulator is scheduled to be operational in November 1984.
6. Question/Response Appendix: Question 220.15	This commitment concerns providing Spent Fuel Rack analysis, sketches, and mathematical models. This information is provided in Attachment III and will be included in Amendment 8 to the HCGS FSAR.
7. Question/Response Appendix: Question 270.2	This commitment concerns providing a preliminary summary report describing the HCGS Environmental Qualification Program for electrical equipment. This summary report is provided in letter; R. L. Mittl (PSE&G) to A. Schwencer (NRC), dated August 24, 1984. Reference to this information is provided in Attachment IV and will be included in Amendment 8 to the HCGS FSAR.
8. Question/Response Appendix: Question 410.38	This commitment concerns providing Spent Fuel Pool criticality information. This information is provided in Attachment V and will be included in Amendment 8 to the HCGS FSAR.
9. Question/Response Appendix: Question 421.13a	This commitment concerns testing of SSLIS and AIS isolation systems. This information is provided in Amendment 7 to the HCGS FSAR.

FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
10. Question/Response Appendix: Question 430.32	This commitment concerns review of inverters to required voltage range. This information is provided as response to DSER Open Item No. 258 in letter; R. L. Mittl (PSE&G) to A. Schwencer (NRC), dated August 1, 1984, and will be included in Amendment 8 to the HCGS FSAR.
11. Question/Response Appendix: Question 430.33	This commitment concerns tests and analysis of inverters as isolation devices. This information is provided as response to DSER Open Item No. 259 in letter; R. L. Mittl (PSE&G) to A. Schwencer (NRC), dated October 3, 1984, and will be included in Amendment 8 to the HCGS FSAR.
12. Supplemental Request for Additional Information: SRAI (5)	This commitment concerns verifying seismic and dynamic qualification and installation of 85-90% of safety-related equipment. This information is provided in letter; R. L. Mittl (PSE&G) to A. Schwencer (NRC), dated October 19, 1984. Reference to this information is provided in Attachment VI and will be included in Amendment 9 to the HCGS FSAR.
13. DSER Open Item No. 103	This commitment concerns updating FSAR Section 3.10 to show extent of operational testing. This information will be provided in March 1985.
14. DSER Open Item No. 189	This commitment concerns providing documentation to NRC regarding qualification testing performed on isolation systems. This information will be provided in March 1985.

ATTACHMENT II

environment is to establish the qualified life by analysis, including the operability requirements during and after the DBE periods, for the component materials with "significant aging mechanisms" as defined in Section 4.4.1 of IEEE STD-627-1980. Nonmetallic materials with a qualified life greater than 40 years are not considered to be susceptible to significant age degradation. Nonmetallic parts used in mechanical equipment include gaskets, diaphragms, seals, lubricating oil or grease, fluids for hydraulic systems, flexible hoses and packing.

Environmental qualification of mechanical equipment is not intended to replace or modify compliance required by adherence to other applicable codes prepared by organizations such as the ASME, AISC and ACI which are the recognized experts in their fields of endeavor.

Insert 1

A tabulation will be provided by September 1984 listing all safety-related mechanical equipment located in a harsh environment. Nonmetallic subcomponents of this equipment will be indicated and their qualified status provided.

The environments for which this equipment is qualified to operate in are identical to those defined for the electrical equipment qualification program.

3.11.2.7 Qualification Methods for NSSS and Non-NSSS Safety-Related Electrical Equipment

3.11.2.7.1 Margin

IEEE-323 and NUREG-0588, Rev. 1, Paragraph 3.(4) are used as a basis for determining margin. The equipment technical specification for safety-related electrical equipment required to be environmentally qualified include conservative environmental conditions which were derived using environmental parameters which contain conservatisms applied during the derivation of local environmental conditions. The equipment vendor determines what margin must be added to allow for variations in production processes, for inaccuracies in the test equipment and for errors associated with defining satisfactory performance.

The qualification documentation for safety-related equipment will include documented provision that adequate margin has been

INSERT 1

A LISTING OF THE MECHANICAL COMPONENTS IN HARSH ENVIRONMENT SELECTED FOR ENVIRONMENTAL QUALIFICATION IS SHOWN ON TABLE 3.11-4. EACH ~~COMPONENT~~ ~~[illegible]~~ NON-METALLIC SUBCOMPONENT OF EACH PIECE OF THIS EQUIPMENT WILL BE ANALYSED FOR ITS QUALIFIED LIFE AS DESCRIBED ABOVE.

RAS 2 of 4

TABLE 3.11-4
MECHANICAL EQUIPMENT SELECTED FOR HARSH
ENVIRONMENT QUALIFICATION

PURCHASE ORDER	COMPONENT	I.D. NUMBER
M-001	Safety Relief Valves	B21-F013
M-001	Main Steam Isolation Valves	B21-F022/F028
M-001	Recirculation Pumps	B31-C001
M-001	Recirc. System Valves (Suction and Discharge)	B31-F023/F031
M-001	Hydraulic Control Units	C11-D001
M-001	CRD Vent Valves	C11-F010/F180
M-001	CRD Drain valves	C11-F011/F181
M-001	SLC Pumps	C41-C001
M-001	RHR Heat Exchanger Relief Valves	E11-B001
M-001	RHR Pumps	E11-C002
M-001	RHR Check Valves	E11-F041/F050
M-001	LPCS Check Valves	E21-F006
M-001	LPCS Pump	E21-C001
M-001	HPCI Pump	E41-C001
M-001	RCIC Pump	E51-C001
M-001	Neutron Monitoring System Valve Ass'y	C51-J004
M-001	RCIC Turbine Assembly	E51-C002
P-301(Q)	Valves	
P-302(Q)	Valves	
P-303A(Q)	Valves	
P-305(Q)	Butterfly Valves	
P-366(Q)	Check Valves	
P-401D	Snubbers	
M-070(Q)	SACS Pumps	
M-082(Q)	Fuel Pool Pumps	
M-141	Relief Valves	
M-150(Q)	Vacuum Relief Valves	
M-713(Q)	Centrifugal Fans	
J-601(Q)	Control Valves	
J-605(Q)	Valves	
J-703(Q)	Excess Flow Check Valves	
J-705(Q)	Instrument Valves	
J-715(Q)	Instrument Valves	

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ATTACHMENT III

QUESTION 220.15 (SECTION 3.8.4)

Provide sketches of the mathematical models used in the design of spent fuel racks. Describe in detail, the methods of analysis by which seismic and other loads are applied to the racks and the pool.

