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SCE&G®
A SCANA COMPANY

July 24, 2001
RC-01-0135

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Ms. K. R. Cotton

Gentlemen:

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50/395
TECHNICAL SPECIFICATION AMENDMENT REQUEST TSP 99-0090
SPENT FUEL POOL STORAGE EXPANSION

South Carolina Electric & Gas Company (SCE&G), acting for itself and as agent for South Carolina Public Service Authority, hereby requests an amendment to the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS). This request is being submitted pursuant to 10 CFR 50.90.

This proposed change will increase the spent fuel pool storage capacity by replacing all eleven existing rack modules with twelve new high density storage racks. These new storage racks will be manufactured and installed by Holtec International.

VCSNS is projected to lose full core offload capacity in the Spent Fuel Pool following Cycle 17, which ends in Spring 2008. The rerack will increase the storage capacity from 1,276 storage cells to 1,712 storage cells. The degrading Boraflex neutron absorbing material in the existing racks will be replaced by Boral material that will be used in the new racks. This additional storage capacity will allow continued full core offload capability through the end of Cycle 24, in 2018, without any restrictions from spent fuel storage capacity limitations.

SCE&G desires that this amendment request be approved by August 30, 2002, to permit implementation of the change, prior to the commencement of rack installation, scheduled to start September 30, 2002. Based on the installation schedule, completion of the installation phase will be completed in time for Refuel 14, scheduled to start in Fall 2003.

There are no commitments made in this Technical Specification change request.

There are significant changes required to be made to the FSAR sections. FSAR Sections 9.1 and 15 were reviewed. Changes to the Sections will be implemented, as appropriate, upon approval of this request. The FPER was reviewed but was not affected.

Other TS changes in review that will affect or be affected by this change request are TSP 99-0263 and TSP 00-0041. Should these changes be approved prior to the approval of this amendment request, corrected pages (3/4 7-40 [3/4 9-12], 3/4 7-41 [3/4 9-13], B 3/4 9-2) will be submitted for inclusion into this package.

This proposed amendment has been reviewed and approved by the Plant Safety Review Committee and the Nuclear Safety Review Committee.

A001

The TS amendment request is contained in the following attachments:

Attachment I	Explanation of Changes Summary and Affected Pages
Attachment II	Safety Evaluation
Attachment III	No Significant Hazards Evaluation
Attachment IV	Commitments to Ensure Equipment Operability
Attachment V	Affidavit per 10 CFR 2.790
Attachment VI	"Spent Fuel Storage Expansion Report," Proprietary version
Attachment VII	"Spent Fuel Storage Expansion Report," non-Proprietary version

A copy of this application and associated attachments is being provided to the designated South Carolina State official in accordance with 10 CFR 50.91.

Should you have questions, please call Mr. Philip A. Rose at (803) 345-4052.

I certify under penalty of perjury that the foregoing is true and correct.

Very truly yours,

Gregory Halnon for SAB

Stephen A. Byrne

PAR/SAB/dr
Attachments (7)

c: N. O. Lorick (w/o Attachments V, VI, VII)
N. S. Carns
T. G. Eppink (w/o Attachments)
R. J. White
NRC Resident Inspector
W. R. Higgins
L. A. Reyes
K. M. Sutton
T. P. O'Kelley
S. H. Pellet (w/o Attachments V, VI, VII)
W. Herwig (w/o Attachments V, VI, VII)
D. V. Bryson (w/o Attachments V, VI, VII)
D. D. Kraus (w/o Attachments V, VI, VII)
P. Ledbetter (w/o Attachments V, VI, VII)
RTS (TSP 99-0090)
File (813.20)
DMS (RC-01-0135)

STATE OF SOUTH CAROLINA :
: **TO WIT :**
COUNTY OF FAIRFIELD :

I hereby certify that on the 24th day of July 2001, before me, the subscriber, a Notary Public of the State of South Carolina personally appeared Gregory H. Halnon, being duly sworn, and states that he is General Manager, Nuclear Plant Operations of the South Carolina Electric & Gas Company, a corporation of the State of South Carolina, that he provides the foregoing response for the purposes therein set forth, that the statements made are true and correct to the best of his knowledge, information, and belief, and that he was authorized to provide the response on behalf of said Corporation.

WITNESS my Hand and Notarial Seal

Notary Public

My Commission Expires

July 13, 2005
Date

Attachment To License Amendment No. XXX
To Facility Operating License No. NPF-12
Docket No. 50-395

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
VIII	VIII
X	X
XIV	XIV
XV	XV
NA	3/4 7-39
NA	3/4 7-40
NA	3/4 7-41
NA	3/4 7-42
NA	3/4 7-43
NA	3/4 7-44
3/4 9-3	3/4 9-3
NA	3/4 9-3a
3/4 9-11	NA
3/4 9-12	NA
3/4 9-13	NA
3/4 9-14	NA
3/4 9-15	NA
3/4 9-16	NA
B 3/4 7-6	B 3/4 7-6
N/A	B 3/4 7-7
B 3/4 9-1	B 3/4 9-1
B 3/4 9-2	B 3/4 9-2
B 3/4 9-3	N/A
5-6	5-6
5-7	5-7
5-8	5-8
5-9	5-9

SCE&G -- EXPLANATION OF CHANGES SUMMARY

Page	Affected Section	Bar #	Description of Change	Reason for Change
Index VIII	3/4.7.10, 3/4.7.11, 3/4.7.12, 3/4.7.13	1	Sections 3/4.7.10, 3/4.7.11, and 3/4.7.12 moved from Section 3/4.9. Section 3/4.7.13 is new specification.	
Index X	3/4.9.10, 3/4.9.11, 3/4.9.12	1	Sections 3/4.9.10, 3/4.9.11, and 3/4.9.12 moved to Section 3/4.7.	
Index XIV	3/4.7.10, 3/4.7.11, 3/4.7.12, 3/4.7.13	1	Bases Sections 3/4.7.10, 3/4.7.11, and 3/4.7.12 moved from Bases Section 3/4.9. Bases Section 3/4.7.13 is new specification.	
Index XV	3/4.9.6	1	3/4.9.6 Manipulator Crane moved to Page B 3/4 9-2 due to pagination.	
	3/4.9.10, 3/4.9.11, 3/4.9.12	2	Bases Sections 3/4.9.10, 3/4.9.11, and 3/4.9.12 moved to Bases Section 3/4.7.	
3/4 7-39	3/4.7.10	1	Move 3/4.9.10 from REFUELING OPERATIONS (3/4.9) to PLANT SYSTEMS (3/4.7)	Remove information that is not specific to Refueling Operations and conform with NUREG 1431 organization.
3/4 7-40	3/4.7.11	1	Move 3/4.9.11 from REFUELING OPERATIONS to PLANT SYSTEMS	Remove information that is not specific to Refueling Operations and conform with NUREG 1431 organization.
3/4 7-41	3/4.7.11	1	Move 3/4.9.11 from REFUELING OPERATIONS to PLANT SYSTEMS	Remove information that is not specific to Refueling Operations and conform with NUREG 1431 organization.
3/4 7-42	3/4. 7.12	1	Move 3/4.9.12 from REFUELING OPERATIONS to PLANT SYSTEMS	Remove information that is not specific to Refueling Operations and conform with NUREG 1431 organization.

Page	Affected Section	Bar #	Description of Change	Reason for Change
3/4 7-43	3/4.7.12	1	New figure to define acceptable burn-up vs. initial enrichment requirement for Region 2	Criticality analysis for re-rack revised burn-up figure in section 9.
3/4 7-44	3/4.7.13	1	Add new Specification for boron concentration limit during non-refueling fuel evolutions.	Criticality analysis for re-rack determined need for minimum dissolved boron in the event a fuel handling accident should occur.
3/4 9-3	3/4.9.3	1	Reduce the minimum incore hold time before fuel can begin offloading. Establish correlation to Component Cooling Water temperature	Station desire to assist in shorter outages.
3/4 9-3a	3/4.9.3	1	Adding new figure of incore hold time vs. Component Cooling Water temperature	Provide consistent requirements for refueling operations.
3/4 9-11	3/4.9.10	1	Delete specification	Moving specification to Section 3/4.7.
3/4 9-12	3/4.9.11	1	Delete specification	Moving specification to Section 3/4.7.
3/4 9-13	3/4.9.11	1	Delete specification	Moving specification to Section 3/4.7.
3/4 9-14	3/4.9.12	1	Delete specification	Moving specification to Section 3/4.7.
3/4 9-15	3/4.9.12	1	Delete Figure 3.9-1, Replace with Figure 3.7-1	Criticality analysis revised requiring revision to figure.
3/4 9-16	3/4.9.12	1	Delete Figure 3.9-2	Re-rack project eliminates Region 3 storage.
B 3/4 7-6	B 3/4.7.10	1	Moved Bases from B 3/4.9.10 and B 3/4.9.11	Remove information that is not specific to Refueling Operations and conform with NUREG 1431 organization.

Page	Affected Section	Bar #	Description of Change	Reason for Change
B 3/4 7-7	B 3/4.7.12	1	Moved Bases from B 3/4.9.12 and added new B 3/4.7.13.	Remove information that is not specific to Refueling Operations and conform with NUREG 1431 organization. Added new bases B 3/4.7.13 to provide bases for new specification requirements.
B 3/4 9-1	B 3/4.9.3	1	Revised Bases for minimum incore hold time of 72 hours	Provide justification for incore hold time as a function of Component Cooling Water temperature.
B 3/4 9-2	B 3/4.9.9 B 3/4.9.10 B 3/4.9.11	1 2	Revised title. Delete Bases and move to B 3/4.7	Remove information that is not specific to Refueling Operations and conform with NUREG 1431 organization.
B 3/4 9-3	B 3/4.9.12	1	Delete Bases and move to B 3/4.7; deleted page.	Remove information that is not specific to Refueling Operations and conform with NUREG 1431 organization.
5-6	5.3.1	1	Revise maximum nominal fuel enrichment to 4.95 w/o.	Criticality analysis for new fuel storage racks assumes maximum nominal enrichment of 4.95 w/o U-235.
5-7	5.6.1.1	1	Replace Section with revised Section	Provide Spent Fuel Pool information as affected by the Re-Rack project.
	5.6.3	2	Changed storage capacities, 1276 to 1712 cells and moved information to Page 5-7. Left page blank due to Page 5-10.	Revised to reflect re-racked storage capacities.
5-8	Figure 5.6.1	1	Delete Figure 5.6-1; left page blank due to continued page 5-10.	Does not apply to Re-Racked pool Region 1.
5-9			Deleted page; moved information to Page 5-7	Pagination

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

INDEX

BASES

SECTION | PAGE

INDEX

BASES

Delete
moved
To BASES
SECtion 3/4.7

3/4.~~9~~7.10 WATER LEVEL-SPENT FUEL POOL

LIMITING CONDITION FOR OPERATION

3.~~9~~7.10 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

SURVEILLANCE REQUIREMENTS

4.~~9~~7.10 The water level in the spent fuel pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the spent fuel pool.

PLANT SYSTEMS

~~REFUELING OPERATIONS~~

3/4. ~~9~~ 7.11 SPENT FUEL POOL VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3. ~~9~~ 7.11 Two independent spent fuel pool ventilation sub-systems shall be OPERABLE with at least one sub-system in operation.

APPLICABILITY: Whenever irradiated fuel is being moved in the spent fuel pool and during crane operation with loads over the pool.

ACTION:

a. With one spent fuel pool ventilation sub-system inoperable, fuel movement within the spent fuel pool or crane operation with loads over the spent fuel pool may proceed provided the OPERABLE spent fuel pool ventilation sub-system is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no spent fuel pool ventilation sub-system OPERABLE, suspend all operations involving movement of fuel within the spent fuel pool or crane operation with loads over the spent fuel pool.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4. ~~9~~ 7.11 The above required spent fuel pool ventilation sub-systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that each sub-system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 30,000 cfm ± 10%.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989, at a relative humidity of 95% and 30°C with a methyl iodide penetration of <2.5%.

3. Verifying a system flow rate of 30,000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. Prior to the movement of fuel or crane operation with loads over the pool by verifying that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989, at a relative humidity of 95% and 30°C with a methyl iodide penetration of <2.5%. Subsequent to each initial analysis (which must be completed prior to fuel movement or crane operation with loads over the pool), during the period of time in which there is to be fuel or crane movement with loads over the pool, verify charcoal adsorber operation every 720 hours by obtaining and analyzing a sample as described above. These subsequent analyses are to be completed within thirty-one (31) days of sample removal.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA and roughing filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 30,000 cfm ± 10%.

2. Verifying that on a loss of offsite power test signal, the system automatically starts.

3. Verifying that the system maintains the spent fuel pool area at a negative pressure greater than or equal to 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 30,000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 30,000 cfm ± 10%.

