PSE&G Public Service Electric and Gas Company

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Robert L. Mittl General Manager
Nuclear Assurance and Regulation

August 13, 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 JULY 1984

Public Service Electric and Gas Company presently plans to issue Amendment No. 7 to the Hope Creek Generating Station Final Safety Analysis Report in August 1984. This letter is provided, along with three (3) signed originals of the required affidavit, to document the status of Hope Creek Generating Station responses to NRC requests for additional information which were forecasted to be responded to by June and July 1984.

Attachment I is a tabulation of the Hope Creek Generating Station Final Safety Analysis Report commitments for June 1984, and the corresponding resolution for each commitment. Attachments II through IX provide the responses to the questions forecasted to be responded to in June 1984, which will be included in Amendment No. 7.

Attachment X is a tabulation of the Hope Creek Generating Station Final Safety Analysis Report commitments for July 1984, and the corresponding resolution for each commitment. Attachments XI and XII provide responses to questions forecasted to be responded to in July 1984, which will be included in Amendment No. 8.

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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 June 1984
Attachment II - Response to Question 210.12
Attachment III - Response to Question 220.15
Attachment IV - Response to Question 281.2
Attachment V - Response to Question 410.38
Attachment VI - Response to Question 410.39
Attachment VII - Response to Question 410.42
Attachment VIII - Response to Question 421.13
Attachment IX - Response to Question 630.7e&f
Attachment X - Hope Creek Generating Station - FSAR Commitment Status through July 1984
Attachment XI - Response to Question 210.12
Attachment XII - Response to Question 430.19

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

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

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
DOCKET NO. 50-354

PUBLIC SERVICE ELECTRIC AND GAS COMPANY

Public Service Electric and Gas Company hereby submits the enclosed Hope Creek Generating Station Final Safety Analysis Report (FSAR) question responses.

The matters set forth in this submittal are true to the best of my knowledge, information, and belief.

Respectfully submitted,

Public Service Electric and Gas Company

By: Thomas J. Martin
Thomas J. Martin
Vice President -
Engineering and Construction

Sworn to and subscribed before me, a Notary Public of New Jersey, this 13th day of August 1984.

David K. Burd

DAVID K. BURD
NOTARY PUBLIC OF NEW JERSEY
My Comm. Expires 10-23-85

NM 19 1i

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

	FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
1.	Question/Response Appendix: Question 100.6	Re: TMI Item I.C.5: This commitment concerns assuring feedback of operating experience to operational personnel via procedures. This information will be provided in August 1984. Re: TMI Item II.B.3: This commitment concerns assuring compliance of the radioactive gas and liquid sampling system for shielding and source term requirements. This information will be provided in August 1984 and September 1984.
2.	Question/Response Appendix: Question 210.12	This commitment concerns compliance with conditions in conditionally approved Code Cases identified in Reg. Guides 1.84 and 1.85. This information will be submitted in three parts to be provided in July 1984, October 1984, and November 1984. These revised commitment dates are provided in Attachment II and will be included in Amendment 7 to the HCGS FSAR.
3.	Question/Response Appendix: Question 220.10	This commitment concerns detailed procedures for seismic instrumentation inservice surveillance program. This information will be provided in April 1985. This revised commitment date will be included in Amendment 8 to the HCGS FSAR.

FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
4. Question/Response Appendix: Question 220.15	This commitment concerns mathematical models used in the design of Spent Fuel Racks. This information will be provided in September 1984. This revised commitment date is provided in Attachment III and will be included in Amendment 7 to the HCGS FSAR.
5. Question/Response Appendix: Question 220.16	This commitment concerns the Spent Fuel Racks and their conformance with subsection NF of the ASME Code. This information is provided in Amendment 6 to the HCGS FSAR.
6. Question/Response Appendix: Question 260.15	This commitment concerns revising FSAR Section 1.8 to reflect conformance with listed Reg. Guides which are applicable during operations phase. This information will be provided in August 1984.
7. Question/Response Appendix: Question 281.2	This commitment concerns providing FSAR Figures 9.1-3 and 9.1-4 which illustrate Spent Fuel Rack design and arrangement. This information is provided in Attachment IV and will be included in Amendment 7 to the HCGS FSAR.
8. Question/Response Appendix: Question 281.9	This commitment concerns limits for dissolved and suspended solids in purified condensate. This information will be provided in December 1984. This revised commitment date will be included in Amendment 8 to the HCGS FSAR.

	FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
9.	Question/Response Appendix: Question 281.11	This commitment concerns chemistry sampling for the Spent Fuel Pool Cleanup System. This information will be provided in September 1984. This revised commitment date will be included in Amendment 8 to the HCGS FSAR.
10.	Question/Response Appendix: Question 281.14	This commitment concerns the materials monitoring program for the Spent Fuel Pool. This information will be provided in August 1984.
11.	Question/Response Appendix: Question 281.15	This commitment concerns information on the Post Accident Sampling System which demonstrates compliance with NUREG-0737, Item II.B.3.This information will be provided in August 1984 and September 1984.
12.	Question/Response Appendix: Question 410.26	This commitment concerns information requested in G.L. 81-34 regarding BWR Scram System Piping. This information will be provided within 60 days of NRC acceptance of the BWROG position. This revised commitment date will be included in Amendment 8 to the HCGS FSAR.
13.	Question/Response Appendix: Question 410.38	This commitment concerns the criticality of the Spent Fuel Pool. This information will be provided in September 1984. This revised commitment date is provided in Attachment V and will be included in Amendment 7 of the HCGS FSAR.

FSAR COMMITMENT LOCATION		COMMITMENT RESOLUTION
14.	Question/Response Appendix: Question 410.39	This commitment concerns Spent Fuel Rack design details. This information is provided in Attachment VI and will be included in Amendment 7 of the HCGS FSAR.
15.	Question/Response Appendix: Question 410.42	This commitment concerns the highest anticipated assembly average enrichment of U235 used in the Spent Fuel Rack criticality analysis. This information is provided in Attachment VII and will be included in Amendment 7 of the HCGS FSAR.
16.	Question/Response Appendix: Question 410.91	This commitment concerns the ability of check valves in the Equipment and Floor Drain System to maintain a functional pressure boundary. This information will be provided in August 1984.
17.	Question/Response Appendix: Question 410.93	This commitment concerns seismic qualifications of check valves in drainage systems. This information will be provided in August 1984.
18.	Question/Response Appendix: Question 421.13	This commitment concerns testing of I&C isolation systems against the effects of EMI per IEEE 472-1974 and PMC 33.1-1978. This testing has been completed and this information is provided in Attachment VIII and will be included in Amendment 7 of the HCGS FSAR.

FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
19. Question/Response Appendix: Question 421.26	This commitment concerns reactor mode switch contact misoperations. This information will be provided in August 1984.
20. Question/Response Appendix: Question 440.10	This commitment concerns trip settings for the plant leak detection system. This information will be provided in August 1984.
21. Question/Response Appendix: Question 460.16	This commitment concerns the implementation of acceptance criteria for the licensed solid waste burial facility. This information will be provided in March 1985. This revised commitment date will be included in Amendment 8 to the HCGS FSAR.
22. Question/Response Appendix: Question 471.14	This commitment concerns providing the resume for Senior Radiation Protection Supervisor. This information will be provided in August 1984.
23. Question/Response Appendix: Question 630.7e&f	This commitment concerns segments of the plant training program. This information is provided in Attachment IX and will be included in Amendment 7 of the HCGS FSAR.
24. Supplementary Request for Additional Information (5)	This commitment concerns providing a master listing of seismic and dynamic qualification summary and status of safety related equipment. This information has been submitted (Refer to: R. L. Mittl (PSE&G) to A. Schwencer (NRC), dated July 5, 1984).

FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
25. FSAR Table 13.1-4	This commitment concerns providing resumes for Maintenance Manager and Senior Nuclear Maintenance Supervisor. This information will be provided in August 1984.

QUESTION 210.12 (SECTION 5.2)

Table 5.2-2 identifies certain ASME Code Cases that have been used in the construction of components for the Hope Creek Generating Station. A number of these Code Cases are identified in Regulatory Guides 1.84 and 1.85 as conditionally acceptable. That is, Regulatory Position C.1 of each guide identifies additional conditions that should be imposed in addition to those conditions specified in each Code Case. The Code Cases that are identified in Regulatory Guide 1.84 as conditionally acceptable are: 1636/1636-1, 1711, 1734, 1818, N-192, and N-275. The Code Cases that are identified in Regulatory Guide 1.85 as conditionally acceptable are: 1644/1644-3, 4, 6, N-71-9, and N-249. Code Case N-253-1 which is not in the Regulatory Guides is also conditionally acceptable.

Demonstrate that you are in compliance with the additional conditions applicable to each of the above conditionally approved Code Cases that are identified in Regulatory Guides 1.84 and 1.85.

RESPONSE

Code Case 1636-1 (N-70) was invoked in the design of the safety auxiliaries cooling system (SACS) pumps. Regulatory Guide 1.84 states that the design guidance in this code case is acceptable subject to the restriction that the stress limit designations of "Upset," "Emergency," and "Faulted" should be established and justified in the design specification.

HCGS complies with this additional regulatory requirement. The loading combinations are specified in the design specification.

Code Case 1711 (N-100) was invoked in the design of a number of safety-related safety-relief valves. Regulatory Guide 1.84 states the design guidance in this code case is acceptable subject to the requirement that the FSAR demonstrate how the pressure relief function is assured if the stress limits utilized for the upset operating condition are in excess of those specified in the code case.

— Insert A —

~~The applicability of the additional regulatory requirements is being evaluated. A response will be provided in June 1984.~~

Code Cases 1734 and 1818 were invoked in the fabrication of certain pipe supports. Regulatory Guide 1.84 states that the design guidance in these code cases is acceptable subject to the additional welding restrictions found in the regulatory guide.