RESPONSE

~~The requested information will be available by June, 1984, and will be added to Section 3.8.4 and/or 9.1.2 as appropriate.~~

Sections ~~3.8.4.4.1~~ 3.8.4.8.3 and 9.1.2.2.2.2 have been revised and Appendix 9B has been added to provide the requested information.

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.

INSERT C

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:

a. 10CFR50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants".

b. ANSI/ASME N45.2, "Quality Assurance Program Requirements for Nuclear Facilities", and

c. ANSI/ASME NQA-1, "Quality Assurance Program Requirements for Nuclear Power Plants".

3.8.5 FOUNDATIONS

Foundations for all Seismic Category I structures and the turbine building and the administration facility, which are non-Seismic Category I structures, are described in this section.

3.8.5.1 Description of the Foundations

The configuration of the foundation mats for the various structures is shown on Figure 3.8-37.

Reinforced concrete mat foundations are provided for all structures. Except for the station service water system (SSWS) intake structure, the mats rest either on the Vincentown Formation or on engineered structural backfill placed on the Vincentown Formation. The mat and the lean concrete leveling

Insert C

The design, analysis and fabrication of the spent fuel racks conforms with the applicable provisions of Subsection NF. See Appendix 9B for a description of the design, analysis and construction of the racks.

9.1.2.2.2.2 High Density Spent Fuel Storage Racks

High density spent fuel storage racks in the fuel pool store spent fuel transferred from the reactor vessel. These are top-entry racks.

The spent fuel storage racks are of freestanding design and are not attached to either the fuel pool wall or the fuel pool liner plate. The racks are constructed of stainless steel, and the neutron absorber is Boral. See Figure 9.1-3 for design details of a typical rack and the special rack.

See Appendix 9B for a description of the design, analysis and construction of the spent fuel storage racks

APPENDIX 9B

DESIGN, ANALYSIS AND CONSTRUCTION
OF ~~HIGH DENSITY~~ SPENT FUEL STORAGE RACKS

9B.1 SCOPE

This appendix describes the design, analysis and construction of the spent fuel racks.

9B.2 DESCRIPTION OF SPENT FUEL POOL AND RACKS

Section 9.1.2.2 contains a description of the spent fuel storage facility including the high density spent fuel storage racks. The spent fuel racks are of free standing design and are not attached to either the fuel pool wall or the fuel pool liner plate. Figures 1.2-10 and 1.2-32 show the spent fuel pool in relation to other plant structures. Figures 9.1.3 and 9.1.4 show details of the spent fuel racks.

The spent fuel racks are designed to withstand the postulated drop of a fuel bundle. Section 9.1.5 contains a description of the overhead heavy load handling systems for the reactor building polar crane including figures showing load paths for the crane.

9B.3 APPLICABLE CODES, STANDARDS AND SPECIFICATIONS

All parts of the spent fuel racks, except the adjusting screws in the feet of each module and the poison material, are made from ASTM A240, Type 304L stainless steel. The adjusting screws are made from ASTM A564, Type 630 stainless steel. Boral is the poison material.

Design, fabrication and installation of the spent fuel racks are performed based upon Subsection NF requirements of Reference 9B-1 for Class 3 component supports.

9B.4 SEISMIC AND IMPACT LOADS

The seismic input for the spent fuel racks consists of floor response spectra for the spent fuel pool slab. Floor response spectra are developed from ground response spectra which comply with the requirements of Regulatory Guides 1.60 and 1.61. Acceleration time histories are developed for two horizontal directions and one vertical direction from the floor response spectra. These three time histories are imposed simultaneously. The peak responses from each direction are combined by square root of the sum of the squares in accordance with Regulatory Guide 1.92.

Impact loads due to fuel rattling are calculated using methods described in Section 9B.6. Impact loads are considered for local as well as overall effects on the rack design.

9B.5 LOADS AND LOAD COMBINATIONS

Loads and load combinations are in agreement with Table 1 of Reference 9B-2. Thermal effects are included by using decreased material properties at the applicable temperature level. Since the racks are free standing, there are no thermal stresses.

(typist - change "3D" to "three dimensional" in all places)

9B.6 DESIGN AND ANALYSIS PROCEDURES

Each fuel rack is idealized as a 3D finite element model using the ANSYS computer program. Figure 9B-1 shows a five canister portion of a rack. The canisters and bottom grid plate are modeled with plate elements. The perimeter bar, which secures the canisters at the top, and the stiffening bars for the grid plate are modeled with beam elements. The thin stainless steel wrapper containing the neutron absorber and the stainless steel panels used to close off the alternate cavities are not modeled but their masses are included. The fuel assemblies are modeled as beam elements.

Figure 9B-2 shows a double rack model in schematic form. 3D interface elements are used to represent the fuel-to-canister clearance as well as the rack-to-rack gap. These nonlinear elements reproduce forces due to fuel rattling and possible rack-to-rack interaction. 3D gap elements with material properties based on the interface friction coefficients are used to simulate the corner supporting feet which may slide or lift off the pool floor. Two bounding values of friction coefficient (0.2 and 0.8) are used in order to identify the mos critical conditions for sliding and for maximum reactions at the support feet.