PLANT SYSTEMS

~~REFUELING OPERATIONS~~

3/4.~~9~~7.12 SPENT FUEL ASSEMBLY STORAGE

LIMITING CONDITION FOR OPERATION

3.~~9~~7.12 The combination of initial enrichment and cumulative burnup for spent fuel assemblies stored in Region~~s~~ 2 ~~and 3~~ shall be within the acceptable domain of Figure ~~3.9-1 for Region 2 and Figure 3.9-2 for Region 3.~~ 3.7-1

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

ACTION:

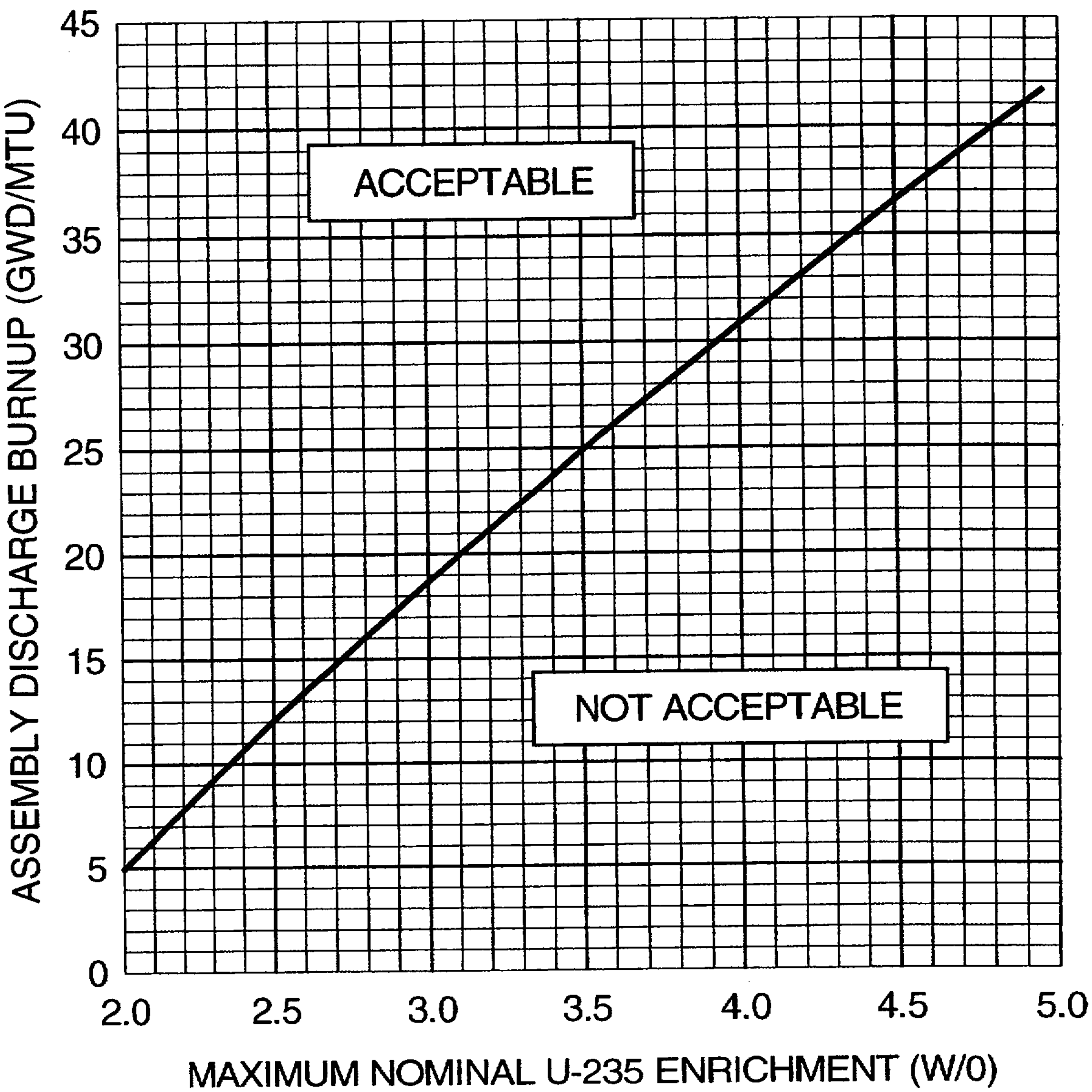
a. With the requirements of the above specification not satisfied, suspend all other movement of fuel assemblies and crane operations with loads in the fuel storage areas and move the non-complying fuel assemblies to Region 1. Until these requirements of the above specification are satisfied, boron concentration of the spent fuel pool shall be verified to be greater than or equal to 2000 ppm at least once per 8 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable

SURVEILLANCE REQUIREMENTS

4.~~9~~7.12 The burnup of each ~~spent~~ fuel assembly stored in Region~~s~~ 2 ~~and 3~~ shall be ascertained by careful analysis of its burnup history prior to storage in Region 2. ~~or 3.~~ A complete record of such analysis shall be kept for the time period that the ~~spent~~ fuel assembly remains in Region ~~2 or 3~~ 2 of the spent fuel pool.

NEW FIGURE



Notes: 1. Fuel assemblies with enrichments less than 2.0 W/0 must meet the burnup requirements of 2.0 W/0 U-235 assemblies.

2. Use of the following polynomial fit is acceptable, where E = Enrichment (W/0):

Assembly Discharge Burnup = $0.1246\,E^3 - 1.91\,E^2 + 20.9205\,E - 30.2482$

FIGURE 3.7-1 REQUIRED FUEL ASSEMBLY BURNUP AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 2

NEW PAGE

PLANT SYSTEMS

3/4.7.13 SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITIONS FOR OPERATION

3.7.13 The boron concentration in the spent fuel pool, the fuel transfer canal, and the cask loading pit shall be maintained at a boron concentration greater than or equal to 500 ppm.

Applicability:

Whenever new or irradiated fuel is being moved (non-refueling movement) in the spent fuel pool, fuel transfer canal, or cask loading pit.

Action:

With the requirements of the above not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel pool, the fuel transfer canal, and the cask loading pit until the boron concentration in the area where fuel is being moved shall be verified greater than or equal to 500 ppm.

Surveillance Requirements:

4.7.13 The boron concentration of the spent fuel pool, fuel transfer canal, or cask loading pit shall be determined by chemical analysis at least once per 72 hours when moving new or irradiated fuel in the spent fuel pool, transfer canal, or cask loading pit.

REFUELING OPERATIONS

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

a period of time within the acceptable domain of Figure 3.9-1, but not less than 72 Hours.

3.9.3 The reactor shall be subcritical for ~~at least 100 hours~~.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than ~~100~~ 72 hours, immediately suspend all ~~operations involving~~ movement of irradiated fuel in the reactor pressure vessel. With the reactor subcritical for greater than 72 Hours but ~~less than~~ not within the acceptable domain of Figure 3.9-1, immediately suspend movement of irradiated fuel in the reactor pressure vessel.

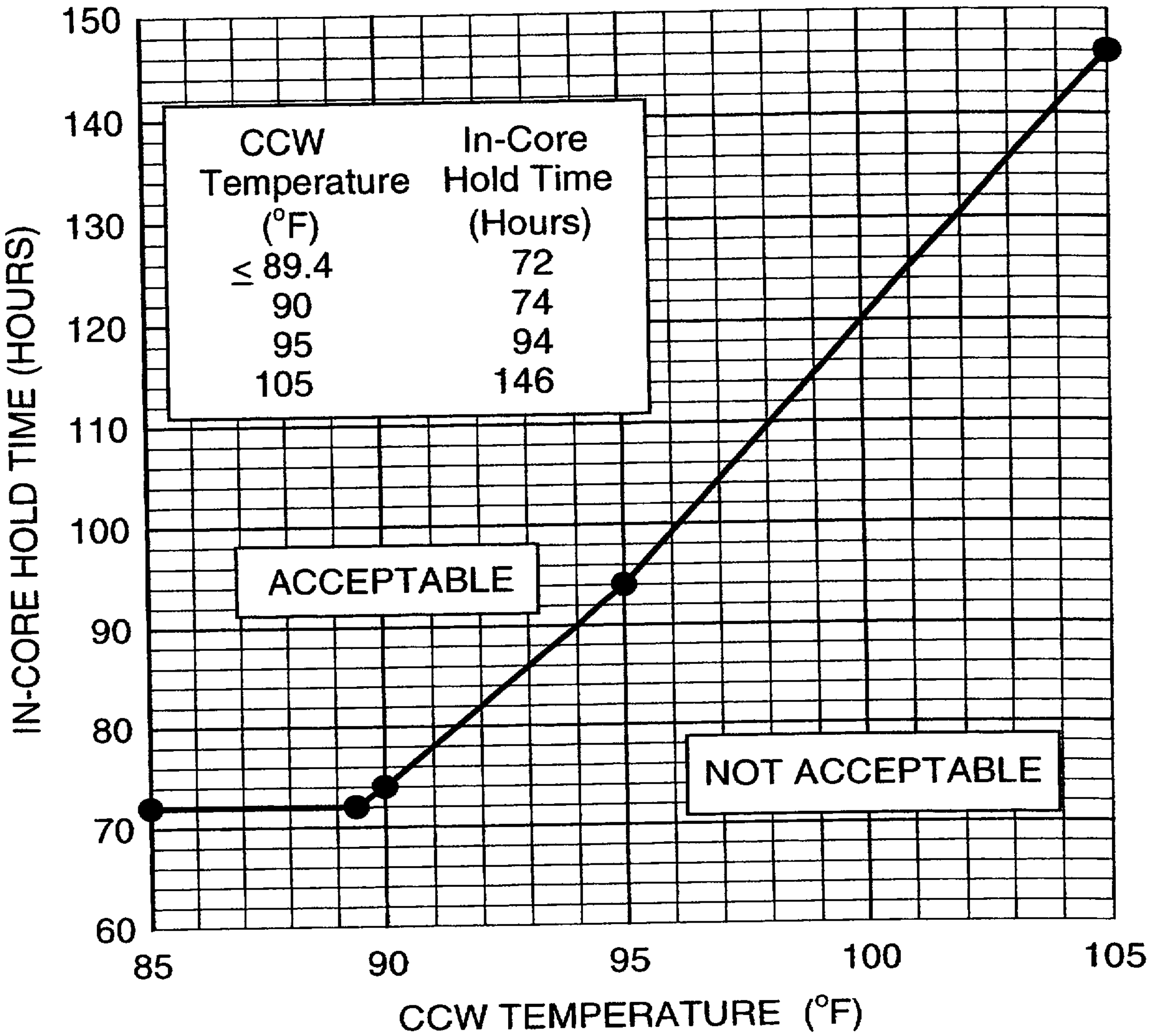
SURVEILLANCE REQUIREMENTS

~~4.9.3 The reactor shall be determined to have been subcritical for at least 100 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.~~

4.9.3.1 The reactor shall be determined to have been subcritical for a period of time within the acceptable domain of Figure 3.9-1 by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

4.9.3.2 Prior to moving irradiated fuel from the reactor pressure vessel, and at least once every 12 hours during movement of irradiated fuel, verify the CCW temperature at the inlet ~~of~~ to the Spent Fuel Pool Cooling System heat exchanger is within the acceptable domain of Figure 3.9-1.

NEW FIGURE



Note: The use of linear interpolation between CCW temperatures reported above is acceptable to determine the minimum incore hold time.

FIGURE 3.9-1 REQUIRED IN-CORE HOLD TIME AS A FUNCTION OF COMPONENT COOLING WATER (CCW) TEMPERATURE

3/4.9.10 WATER LEVEL-SPENT FUEL POOL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level in the spent fuel pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the spent fuel pool.

DELETE 3/4.9.10.
MOVED IN ENTIRETY TO 3/4.7.10

DELETE 3/4.9.11
MOVED IN ENTIRETY TO 3/4.7.11

3/4.9.11 SPENT FUEL POOL VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.11 Two independent spent fuel pool ventilation sub-systems shall be OPERABLE with at least one sub-system in operation.

APPLICABILITY: Whenever irradiated fuel is being moved in the spent fuel pool and during crane operation with loads over the pool.

ACTION:

a. With one spent fuel pool ventilation sub-system inoperable, fuel movement within the spent fuel pool or crane operation with loads over the spent fuel pool may proceed provided the OPERABLE spent fuel pool ventilation sub-system is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no spent fuel pool ventilation sub-system OPERABLE, suspend all operations involving movement of fuel within the spent fuel pool or crane operation with loads over the spent fuel pool.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The above required spent fuel pool ventilation sub-systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that each sub-system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 30,000 cfm ± 10%.

DELETE 3/4.9.11.
MOVED IN ENTIRETY TO 3/4.7.11

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989, at a relative humidity of 95% and 30°C with a methyl iodide penetration of <2.5%.

3. Verifying a system flow rate of 30,000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. Prior to the movement of fuel or crane operation with loads over the pool by verifying that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989, at a relative humidity of 95% and 30°C with a methyl iodide penetration of <2.5%. Subsequent to each initial analysis (which must be completed prior to fuel movement or crane operation with loads over the pool), during the period of time in which there is to be fuel or crane movement with loads over the pool, verify charcoal adsorber operation every 720 hours by obtaining and analyzing a sample as described above. These subsequent analyses are to be completed within thirty-one (31) days of sample removal.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA and roughing filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 30,000 cfm ± 10%.

2. Verifying that on a loss of offsite power test signal, the system automatically starts.

3. Verifying that the system maintains the spent fuel pool area at a negative pressure greater than or equal to 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 30,000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 30,000 cfm ± 10%.