HCGS complies with these additional regulatory requirements. The applicable design specifications permit the use of Code Case 1734

and/or 1818 subject to the limitations recommended by Regulatory Guide 1.84.

Code Case N-192 was invoked in the fabrication of certain flexible metal instrument hose assemblies and on certain standby diesel generator skid-to-facility connectors. Regulatory Guide 1.84 states that this code case is acceptable subject to the requirement that the applicant should provide design data to demonstrate compliance with Paragraph NC/ND-3649.

July Information to comply with this additional regulatory requirement will be provided in ~~June~~ 1984, under separate cover. This information is considered to be proprietary.

Code Case N-275 was invoked in the fabrication of certain safety-related pipe. Regulatory Guide 1.84 states that the design guidance in this code case is acceptable subject to the additional welding restrictions in the regulatory guide.

HCGS complies with these additional regulatory requirements. The HCGS piping design specification permits the use of Code Case N-275 subject to the limitations recommended by Regulatory Guide 1.84.

Code Case 1644 and its various revisions has been invoked in numerous applications. Regulatory Guide 1.85 states that this code case is acceptable subject to the limitations on maximum ultimate tensile strength and, in the case of Code Case 1644-9 (N-71-9), the additional requirements for electrode dispersal.

— Insert B —

~~HCGS is currently evaluating the applicability of the additional maximum ultimate strength limitation in view of the concerns with material brittleness and stress corrosion cracking. A response will be provided in June 1984.~~

Use of Code Case 1644-9 (N-71-9) is subject to the additional precautions cited in Regulatory Guide 1.85.

Use of Code Case N-249 is permitted for the containment hydrogen recombiner technical specification. To date, this code case has not been invoked.

Code Case N-253-1 provides rules for the construction of ASME components which experience elevated temperatures. This code case was invoked in the design of the containment hydrogen recombiners. This code case was invoked on HCGS because there are portions of the containment hydrogen recombiners that operate at temperatures in excess of 800°F.

- Insert A

210.12

~~We have completed the reviewed~~ all of the safety-relief valves at HCGS that use Code Case 1711 (N-100), with the exception of the 10 valves found on the primary containment instrument gas compressor skids and the 2 on the instrument gas receiver tanks have been reviewed. We have determined that all the valves were specified as active. This designation requires that the stress limits for the faulted condition meet normal loading condition allowables. Thus, the additional requirements in Regulatory Guide 1.84 are not applicable.

 - Insert A -

A We are still evaluating the applicability of the additional requirements to the remaining valves. A revised response will be provided in October 1984.

- insert B -

Most of the HCGS specifications which invoke Code Case 1644 (N-71) invoke a specific revision ~~to~~ of the Code Case ~~1644 (N-71)~~, and direct the vendor to comply with the additional regulatory requirements. For these ~~few~~ specifications that did not ~~specify~~ identify a specific revision ~~or~~ and direct conformance with the additional regulatory requirements, a review has been performed to confirm that the ultimate tensile strength of the materials is below 170 KSI.

An ~~We are~~ evaluation is ~~still~~ underway to determine the applicability of the electrode dispersal requirements and a revised response will be provided in November 1984.

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

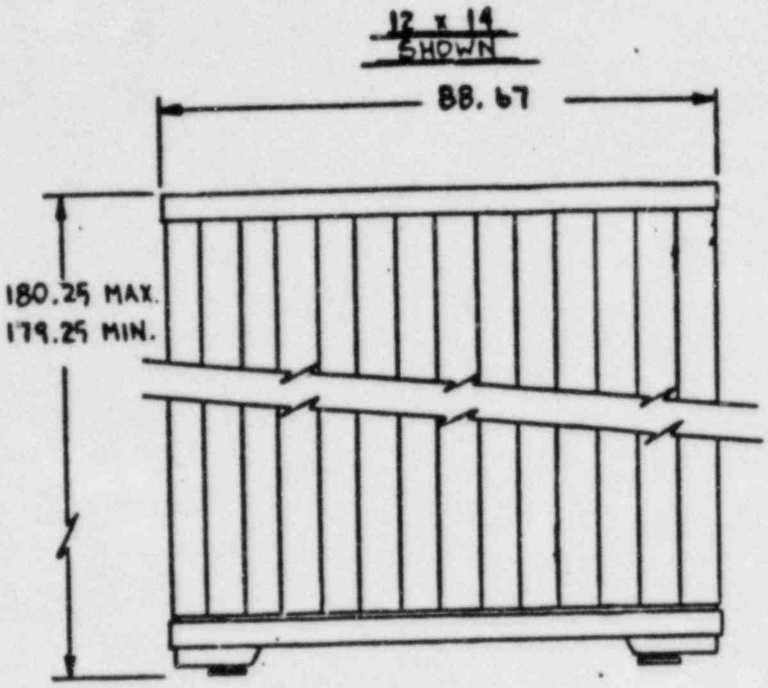
The requested information will be available by ~~June~~ September, 1984, and will be added to Section 3.8.4 and/or 9.1.2 as appropriate.

QUESTION 281.2 (SECTION 9.1.2)

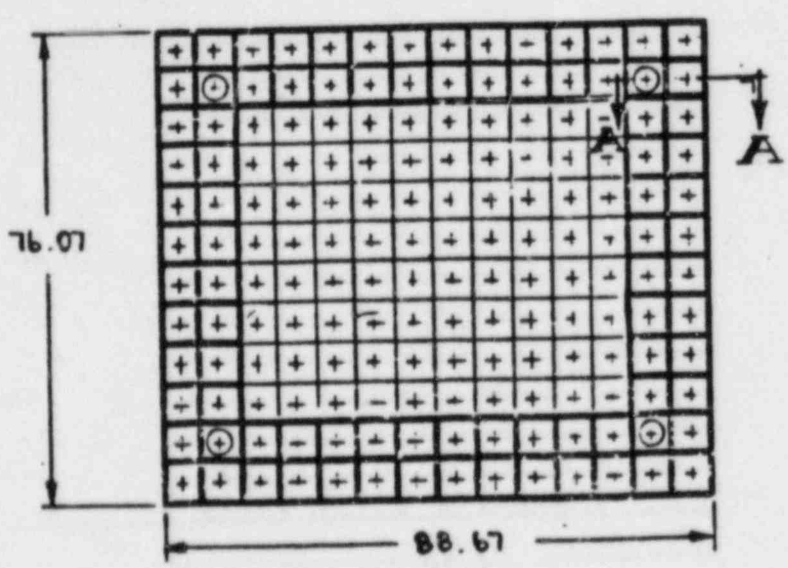
Figures 9.1-3 and 9.1-4 are shown as to be supplied later. Supply these figures or a date by which they will be provided.

RESPONSE

Section 9.1.2 has been revised to provide Figure 9.1-3, A Typical Spent Fuel Rack, and Figure 9.1-4, Spent Fuel Rack Arrangement In Fuel Pool. ~~will be provided by June 1984~~.