Structural damping coefficients of 2 per cent for OBE and 4 per cent for SSE are used, except that impact damping of 10 per cent of critical is used for the gap elements since impact dissipates substantial amounts of energy. With 20 feet of submergence, sloshing effects are negligible and therefore are neglected. Fluid damping effects are also neglected. To simulate the immersion effects, all the internal water entrapped within the rack envelope is added to the horizontal mass. The external water between adjacent racks is modeled using the hydrodynamic coupling element shown in Figure 9A-2.

A parametric study, which considers varying amounts of fuel in a single rack, is conducted to determine which of the following conditions should be considered in order to maximize the seismic response of the racks.

- rack empty
- rack one-third full

- rack two-thirds full
- rack full

For the partially loaded conditions, eccentricity of the fuel on one side of the rack is considered.

9A.7 STRUCTURAL ACCEPTANCE CRITERIA

Allowable stresses are in agreement with Table 1 of Reference 9A-2. Stress levels for beam elements comply with the requirements of Appendix XVII to Reference 9A-1. Stress levels for plate elements comply with rules for plate and shell type supports since stress fields in these components are biaxial.

For the load drop condition, local permanent deformation possibly requiring repair is permissible provided that overall stresses do not exceed values permitted for Level D service limits and the resulting deformation does not permit the fuel configuration K_{eff} to exceed 0.95.

9B.8 MATERIALS, QUALITY CONTROL AND SPECIAL CONSTRUCTION TECHNIQUES

Materials are described in Section 9B.3. Quality control procedures for materials, fabrication and design control and verification comply with ANSI N45.2. Conventional construction methods are used.

As described in Section 9.1.2.2.2.2, approximately 25 per cent of the total spent fuel storage capacity will be provided by racks installed prior to initial plant operation. The remaining racks will be installed later. The initially installed racks are generally located at the north end of the spent fuel pool. Therefore, the additional racks can be installed later without being transported over existing racks which contain spent fuel.

9B.9 REFERENCES

9B-1 ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1980 Edition, Summer 1982 Addenda.

9B-2 NRC NUREG-0800, SRP Section 3.8.4, Appendix D, Rev. 0, July 1981.