3/4.9.12 SPENT FUEL ASSEMBLY STORAGE

LIMITING CONDITION FOR OPERATION

3.9.12 The combination of initial enrichment and cumulative burnup for spent fuel assemblies stored in Regions 2 and 3 shall be within the acceptable domain of Figure 3.9-1 for Region 2 and Figure 3.9-2 for Region 3.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all other movement of fuel assemblies and crane operations with loads in the fuel storage areas and move the non-complying fuel assemblies to Region 1. Until these requirements of the above specification are satisfied, boron concentration of the spent fuel pool shall be verified to be greater than or equal to 2000 ppm at least once per 8 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable

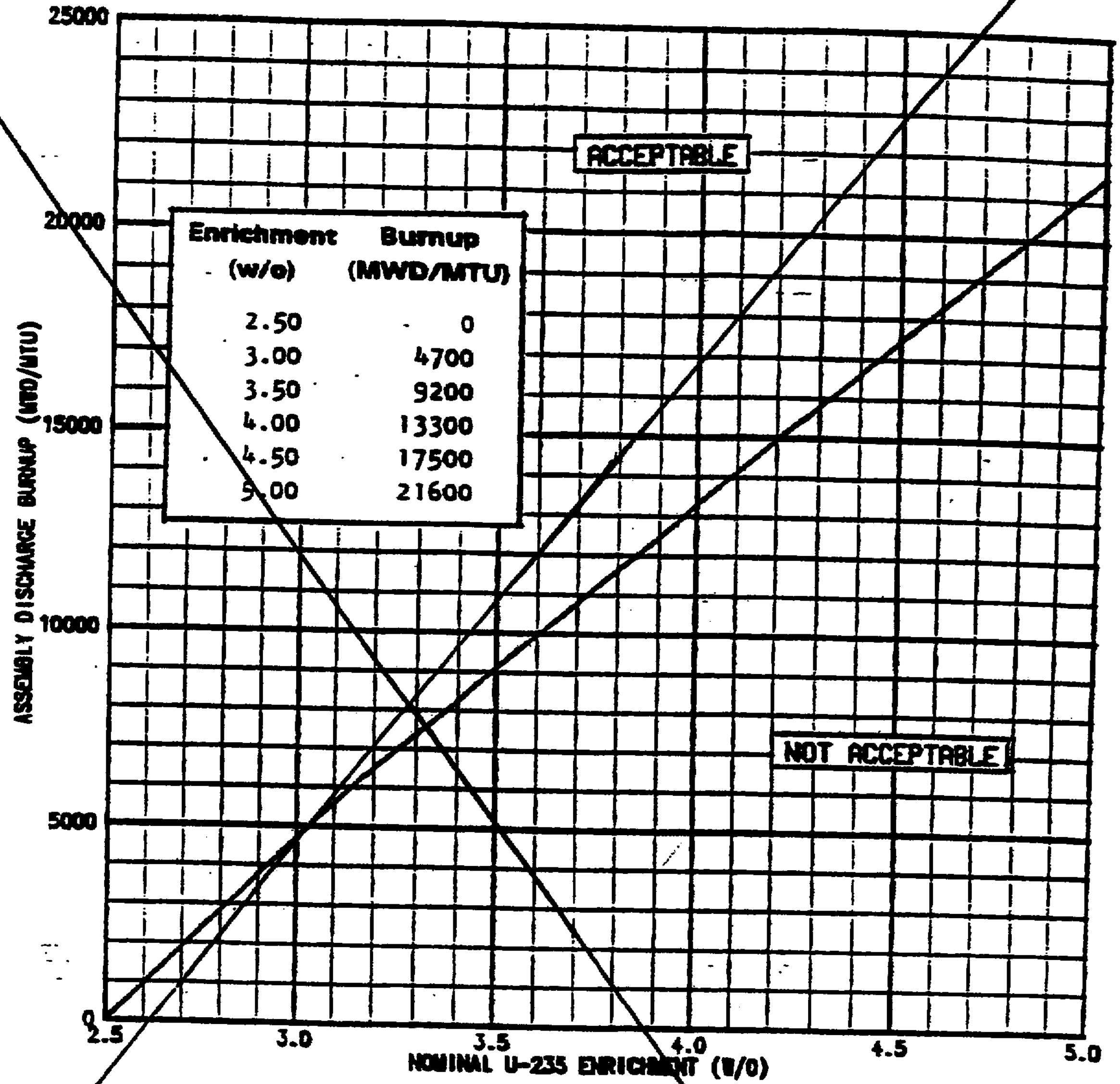
SURVEILLANCE REQUIREMENTS

4.9.12 The burnup of each spent fuel assembly stored in Regions 2 and 3 shall be ascertained by careful analysis of its burnup history prior to storage in Region 2 or 3. A complete record of such analysis shall be kept for the time period that the spent fuel assembly remains in Region 2 or 3 of the spent fuel pool.

DELETE 3/4.9.12.
REVISED AND MOVED TO 3/4.7.12

8

DELETE THIS FIGURE

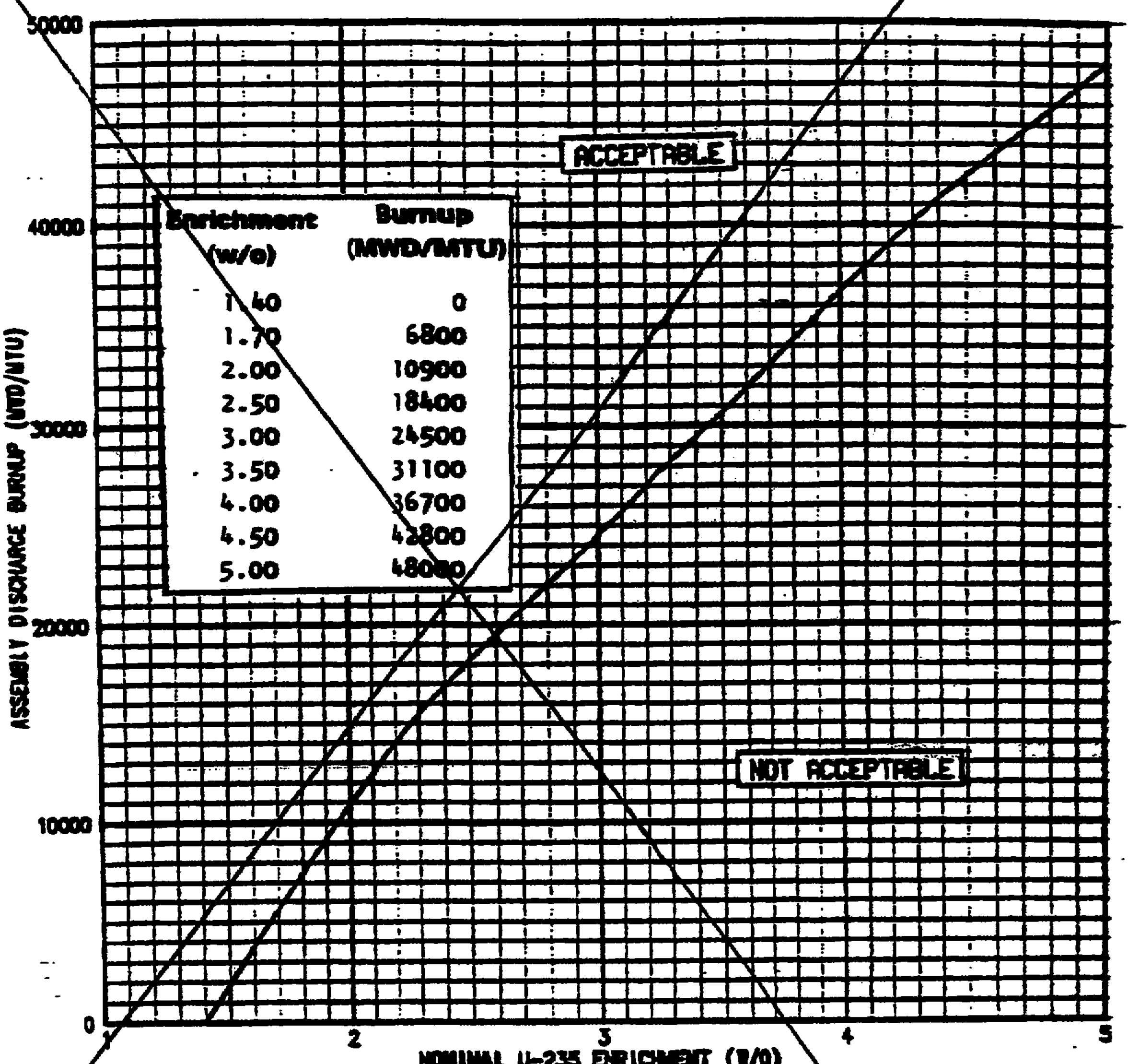


Enrichment (w/o)	Burnup (MWD/MTU)
2.50	0
3.00	4700
3.50	9200
4.00	13300
4.50	17500
5.00	21600

Note: The use of linear interpolation between the minimum burnups reported above is acceptable.

FIGURE 3.9-1 MINIMUM REQUIRED FUEL ASSEMBLY BURNUP AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 2

DELETE THIS FIGURE



Enrichment (w/o)	Burnup (MWD/MTU)
1.40	0
1.70	6800
2.00	10900
2.50	18400
3.00	24500
3.50	31100
4.00	36700
4.50	42800
5.00	48000

Note: The use of linear interpolation between the minimum burnups reported above is acceptable.

FIGURE 3.9-2 MINIMUM REQUIRED FUEL ASSEMBLY BURNUP AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 3

3/4.7.8 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e. sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.9 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of 2°F.

~~3/4.9.9 and~~ 3/4.~~9~~7.10 WATER LEVEL - ~~REACTOR VESSEL and~~ SPENT FUEL POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.~~9~~7.11 SPENT FUEL POOL VENTILATION SYSTEM

The limitations on the spent fuel pool ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

3/4.~~9~~7.12 SPENT FUEL ASSEMBLY STORAGE

~~The restrictions placed on spent fuel assemblies stored in Regions 2 and 3 of the spent fuel pool ensure inadvertent criticality will not occur.~~

The restrictions placed on spent fuel assemblies in Region 2 of the spent fuel pool ensure K_{eff} remains less than 0.95. The minimum burnup bounds the use of Burnable Poison Rod Assemblies (BPRA), Wetted Annular Burnable Absorbers(WABA), Integral Fuel Burnable Absorbers (IFBA), and Erbia.

An axial burnup shape penalty is also included in the burnup requirement.

3/4.7.13 SPENT FUEL POOL BORON CONCENTRATION

A minimum boron concentration is required in the spent fuel pool, fuel transfer canal, or cask loading pit whenever new 4.95 W/0 fuel is being moved to ensure K_{eff} remains less than 0.95 during this normal condition of fuel movement.

The minimum boron concentration in the spent fuel pool, fuel transfer canal, or cask loading pit also is sufficient to maintain K_{eff} less than 0.95 for postulated accident condition consisting of a dropped or a mispositioned fuel assembly.

Sampling to determine boron concentration is required only for those specific areas where fuel is being moved, e.g. in the spent fuel pool, in the fuel transfer canal, or in the cask loading area.

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The value of 0.95 or less for K_{eff} includes a 1 percent delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value of 2000 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron. Valves in the reactor makeup system are required to be closed to minimize the possibility of a boron dilution accident.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

DELETE AND REPLACE WITH INSERT

~~The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.~~

3/4.9.4 REACTOR BUILDING PENETRATIONS

The requirements on reactor building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of reactor building pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

MOVED TO B 3/4 9-2 FOR REPAGINATION

3/4.9.6 MANIPULATOR CRANE

The OPERABILITY requirements for the manipulator cranes ensure that: 1) manipulator cranes will be used for movement of control rods and fuel assemblies, 2) each crane has sufficient load capacity to lift a control rod and fuel assembly,

3/4.9.3 DECAY TIME

The minimum time of 72 hours for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. The minimum decay time of 72 hours is consistent with the assumptions used in the accident analysis.

The tabulated hold times associated with Component Cooling Water (CCW) temperature ensure that the spent fuel heat load is reduced sufficiently to allow the spent fuel pool cooling system to maintain the bulk pool temperature below 170°F. These hold times ensure that adequate cooling is provided to the Spent Fuel Pool under the highest possible heat load conditions. The hold times are based on the performance of the cooling system, which is dependent upon CCW temperature and recognizes that the spent fuel pool cooling system is capable of increased flow rates up to 2400 gpm during single loop operation. This higher flow rate may be required when only a single cooling loop is operable during a refueling outage.

The CCW temperature limits defined in Figure 3.9-1 are adjusted for uncertainty in the implementing procedure.

REFUELING OPERATIONS

BASES

MANIPULATOR CRANE (Continued)

and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal loop be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and 2) sufficient coolant circulation is maintained thru the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.8 REACTOR BUILDING PURGE SUPPLY AND EXHAUST ISOLATION SYSTEM

The OPERABILITY of this system ensures that the reactor building vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the reactor building. The OPERABILITY of this system is required to restrict the release of radioactive material from the reactor building atmosphere to the environment.

3/4.9.9 ~~and 3/4.9.10~~ WATER LEVEL - REACTOR VESSEL ~~and SPENT FUEL POOL~~

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

~~3/4.9.11~~ SPENT FUEL POOL VENTILATION SYSTEM

Delete AND Move To B 3/4 7.11

The limitations on the spent fuel pool ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

DELETE

3/4.9.12 SPENT FUEL ASSEMBLY STORAGE

The restrictions placed on spent fuel assemblies stored in Regions 2 and 3 of the spent fuel pool ensure inadvertent criticality will not occur.

REVISED AND MOVED TO BASES 3/4.7.12

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies. Each fuel assembly shall consist of 264 Zircaloy-4 or ZIRLO(TM) clad fuel rods with an initial composition of uranium dioxide with a maximum nominal enrichment of ~~5.0~~ 4.95 weight percent U-235 as fuel material. Limited substitutions of Zircaloy-4, ZIRLO(TM) and/or stainless steel filler rods for fuel rods, if justified by a cycle specific reload analysis using an NRC-approved methodology, may be used. Fuel assembly configurations shall be limited to those designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or cycle-specific reload analyses to comply with all fuel safety design bases. Reload fuel shall contain sufficient integral fuel burnable absorbers such that the requirements of Specifications 5.6.1.1a.2 and 5.6.1.2.b are met. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core locations.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 48 full length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 9914 ± 100 cubic feet at an indicated T_{avg} of 587.4°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

DELETE AND REPLACE WITH INSERT

5.6.1.1 The spent fuel storage racks consist of 1276 individual cells, each of which accommodates a single assembly. The cells are grouped into 3 regions. The spent fuel storage racks are designed and shall be maintained with a K_{eff} less than or equal to 0.95 when flooded with unborated water, which includes conservative allowances for uncertainties and biases. This is ensured by maintaining the following for each region:

a. REGION 1 - designated for storage of fresh fuel assemblies and freshly discharged fuel assemblies.

   1. A nominal 10.4025 inch center-to-center distance between fuel assemblies placed in the storage rack.

   2. A nominal enrichment of 5.0 weight percent U-235 with a minimum number of integral fuel burnable absorbers as shown on Figure 5.6-1. The Integral Fuel Burnable Absorbers (IFBA) rod requirements shown in Figure 5.6-1 are based on a nominal IFBA linear B^{10} loading of 1.50 mg-B^{10}/inch (1.0X). For higher IFBA loadings up to 3.00 mg-B^{10}/inch (2.0X), the required number of IFBA rods may be reduced by the ratio of the increased B^{10} loading to the nominal 1.50 mg-B^{10}/inch loading. The poison length of the IFBA rods is greater than or equal to 108 inches.

b. REGION 2 - designated for storage of discharged fuel assemblies.