SIDE VIEW

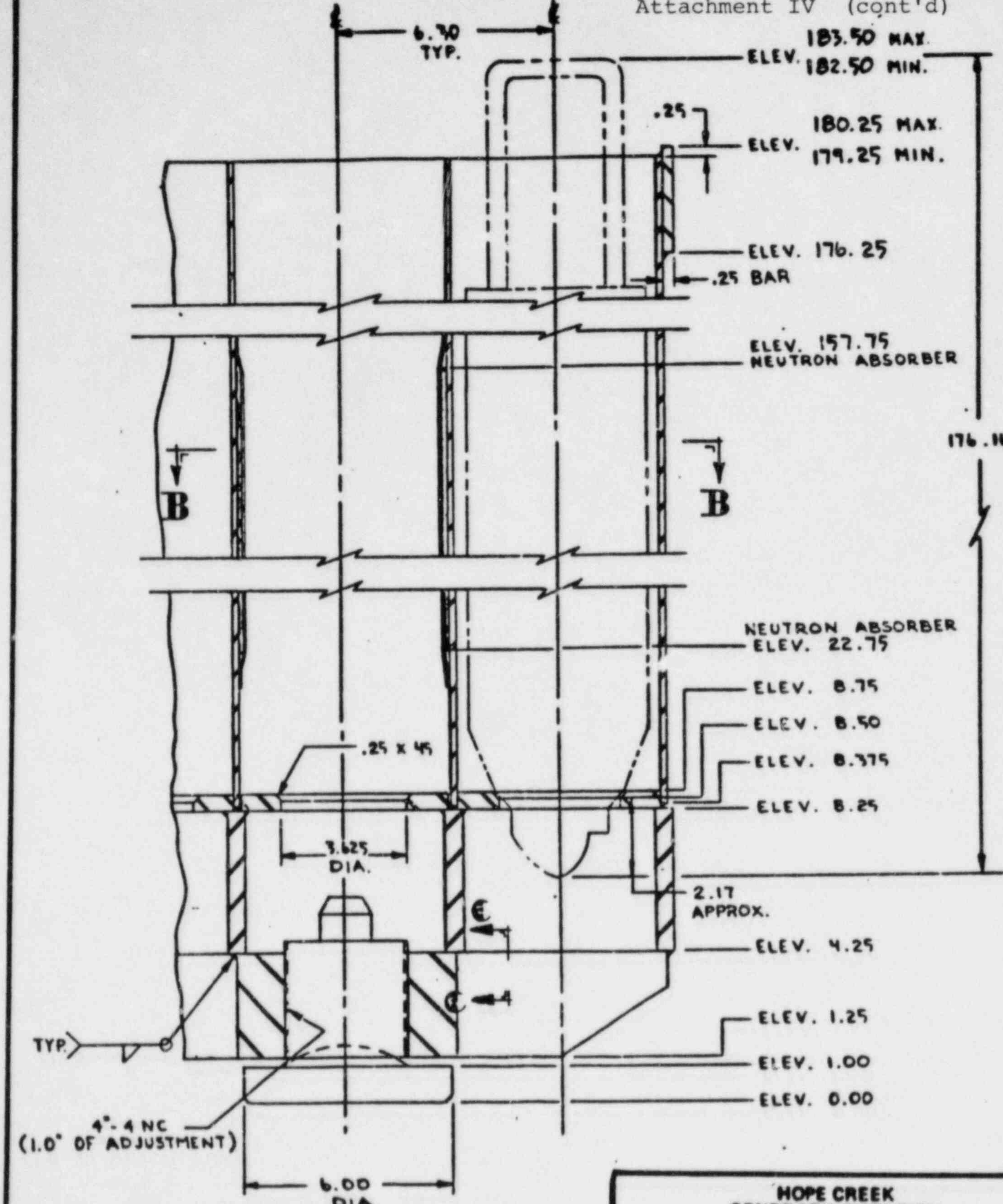


BOTTOM VIEW

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

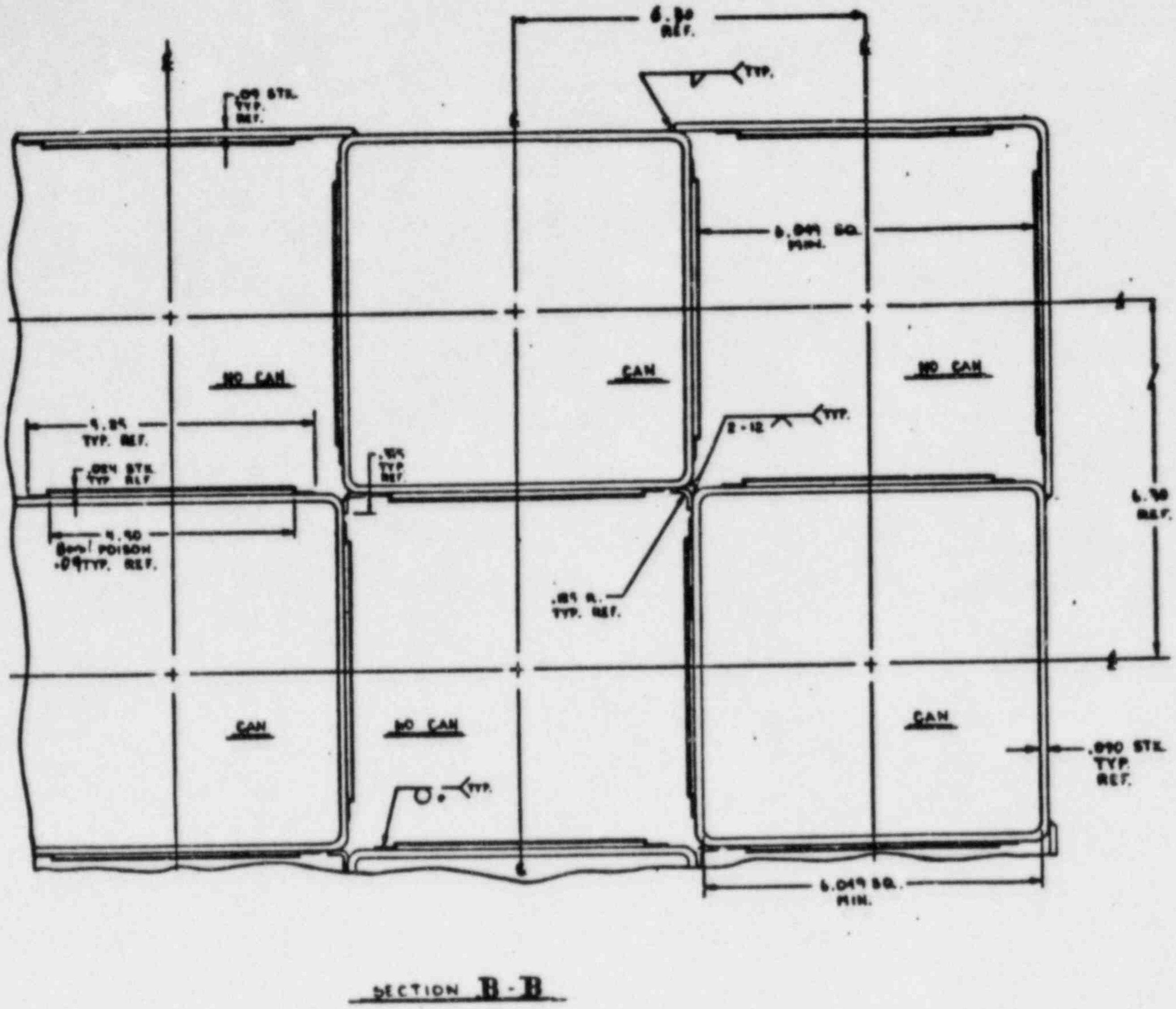
A TYPICAL SPENT FUEL RACK

FIGURE 9.1-3 sheet 1 of 4



SECTION A-A
SHOWN @ MIN. HEIGHT

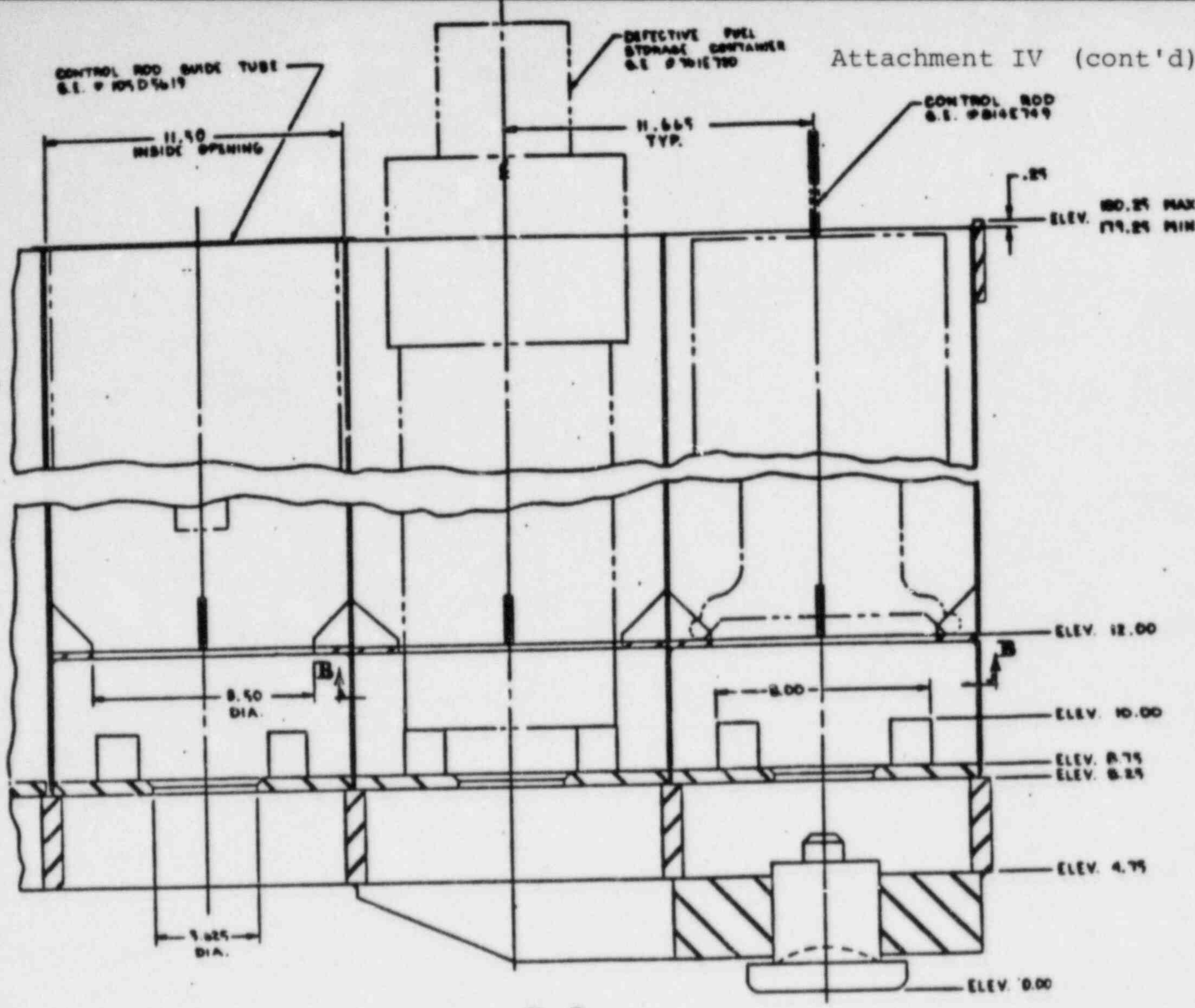
HOPE CREEK GENERATING STATION FINAL SAFETY ANALYSIS REPORT
A TYPICAL SPENT FUEL RACK
FIGURE 9.1-3 sheet 2 of 4



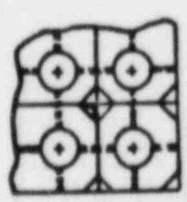
HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