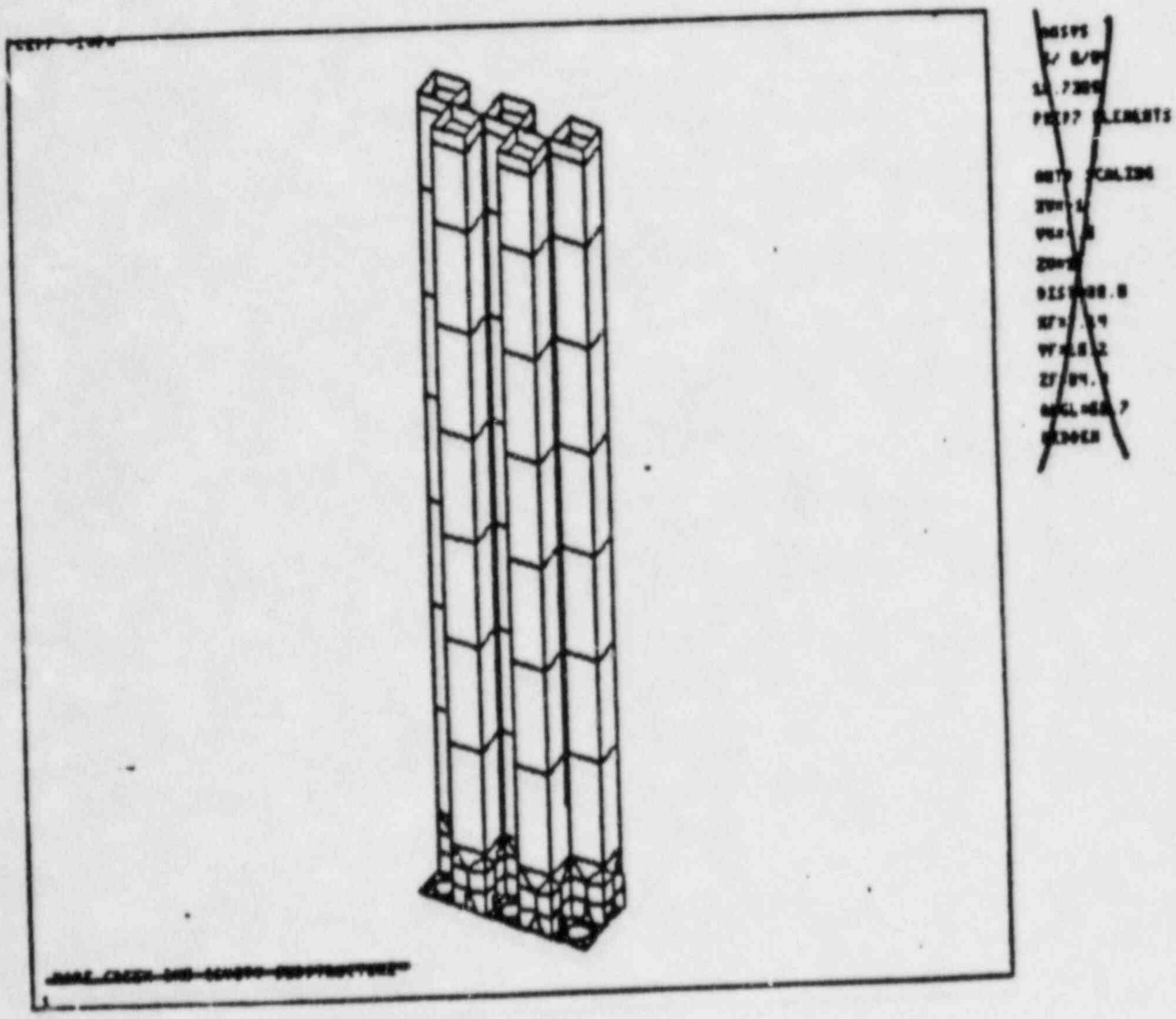

Figure 9B-1 Model for canister and portion of bottom grid

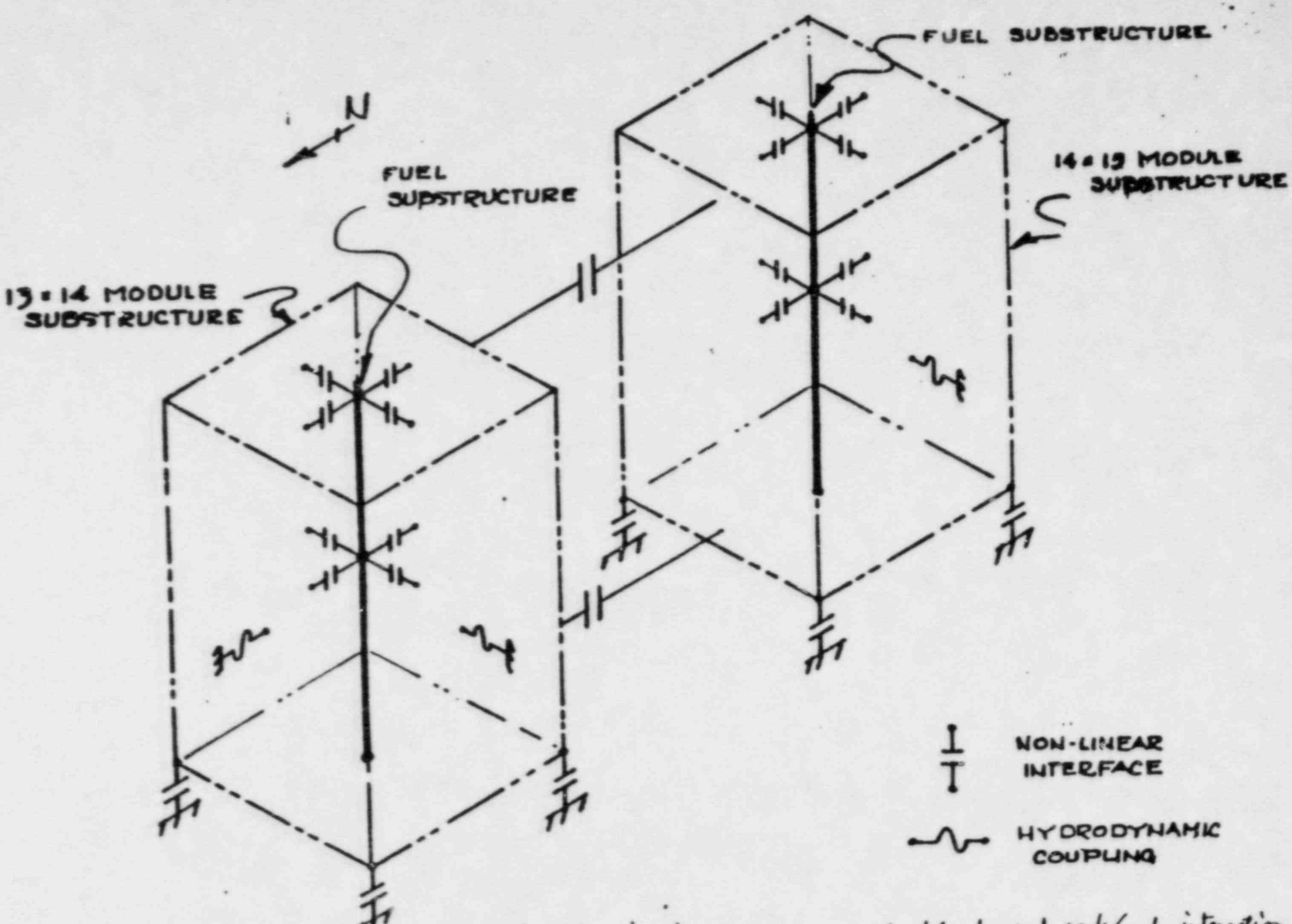

Figure 9B-2 Double rack model for detail seismic analyses on fuel/rack and rack/rack interaction

ATTACHMENT IV

QUESTION 270.2 (SECTION 3.11)

Prior to the completion of our review of your license application, it is necessary that we establish that you comply with the Commission's requirements applicable to environmental qualification contained in 10 CFR 50.49 for electrical equipment important to safety; GDC 4, Appendix A, 10 CFR 50; and Appendix B, 10 CFR 50, Sections III, XI, XVII.

As a result of the issuance of Section 50.49 of 10 CFR Part 50, some of the information requested in SRP3.11 and R.G. 1.70, Section 3.11, is no longer required for staff review. Other new information is required, however, and is defined in this guidance. By utilizing these guidelines to demonstrate compliance with the Commission's regulations, applicants can significantly reduce the need for requests for additional information from the NRC staff. The information required may be submitted in Section 3.11 of the FSAR or in a separate submittal. If the latter approach is chosen, Section 3.11 should reference the information in the environmental qualification program submittal.