   1. A nominal 10.4025 x 10.1875 inch center-to-center distance between fuel assemblies placed in the storage rack.

   2. A maximum nominal enrichment of 2.5 weight percent U-235 with no burnup and up to 5.0 weight percent U-235 with a minimum burnup of up to 21,600 MWD/MTU, as specified in Figure 3.9-1.

c. REGION 3 - designated for storage of discharged fuel assemblies.

   1. A nominal 10.116 inch center-to-center distance between fuel assemblies placed in the storage rack.

   2. A maximum nominal enrichment of 1.4 weight percent U-235 with no burnup and up to 5.0 weight percent U-235 with a minimum burnup of up to 48,000 MWD/MTU, as specified in Figure 3.9-2.

5.6.1.2 The new fuel storage racks consist of 60 individual cells, each of which accommodates a single assembly. The new fuel pit storage racks are designed and shall be maintained with a K_{eff} less than or equal to 0.95 when flooded with unborated water and less than or equal to 0.98 for low density optimum moderation conditions, including conservative allowances for uncertainties and biases. This is ensured by maintaining:

a. A nominal 21 inch center-to-center distance between new fuel assemblies placed in the storage rack.

b. A nominal enrichment of 5.0 weight percent U-235.

Insert for 5.6.1.1

5.6.1.1 The spent fuel storage racks consist of 1712 individual storage cells. The cells are grouped into two regions, which are determined based on storage cell spacing as defined below. The spent fuel storage racks are designed, and shall be maintained, with a K_{eff} less than or equal to 0.95 when flooded with unborated water, which includes conservative allowances for uncertainties and biases. This is ensured by maintaining the following for each region:

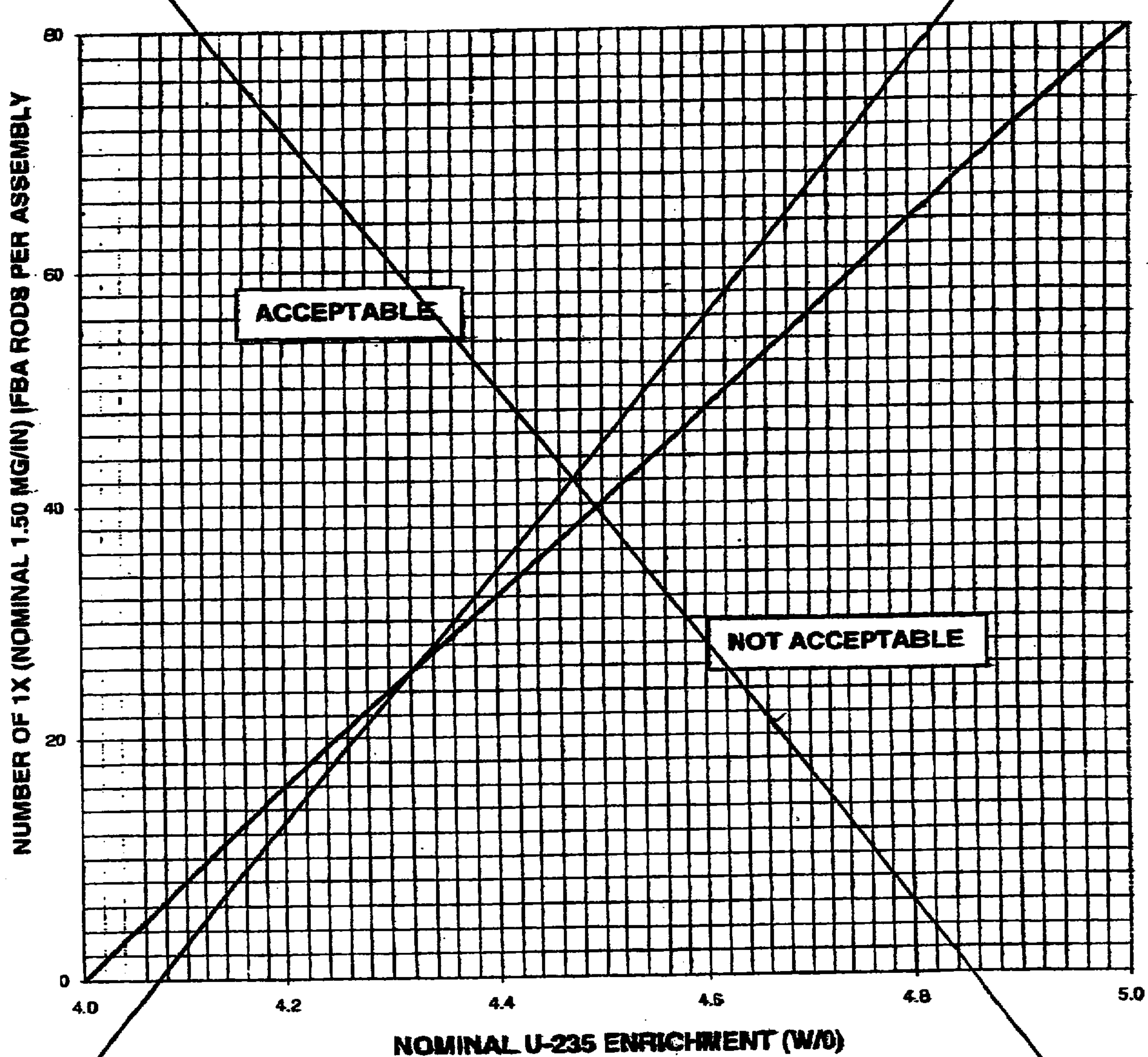
a. Region 1- designated for storage of fresh fuel assemblies and fuel assemblies with a cumulative burnup less than the required cumulative burnup for storage in Region 2.

   1. A nominal 10.867 inch center-to-center distance between fuel assemblies placed in the storage rack.
   2. A maximum nominal initial enrichment of 4.95 weight percent U-235.

b. Region 2 – designated for storage of discharged fuel assemblies.

   1. A nominal 9.07 inch center-to-center distaqnce between fuel assemblies placed in the storage rack.
   2. A cumulative burnup within the acceptable domain defined by Figure 3.7-1.

DELETE FIGURE 5.6-1



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION
Region 1 Minimum
IFBA Requirements
FIGURE 5.6-1

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DRAINAGE

5.6.2 The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 460'3".

CAPACITY

5.6.3 The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than ~~1276~~ 1712 fuel assemblies, ~~242 in Region 1, 99 in Region 2, and 935 in Region 3.~~ with 200 assemblies in Region 1 and 1512 assemblies in Region 2.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

BASES

3/4.7.10 WATER LEVEL-SPENT FUEL POOL

LIMITING CONDITION FOR OPERATION

3.7.10 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

SURVEILLANCE REQUIREMENTS

4.7.10 The water level in the spent fuel pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the spent fuel pool.

PLANT SYSTEMS

3/4.7.11 SPENT FUEL POOL VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.11 Two independent spent fuel pool ventilation sub-systems shall be OPERABLE with at least one sub-system in operation.

APPLICABILITY: Whenever irradiated fuel is being moved in the spent fuel pool and during crane operation with loads over the pool.

ACTION:

a. With one spent fuel pool ventilation sub-system inoperable, fuel movement within the spent fuel pool or crane operation with loads over the spent fuel pool may proceed provided the OPERABLE spent fuel pool ventilation sub-system is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no spent fuel pool ventilation sub-system OPERABLE, suspend all operations involving movement of fuel within the spent fuel pool or crane operation with loads over the spent fuel pool.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11 The above required spent fuel pool ventilation sub-systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that each sub-system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 30,000 cfm ± 10%.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989, at a relative humidity of 95% and 30°C with a methyl iodide penetration of <2.5%.

3. Verifying a system flow rate of 30,000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. Prior to the movement of fuel or crane operation with loads over the pool by verifying that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989, at a relative humidity of 95% and 30°C with a methyl iodide penetration of <2.5%. Subsequent to each initial analysis (which must be completed prior to fuel movement or crane operation with loads over the pool), during the period of time in which there is to be fuel or crane movement with loads over the pool, verify charcoal adsorber operation every 720 hours by obtaining and analyzing a sample as described above. These subsequent analyses are to be completed within thirty-one (31) days of sample removal.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA and roughing filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 30,000 cfm ± 10%.

2. Verifying that on a loss of offsite power test signal, the system automatically starts.

3. Verifying that the system maintains the spent fuel pool area at a negative pressure greater than or equal to 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 30,000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 30,000 cfm ± 10%.

3/4.7.12 SPENT FUEL ASSEMBLY STORAGE

LIMITING CONDITION FOR OPERATION

3.7.12 The combination of initial enrichment and cumulative burnup for spent fuel assemblies stored in Region 2 shall be within the acceptable domain of Figure 3.7-1.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

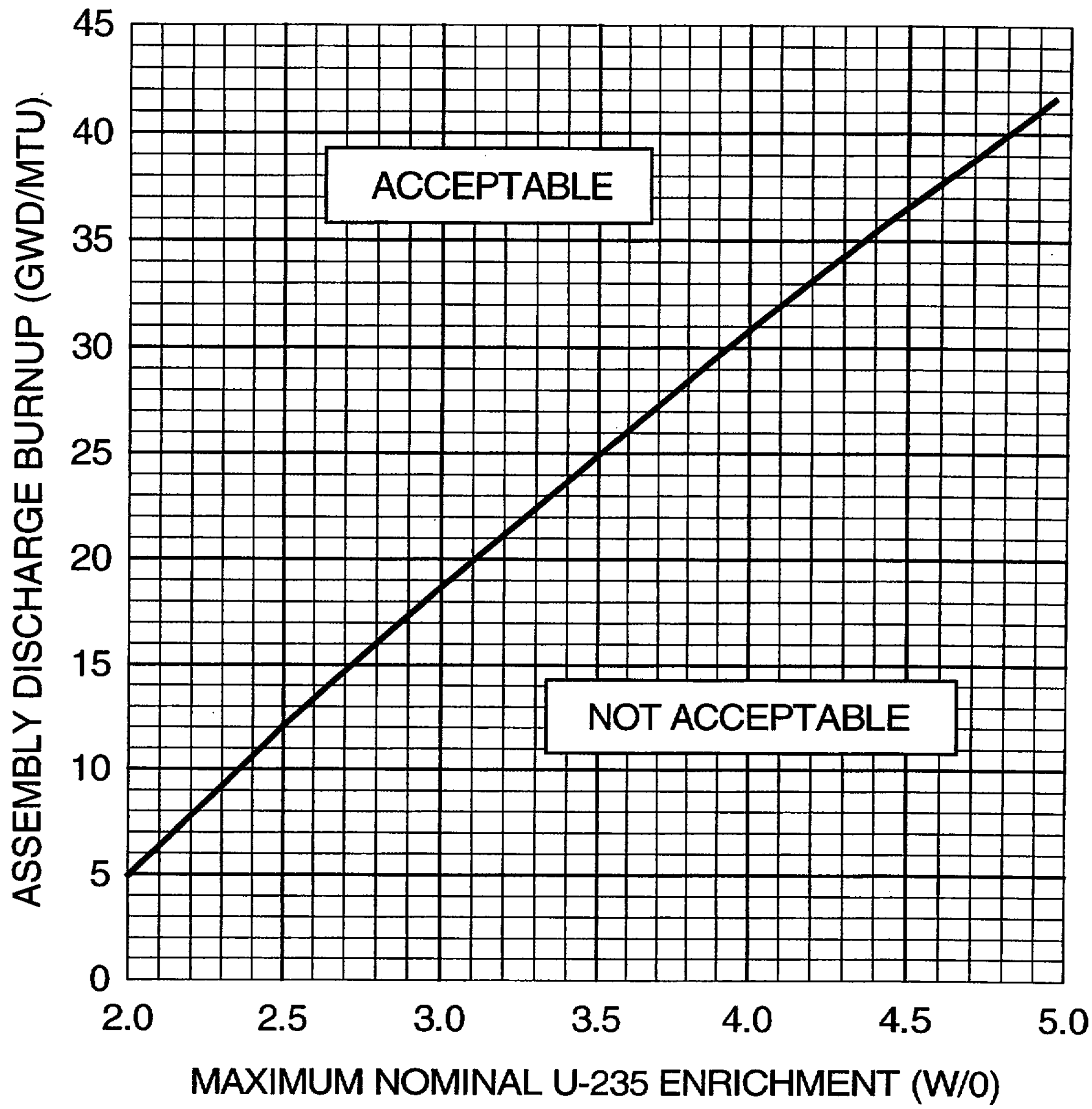
ACTION:

a. With the requirements of the above specification not satisfied, suspend all other movement of fuel assemblies and crane operations with loads in the fuel storage areas and move the non-complying fuel assemblies to Region 1. Until these requirements of the above specification are satisfied, boron concentration of the spent fuel pool shall be verified to be greater than or equal to 2000 ppm at least once per 8 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.12 The burnup of each fuel assembly stored in Region 2 shall be ascertained by careful analysis of its burnup history prior to storage in Region 2. A complete record of such analysis shall be kept for the time period that the fuel assembly remains in Region 2 of the spent fuel pool.



Notes: 1. Fuel assemblies with enrichments less than 2.0 W/0 must meet the burn-up requirements of 2.0 W/0 U-235 assemblies.

2. Use of the following polynomial fit is acceptable, where E = Enrichment (W/0):

Assembly Discharge Burnup = $0.1246\ E^3 - 1.91\ E^2 + 20.9205\ E - 30.2482$

FIGURE 3.7-1 REQUIRED FUEL ASSEMBLY BURN-UP AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 2

3/4.7.13 SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.7.13 The boron concentration in the spent fuel pool, the fuel transfer canal, and the cask loading pit shall be maintained at a boron concentration greater than or equal to 500 ppm.

APPLICABILITY: Whenever new or irradiated fuel is being moved (non-refueling movement) in the spent fuel pool, fuel transfer canal, or cask loading pit.