A TYPICAL SPENT FUEL RACK

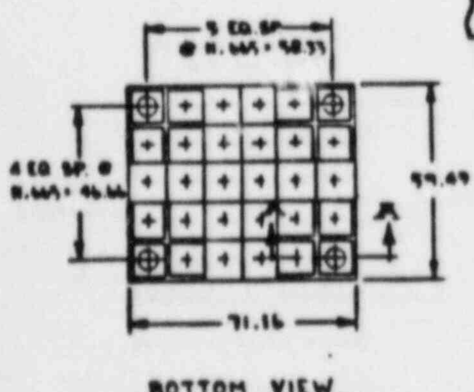
FIGURE 9.1-3 Sheet 3 of 4



SECTION A-A
SHOWN @ MIN HEIGHT



SECTION B-B
SCALE : NONE



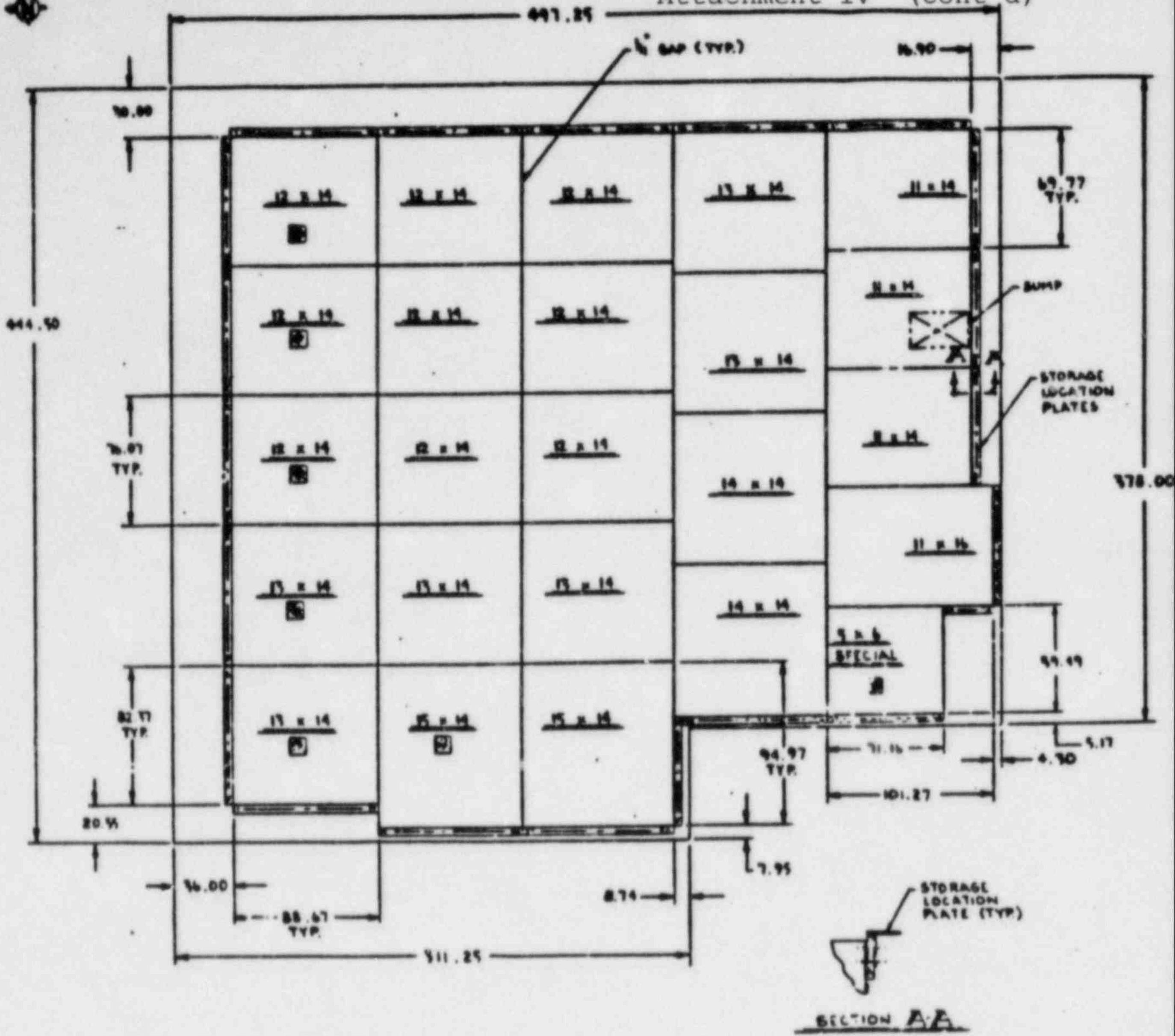
BOTTOM VIEW
SCALE : NONE

NOTES :
1. REGULAR FUEL BUNDLES COULD ALSO BE STORED IN THIS RACK ON A TEMPORARY BASIS IF NECESSARY.
2. 5 x 6 SPECIAL RACK FOR STORAGE OF CONTROL ROD G.E. # 814E749 AND DEFECTIVE FUEL STORAGE CONTAINER G.E. # 761E780 AND CONTROL ROD GUIDE TUBE G.E. # 105D5619

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

A Typical Spent Fuel Rack

FIGURE 9.1-3 : Sheet 4 of 4



TYPE OF MODULE	QTY OF MODULES	QTY OF CAVITIES	QTY OF SPECIAL CAVITIES
SPECIAL 5 x 6	1	—	30
11 x 14	3	462	—
11 x 16	1	176	—
12 x 14	9	1512	—
13 x 14	6	1092	—
14 x 14	2	392	—
14 x 15	2	420	—
TOTAL	24	4054	30

NOTES:

1. ☐ - DENOTES RACKS FOR BASE CAPACITY
2. 5 x 6 SPECIAL RACK IS FOR STORAGE OF CONTROL RODS, GUIDE TUBES AND DEFECTIVE FUEL CONTAINERS
3. 1/4" GAP TYP. BETWEEN MODULES.

HOPE CREEK GENERATING STATION

FINAL SAFETY ANALYSIS REPORT

SPENT FUEL RACK ARRANGEMENT IN FUEL POOL

FIGURE 9.1-4

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 ~~June~~ September 1984, and will be added to Section 9.1.2.

QUESTION 410.39 (SECTION 9.1.2)

Without the spent fuel storage rack design details, the results of an analysis of impact onto the racks and the bundle-to-bundle fuel spacing, the staff cannot make any favorable conclusions about the design. Provide this information in the FSAR.

RESPONSE

Section 9.1.2 and Figure 9.1-3 have been revised to include the spent fuel rack design details. ~~will be available by June 1984, and will be added to Section 9.1.2.~~

CHAPTER 9

FIGURES

1. Normal storage conditions exist when the fuel storage racks are located in the pool and are covered with about 25 feet of water for radiation shielding, and with the maximum number of fuel assemblies or bundles in their design storage position.

2. An abnormal storage condition may result from accidental dropping of an empty fuel rack, or from damage caused by the horizontal movement of fuel handling equipment without first disengaging the fuel from the hoisting equipment.

b. It is assumed that the storage array is infinite in all directions. Since no credit is taken for leakage, the values reported as effective neutron multiplication factors are in reality infinite neutron multiplication factors. The biases between the calculated results and experimental results and the uncertainty involved in the calculations, as well as other uncertainties, are taken into account as part of the calculational procedure to ensure that the specified K_{eff} limits are met.

c. The racks are designed to protect the fuel assemblies from physical damage caused by impact from fuel assemblies. The rack design would prevent the release of radioactive materials in excess of 10 CFR 20 and 10 CFR 100 allowances under normal and abnormal storage conditions.

d. The racks are constructed in accordance with the QA requirements of 10 CFR 50, Appendix B.

e. The spent fuel storage racks are constructed in accordance with Seismic Category I requirements. The applicable code for the design of racks is ASME Section III, Subsection NF.

f. Spent fuel storage space is provided in the fuel storage pool to accommodate ~~at least 4.0~~ 5.3 core loads of fuel assemblies.

g. Spent fuel storage racks are designed and arranged so that the fuel assemblies can be handled efficiently during refueling operations.

h. The spent fuel storage facility and all piping connections are designed to prevent a loss of cooling water from the spent fuel pool that could uncover the stored fuel.

i. The spent fuel storage facility is designed to prevent criticality of stored fuel under adverse environmental and postulated fuel handling accident conditions.

j. Shielding for the stored spent fuel assemblies is designed to protect plant personnel from exposure to direct radiation greater than that permitted for continuous occupational exposure during normal operations.

k. The spent fuel storage facility is designed to remain functional during and following a safe shutdown earthquake (SSE).

l. Failures of systems or structures not designed to Seismic Category I standards and located in the vicinity of the spent fuel storage facility do not increase the K_{eff} established by design.

m. The spent fuel pool is designed to withstand thermal stresses resulting from the pool water boiling.

n. The rack design prevents accidental insertion of fuel assemblies between adjacent racks.

o. The spent fuel storage facility is designed so that failure of structures, systems, or components that are not Seismic Category I will not result in a loss of function of the facility.

p. The spent fuel storage area is designed for fuel having an average U235 enrichment of 3.4 weight percent.

The following design bases for the spent fuel and new fuel storage facilities are discussed in the sections indicated below:

a. Seismic and system quality group classifications - Sections 3.2.1 and 3.2.2

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 ~~details of rack construction will be provided prior to fuel load.~~ neutron absorber is Boral. See Figure 9.1-3 for ~~construction~~ design details of a typical rack and the special rack.

30

The spent fuel pool has been designed for a storage capacity for
4054 ~~at least 3668~~ fuel assemblies, plus ~~27~~ multipurpose cavities for storage of control rods, control rod guide tubes, and defective fuel containers. For initial plant operation, a storage space
1078 for ~~about 1044~~ fuel assemblies ~~will be~~ provided and the remaining storage capacity will be added later.

, plus 30 multipurpose cavities

~~The superstructure of the reactor building~~ serves as a low leakage barrier to provide atmospheric isolation ~~of~~ the spent fuel storage pool and associated fuel handling area.
is located within the reactor building ~~secondary containment~~ which

9.1.2.2.2.3 Refueling Area Cavities

As shown on Figure 9.1-2, the cask loading pit and the reactor well are adjacent to the spent fuel pool. The dryer and separator pool is adjacent to the reactor well. Like the spent fuel pool, these cavities are lined with stainless steel plate and are provided with liner leakage collection systems. The reactor well and the cask loading pit are connected to the spent fuel pool by fuel transfer canals approximately 4-feet wide. Each canal is provided with two gates and concrete plugs to prevent loss of water from the spent fuel pool during periods when the adjacent cavity is not filled with water.