RESPONSE

The HCGS Environmental Qualification Summary Report, Rev. 0, submitted via letter from R.L. Mittl, PSE&G, to A. Schwencer, NRC, dated August 24, 1984 ~~[illegible]~~ describes in detail the HCGS Environmental Qualification Program for electrical equipment. It is the intent of PSE&G to answer NRC questions and to amend~~ing~~ the FSAR to clarify any given position prior to the submittal of ~~this~~ the final version of this report, ~~[illegible]~~ which will contain the information ~~[illegible]~~ specified in NUREG-0588 as modified by 10 CFR 50.49. ~~[illegible]~~

ATTACHMENT V

QUESTION 410.38 (SECTION 9.1.2)

Insufficient information is provided for review of the criticality of the spent fuel pool. The design bases are acceptable with respect to criticality. The information required for the review is promised for later. Such information should include the following:

a. Sufficient structural detail to permit an independent calculation of the criticality of the racks.

b. A description of the calculational methods used along with the results of the verification of the methods. This may be by reference to documents previously submitted by the organizations doing the analysis.

c. A tabulation of the nominal value of k effective of the racks along with the various uncertainties and biases considered in the analysis.

d. A tabulation of the reactivity effect of each of the abnormal (accident) situations considered.

RESPONSE

~~Sufficient information for review of the criticality of the spent fuel pool, including that listed above will be available by September 1984, and will be added to Section 9.1.2.~~

Section 9.1.2.3.3 has been revised to include the information requested above.

i. The maximum stress in the fully loaded rack in a faulted condition will be provided prior to fuel load.

j. The spent fuel storage racks also have the capability of storing control rod guide tubes, control rods, and defective fuel containers. When the spent fuel is stored in the spaces provided for storing the above the K_{eff} does not exceed 0.95.

k. Several design features reduce the possibility of heavy objects dropping into the fuel pool. The main and auxiliary hoists of the reactor building polar crane are single-failure proof. In addition, the main hoist is physically prevented from traveling in the truncated segment shown on Figure 9.1-31 by mechanical stops on the girders of the polar crane. The crane design is discussed in Section 9.1.5. The removable guardrail and the four-inch curb around the refueling cavities further limit the possibility of heavy objects dropping into the fuel pool.

l. The fuel storage pool has water shielding for the stored spent fuel. Liquid level sensors are installed to detect a low pool water level. Makeup water is available to ensure that the fuel will not be uncovered should a leak occur.

m. Since the fuel racks are made of noncombustible material and are stored underwater, there is no potential fire hazard. The large water volume also protects the spent fuel storage racks from potential pipe breaks and associated jet impingement loads.

9.1.2.3.3 INSERT A

9.1.2.4 Spent Fuel Rack Inservice Inspection

~~An inservice inspection program is in effect throughout the life of the racks to ensure that the quality of the poisoned racks is~~

Insert B .

INSERT A page 1 of 16

9.1.2.3.3 CRITICALITY ANALYSIS AND RESULTS

The criticality analysis was performed using the input parameters contained in Table 9.1-19. Figure 9.1-20 shows the reference geometry used in the criticality analysis, the zero flux boundary and the postulated dropped fuel assembly.

The criticality analysis is based on new fuel with a nominal, flat U-235 enrichment of 3.4 w/o. No credit is taken for the burnable poison fuel rods which may be present in the fuel assemblies. The analysis simulates an infinite array of fuel of infinite length, AND ~~The analysis~~ uses Utility Associates International's (UAI's) diffusion theory model, CHEETAH-B/CORC-BLADE/PDQ7 as the main working model. The analysis includes the various criticality safety-related aspects of the rack design, including various sensitivity calculations. The Monte Carlo transport model, AMPX/KENO -IV, is used as the verification model to verify the reactivity of the nominal rack design.

UAI performed similar criticality analyses for Limerick and Susquehanna.

The analysis includes all the normal, abnormal, and accident conditions described in Section 9.1.2.3.1.

Table 9.1-20 summarizes the nominal value of K effective of the racks under normal, abnormal, and accident conditions. The various uncertainties and biases considered in the analysis are also included.

~~3.0~~ CALCULATIONAL MODELS

This section presents a description of the calculational models and the basic assumptions used in this criticality analysis.

~~3.1~~ The Working Model

The criticality analysis for the Hope Creek BWR spent fuel racks employs the CHEETAH-B/CORC-BLADE/PDQ-7 model as the basic engineering tool. CHEETAH-B~~(1)~~ is UAI's BWR lattice code based on the original LEOPARD code and uses a modified ENDF/B-II cross section library. CORC-BLADE~~(3)~~ generates equivalent diffusion theory cross sections for the control blade. The PDQ-7~~(4)~~ program is the well-known few-group spatial diffusion theory code widely used by the industry. The CHEETAH-B/CORC-BLADE/PDQ-7 model, which is also a part of the LEAHS (Lifetime Evaluation and Analysis of Heterogeneous Systems) nuclear analysis series of Control Data Corporation, has been extensively tested through benchmarking calculations of measured criticals as well as through core physics calculations for several operating power reactors.

A zero current boundary condition was applied to the four sides of the unit reference storage rack cavity ~~(Figure 2)~~ to produce an infinite array effect. The two-dimensional, PDQ-7 calculations were made for four neutron energy groups, two mesh intervals per fuel pin, a flat U-235 enrichment description and a zero axial buckling to simulate infinite fuel length.

~~3.2~~ The Verification Model

The verification calculation employs the KENO-IV~~(5)~~/AMPX~~(6)~~ model. The basic neutron cross section data comes from the master libraries of AMPX - a 123 group GAM-THERMOS neutron library prepared from ENDF/B version II data. The NITAWL module of the AMPX program is used to perform a Nordheim integral treatment of the U-238 resonances accounting for the self-shielding effect. The working library produced by the NITAWL/AMPX module retains the 123 group energy structure and is used directly by KENO-IV.

In the KENO-IV calculation, the spent fuel rack geometry including each fuel and water rod cell is represented discretely. To simulate the arrangement of a large number of storage rack units, and for a non-leakage condition in the axial directions, a specular reflective condition is applied to all six sides of the reference case storage rack cavity ~~(Figure 2)~~.