ACTION:

With the requirements of the above not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel pool, the fuel transfer canal, and the cask loading pit until the boron concentration in the area where fuel is being moved shall be verified greater than or equal to 500 ppm.

SURVEILLANCE REQUIREMENTS

4.7.13 The boron concentration of the spent fuel pool, fuel transfer canal, or cask loading pit shall be determined by chemical analysis at least once per 72 hours when moving new or irradiated fuel in the spent fuel pool, transfer canal, or cask loading pit.

REFUELING OPERATIONS

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical a period of time within the acceptable domain of Figure 3.9-1, but not less than 72 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

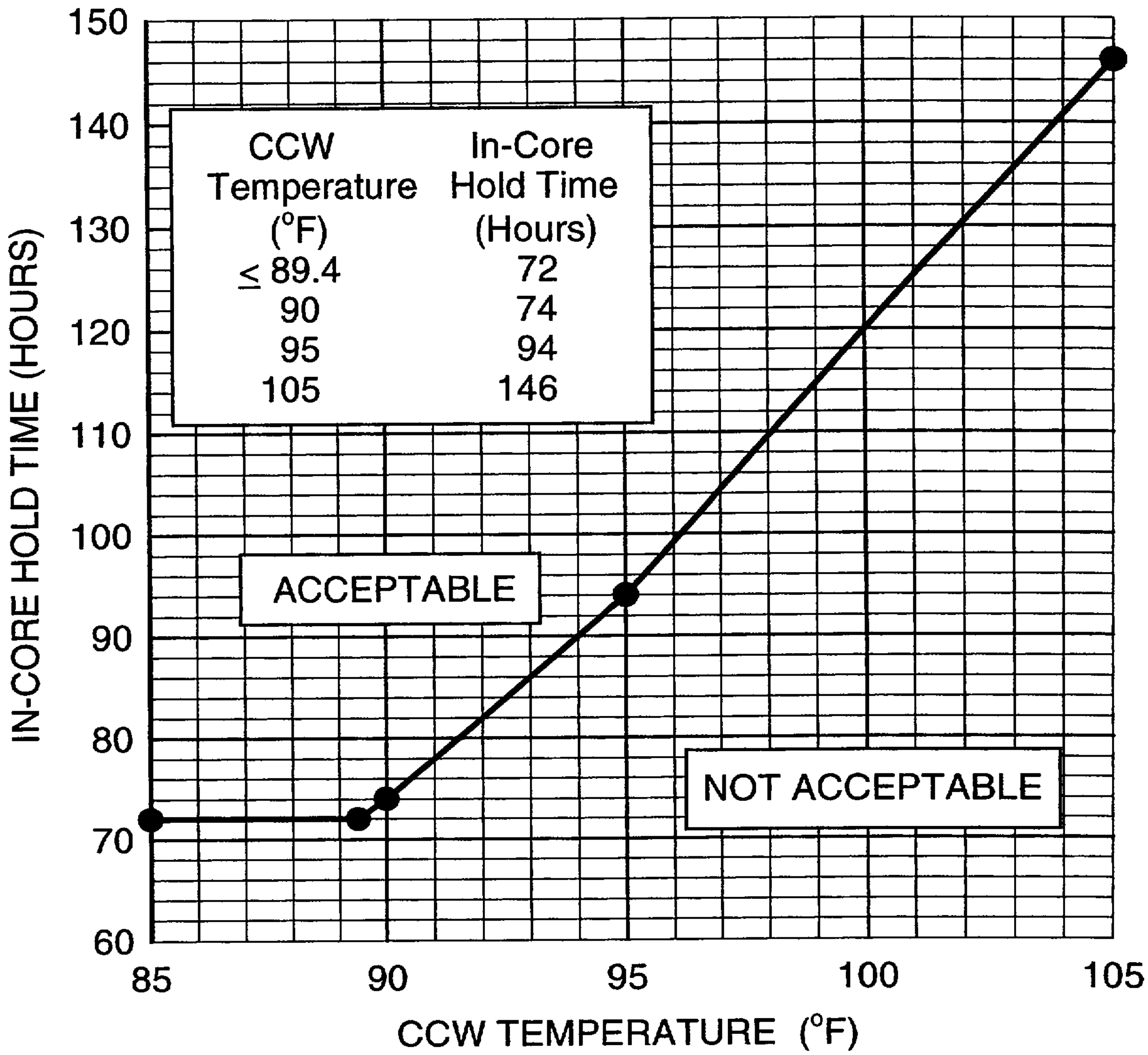
ACTION:

With the reactor subcritical for less than 72 hours, immediately suspend all movement of irradiated fuel in the reactor pressure vessel. With the reactor subcritical for greater than 72 hours but not within the acceptable domain of Figure 3.9-1, immediately suspend movement of irradiated fuel in the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.3.1 The reactor shall be determined to have been subcritical for a period of time within the acceptable domain of Figure 3.9-1 by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

4.9.3.2 Prior to moving irradiated fuel from the reactor pressure vessel, and at least once every 12 hours during movement of irradiated fuel, verify the CCW temperature at the inlet to the Spent Fuel Pool Cooling System heat exchanger is within the acceptable domain of Figure 3.9-1.



Note: The use of linear interpolation between CCW temperatures reported above is acceptable to determine the minimum incore hold time.

FIGURE 3.9-1 REQUIRED IN-CORE HOLD TIME AS A FUNCTION OF COMPONENT COOLING WATER (CCW) TEMPERATURE

3/4.7.8 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e. sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.9 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of 2°F.

3/4.7.10 WATER LEVEL - SPENT FUEL POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.7.11 SPENT FUEL POOL VENTILATION SYSTEM

The limitations on the spent fuel pool ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analysis.

3/4.7.12 SPENT FUEL ASSEMBLY STORAGE

The restrictions placed on spent fuel assemblies in Region 2 of the spent fuel pool ensure K_{eff} remains less than 0.95. The minimum burnup bounds the use of Burnable Poison Rod Assemblies (BPRA), Wetted Annular Burnable Absorbers (WABA), Integral Fuel Burnable Absorbers (IFBA), and Erbia.

An axial burnup shape penalty is also included in the burnup requirement.

3/4.7.13 SPENT FUEL POOL BORON CONCENTRATION

A minimum boron concentration is required in the spent fuel pool, fuel transfer canal, or cask loading pit whenever new 4.95 W/0 fuel is being moved to ensure K_{eff} remains less than 0.95 during this normal condition of fuel movement.

The minimum boron concentration in the spent fuel pool, fuel transfer canal, or cask loading pit also is sufficient to maintain K_{eff} less than 0.95 for the postulated accident conditions consisting of a dropped or a mispositioned fuel assembly.

Sampling to determine boron concentration is required only for those specific areas where fuel is being moved, e.g. in the spent fuel pool, in the fuel transfer canal, or in the cask loading area.

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The value of 0.95 or less for K_{eff} includes a 1 percent delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value of 2000 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron. Valves in the reactor makeup system are required to be closed to minimize the possibility of a boron dilution accident.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum time of 72 hours for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. The minimum decay time of 72 hours is consistent with the assumptions used in the accident analysis.

The tabulated hold times associated with Component Cooling Water (CCW) temperature ensure that the spent fuel heat load is reduced sufficiently to allow the spent fuel pool cooling system to maintain the bulk pool temperature below 170°F. These hold times ensure that adequate cooling is provided to the Spent Fuel Pool under the highest possible heat load conditions. The hold times are based on the performance of the cooling system, which is dependent upon CCW temperature and recognizes that the spent fuel pool cooling system is capable of increased flow rates up to 2400 gpm during single loop operation. This higher flow rate may be required when only a single cooling loop is operable during a refueling outage.

The CCW temperature limits defined in Figure 3.9-1 are adjusted for uncertainty in the implementing procedure.

3/4.9.4 REACTOR BUILDING PENETRATIONS

The requirements on reactor building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of reactor building pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

The OPERABILITY requirements for the manipulator cranes ensure that:
1) manipulator cranes will be used for movement of control rods and fuel assemblies,
2) each crane has sufficient load capacity to lift a control rod and fuel assembly, and
3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal loop be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and 2) sufficient coolant circulation is maintained thru the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.8 REACTOR BUILDING PURGE SUPPLY AND EXHAUST ISOLATION SYSTEM

The OPERABILITY of this system ensures that the reactor building vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the reactor building. The OPERABILITY of this system is required to restrict the release of radioactive material from the reactor building atmosphere to the environment.

3/4.9.9 WATER LEVEL - REACTOR VESSEL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies. Each fuel assembly shall consist of 264 Zircaloy-4 or ZIRLO(TM) clad fuel rods with an initial composition of uranium dioxide with a maximum nominal enrichment of 4.95 weight percent U-235 as fuel material. Limited substitutions of Zircaloy-4, ZIRLO(TM) and/or stainless steel filler rods for fuel rods, if justified by a cycle specific reload analysis using an NRC-approved methodology, may be used. Fuel assembly configurations shall be limited to those designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or cycle-specific reload analyses to comply with all fuel safety design bases. Reload fuel shall contain sufficient integral fuel burnable absorbers such that the requirements of Specifications 5.6.1.1a.2 and 5.6.1.2.b are met. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core locations.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 48 full length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 9914 ± 100 cubic feet at an indicated T_{avg} of 587.4°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks consist of 1712 individual storage cells. The cells are grouped into two regions, which are determined based on storage cell spacing as defined below. The spent fuel storage racks are designed, and shall be maintained, with a K_{eff} less than or equal to 0.95 when flooded with unborated water, which includes conservative allowances for uncertainties and biases. This is ensured by maintaining the following for each region:

a. REGION 1 - designated for storage of fresh fuel assemblies and fuel assemblies with a cumulative burnup less than the required cumulative burnup for storage in Region 2.

   1. A nominal 10.867 inch center-to-center distance between fuel assemblies placed in the storage rack.
   2. A maximum nominal initial enrichment of 4.95 weight percent U-235.

b. REGION 2 - designated for storage of discharged fuel assemblies.

   1. A nominal 9.07 inch center-to-center distance between fuel assemblies placed in the storage rack.
   2. A cumulative burnup with the acceptable domain defined by Figure 3.7-1.

5.6.1.2 The new fuel storage racks consist of 60 individual cells, each of which accommodates a single assembly. The new fuel pit storage racks are designed and shall be maintained with a K_{eff} less than or equal to 0.95 when flooded with unborated water and less than or equal to 0.98 for low density optimum moderation conditions, including conservative allowances for uncertainties and biases. This is ensured by maintaining:

a. A nominal 21 inch center-to-center distance between new fuel assemblies placed in the storage rack.

b. A nominal enrichment of 5.0 weight percent U-235.

DRAINAGE

5.6.2 The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 460'3".

CAPACITY

5.6.3 The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 1712 fuel assemblies, with 200 assemblies in Region 1 and 1512 assemblies in Region 2.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

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SAFETY EVALUATION
FOR SPENT FUEL POOL STORAGE EXPANSION
THE VIRGIL C. SUMMER NUCLEAR STATION
TECHNICAL SPECIFICATIONS

Description of Amendment Request

The Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS), are being revised to permit increased storage (re-racking) in the Spent Fuel Storage pool. This change request also affects the storage requirements as well as the incore hold time before fuel can be moved out of the reactor vessel.

The expansion will increase the total storage space from 1276 to 1712 storage cells, which is an increase of 436 cells and will extend the capability for full core offload from 2008 to 2018. Only two rack types will be used, as opposed to the three types currently in the pool. Region 1 will allow for storage (200 cells) for enrichments up to 4.95 nominal weight percent (w/o) U-235 without regard to fuel burn-up. Region 2 will allow storage of assemblies (1512 cells) that meet minimum burn-up requirements for unrestricted storage. Boral will be used as the active neutron absorbing poison in both regions.

A minimum boron concentration of 500 ppm is proposed that will ensure reactivity is maintained less than design limits in the event of a fuel handling accident anytime during the cycle but is specifically applicable to non-refueling outage evolutions.

A minimum incore hold time of 72 hours is proposed which will be a significant contributor to shorter outages. This minimum hold time is directly related to the temperature of the Component Cooling Water (CCW) and the capacity to remove the decay heat generated while in the reactor.

Additionally, Specifications 3.9.10, 3.9.11, and 3.9.12 are to be moved out of the Refueling Operations Section (3.9) and into the Plant Systems Section (3.7) since they are not specific to refueling operations and to conform to the Improved Standardized Technical Specifications – NUREG 1431.

Background

V. C. Summer Nuclear Station currently expects to lose the capacity for full core offload during refueling operations in 2008 (after Cycle 17). It is not likely that the Department of Energy will have a facility open in time to prevent this loss of capability.

SCE&G has evaluated spent fuel storage options that have been licensed by the NRC and which are currently feasible for use at the VCSNS site. The evaluation concluded that re-racking the spent fuel pool is currently the most cost effective alternative. Re-racking would provide an increase in storage capacity, which would maintain the plant's capability to accommodate a full core discharge, until the end of Cycle 24 in 2018.

The proposed change would remove the 11 current storage racks and replace them with 12 higher density racks. Fuel shuffling during the effort will require that one rack be temporarily placed in the cask loading pit. A temporary gantry crane will be utilized to remove the old storage racks and install the new racks. Evaluations have been performed to ensure that the project can be completed safely without violating any design limits.

Safety Evaluation

The planned expansion of the storage capacity involves replacing the 11 existing racks in the Spent Fuel Pool with ten new Region 2 high-density rack modules and two new Region 1 rack modules for a total of 1712 storage cells. Each region is characterized by a nominal center-to-center spacing of the storage cells.

Rack modules in both regions will be free-standing and self supporting. The new Region 2 modules will be separated by a gap of approximately 1.0 inches from one another. There will be a nominal gap of 2.5 inches between the two Region 1 racks and between the Region 1 and Region 2 racks. Along the pool walls, a nominal gap will also be provided which varies between 2.0 inch and 3.8 inches.

With the expanded capacity, the Spent Fuel Pool cooling system will be required to remove an increased heat load while maintaining the pool water temperature below the design limit. The maximum heat load typically develops from the residual heat in the pool after full-core-discharge completely fills the spent fuel racks.