The cask loading pit is designed to permit the underwater loading of spent fuel assemblies into spent fuel shipping casks. The pit can be drained of water during periods when cask loading operations are not being performed. The spent fuel shipping cask can be decontaminated either in the cask loading pit or in the cask washdown area on the refueling floor adjacent to the cask loading pit.

The reactor well is a circular cavity located directly above the primary containment. Removal of the drywell head and reactor vessel head provides direct access from the reactor well to the inside of the reactor vessel. The reactor well is filled with water during transfer of fuel assemblies from the reactor vessel to the spent fuel pool. Seals are provided at the bottom of the reactor well between the drywell and reactor well wall and between the reactor vessel and the drywell to prevent water leakage.

The dryer and separator pool provides for storage of the steam dryer and steam separator when they are removed from the reactor vessel. The dryer and separator pool is connected to the reactor

well to permit underwater transfer of components between the two cavities. Concrete seal plugs are provided to minimize water loss during normal and abnormal storage conditions when the reactor well is not filled with water. Gaskets are attached to the horizontal surfaces of the seal plugs to further reduce water loss.

9.1.2.2.2.4 Other Features

The spent fuel pool area ventilation system is discussed in Section 9.4.2.

The area radiation and airborne radioactivity monitoring instrumentation is described in Section 12.3.4.

9.1.2.3 Safety Evaluation

9.1.2.3.1 Criticality Control

Geometrically safe configurations of fuel stored in the spent fuel array, and poison materials, are employed to ensure that K_{eff} will not exceed 0.95 under any normal or abnormal storage condition. To ensure that the design criteria are met, the following normal and abnormal spent fuel storage conditions ~~were~~ are analyzed: X

a. Normal positioning of fuel assemblies in the spent fuel storage array

b. Eccentric positioning of fuel assemblies in the spent fuel storage array

c. Normal storage array of ruptured fuel

d. Moving or placing a fuel bundle along the outside of storage racks

e. Deleted

f. Spent fuel bundle falling onto the rack with spent fuel

g. ~~Empty fuel rack falling onto the rack with spent fuel.~~
Deleted

9.1.2.3.2 High Density Spent Fuel Rack Design Criteria

The principal design criteria of the spent fuel racks are as follows:

a. Up to 4084, ~~At least 3668~~ fuel assemblies may be stored in the fuel pool.

b. The storage racks provide an individual storage compartment for each fuel assembly. The fuel assemblies are stored in a vertical position with the lower tie plate engaged in a captive slot in the lower fuel rack support plate.

c. The weight of the fuel assembly is held by the lower rack support plate.

d. The spent fuel storage racks are made from 304L stainless steel.

e. The minimum center-to-center spacing for the fuel assembly ~~between rows, and the minimum center-to-center spacing~~ within the rows, ~~will be provided prior to fuel load.~~ is shown on Figure 9.1-3. Fuel assembly placement between rows is not possible.

f. ~~Lead-in and lead-out guides at the top of the racks provide guidance of the fuel assembly during insertion or withdrawal.~~
Deleted

g. The impact force considered in the rack design will be provided prior to fuel load.

h. The storage rack is designed to withstand a pull-up force of 4000 pounds and a horizontal force of 1000 pounds. There are no readily available forces in excess of 1000 pounds. ~~The racks are designed with lead-outs to prevent sticking. However,~~ in the event of a stuck fuel assembly, the maximum lifting force ~~in~~

~~excess of 1000 pounds. The racks are designed with lead-outs to prevent sticking. However, in the event of a stuck fuel assembly, the maximum lifting force~~ of the fuel handling platform grapple, assuming limit switches fail, is 3000 pounds.

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.

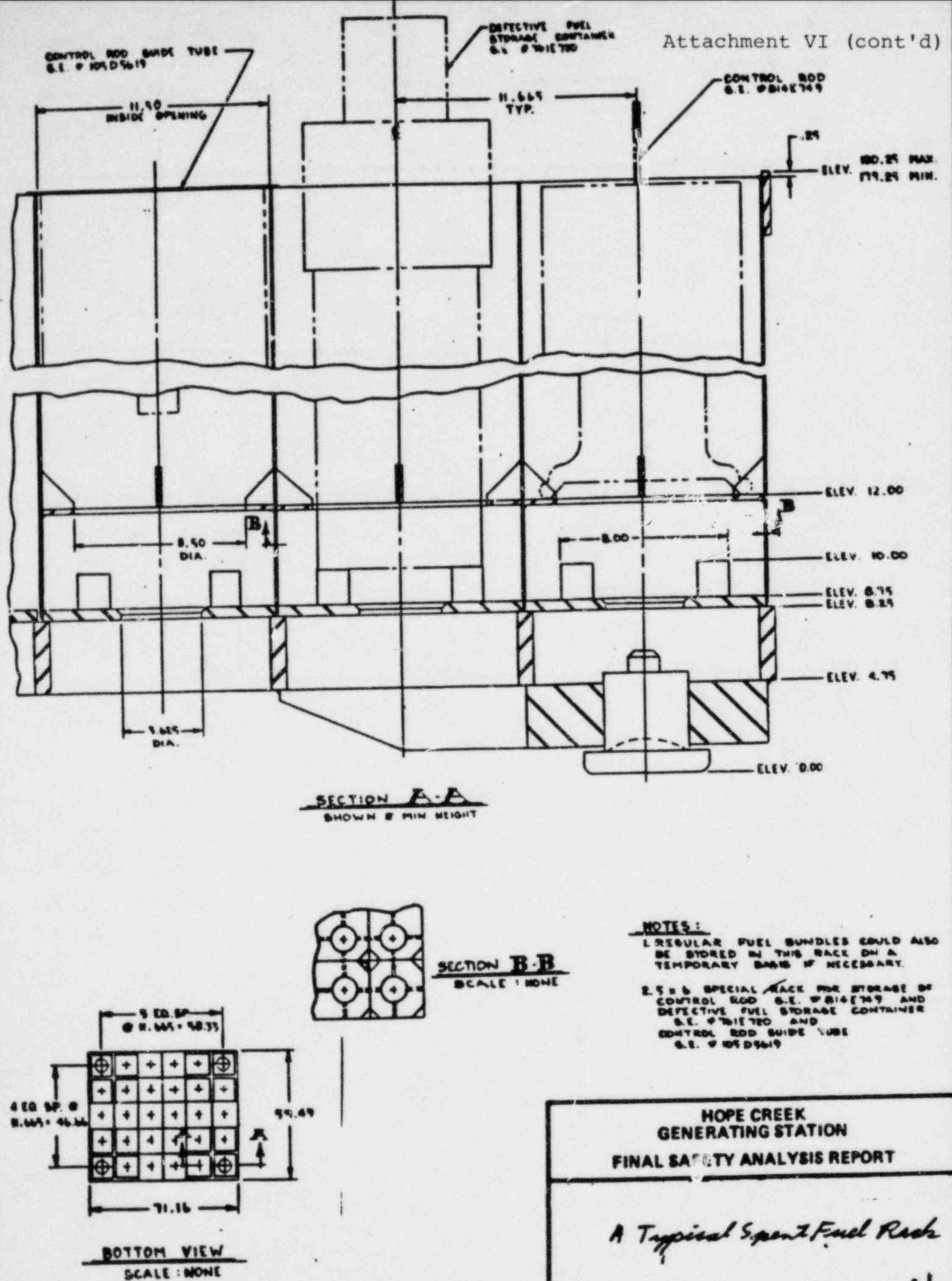
k. Several design features reduce the possibility of heavy objects dropping into the fuel pool. The main and auxiliary hoists of the reactor building polar crane are single-failure proof. In addition, the main hoist is physically prevented from traveling in the truncated segment shown on Figure 9.1-31 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.