~~3.0~~ Basic Assumptions

To ensure that the analysis follows a conservative approach and conforms to the general guidelines of criticality safety analysis in Reference ~~4~~ 9.1-10, the calculations are performed with the following assumptions:

1. A flat 3.4 w/o distribution in an 8x8 bundle, with U-234 neglected
2. Fresh fuel, no burnable poison
3. Minor structural members replaced by water, i.e., spacer grids
4. Fresh water
5. Fuel is channeled.

Insert A page 4 of 16

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~~UNI 84-423~~
~~Revision 0~~
~~June 8, 1984~~

9.1.2.3.3.1 ~~4.0~~ REFERENCE CASE CALCULATIONS

Physical Parameters and the Basic Storage Rack Cavity Geometry

The reference storage rack cavity ~~(Figure 2)~~ has a pitch of 6.308" $\pm$ 0.030". The stainless steel canister has a nominal inside clearance of 6.080 to accommodate 8x8 fuel assembly channeled in 0.080" thick Zircaloy-4. Plates of the neutron absorber material Boral, consisting of B_4C in an aluminum matrix core and clad with an aluminum sheath, are fastened to the outside of the canister. The Boral plate has a nominal total thickness of 95 mils and a minimum B-10 density of 0.028 g/cm^2. Table ~~1~~ 9.1-19 contains the values of the input parameters used in the analysis.

The rack must accommodate both channeled and unchanneled fuel. Studies reveal that the channeled fuel in the rack is more reactive than the unchanneled fuel. Taking the conservative approach, the study here involves channeled fuel (except in the accident condition where the dropped fuel is unchanneled in order to permit the closest contact between the dropped fuel assembly and the rack).

Two small, but non-conservative changes were made to the reference case in order to facilitate modeling. First, the boral width was set at 4.48" instead of 4.465". Second, the stainless steel flanges used in welding the outer wrapper to the inner can were deleted. An adjustment was made using PDQ to account for these differences.

Results of the Reference Case Calculations

Using the input data from Table ~~1~~ 9.1-19 and Figure ~~2~~ 9.1-30 (except as noted above), the K_{eff} values of the reference case at 68°F were calculated for the calculational model described ~~in Section 3.0.~~ previously.
The results are:

HCGS FSAR

	PDQ-7	KENO-IV
k_{eff}, reference calculation	0.9229	0.9306 ± 0.0042
95% confidence interval	______	0.9222 - 0.9390

Insert A page 6 of 16

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9.1.2.3.3.2

[redacted] SENSITIVITY AND TOLERANCE REACTIVITY CALCULATIONS

[redacted] Temperature Effect

Using the reference storage rack cavity geometry, the temperature of the fuel and pool water was varied. In addition to the nominal 68°F, 40°F and 212°F were studied and the results of the CHEETAH-B/CORC-BLADE/PDQ-7 runs are given on Table 9.1-20 [redacted] As shown, reactivity decreases continuously as temperature increases from 40°F.

[redacted] Void Effect

The effect of boiling (assuming equal voids inside and outside of the rack) was studied by varying the voids from 0% to 20% at a temperature of 212°F with the reference geometry. The CHEETAH-B/CORC-BLADE/PDQ-7 results are shown in [redacted] Table 9.1-20 [redacted]. As indicated, k_{eff} decreases continuously as the void fraction increases.

[redacted] Pitch Sensitivity

The rack design permits the storage cavity pitch to differ from the 6.308" nominal value by ±0.030". The pitch sensitivity calculations of this analysis show the reactivity effect of these tolerance components as well as the reactivity pitch sensitivity by expanding the calculational range from -0.060" to +.030" at .030" intervals. The results, which are [redacted] tabulated in Table 9.1-20 [redacted], indicate that in the neighborhood of the nominal pitch, the pitch reactivity coefficient is about .15%Δk per .030" pitch change.

HCGS FSAR

Effect of Boron

The Boral Plates which separate two adjacent fuel assemblies have a nominal thickness of .095" (consisting of an 73 mil core and 11 mil aluminum sheaths) a nominal width of 4.465" and an overall length of 11 feet 3 inches. The minimum B-10 loading is 0.028 g/cm^2.

(a) Boron Width Tolerance

The effect of reducing the Boral width was examined. The PDQ-7 calculation for the reference case configuration with the Boral width reduced by 0.0625" yielded k_∞ = 0.92641. Hence, the reactivity increases due to the -0.0625" tolerance on Boral width is Δk = +0.0029.

(b) Boron Density

The boron density was maintained at .028 g/cm^2 for all calculations. This areal density is the minimum density allowed by manufacturing design specifications.

(c) Boral Core Thickness Variation

The sensitivity to the Boral core thickness was determined by calculations in which the thickness varied from 61 mils to 80 mils (the aluminum sheaths were varied within tolerance to obtain the worst case core thickness). The results, tabulated in Table 9.1-30 [redacted] show a continuous increase in reactivity as the core thickness increases. This is due to the fact that the areal density is held constant, so an increase in thickness reduces volumetric density and, to a small degree, the boral effectiveness.

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[redacted] Dimensional and Positional Tolerances

The total Δk bias for dimensional and positional tolerances are calculated from five separate contributions:

(i) Pitch Reduction
(ii) Boral Width Reduction
(iii) Inter-Cavity Spacing Reduction
(iv) Off-center Loading
(v) Boral Thickness Increase

(i) Pitch Reduction. The effect of reducing the center-to-center spacing of the rack cavities is obtained from [redacted] Table 9.1-20 ~~results of Section 5.3. From either Table 5 or Figure 5,~~ and is [redacted] $\Delta k_1 = 0.0015$.

(ii) Boral Width Reduction. The Δk bias due to reducing the Boral width by its tolerance, 0.0625" is obtained from [redacted] Table 9.1-20 and is $\Delta k_2 = 0.0029$.