The Spent Fuel Pool thermal performance, criticality, and seismic response have been re-analyzed considering the increased storage capacity and fuel enrichment. The results of these analyses have shown that the pool storage systems remain adequate.

The Significant Hazards Consideration (SHC), contained in Attachment III, and the attached Licensing Report (Attachment V) address the safety issues arising from the proposed modification and revisions to the Technical Specifications. The scope of the technical analysis supporting this evaluation focused mainly on the final configuration of the expanded storage space. The transition to the final configuration involving some intermediate stages during the pool re-configuring is also included in the evaluation.

Mechanical Design Evaluation

The new fuel rack designs have been evaluated with respect to the mechanical and material qualifications, neutron poison, fuel handling qualifications, fuel interfaces, and accident considerations.

The principal construction materials for the new racks will be SA240 Type 304L stainless steel, or plate stock, and SA564-630 precipitation hardened stainless steel for the adjustable support spindles. The rack designs, material selection and fabrication process will comply with the applicable ASTM Standards A240, A276, A479, A564 and others, for service in the nuclear and the boric acid environments. The governing quality assurance requirements for fabrication of the racks are compatible with the quality assurance and quality control of 10CFR50, Appendix B requirements.

For primary nuclear criticality control in the new racks, a fixed neutron absorber will be used, integrated within the rack structure. The absorber, trade name Boral, is a boron carbide and aluminum-composite sandwich. Boral is chemically inert and has a long history of applications in the Spent Fuel Pool environments where it has maintained its neutron attenuation capability under thermal loads. Boral is manufactured under the control of a quality assurance program, which conforms to the requirements of 10CFR50, Appendix B.

The installation of the new rack modules will preserve space for thermal expansion and seismic movement. The support legs on the racks will allow for remote leveling and alignment of the rack modules to accommodate variations in the floor flatness. A thick bearing pad will be interposed between the rack pedestals and the floor to distribute the dead load over a wider support area.

The rack structural performance with respect to the impact and tensile loads, as well as the subcritical configuration, has been analyzed. The analysis included an accidental drop of a fuel assembly during movement to a storage location. It has been shown that these accidents will not invalidate the mechanical design and material selection criteria to safely store spent fuel in a coolable and subcritical configuration in any region. The storage rack structural integrity, and thus the fuel configuration, will be maintained. The fuel will remain subcritical.

Criticality Considerations

The new spent fuel racks are designed to maintain the required subcriticality margin when fully loaded with enriched fuel and submerged in unborated water at a temperature corresponding to the highest reactivity. For reactivity control in the racks, Boral panels will be used. The panels have been sized to sufficiently shadow the active fuel height of all assembly designs stored in the pool. The panels will be held in place and protected against damage by a stainless steel jacket, which will be stitch welded to the cell walls. The panels will be mounted on the exterior or on the interior of the cells, in an alternating pattern.

The storage of spent fuel in each region will be controlled by the criteria defining the maximum permissible reactivity. Region 1 will store the most reactive fresh fuel with an enrichment of up to 4.95 nominal w/o U-235, or spent fuel regardless of the burn-up history. These modules have been designed to accommodate an emergency offload. Region 2 storage will also accommodate fuel of 4.95 nominal w/o enrichment, but will be subject to burn-up limits. If the assembly does not meet the requirements for unrestricted storage in Region 2, then it must be stored in Region 1.

The NRC guidelines and the ANSI standards specify that the margin of safety for criticality be maintained by having the maximum neutron multiplication factor, K_{eff} less than or equal to 0.95, including uncertainties, for all normal and accident conditions. The analysis has shown that this criterion is always maintained under all postulated accidents. The accidents and malfunctions evaluated included a dropped fuel assembly on top of the fuel rack; impact on criticality of water temperature and density effects; and impact on criticality of eccentric positioning of a fuel assembly within the rack.

Thermal Hydraulics and Pool Cooling

A comprehensive thermal-hydraulic evaluation of the Spent Fuel Pool racks under the expanded storage configuration has been performed to analyze their thermal performance. Evaluations performed for the Spent Fuel Pool cooling system conservatively considered a total storage capacity of 1712 assemblies. This capacity is based on projected fuel discharges and a full core discharge occurring at the end of cycle 24 in 2018.

The calculation of the long-term decay heat for thermal analysis of the pool was performed using the industry-standard ORIGEN2 isotope depletion and generation code developed by the Oak Ridge National Laboratory and took into account all past discharges and the predicted heat load for each newly discharged fuel batch. The fuel discharge plan considered 18-month fuel cycles with 72 assemblies discharged each outage. The time-variant decay heat, generated by the most recent outage discharge, will be assumed to take place after the shortest period of cooling time allowed by the Technical Specifications and with the highest rate of fuel transfer from the vessel to the pool to maximize the heat addition.

Several discharge scenarios were considered with both partial and full core discharges to the pool coupled with both two and one cooling trains operable. A bulk pool maximum temperature of 170°F was chosen as the acceptable pool water temperature based on cooling system performance parameters and the pool structure evaluation.

Recognizing that the bulk pool temperature is dependent upon both the spent fuel decay heat load and the performance of the cooling system, parametric studies were performed to establish the bulk temperature relationship to the Component Cooling Water temperature at the inlet of the spent fuel pool heat exchangers. This provided an accurate means to determine the in-core hold time (and corresponding reduction in fuel decay heat load) necessary to maintain bulk pool temperatures at or below the selected maximum temperature of 170°F.

The local water temperature determinations are performed assuming that the pool is at its peak bulk temperature. The worst location was identified as the cell with the hottest assembly and the most restrictive convective flow. A conservative value for the axial peaking power factor was used. The local analysis was extended to include the effects of a partially blocked exit flow, postulated from an accidentally dropped assembly on top of the rack. In all cases analyzed, the heat transfer model conservatively accounted for an additional resistance from the fouling of the heat transfer surface in the heat exchangers and performance loss due to plugged tubes.

The calculated maximum local water temperature is determined to be 192.7°F in the hottest channel and coincides in time with the highest pool bulk temperature. The maximum fuel cladding temperature at the same location is calculated to be 230.4°F. These results conservatively assume a dropped fuel assembly blocking the exit of the cell. The local boiling point at the top of the fuel, based on the minimum water level in the pool as required by the Technical Specifications, will be approximately 240°F which indicates that the channel will remain in subcooled flow, thus minimizing the potential for fuel damage.

An evaluation of the Fuel Handling Building's heating, ventilation, and air conditioning (HVAC) system was performed for the limiting conditions of normal pool heat load (full core discharge scenarios). This evaluation has determined that the air temperature directly above the pool surface will be below 114°F. The calculations of passive losses (i.e., heat and moisture transfer) from the pool surface appropriately recognize this bounding air temperature.

Seismic and Structural Evaluation

A complete re-evaluation of the mechanical and civil structures, to address the structural issues resulting from the expansion of the pool storage capacity, has been performed. The analysis considered the loads from seismic, thermal, and mechanical forces to determine the margin of safety in the structural integrity of the fuel racks, the Cask Pit platform, the SFP and liner, and the Fuel Handling Building. The loads, load combinations, and acceptance criteria were based on the ASME Section III, Subsection NF, and on NUREG-0800, SRP Section 3.8.4, Appendix D.

a. The storage rack evaluation

The final configuration of the pool will consist of free standing and self-supporting style rack storage modules. The seismic analysis is performed using both whole pool multi-rack analysis and single rack analysis. These analyses were based on the simulation of the Safe Shutdown Earthquake (SSE) and the Operating Basis Earthquake (OBE) in accordance with SRP 3.7.1 requirements. Separate models were developed for the analysis of the whole pool configuration and the individual racks. The rack modules in the whole pool configuration were analyzed as completely full. The single rack analyses considered various rack loadings (full and several eccentric loading configurations), coefficients of friction at the base of the pedestals, motion in-phase and out-of-phase with adjacent racks, and the highest aspect ratio (height to width) and largest racks. Parametric studies were performed for these various rack attributes, primarily to study rack behavior under the plant specific dynamic conditions, to assess the possibility of rack overturning and determine the largest possible top-of-rack displacement. A total of 163 single rack simulations were performed, including one with an assumed 2000 pound mass located at the top of the rack.

The results indicate that the maximum seismic displacements do not pose any threat of impacts between the top of racks or with the pool walls. The resultant member and weld stresses in the racks are all below the allowable stresses, with a safety factor of at least 1.13. This minimum calculated safety factor is associated with the pedestal support female thread shear stress. The minimum safety factor for the cell membrane material and associated welding is 1.2. The racks will remain functional during and after a Safe Shutdown Earthquake.

The rack analysis provides pedestal to bearing pad impact loads resulting from lift-off and subsequent resettling during dynamic events. The pool floor stresses were evaluated for these impact loads and determined to remain within allowable limits even when considering the worst case pedestal location with respect to leak chases.

In addition to the seismic evaluations, the storage racks were also analyzed for all postulated accident conditions. A fuel handling accident involving a fuel assembly dropped from the Spent Fuel Bridge Crane highest possible lift point would not compromise the integrity of the rack. Permanent deformation of the rack would be limited to the top region only. This is acceptable since the rack cross-sectional geometry at the active fuel height is not altered. Thus, the functionality of the rack is not affected.

The Cask Pit platform is designed to support the storage rack in the Cask Pit and maintain the elevation of the top of this racks level with those in the Spent Fuel Pool. The platform is designed in accordance with ASME Section III, Subsection NF based on maximum calculated pedestal loadings from the supported storage racks.

b. Pool and Fuel Handling Building structural evaluation

The Spent Fuel Pool is a cast-in-place steel lined reinforced concrete tank structure that provides space for storage of spent fuel assemblies. The Spent Fuel Pool is located at the south-west corner of the of the Fuel Handling Building (FHB). The pool west wall is part of the FHB outer wall. The Fuel Handling Building consists of four reinforced concrete walls and a base slab that are lined with a stainless steel liner. The pool south and west walls are supported by three reinforced concrete columns and Transfer Canal mass foundation that rest on the fill mats of the adjacent Auxiliary Building and Reactor Building. The remaining part of the Spent Fuel Pool is supported by a system of caissons that extend below the pool slab, through the supporting soil, and into the underlying bedrock. Floor slabs at the ground elevation and pool walls top elevation provide horizontal bracing to the pool north and south walls. The Fuel Handling Building is designed as a seismic Class I structure.

The pool structure and appropriate portions of the Fuel Handling Building have been analyzed using a 3-D finite element model with static equivalent loads applied to envelope the rack and hydrodynamic loads. The individual loads and load combinations used were in accordance with NUREG-0800, SRP Section 3.8.4 and based on the "ultimate strength" design method. The primary loads considered were:

- the dead weight of the concrete structure, fully loaded racks, and the water,
- rack seismic loads developed from the whole pool multi-rack simulations,
- pool structure self weight excitation with g-values equal to the magnitudes of the maximum floor accelerations at the pool floor slab elevation
- hydrostatic pressure force lateral to the walls,
- hydrodynamic coupling forces applied to the lower portion of the wall and water slosh pressures on top portion of the wall,
- bounding thermal loads producing the largest temperature gradient across the thickness of the wall and the slab,

In addition to the loads described above, the pool structure and liner were also analyzed for mechanical loads under accident conditions. Analyses were also performed on liner fatigue considering both temperature and seismic cycles. The result of the analyses performed on the Spent Fuel Pool and Fuel Handling Building indicate that under all postulated loadings the structural components, floor slabs, pool walls, supporting columns, liner and its anchorages will be subjected to stresses or strains within acceptable limits.

Radiological Considerations

Radiological consequences of accidents in the Fuel Handling Building have been evaluated. The fuel handling accident considers the release of the gaseous fission products contained in the fuel/cladding gaps of a peak-power 264-rod fuel assembly plus 50 rods in an impacted assembly, for a total of 314 rods. The assemblies are considered burned to 70,000 MWD/MTU, and the drop accident occurs 72 hours after reactor shutdown. This represents an increase in burn-up and a reduction in the in-core hold time from the previous hold time of 100 hours. The changes in source term have been re-evaluated and have been shown to be acceptable.

For the fuel handling accident in the Reactor Building, the release path of radionuclides would not normally pass through charcoal filters. The whole-body and skin doses would be the same as the doses for the accident in the Fuel Handling Building, because those doses are caused by radionuclides that, in the Fuel Handling Building accident, were not affected by the charcoal filters in the building exhaust. The hypothetical thyroid dose would be higher than those determined for the accidental assembly drop in the Fuel Handling Building and higher than the criterion of the Standard Review Plan. However, as described in Section 15.4.5.1.4 of the

FSAR, instrumentation is available to detect the release of radioactivity and close the Reactor Building Purge system. This action essentially precludes any radioactive release to the environment for this accident.

A rack drop involving radiological consequences is precluded, since all rack movement during the removal and installation phase will follow safe load paths that prevent heavy loads from being transported over the stored spent fuel. Thus, there are no credible radiological consequences from this accident.

There has been no steady long-term increase of radiological conditions in the Spent Fuel Pool resulting from the radionuclides within the fuel as more spent fuel is added to the pool. The radiological conditions within the building are typically dominated by the most recent batch of the spent fuel from a full-core-discharge. The radioactive inventory of the older fuel that will increase with the expanded storage capacity will be insignificant compared to that of the recent offload.

Since the new storage racks will be located in closer proximity to the Spent Fuel Pool walls, an increase in the adjacent radiological doses is expected. Radiological analyses have shown that the dose levels adjacent to all pool areas will remain within acceptable levels.

<u>Supporting Analysis</u>

For supplemental information on the V.C. Summer Spent Fuel Pool storage expansion proposed license amendment, refer to the attached Licensing Report. Two versions of the report are attached. The version included in Attachment V contains complete documentation for all sections of the report, including some information, which is considered proprietary pursuant to 10CFR2.790. South Carolina Electric & Gas (SCE&G) requests that this version be withheld from public viewing. The version included as Attachment VI is identical, except that proprietary information has been removed and replaced by a note of explanation at each location where information has been omitted. SCE&G offers this additional version for the purposes of public review.