9.1.2.4 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

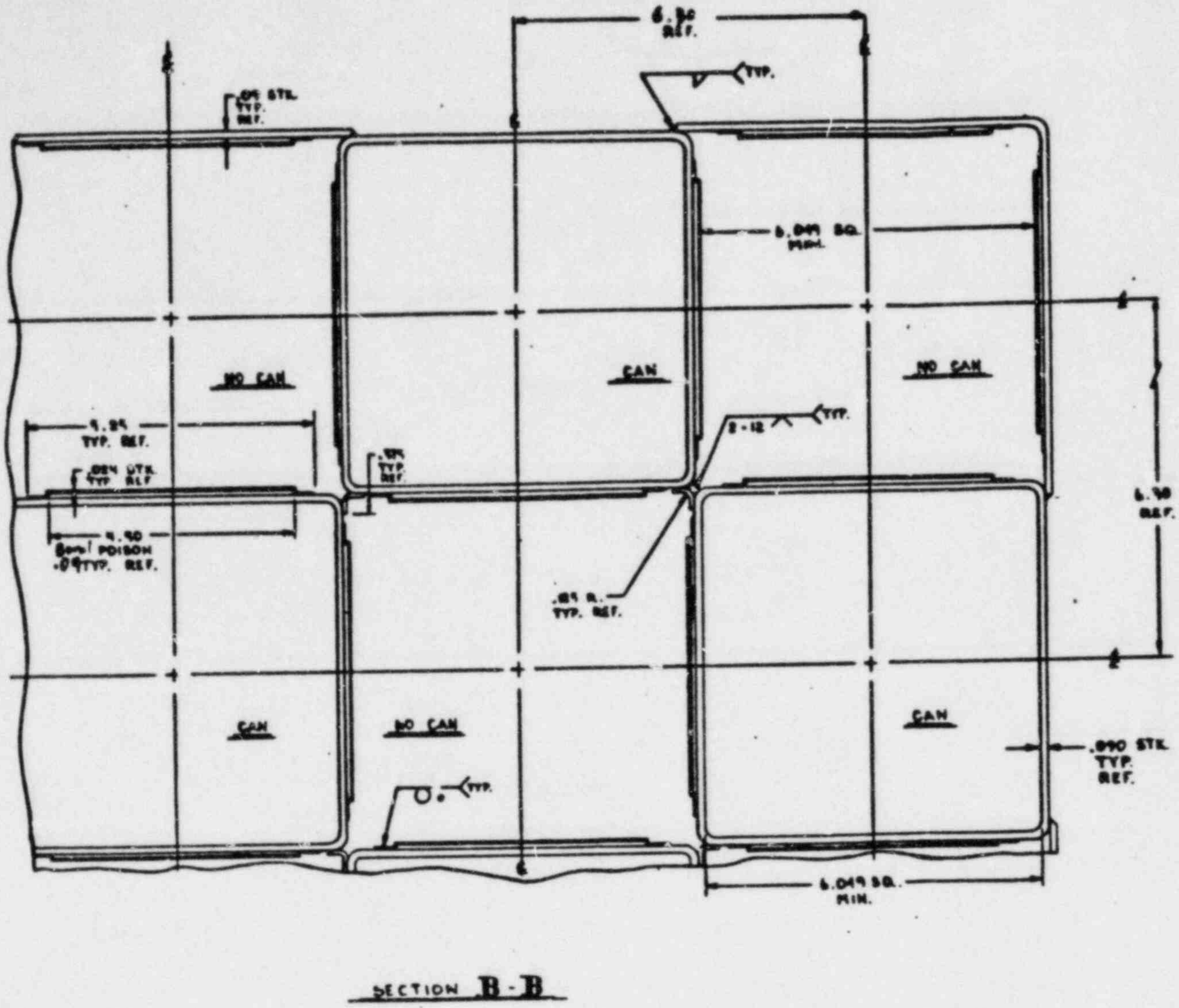


HOPE CREEK GENERATING STATION

FINAL SAFETY ANALYSIS REPORT

A Typical Spent Fuel Rack

FIGURE 9.1-3 sheet 4 of 4



HOPE CREEK
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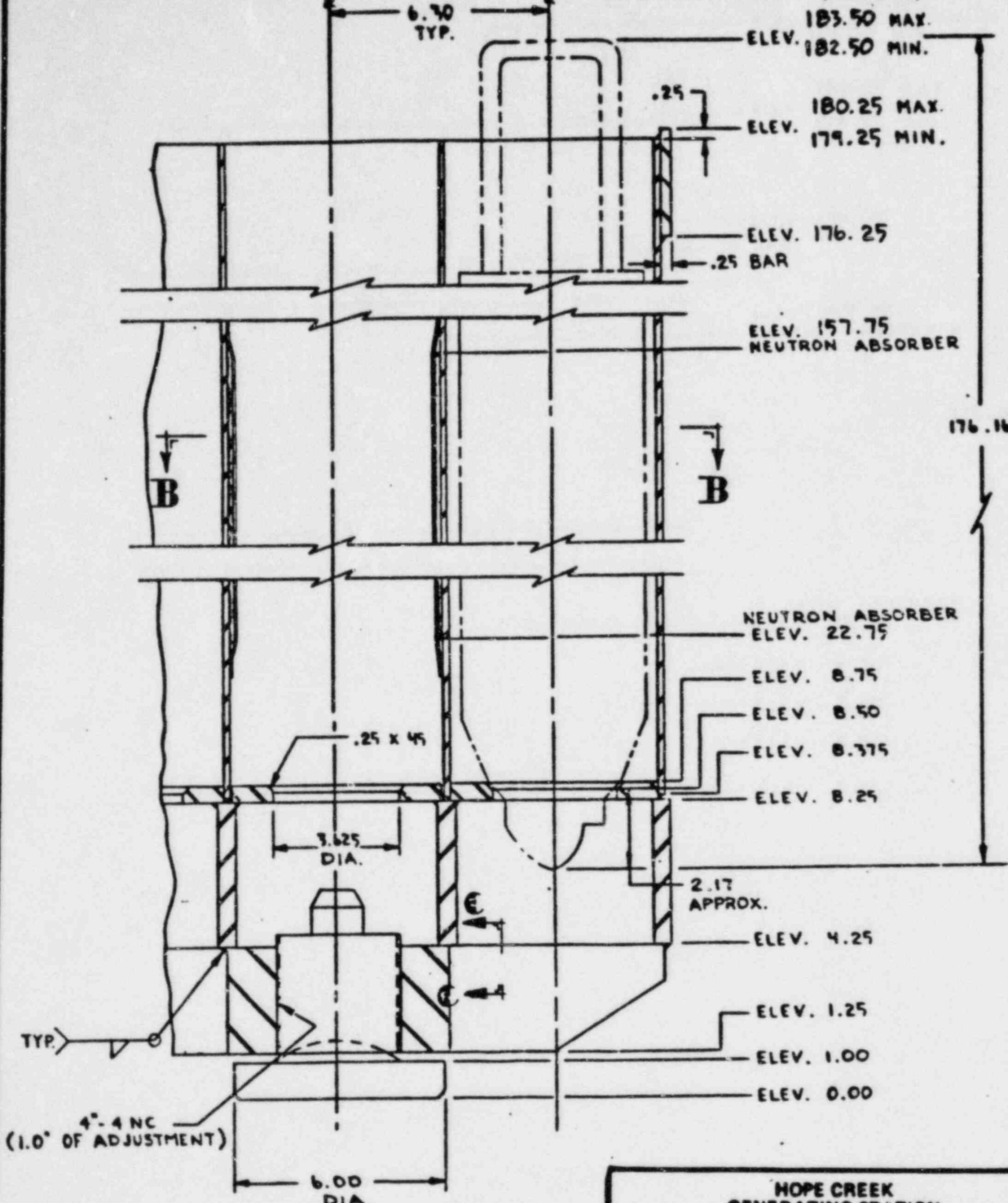
A TYPICAL SPENT FUEL RACK

FIGURE 9.1-3 Sheet 3 of 4



SECTION A-A
SHOWN @ MIN. HEIGHT

HOPE CREEK
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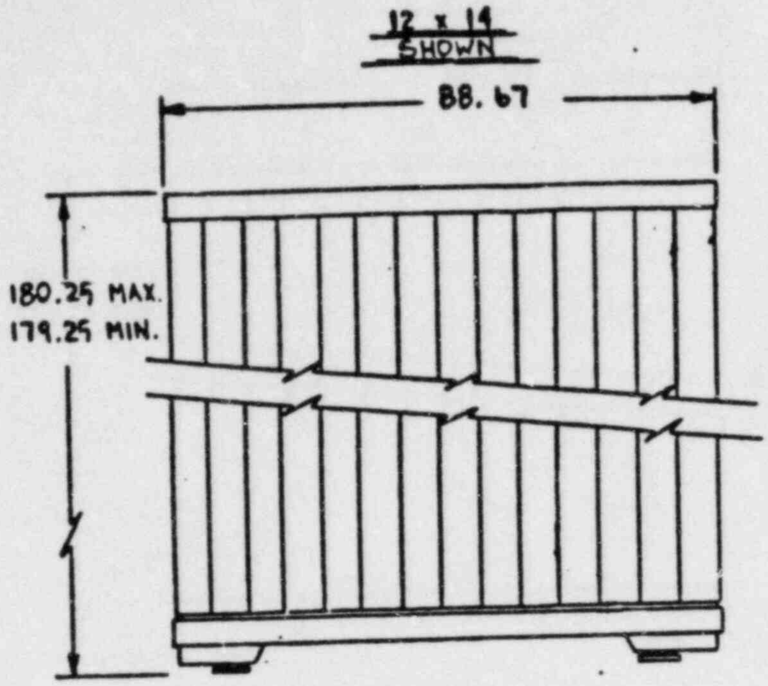
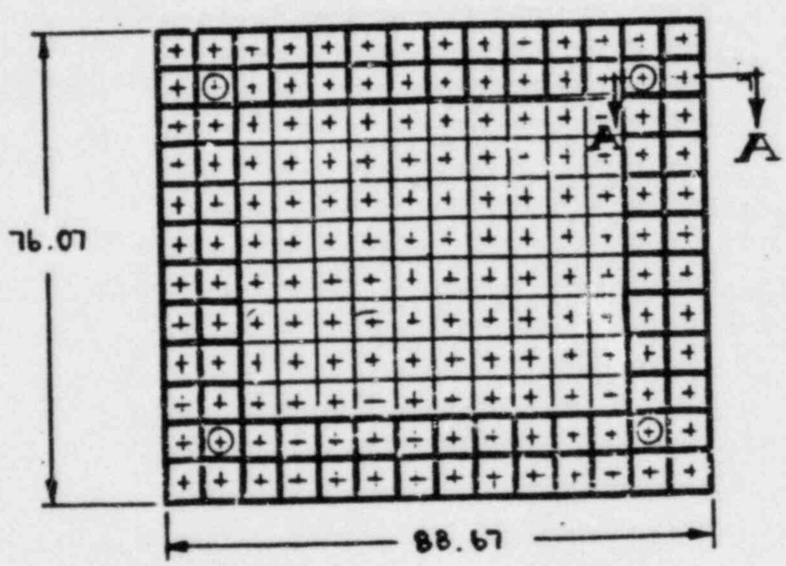
A TYPICAL SPENT FUEL RACK

FIGURE 9.1-3 sheet 2 of 4



SIDE VIEW



BOTTOM VIEW

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

A TYPICAL SPENT FUEL RACK

FIGURE 9.1-3 sheet 1 of 4

QUESTION 410.42 (SECTION 9.1.2)

Specify the weight percent of U235 which corresponds to the highest anticipated enrichment used in the criticality calculations for the new and spent fuel storage areas.