(iii) Inter-Cavity Spacing Reduction. Any seismic effect that may reduce the separation distance between adjacent cavities can be determined from the pitch sensitivity study. [redacted] Bringing two adjacent cavities closer by 0.048" results in the canisters touching and a reactivity increase $\Delta k_1 = 0.0023$ (from Table 9.1-20 [redacted]). Since this reduction is the maximum reduction of pitch possible in this design, this effect will not be added to item (i), but will replace it.

(iv) Off-Center Loading. The free space existing between a properly center LOADED fuel assembly and the top casting allows an assembly to be loaded off-center in a cavity. It was

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shown that this condition causes no adverse reactivity effect since the resulting k_{eff} for off-centered loading is less than that for properly centered assemblies.

(v) Boral Thickness Increase. The worst case boral core thickness reactivity effect calculated due to manufacturing tolerance stackup (.080") is obtained from ~~Section 5.4(c)~~ Table 9.1-20 and is $\Delta k_5 = .0001$.

The above positive Δk contributions are statistically combined to give the total Δk bias for mechanical and seismic uncertainties.

$$\Delta k = \sqrt{(\Delta k_1)^2 + (\Delta k_2)^2 + (\Delta k_5)^2} = 0.0037$$

9.1.2.3.3 SPECIAL CASES

Grappler Drop Accident

The accident considered is the inadvertent drop of the assembly grappler used in lifting assemblies within the spent fuel pool. In this accident, the grappler is dropped in such a way that assemblies in adjacent rack cavities are displaced such that they are resting in an off-center loading arrangement. The reactivity effect for this off-center arrangement was discussed in Section 9.1.2.3.3.2(iv)

Assembly Drop Accident

(a) Single Assembly Dropped on Top of Rack. No adverse reactivity effect is expected from dropping a fuel assembly on top of a fully loaded storage rack during fuel handling because of the large water thickness (~14 inches) existing between the top of the assemblies already inside the cavities and the dropped assembly resting on top of the rack. Moreover the PDQ-7 model assumes an infinite fuel length in the axial direction.

(b) Single Assembly Next to Rack. The dropping of an assembly outside the rack is a possible event because of the unobstructed water area existing between the periphery of the storage racks and the side walls of the pool.

A conservative analysis to evaluate this situation is illustrated in Figure 9.1-20. An assembly, presumed to be

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dropped during handling, lodges paralled to an assembly in the outer cavity with no Boral slab separating the two assemblies. The dropped assembly is unchanneled to permit the closest contact with the rack. ~~and is 31 inches away from the edge of the pool.~~ The dimensions used are those of the reference case. This arrangement of the dropped fuel assembly with a 3 1/2 x 3 finite fuel rack is reflected on three sides as indicated in Figure 9.1-20; the fourth side is a zero flux boundary. The k_{eff} result for this case was 0.9128. The result for the same geometry without the dropped fuel was 0.9064 giving an increase of reactivity of $\Delta k = 0.0063$ for the above dropped assembly configuration. ~~This increase is included in the final k_{eff} tabulation (section 3.0).~~

~~6.~~ Assembly Moving Between Two Storage Racks

The rack structural design does not allow sufficient room to fit a fuel assembly between any two of the high density spent fuel racks. Therefore, the movement of assemblies between racks is precluded.

~~Multiple~~ Assemblies Placed Adjacent to the Rack

Special consideration was given to the accident condition of the placemnt of two or three bundles ~~12 inches from the rack edge, between the rack~~ and the pool wall. A comparison of cases ~~run~~, summarized below, show ~~little~~ or no interaction between the bundles outside the rack and the fuel in the rack.

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6.4.1 Three Bundles Arranged in a "T"

Figure 8 shows the PDQ-7 model used for the study. Table 5 includes the resulting K-eff for the T arrangement. For this case the reactivity is not greater than the rack design limit of .95.

6.4.2 Three Bundles Arranged in a Line

Figure 9 shows the PDQ-7 model used for the study. Table 5 includes the resulting K-eff for the linear arrangement. For this case the reactivity is not greater than the rack design limit of .95.

The K-eff on Table 5 is lower than that of the rack with no assemblies adjacent to it. The reason for this is that the rack is uncoupled from the bundles by the twelve inches of water separating them. In the case of the linear bundle arrangement, the bundles outside of the rack are less reactive than the rack itself. In the case of the "T" bundle arrangement, the bundles outside the rack are more reactive than the rack.

6.4.3 Two Bundles

Since two bundles adjacent to the rack is less reactive than three bundles adjacent to the rack, this case was not analyzed.

9.1.2.3.3.4 [redacted] New Fuel Storage in the Spent Fuel Racks

The feasibility of storage of fresh fuel in the high density spent fuel racks was analyzed. Storage of new fuel in the mist, partly flooded, and dry conditions are addressed below.

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25% Mist Condition

The storage of new fuel of uniform 3.4 w/o U-235 enrichment in the high density spent fuel rack in a 25% aqueous mist environment was analyzed with the KENO model ~~(refer to Figure 2)~~. The resulting k_∞ and 95% confidence interval are shown below:

	K_∞	95% Confidence Interval
25% Mist	.6375±.0054	.6267- .6483

Dry Condition

UAI experience in the analysis of poisoned rack criticality indicates that the fully flooded rack configuration is the most reactive with reacitivy decreasing with a decrease in moderator density. The 25% mist condition analysis confirms this as shown below. For this reason a dry condition analysis was not performed since it too will be less reactive than the flooded condition.

Moderator Density	K_∞	95% Confidence Interval
Reference Case: 1.00 g/cm^3	.9306±.0042	.9223-.9389
25% Mist Condition: 0.25 g/cm^3	.6375±.0054	.6267-.6483

Partly Flooded Condition

The totally flooded condition as analyzed in the reference case is more reactive than that of the partly flooded condition.