NO SIGNIFICANT HAZARDS EVALUATION FOR SPENT FUEL POOL STORAGE EXPANSION THE VIRGIL C. SUMMER NUCLEAR STATION TECHNICAL SPECIFICATIONS

Description of Amendment Request

The Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS), are being revised to permit increased storage (re-racking) in the Spent Fuel Storage pool. This change request also affects the storage requirements as well as the incore hold time before fuel can be moved out of the reactor vessel.

The expansion will increase the total storage space from 1276 to 1712 storage cells, which is an increase of 436 cells and will extend the capability for full core offload from 2008 to 2018. Only two rack types will be used, as opposed to the three types currently in the pool. Region 1 will allow for storage (200 cells) for enrichments up to 4.95 nominal weight percent (w/o) U-235 without regard to fuel burn-up. Region 2 will allow storage of assemblies (1512 cells) that meet minimum burn-up requirements for unrestricted storage. Boral will be used as the active neutron absorbing poison in both regions.

A minimum boron concentration of 500 ppm is proposed that will ensure reactivity is maintained less than design limits in the event of a fuel handling accident anytime during the cycle but is specifically applicable to non-refueling outage evolutions.

A minimum incore hold time of 72 hours is proposed which will be a significant contributor to shorter outages. This minimum hold time is directly related to the temperature of the Component Cooling Water (CCW) and the capacity to remove the decay heat generated while in the reactor.

Additionally, Specifications 3.9.10, 3.9.11, and 3.9.12 are to be moved out of the Refueling Operations Section (3.9) and into the Plant Systems Section (3.7) since they are not specific to refueling operations and to conform to the Improved Standardized Technical Specifications – NUREG 1431.

Background

V. C. Summer Nuclear Station_currently expects to lose the capacity for full core offload during refueling operations in 2008 (after Cycle 17). It is not likely that the Department of Energy will have a facility open in time to prevent this loss of capability.

SCE&G has evaluated spent fuel storage options that have been licensed by the NRC and which are currently feasible for use at the VCSNS site. The evaluation concluded that re-racking the spent fuel pool is currently the most cost effective alternative. Re-racking would provide an increase in storage capacity, which would maintain the plant's capability to accommodate a full core discharge, until the end of Cycle 24 in 2018.

The proposed change would remove the 11 current storage racks and replace them with 12 higher density racks. Fuel shuffling during the effort will require that one rack be temporarily placed in the cask loading pit. A temporary gantry crane will be utilized to remove the old storage racks and install the new racks. Evaluations have been performed to ensure that the project can be completed safely without violating any design limits.

Basis for No Significant Hazards Consideration Determination

South Carolina Electric & Gas Company (SCE&G) has evaluated the proposed changes to the VCSNS TS described above against the Significant Hazards Criteria of 10 CFR 50.92 and has determined that the changes do not involve any significant hazard. The following is provided in support of this conclusion.

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

   In the analysis of the safety issues concerning the expanded pool storage capacity, the following previously postulated accident scenarios have been considered:

   a. A spent fuel assembly drop in the Spent Fuel Pool

   b. Loss of Spent Fuel Pool cooling flow

   c. A seismic event

   d. Misloaded fuel assembly

   The probability that any of the accidents in the above list can occur is not significantly increased by the modification itself. The probabilities of a seismic event or loss of Spent Fuel Pool cooling flow are not influenced by the proposed changes. The probabilities of accidental fuel assembly drops or misloadings are primarily influenced by the methods used to lift and move these loads. The method of handling loads during normal plant operations is not significantly changed, since the same equipment (i.e., Spent Fuel Bridge Crane) and procedures will be used. Since the methods used to move loads during normal operations remain nearly the same as those used previously, there is no significant increase in the probability of an accident.

   During rack removal and installation, all work in the pool area will be controlled and performed in strict accordance with specific written procedures. Any movement of fuel assemblies required to be performed to support the modification (e.g., removal and installation of racks) will be performed in the same manner as during normal fuel handling operations. Shipping cask movements will not be performed during the modification period.

Accordingly, the proposed modification does not involve a significant increase in the probability of an accident previously evaluated.

The consequences of the previously postulated scenarios for an accidental drop of a fuel assembly in the Spent Fuel Pool have been re-evaluated for the proposed change. The results show that the postulated accident of a fuel assembly striking the top of the storage racks will not distort the racks sufficiently to impair their functionality. The minimum subcriticality margin, K_{eff} less than or equal to 0.95, will be maintained. The structural damage to the Fuel Handling Building, pool liner, and fuel assembly resulting from a fuel assembly drop striking the pool floor or another assembly located within the racks is primarily dependent on the mass of the falling object and the drop height. Since these two parameters are not changed by the proposed modification, the postulated structural damage to these items remains unchanged. The radiological dose at the exclusion area boundary will increase due to the changes in in-core hold time and burnup. The previously calculated doses to thyroid and whole body were 10.6 and 0.52 rem, respectively. The new Exclusion Area Boundary (EAB) thyroid and whole body doses based on the proposed change will be 12.97 and 0.678 rem, respectively. These dose levels will remain "well within" the levels required by 10CFR100, paragraph 11, as defined in Section 15.7.4.II.1 of the Standard Review Plan. Therefore, the increase in dose is not considered a significant increase in consequence.

The consequences of the previously postulated scenarios for an accidental drop of a fuel assembly in the Reactor Building have also been re-evaluated for the proposed change to assess the affect of higher burnup and shorter cooling time. The proposed re-racking does not affect the fuel assembly mass or drop height parameters. Therefore, the previously determined fuel damage and resulting criticality assessments remain unchanged. However, the radiological dose at the exclusion area boundary will increase due to the changes in in-core hold time and burnup. The previously calculated doses to thyroid were 211 rem. With no action to limit the consequences of the fuel handling accident in the reactor building, the new EAB thyroid dose would be 259 rem. The whole-body would be the same as the doses for the accident in the fuel handling building, since those doses are caused by radionuclides that, in the Fuel-Handling-Building accident, were not affected by the charcoal filters in the building exhaust. This hypothetical thyroid dose would be higher than the criterion of the Standard Review Plan. However, as described in Section 15.4.5.1.4 of the VCSNS FSAR, instrumentation is available to detect the release of radioactivity and to close the Reactor Building Purge System. This action essentially precludes any radioactive release to the environment for this accident. Thus, the results of the postulated fuel drop accidents remain acceptable and do not represent a significant increase in consequences from any of the same previously evaluated accidents that have been reviewed and found acceptable by the NRC.

The consequences of a loss of Spent Fuel Pool cooling have been evaluated and found to have no increase. The concern with this accident is a reduction of Spent Fuel Pool water inventory from bulk pool boiling resulting in uncovering fuel assemblies. This situation could lead to fuel failure and subsequent significant increase in offsite dose. Loss of spent fuel pool cooling at V.C. Summer is mitigated in the usual manner by ensuring that a sufficient time lapse exists between the loss of forced cooling and uncovering fuel. This period of time is compared against a reasonable period to re-establish cooling or supply an alternative water source. Evaluation of this accident usually includes determination of the time to boil. This time period is much less than the onset of any significant increase in offsite dose, since once boiling begins it would have to continue unchecked until the pool surface was lowered to the point of exposing active fuel. The time to boil represents the onset of loss of pool water inventory and is commonly used as a gage for establishing the comparison of consequences before and after a reracking project. The heat up rate in the Spent Fuel Pool is a nearly linear function of the fuel decay heat load. The fuel decay heat load will increase subsequent to the proposed changes because of the increase in the number of assemblies, shorter hold times, and higher fuel burn-ups. The thermal-hydraulic analysis determined that the minimum time to boil is more than two hours subsequent to complete loss of forced cooling and a minimum of 24 hours between loss of forced cooling and a drop of water level to within 10 feet of the top of the racks. In the unlikely event that all pool cooling is lost, sufficient time will still be available subsequent to the proposed changes for the operators to provide alternate means of cooling before the water shielding above the top of the racks falls below 10 feet. Therefore, the proposed change represents no increase in the consequences of loss of pool cooling.

The consequences of a design basis seismic event are not increased. The consequences of this accident are evaluated on the basis of subsequent fuel damage or compromise of the fuel storage or building configurations leading to radiological or criticality concerns. The new racks have been analyzed in their new configuration and found safe during seismic motion. Fuel has been determined to remain intact and the storage racks maintain the fuel and fixed poison configurations subsequent to a seismic event. The structural capability of the pool and liner will not be exceeded under the appropriate combinations of dead weight, thermal, and seismic loads. The Fuel Handling Building structure will remain intact during a seismic event and will continue to adequately support and protect the fuel racks, storage array, and pool moderator/coolant. Thus, the consequences of a seismic event are not increased.

Fuel misloading accidents were previously postulated occurrences. The consequence of this type of accident has been analyzed for the worst possible storage configuration subsequent to the proposed modification and it has been shown that the consequences remain acceptable with respect to the same criteria used previously. Therefore, there is no increase in consequences.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

To assess the possibility of new or different kind of accidents, a list of the important parameters required to ensure safe fuel storage was established. Safe fuel storage is defined here as providing an environment which would not present any significant threats to workers or the general public. In other words, meeting the requirements of 10CFR100 and 10CFR20. Any new events, which would modify these parameters sufficiently to place them outside of the boundaries analyzed for normal conditions and/or outside of the boundaries previously considered for accidents, would be considered a new or different accident. The criticality and radiological safety evaluations were reviewed to establish the list of important parameters. The fuel configuration and the existence of the moderator/coolant were identified as the only two parameters important to safe fuel storage. Significant modification of these two parameters represents the only possibility of an unsafe storage condition. Once the two important parameters were established, an additional step was taken to determine what events (which were not previously considered) could result in changes to the storage configuration or moderator/coolant presence during or subsequent to the proposed changes. This process was adopted to ensure that the possibility of any new or different accident scenario or event would be identified.

Due to the proposed changes, an accidental drop of a rack module during construction activity in the pool was considered as the only event, which might represent a new or different kind of accident.

A construction accident of a rack dropping onto stored spent fuel or the pool floor liner is not a postulated event due to the defense-in-depth approach to be taken, as discussed in detail within Section 3.5 of the attached Licensing Report (Attachment V). A new temporary crane, hoist, and rack lifting rig will be introduced to remove the existing racks and install the new racks. These temporary lift items have been designed to meet the requirements of NUREG 0612 and ANSI N14.6. A rack drop event is commonly referred to as a "heavy load drop" over the pools. Racks will not be allowed to travel over any racks containing fuel assemblies, thus a rack drop onto fuel is precluded. A rack drop to the pool liner is not a postulated event, since all of the mechanical lifting components either provide redundancy in load path or are designed with safety margins greater than a factor of ten. All movements of heavy loads over the pool will comply with the applicable administrative controls and guidelines (i.e. plant procedures, NUREG 0612, etc.). Nevertheless, the analysis of a rack dropping to the liner has been performed and shown to be acceptable. A rack drop would not alter the storage configuration or moderator/coolant presence. Therefore, the rack drop does not represent a new or different kind of accident.

The proposed change does not alter the operating requirements of the plant or of the equipment credited in the mitigation of the design basis accidents. The proposed change does not affect any of the important parameters required to ensure safe fuel storage. Therefore, the potential for a new or previously unanalyzed accident is not created.

3. *Does this change involve a significant reduction in margin of safety?*

The function of the Spent Fuel Pool is to store the fuel assemblies in a subcritical and coolable configuration through all environmental and abnormal loadings, such as an earthquake or fuel assembly drop. The new rack design must meet all applicable requirements for safe storage and be functionally compatible with the Spent Fuel Pool.

SCE&G has addressed the safety issues related to the expanded pool storage capacity in the following areas:

1. Material, mechanical and structural considerations
2. Nuclear criticality
3. Thermal-hydraulic and pool cooling

The mechanical, material, and structural designs of the new racks have been reviewed in accordance with the applicable provisions of the NRC Guidance entitled, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications". The rack materials used are compatible with the spent fuel assemblies and the Spent Fuel Pool environment. The design of the new racks preserves the proper margin of safety during abnormal loads such as a dropped assembly and tensile loads from a stuck assembly. It has been shown that such loads will not invalidate the mechanical design and material selection to safely store fuel in a coolable and subcritical configuration.

The methodology used in the criticality analysis of the expanded Spent Fuel Pool meets the appropriate NRC guidelines and the ANSI standards (GDC 62, NUREG 0800, Section 9.1.2, the OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, Reg. Guide 1.13, and ANSI ANS 8.17). The margin of safety for subcriticality is maintained by having the neutron multiplication factor equal to, or less than, 0.95 under all normal storage, fuel handling, and accident conditions, including uncertainties.

An additional Technical Specification has been added to require a minimum of 500 ppm boron whenever new or irradiated fuel is being moved (non-refueling movement) in the spent fuel pool, fuel transfer canal, or cask loading pit. This minimum boron concentration will ensure that the fuel remains subcritical under any normal fuel handling or misloading/mispositioning accidents.

The criterion of having the neutron multiplication factor equal to, or less than, 0.95 during storage or fuel movement is the same as that used previously to establish criticality safety evaluation acceptance. Therefore, the accepted margin of safety remains the same.

The thermal-hydraulic and cooling evaluation of the pool demonstrated that the pool can be maintained below the specified thermal limits under the conditions of the maximum heat load and during all credible accident sequences and seismic events. The pool temperature will not exceed 170°F during the worst single failure of a cooling pump. The maximum local water temperature in the hot channel will remain below the boiling point. The fuel will not undergo any significant heat up after an accidental drop of a fuel assembly on top of the rack blocking the flow path. A loss of cooling to the pool will allow sufficient time (24 hours) for the operators to intervene and line up alternate cooling paths and the means of inventory make-up before the water shielding above the top of the racks falls below 10 feet. The thermal limits specified for the evaluations performed to support the proposed change are the same as those which were used in the previous evaluations. Therefore, the accepted margin of safety remains the same.