RESPONSE

The highest anticipated assembly average enrichment of U235 used in the spent fuel rack criticality analysis is 3.4 weight percent. ~~will be available by June, 1984, and will be added to Section 9.1.2.~~

The new fuel storage area criticality analysis is based on a maximum anticipated reactivity of the fuel and not on enrichment alone. The use of reactivity takes into account the combined effects of enrichment, enrichment distribution, gadolinia, gadolinia distribution, and fuel lattice geometry (i.e., water rods and fuel rod pitch). The HCGS new-fuel storage racks were designed to store fuel which has an infinite neutron multiplication factor of ≤1.31, in the uncontrolled reactor core geometry.

This fuel reactivity limit bounds all existing and expected GE fuel designs. The use of the maximum reactivity of the fuel, instead of enrichment alone, is consistent with the NRC Standard Review Plan 9.1.1, as well as industry standards such as ANSI Standards N208, N209, and N210.

Seismic qualification for this isolation system is in accordance with qualification procedures and acceptance criteria defined in IEEE Standard 344-1975, and implemented by Regulatory Guide 1.100, Revision 1.

This isolation system is located in and qualified for a mild environment as defined in Sections 3.11.2.4 and 3.11.2.5. The worst-case specified environmental conditions in which this isolation system is designed to operate are as follows:

Pressure: Atmospheric plus fractional inch of H_2O

Temperature:	104°F maximum 40°F minimum 83°F ± 2°F	these conditions may exist 24 hours per year

Relative Humidity: 50% maximum (summertime)
20% minimum (wintertime)

Nuclear Radiation: 175 Rads Carbon (40 year TID)
88 Rads Carbon - Beta (180 day TID)
2.5 Rads Carbon - Gamma (180 day TID)

(TID = Total Integrated Dose)

Testing performed in accordance with SAMA Standard PMC 33.1-1978 ~~will be completed by June, 1984~~, to ensures that this isolation system is adequately protected against the effects of electromagnetic interference (EMI).

Testing performed in accordance with IEEE Standard 472-1974 ~~will be completed by June, 1984, to~~ ensures that this isolation system is adequately protected against the effects of short-circuit failures, voltage faults and/or surges.

b. Computer Products Inc. (CPI) Emergency Response Facilities Data Acquisition System (ERFDAS) - this system utilizes the CPI real time peripheral (RTP) system for 1E to non-1E isolation. The basic components of the RTP system are analog and digital surge cards (qualified to IEEE Standard 472-1974 requirements), analog input cards and optically isolated digital input cards, distributed input/output controllers (DIOC) and transformer-coupled multi-drop limited distance modems (MDLDM). The MDLDMs provided the 1E to non-1E isolation. Data transmission to receiving MDLDMs is by twisted-shielded pairs.

Seismic qualification for this isolation system is in accordance with qualification procedures and acceptance criteria defined in IEEE Standard 344-1975, and implemented by Regulatory Guide 1.100, Revision 1.

Attachment VIII (cont'd)

This isolation system is located in and qualified for a mild environment as defined in Sections 3.11.2.4 and 3.11.2.5. The worst-case specified environmental conditions in which this isolation system is designed to operate are as follows:

Pressure: Atmospheric plus fractional inch of H_2O

Temperature:	104°F maximum 40°F minimum 83°F ± 2°F	these conditions may exist 24 hours per year

Relative Humidity: 50% maximum (summertime)
20% minimum (wintertime)

Nuclear Radiation: 175 Rads Carbon (40 year TID)
88 Rads Carbon - Beta (180 day TID)
2.5 Rads Carbon - Gamma (180 day TID)

Testing performed in accordance with SAMA Standard PMC 33.1-1978 ~~will be completed by August, 1984, to~~ ensures that this isolation system is adequately protected against the effects of electromagnetic interference (EMI).

Testing performed in accordance with IEEE Standard 472-1974, ~~will be completed by February, 1984, to~~ ensures that this isolation system is adequately protected against the effects of short-circuit failures, voltage faults and/or surges. These tests are ~~will be~~ performed on the analog and digital surge cards and the transmit/receive circuits of the MDLDMs.

c. Technology for Energy Corporation (TEC) Radiation Monitoring System (RMS) - this system utilizes three separate isolation methods depending upon the type of isolation required:

1) 1E to 1E isolation - for this type of isolation, Hewlett Packard HFBR 1000 and HFBR 2001 isolators are used. Optical coupling is used to provide the isolation.

2) 1E to non-1E annunciator outputs - for this type of isolation, Agastat Model EGP isolation relays are used. Relay coil to contact separation provides the isolation.

3) 1E to non-1E communication - for data transmission between the TEC 1E microprocessor and the non-1E host computer, TEC Synchronous Data Link Control, serial communications modules 600-1200 are used. Transformer coupling provides the isolation for the transmit circuits. Optical coupling provides the isolation for the receive circuits.

Seismic qualifiction for these isolation systems is in accordance with qualification procedures and acceptance criteria defined in IEEE Standard 344-1975, and implemented by Regulatory Guide 1.100, Revision 1.

These isolation systems are located in and qualified for a mild environment as defined in Sections 3.11.2.4 and 3.11.2.5. The worst-case specified environmental conditions in which these isolation systems are designed to operate are as follows:

Temperature:	104°F maximum 40°F minimum 76°F ± 2°f	these conditions may exist 24 hours per year
Relative Humidity:	50% maximum 20% minimum	

Testing performed in conformance with Military Standards 461B and 462 on the effects of EMI ~~will be completed by July, 1984.~~ ensures that these isolation systems are adequately protected against the effects of EMI

Testing performed in accordance with IEEE Standard 472-1974~~, will be completed by July, 1984, to~~ ensures that these isolation systems are adequately protected against the effects of short-circuit failures, voltage faults and/or surges.

d. Remote control panels - two isolation methods are provided for remote control panels requiring 1E to non-1E isolation.

1) Digital 1E to non-1E isolation - for this type of isolation, Struthers Dunn type 219, Allen Bradley model 700-200A12P, and General Electric model HEA99 isolation relays are used. Relay coil to contact separation provides the isolation.

2) Analog 1E to non-1E isolation - for this type of isolation, TEC analog isolators, model 156, are used. Transformer coupling is used to provide the isolation.

Seismic qualification for these isolation systems is in accordance with qualification procedures and acceptance criteria defined in IEEE Standard 344-1975, and implemented by Regulatory Guide 1.100, Revision 1.

The Struthers Dunn type 219 and General Electric model HEA99 isolation relays are located in and qualified for a mild environment as defined in Sections 3.11.2.4 and 3.11.2.5. The worst-case specified environmental conditions in which these isolation relays are designed to operate are as follows:

Struther Dunn Type 219

Nuclear radiation: 200 Rads (40 year TID)

No testing was conducted on the effects of EMI on the Struthers Dunn type 219, Allen Bradley model 700-200A12P, or General Electric model HEA99 isolation relays. By design, these relays should be immune to the effects of EMI.

Generic EMI susceptibility and emissions test were conducted on the TEC model 156 analog isolators following procedure 156-QP-04, "Electromagnetic Interference (EMI) Test for TEC Model 156 Analog Signal Isolator Module," which is Appendix B to test report 31041-QP-01, "Qualification Test Report for Environmental and Seismic Testing of the TEC Model 158 Analog Isolation System." Results of these tests are available for review at Technology for Energy Corporation, Knoxville, Tennessee.

Testing performed in accordance with IEEE Standard 472-1974 ~~will be performed to~~ ensures that the Struthers Dunn Type 219 ~~and General Electric model HEA99~~ isolation relays are adequately protected against the effects of short-circuit failures, voltage faults and/or surges.

— INSERT B —

The following test was performed on the Allen Bradley model 700P-200A12P isolation relay to ensure adequate protection against the effects of short circuit failures, voltage faults and/or surges:

1) Test type - 100% high potential test

2) Test characteristics - 2700 V applied for one second between points of opposite polarity and to ground.

Testing performed in accordance with IEEE Standard 472-1974 ~~will be performed to~~ ensures that the TEC model 156 analog isolators are adequately protected against the effects of short circuit failures, voltage faults and/or surges.

e. Equipment air lock isolation dampers HD-9450A and B interlock with receiving bay door #4323A - Potter Brumfield model MDR isolation relays are utilized to provide both non-1E to 1E and 1E to non-1E isolation as shown below:

1) Non-1E to 1E - receiving bay door #4323A (non-1E coil) permissive to equipment air lock isolation dampers HD-9450A and B (1E contact)

2) 1E to non-1E - equipment air lock isolation dampers HD-9450A and B (1E coil) permissive to receiving bay door #4323A (non-1E contact)

These two relays were purchased from General Electric.

—INSERT B—

Testing performed in accordance with General Electric Power Systems Management Business Department document MIL 82-12, dated July 24, 1982, ensures that the General Electric model HEA99 isolation relays are adequately protected against the effects of short-circuit failures, voltage faults and/or surges.

f. Startup Transient Monitoring System (STMS) - The qualification requirements of isolation devices, used by the STMS are described in Section 7.5.1.3.5.

— INSERT A —

NSSS:

The isolation devices used to electrically separate nonessential and essential circuits are pursuant to the guidelines of IEEE Standard 384. Both relay and optical isolation devices are employed. The optical isolators utilize a fiber-optic light pipe to electrically separate the input from the output. For example, an essential logic signal activates a light emitting diode, the light is transmitted through the light pipe to a photo switch and the switch changes state on receipt of the light signal and either blocks or transmits.

The relay isolation devices provide the same degree of separation and are used typically for control voltage separation applications, i.e., 120-Vac and 125 Vdc essential to nonessential and redundant essential circuits. The relays are mounted so that a metal barrier separates the coil from the contacts with a minimum distance of one inch between the coil and barrier and between the contact and barrier.