9.1.2.3.3.5 Special Spent Fuel Rack Storage

~~Included with the regular borated spent fuel racks as shown in Figure X~~, a 5x6 non-borated special rack is to be installed in the ~~Hope Creek~~ HCGS spent fuel pool. Storage of control rods, control rod guide tubes and defective fuel is provided for by this special rack. This rack was analyzed for storage of ruptured fuel as shown in Figure 9.1-19. Special rack input parameters are summarized in Table 9.1-21.

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The storage of ruptured fuel is a more reactive evaluation than that of control rods or control rod guide tubes.

[redacted] Storage of Ruptured Fuel in the Fully Flooded Special Rack

The storage of ruptured fuel assemblies within defective fuel storage containers inserted into the special ~~6 x 6 un-borated~~ rack was analyzed using the CHEETAH-B/PDQ-7 diffusion theory model. The case was analyzed as an infinite array in order to simulate storage of 30 ~~27 or more~~ ruptured fuel assemblies in the special rack. The resulting K_{eff} for this case was .6589. Considering that this K_{eff} accounts for no radial or axial leakage, the reactivity for the storage of fuel in the special rack is well below the design limit K_{eff} of .95.

Storage of undamaged fuel within the special rack is less reactive than storage of damaged fuel. This is due to the fact that in the ruptured fuel case, the defective fuel storage container displaces water. For this reason, the storage of undamaged fuel was not analyzed.

[redacted] Storage of Ruptured Fuel in the Drained Special Rack

The accident being considered here is the unlikely event that the spent fuel pool is drained while ruptured fuel remains stored in the special rack. Under these conditions, the rack would be drained of water and replaced with air. The defective fuel storage contained would, however, remain

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9.1.2.3.3.6 SUMMARY AND CONCLUSION

The final result as calculated by both the working model (CHEETAH-B/CORCBLADE/PDQ-7) and the verification model (AMPX/KENO-IV) is summarized in this section and compared to the NRC regulation k_{eff} limit of 0.950. The "Reference Case" referred to in this report uses the nominal dimensions given in Figure 9.1-20 and Table 9.1-19 without the dimensional and material tolerances included.

Results of the Transport Monte Carlo (AMPX/KENO-IV) Verification Calculations and the Calculational Bias

k_{eff}, Reference Case 9.1.2.3.3.1 [redacted]	0.9306 ± 0.0042
Benchmark bias, Δk	-0.001
	.9296 ± 0.0042
95% Confidence Interval k_{eff}	0.9212 - 0.9380

The bias of the KENO-IV vs. measurement is based on criticality experiments performed with fixed neutron poisons [redacted]. These experiments were chosen because they approach the fuel storage rack configuration in that they used fixed poison plates between fuel rod clusters. The result of the benchmark calculations was that the KENO-IV results were 0.001Δk above the measured value. This demonstrates a negative bias of 0.001Δk.

Summary of Results

k_{eff}, adjusted (KENO [redacted])	0.9296 ± 0.0042
Dimensional and Positional Tolerance, Δk (PDQ [redacted])	0.0037
PDQ correction for non-conservative assumptions in the reference case, Δk (PDQ [redacted])	0.0006
Dropped Assembly, Δk (PDQ [redacted])	0.0063

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Final K_{eff}	0.9402 ± 0.0042
95% Confidence Interval	0.9318 - 0.9486
Design Limit, k_{eff}	~~0.9318 - 0.9486~~ 0.950

The final k_{eff} value (0.9486) includes all the design specification tolerances, the postulation of a dropped fuel assembly, the model bias, and the 95% confidence interval from the KENO calculations. However, the negative reactivity effect (~ 0.5% Δk) due to the presence of U-234 and the parasitic structure materials (i.e., spacer grids) in each assembly was not included.

ATTACHMENT VI

(5) Equipment Qualifications: Seismic and Environmental - Seismic and Dynamic Qualification and Qualification of Mechanical Equipment reviews consist of two elements: a review of the FSAR, and a detailed onsite audit. The information required for the review is included as Enclosure 9 and its attachments. The information will not be required until the first calendar quarter of 1984.

RESPONSE

At least ~~It is expected that~~ 85 ~~to 90~~ percent of the safety related equipment will be qualified (seismic and dynamic qualification) and installed by December ~~the third quarter of~~ 1984. An updated ~~A~~ status summary list has been provided (letter from R.L. Mittl, PSE&G to A. Schwencer, NRC, dated October 19 ~~July 5~~, 1984) for all safety related equipment for the SQRT and PVORT audits in anticipation of the NRC audits being conducted in the month of January 1985. ~~the third quarter of 1984. A partial status summary list has been submitted under separate cover (letter from R. L. Mittl, PSE&G to A. Schwencer, NRC, dated April 13, 1984).~~

The format of the status summary list follows the sample provided in enclosure 9 of the NRC acceptance review letter dated June 23, 1983, with the following modification. The list ~~will be~~ is assembled on a purchase order basis, with individual tag number identified. The items on the list ~~will~~ have system designators identified.

SQRT and PVORT forms for all safety related equipment will be available for NRC review during the audit. The PVORT forms will contain information which is currently available per the purchase specifications for the subject components.

Information on seismic and dynamic qualification test programs for non-NSSS supplied components has been submitted under separate cover (letter from R.L.Mittl, PSE&G, to M.A. Schwencer, NRC, dated October 5, 1983). The seismic and dynamic qualification test program for NSSS supplied components has been completed.

Seismic and environmental qualification of the containment vent and purge valves is discussed in Section 6.2.5.2.2.

However, additional seismic testing is scheduled to be done under the NSSS environmental qualification program. This schedule has been submitted under separate cover.