Summary of Purchase Specification:

a. RELAY

1. Design Specification
   a) MIL-R-19523
   b) Contact Specification
   c) Coil Specification
   d) Insulation Specification
   e) Design Life
   f) Reliability

2. Class 1E Safety Function
   a) Functional Specification
   b) Reliability

3. Qualification Testing
   a) Ambient and Design Environments
   b) Application Configuration

b. ISOLATOR

1. Bill of Material

2. Purchase part drawings 204B6186 and 204B6188

3. Qualification Testing
   a) Tested as a panel subassembly

— INSERT A —

Qualification testing for the isolation systems identified in parts a through f is ongoing. Reports documenting the results of this testing will be made available for review by the NRC when received. This is anticipated to be by September, 1984, for equipment identified in parts a, b, d, e, and f, and January, 1985, for the TEC RMS equipment identified in part c.

each individual will meet the education and experience requirements of ANS/ANSI 3.1 - 1981 prior to initiate fuel loading.

In general, the personnel assigned to the licensed operator training come from one of the following areas:

1. Degreed engineer
2. Previously licensed (BWR/PWR)
3. Navy nuclear plant operator
4. Fossil plant operator
5. Salem EO upgrade

In general, personnel assigned to the non-licensed operator training will come from one of the following areas:

1. Qualified utility/equipment operator from Salem Generating Station
2. Navy nuclear plant operator
3. Fossil plant operator

Insert A'

b. Training on the HCGS plant specific procedures and technical specifications will be conducted as the procedures become available. These procedures are under development and will become available at various intervals throughout the training period. To ensure that all licensed operator candidates are thoroughly familiar with the procedures and , Appendix 13; technical specifications, ~~an intense pre-license training program~~ Insert B' will be implemented three (3) to six (6) months prior to the license examinations. This training will cover all the HCGS specific operating, abnormal and emergency procedures, administrative and emergency response procedures, technical specifications and low power and surveillance testing procedures. Training will be covered by classroom instruction, in-plant oral examinations, written examinations and performance testing on the Hope Creek specific simulator.

c. Applicable references for each of the segments outlined in the appendices are shown on the appropriate cover sheet of each appendix.

d. Training segments which include 10CFR Part 55 Section 21, 22 and 23 are identified in Appendix 13A, 13C, 13E, 13F and 13G.

e. The following segments of the training program are still under development:

Appendix J - i Cold license operator in-plant training ~~Control room preoperational testing~~ e
Appendix J - ~~Practical work assignments~~ e
~~Appendix K~~ - ~~Pre-license exam audit observation training~~ e
Pre-license examination testing and training

Insert A'

~~In addition~~ These potential license candidates are required to achieve a satisfactory score on a screening examination as a prerequisite to assignment to the operator training program. At present, ~~the~~ Power Operator Service Selection (POSS) is used. Exception ~~to the~~ requirement is made for individual who previously held a NRC license and for degreed personnel.

All prospective employees must participate in a physiological screening process. ~~A~~ The Minnesota Multi-Phasic Personnality Inventory (MMPI) is presently in ~~a~~ use.

b. insert B'

Attachment IX (cont'd)

~~b~~ ... training on plant specific procedures and technical specifications will be incorporated into the training programs outlined in appendices B5, 13 ~~b~~ & 13 i.e. ~~WA~~ ~~[illegible]~~ In addition to this training . . .

f. ~~These segments will be implemented or procedures describing the content of each segment will be developed by June 1984.~~

g. Hot license training for NRC candidates will be conducted to augment the shift staffing allotment, allow for promotion or fill vacancies due to reassignment. This training will utilize a major portion of the existing cold license training program; however, certain areas may be waived based on an individual's prior experience and educational background. Procedures describing the content and administrative requirements will be completed by June 1985.

h. Appendix 13F has been revised to incorporate this response.

~~The course~~

A course description for segments i and j of the training program is contained in Appendices 13i and 13j, respectively.

ATTACHMENT X
HOPE CREEK GENERATING STATION
FSAR COMMITMENT STATUS THROUGH JULY 1984

FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
1. Question/Response Appendix: Question 210.12	This commitment concerns providing design data to demonstrate compliance with Paragraph NC/ND-3649 of Code Case N-192. This information has been provided in the letter, R. L. Mittl (PSEG) to A. Schwencer (NRC), "Compliance with Reg. Guide 1.84", dated July 30, 1984. The information in Attachment XI will be included in Amendment 8 to the HCGS FSAR.
2. Question/Response Appendix: Question 210.20	This commitment concerns providing the method and acceptance criteria for a dynamic analysis of the feedwater check valve response to a feedwater line break outside containment. This information will be provided in August 1984.
3. Question/Response Appendix: Question 410.87	This commitment concerns preparing preventive maintenance procedures for instrument air systems in compliance to ANSI MC11.1-1976. This information will be provided in November 1984. This revised commitment date will be included in Amendment 8 to the HCGS FSAR.
4. Question/Response Appendix: Question 210.12	This commitment concerns itemizing results of a review to determine if bypasses may have to be used for in-service testing of Electric Power and Protection Systems. This information has been provided in the letter; R. L. Mittl (PSEG) to A. Schwencer (NRC), "DSER Open Item Status," dated August 3, 1984.

FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
5. Question/Response Appendix: Question 421.13c	This commitment concerns testing the isolation systems for the Radiation Monitoring System for effects of EMI and protection against effects of short-circuit failures, voltage faults, and/or surges. This information has been provided in the letter; R. L. Mittl (PSEG) to A. Schwencer (NRC), "DSER Open Item Status", dated August 1, 1984.
6. Question/Response Appendix: Question 421.22	This commitment concerns providing information on the capability for the at-power surveillance testing of the instrumentation channels, logic and actuation devices of plant safety systems. This information has been provided in the letter; R. L. Mittl (PSEG) to A. Schwencer (NRC), "DSER Open Item Status," dated August 1, 1984.
7. Question/Response Appendix: Question 421.23	This commitment concerns providing justification of BWR reactor vessel level sensing lines. This information has been provided in the letter; R. L. Mittl (PSEG) to A. Schwencer (NRC), "DSER Open Item Status," dated August 1, 1984.
8. Question/Response Appendix: Question 430.19	This commitment concerns preparing an emergency load sequencer (ELS) system reliability analysis. This information has been provided in the letter; R. L. Mittl (PSEG) to A. Schwencer (NRC), "DSER Open Item Status," dated August 1, 1984. The information in Attachment XII will be included in Amendment 8 to the HCGS FSAR.
9. Question/Response Appendix: Question 630.9	This commitment concerns providing license examination periods in FSAR Figure 13.2-1. This information has been provided in Amendment 5 to the HCGS FSAR.

and/or 1818 subject to the limitations recommended by Regulatory Guide 1.84.

Code Case N-192 was invoked in the fabrication of certain flexible metal instrument hose assemblies and on certain standby diesel generator skid-to-facility connectors. Regulatory Guide 1.84 states that this code case is acceptable subject to the requirement that the applicant should provide design data to demonstrate compliance with Paragraph NC/ND-3649.

Information to comply with this additional regulatory requirement ~~will be provided in June 1984.~~

Code Case N-275 was invoked in the fabrication of certain safety-related pipe. Regulatory Guide 1.84 states that the design guidance in this code case is acceptable subject to the additional welding restrictions in the regulatory guide.

HCGS complies with these additional regulatory requirements. The HCGS piping design specification permits the use of Code Case N-275 subject to the limitations recommended by Regulatory Guide 1.84.

Code Case 1644 and its various revisions has been invoked in numerous applications. Regulatory Guide 1.85 states that this code case is acceptable subject to the limitations on maximum ultimate tensile strength and, in the case of Code Case 1644-9 (N-71-9), the additional requirements for electrode dispersal.

HCGS is currently evaluating the applicability of the additional maximum ultimate strength limitation in view of the concerns with material brittleness and stress corrosion cracking. A response will be provided in June 1984.

Use of Code Case 1644-9 (N-71-9) is subject to the additional precautions cited in Regulatory Guide 1.85.

Use of Code Case N-249 is permitted for the containment hydrogen recombiner technical specification. To date, this code case has not been invoked.

Code Case N-253-1 provides rules for the construction of ASME components which experience elevated temperatures. This code case was invoked in the design of the containment hydrogen recombiners. This code case was invoked on HCGS because there are portions of the containment hydrogen recombiners that operate at temperatures in excess of 800°F.

has been submitted under separate cover (letter from R.L. Mittl (PSE&G), to A. Schwencer (NRC), dated July 30, 1984). Because the information is considered to be proprietary, a request to withhold the information from public disclosure is included in the letter.

f. <u>LOP AFTER LOCA SEQUENCING COMPLETED</u>

If a LOCA signal is still present when the SDG circuit breaker is closed, the LOCA signal overrides the LOP sequencer and starts the LOCA sequencer to apply LOCA loads in the predetermined sequence.

For scenarios '2a' through '2f' above, the PSIS signals are present to prevent the inadvertant starting of equipment before its predetermined sequenced time.

ELS TESTING:

Provisions exist at each of the sequencer cabinets to test the ELSs for 2a through 2f scenarios described above. An alarm is provided in the main control room to indicate that an ELS is being tested. If an actual LOP or LOCA occurs during the testing of an ELS, the sequencer resets automatically and responds to LOP and/or LOCA event.

The ELS system reliability analysis ~~IS FURNISHED~~ has been provided UNDER A SEPARATE COVER. THE ELS SYSTEM RELIABILITY IS ENHANCED BY THE USE OF TWO REDUNDANT MICROPROCESSORS IN EACH OF THE FOUR ELS SYSTEMS.

(See letter; R.L. Mittl (PSEG) to A. Schwencer (NRC), "DSER Open Item Status", dated August 1, 1984)