Thus, it is concluded that the changes do not involve a significant reduction in the margin of safety.

The NRC has provided guidance concerning the application of standards in 10CFR50.92 by providing certain examples (51FR7751, March 6, 1986) of amendments that are considered not likely to involve a SHC. The proposed changes for V.C. Summer are similar to Example (x): an expansion of the storage capacity of Spent Fuel Pool when all of the following are satisfied:

(1) The storage expansion method consists of either replacing existing racks with a design that allows closer spacing between stored spent fuel assemblies or placing additional racks of the original design on the pool floor if space permits.

The V.C. Summer reracking modification involves replacement of the existing racks with a design that will allow closer spacing of the stored fuel assemblies. Also includes installing one new rack in the existing space in the NE corner of the spent fuel pool.

(2) The storage expansion method does not involve rod consolidation or double tiers.

The V.C. Summer reracking does not involve fuel consolidation. The racks will not be double tiered; no fuel assemblies will be stored above other assemblies.

(3) The K_{eff} of the pool is maintained less than, or equal to, 0.95.

The design of the new racks integrates a neutron absorber, Boral, within the racks to allow closer storage of spent fuel assemblies while ensuring that K_{eff} remains less than 0.95 under all conditions. Additionally, the water in the Spent Fuel Pool does contain boron as further assurance that K_{eff} remains less than 0.95.

(4) No new technology or unproven technology is utilized in either the construction process or the analytical techniques necessary to justify the expansion.

The rack vendor has successfully participated in the licensing of numerous other racks of a similar design. The construction process and the analytical techniques of the V.C. Summer pool expansion are substantially the same as in the other completed rerack projects. Thus, no new or unproven technology is used in the V.C. Summer reracking.

Pursuant to 10 CFR 50.91, the preceding analyses provides a determination that the proposed Technical Specifications change poses no significant hazard as delineated by 10 CFR 50.92.

Environmental Assessment

This proposed Technical Specification change has been evaluated against criteria for and identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed change meets the criteria for categorical exclusion as provided for under 10 CFR 51.22(c)(9). The following is a discussion of how the proposed Technical Specification change meets the criteria for categorical exclusion.

10 CFR 51.22(c)(9): Although the proposed change involves change to requirements with respect to inspection, surveillance, or design requirements:

(i) the proposed change involves No Significance Hazards Consideration (refer to the No Significant Hazards Consideration Determination section of this Technical Specification Change Request);

(ii) there are no significant changes in the types or significant increase in the amounts of any effluents that may be released offsite since the proposed change does not affect the generation of any radioactive effluents nor does it affect any of the permitted release paths; and

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Based on the aforementioned and pursuant to 10 CFR 51.22(b), no environmental assessment or environmental impact statement need be prepared in connection with issuance of an amendment to the Technical Specifications incorporating the proposed change.

COMMITMENTS TO ENSURE SAFE OPERATIONS
SPENT FUEL POOL STORAGE EXPANSION
THE VIRGIL C. SUMMER NUCLEAR STATION
TECHNICAL SPECIFICATIONS

Description of Amendment Request

The Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS), are being revised to permit increased storage (re-racking) in the Spent Fuel Storage pool. This change request also affects the storage requirements as well as the incore hold time before fuel can be moved out of the reactor vessel.

The expansion will increase the total storage space from 1276 to 1712 storage cells, which is an increase of 436 cells and will extend the capability for full core offload from 2008 to 2018. Only two rack types will be used, as opposed to the three types currently in the pool. Region 1 will allow for storage (200 cells) for enrichments up to 4.95 nominal weight percent (w/o) U-235 without regard to fuel burn-up. Region 2 will allow storage of assemblies (1512 cells) that meet minimum burn-up requirements for unrestricted storage. Boral will be used as the active neutron absorbing poison in both regions.

A minimum boron concentration of 500 ppm is proposed that will ensure reactivity is maintained less than design limits in the event of a fuel handling accident anytime during the cycle but is specifically applicable to non-refueling outage evolutions.

A minimum incore hold time of 72 hours is proposed which will be a significant contributor to shorter outages. This minimum hold time is directly related to the temperature of the Component Cooling Water (CCW) and the capacity to remove the decay heat generated while in the reactor.

Additionally, Specifications 3.9.10, 3.9.11, and 3.9.12 are to be moved out of the Refueling Operations Section (3.9) and into the Plant Systems Section (3.7) since they are not specific to refueling operations and to conform to the Improved Standardized Technical Specifications – NUREG 1431.

Background

V. C. Summer Nuclear Station_currently expects to lose the capacity for full core offload during refueling operations in 2008 (after Cycle 17). It is not likely that the Department of Energy will have a facility open in time to prevent this loss of capability.

SCE&G has evaluated spent fuel storage options that have been licensed by the NRC and which are currently feasible for use at the VCSNS site. The evaluation concluded that re-racking the spent fuel pool is currently the most cost effective alternative. Re-racking would provide an increase in storage capacity, which would maintain the plant's capability to accommodate a full core discharge, until the end of Cycle 24 in 2018.

The proposed change would remove the 11 current storage racks and replace them with 12 higher density racks. Fuel shuffling during the effort will require that one rack be temporarily placed in the cask loading pit. A temporary gantry crane will be utilized to remove the old storage racks and install the new racks. Evaluations have been performed to ensure that the project can be completed safely without violating any design limits.

Commitments to Ensure Safe Operations

Several of these commitments supercede previously docketed commitments pertaining to spent fuel handling and storage.

The V. C. Summer re-racking project does not involve rod consolidation and the racks will not be doubled tier.

Both Region 1 and Region 2 of the pool will utilize Boral as the active neutron absorbing poison.

Maximum fuel enrichment for either Region 1 or Region 2 will be 4.95 nominal weight/percent (w/0) U-235.

Region 1 will be used for storage of new and spent fuel that does not meet the Region 2 minimum burn-up requirements.

A minimum boric acid concentration of 2,000 parts per million (ppm) will normally be maintained in the spent fuel pool whenever an assembly is moved during refueling.

A temporary crane, hoist, lifting rig will be used during the implementation phase for manipulation of the racks. These items will meet the requirements of NUREG 0612 and ANSI N14.6 as described in the technical report.

All rack movement during the implementation phase will follow safe load paths and comply with applicable administrative controls and guidelines.

The installation of new racks will preserve space for thermal expansion and seismic movement.

A platform will be installed underneath the rack to be temporarily placed in the cask loading pit to allow fuel shuffle to occur at the same elevation as the racks in the spent fuel pool.

Shipping cask movement will not be performed during the re-rack implementation.

The existing Boroflex inspection will cease with the removal of the existing racks.

The movement of spent fuel from Region 1 to Region 2 can be performed anytime after the Technical Specification requirements have been satisfied.

The discharge of spent fuel from the reactor vessel into the spent fuel pool will be performed per Technical Specification requirements.

AFFIDAVIT PURSUANT TO 10CFR 2.790

AFFIDAVIT PURSUANT TO 10CFR2.790

I, Scott H. Pellet, being duly sworn, depose and state as follows:

(1) I am the Project Manager for Holtec International and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the document entitled "Spent Fuel Storage Expansion at Virgil C. Summer for South Carolina Electric & Gas," Holtec Report HI-2012624. The proprietary material in this document is delineated by proprietary designation (i.e., shaded areas), as stated on the report cover sheet.

(3) In making this application for withholding of proprietary information of which it is the owner, Holtec International relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4) and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10CFR Part 9.17(a)(4), 2.790(a)(4), and 2.790(b)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by Holtec's competitors without license from Holtec International constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. Information which reveals cost or price information, production, capacities, budget levels, or commercial strategies of Holtec International, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future Holtec International customer-funded development plans and programs of potential commercial value to Holtec International;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs 4.a, 4.b, 4.d, and 4.e, above.

(5) The information sought to be withheld is being submitted to the NRC in confidence. The information (including that compiled from many sources) is of a sort customarily held in confidence by Holtec International, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by Holtec International. No public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within Holtec International is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his designee), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside Holtec International are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information classified as proprietary was developed and compiled by Holtec International at a significant cost to Holtec International. This information is classified as proprietary because it contains detailed historical data and analytical results not available elsewhere. This information would provide other parties, including competitors, with information from Holtec International's technical database and the results of evaluations performed using codes developed by Holtec International. Release of this information would improve a competitor's position without the competitor having to expend similar resources for the development of the database. A substantial effort has been expended by Holtec International to develop this information.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to Holtec International's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of Holtec International's comprehensive spent fuel storage technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology, and includes development of the expertise to determine and apply the appropriate evaluation process.

The research, development, engineering, and analytical costs comprise a substantial investment of time and money by Holtec International.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

AFFIDAVIT PURSUANT TO 10CFR2.790

Holtec International's competitive advantage will be lost if its competitors are able to use the results of the Holtec International experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to Holtec International would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive Holtec International of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

STATE OF NEW JERSEY)
) ss:
COUNTY OF BURLINGTON)

Scott H. Pellet, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at Marlton, New Jersey, this 3rd day of July, 2001.

Scott H. Pellet

Mr. Scott H. Pellet
Holtec International

Subscribed and sworn before me this 3rd day of July, 2001.

Maria C. Pepe

MARIA C. PEPE
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires April 25, 2005

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SPENT FUEL STORAGE EXPANSION

at

VIRGIL C. SUMMER

for

SOUTH CAROLINA ELECTRIC & GAS

HOLTEC PROJECT NO. 1093

HOLTEC REPORT HI-2012624

REPORT CATEGORY: A

REPORT CLASS: SAFETY RELATED

REVIEW AND CERTIFICATION LOG FOR MULTIPLE AUTHORS

Sheet 1 of 2

REPORT NUMBER: 2012624 PROJECT NUMBER: 1093

Document Portion	REVISION 0		REVISION 1		REVISION 2		REVISION 3	
	Author	Reviewer	Author	Reviewer	Author	Reviewer	Author	Reviewer
Chapter 1	Scott Pellet SHP 6/5/01	C. Bullard C.B. 6/5/01	Scott Pellet SHP 7/2/01	C. Bullard C.B. 7/3/01				
Chapter 2	Scott Pellet SHP 6/5/01	C. Bullard C.B. 6/5/01	Scott Pellet SHP 7/2/01	C. Bullard C.B. 7/3/01				
Chapter 3	Scott Pellet SHP 6/5/01	C. Bullard C.B. 6/5/01	Scott Pellet SHP 7/2/01	C. Bullard C.B. 7/3/01				
Chapter 4	[illegible] 6/5/01	[illegible] DMM 6/5/01	Scott Pellet SHP 7/2/01	C. Bullard C.B. 7/3/01				
Chapter 5	[illegible] ER 6/5/01	[illegible] DMM 6/5/01	Scott Pellet SHP 7/2/01	C. Bullard C.B. 7/3/01				
Chapter 6	Scott Pellet SHP 6/5/01	[illegible] 6/5/01	Scott Pellet SHP 7/2/01	C. Bullard C.B. 7/3/01				
Chapter 7	Scott Pellet SHP 6/5/01	John Zhai J.Z. 6/7/01	Scott Pellet SHP 7/2/01	C. Bullard C.B. 7/3/01				
Chapter 8	Scott Pellet SHP 6/5/01	John Zhai J.Z. 6/7/01	Scott Pellet SHP 7/2/01	C. Bullard C.B. 7/3/01				

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REVIEW AND CERTIFICATION LOG FOR MULTIPLE AUTHORS

REPORT NUMBER: 2012624 PROJECT NUMBER: 1093

Document Portion	REVISION 0		REVISION 1		REVISION 2		REVISION 3	
	Author	Reviewer	Author	Reviewer	Author	Reviewer	Author	Reviewer
Chapter 9	Scott Pellet FOR DR STAN TURNER	Scott Pellet FOR WALTER MITCHELL	Scott Pellet SHP 7/2/01	C. Bullard C.B. 7/3/01.				
Chapter 10	Scott Pellet 6/5/01	C. Bullard C.B. 6/5/01	Scott Pellet SHP 7/2/01	C. Bullard C.B. 7/3/01				
Chapter 11	Scott Pellet 6/5/01	C. Bullard CB 6/5/01.	Scott Pellet SHP 7/2/01	C. Bullard CB. 7/3/01.				
QA APPROVAL	N/A		N/A					
PROJECT MANAGER†	Scott Pellet SHP 6-8-01		Scott Pellet SHP 7-3-01					

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SUMMARY OF REVISIONS

Revision 1 contains the following pages:	
COVER PAGE	1 page
QA AND ADMINISTRATIVE INFORMATION LOG	3 page
TABLE OF CONTENTS	10 pages
1.0 INTRODUCTION	5 pages
2.0 HIGH DENSITY STORAGE RACKS	26 pages
3.0 MATERIAL AND HEAVY LOAD CONSIDERATIONS	18 pages
4.0 CRITICALITY SAFETY EVALUATION	32 pages
--- APPENDIX 4A - BENCHMARK CALCULATIONS	25 pages
5.0 THERMAL-HYDRAULIC CONSIDERATIONS	38 pages
6.0 STRUCTURAL/SEISMIC CONSIDERATIONS	63 pages
7.0 MECHANICAL ACCIDENTS	19 pages
8.0 SPENT FUEL POOL STRUCTURE INTEGRITY CONSIDERATIONS	23 pages
9.0 RADIOLOGICAL EVALUATION	11 pages
10.0 INSTALLATION	10 pages
11.0 ENVIRONMENTAL COST/BENEFIT ASSESSMENT	8 pages
TOTAL	292 pages

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

Tables

TABLE OF CONTENTS

TABLE OF CONTENTS

Figures

TABLE OF CONTENTS

TABLE OF CONTENTS

1.0 INTRODUCTION

Virgil C. Summer Nuclear Station (VCSNS) is projected to lose full core reserve (FCR) in its Spent Fuel Pool (SFP) following the Cycle 17, which ends in Spring 2008. VCSNS intends to rerack the SFP with new racks increasing the current SFP storage capacity of 1,276 storage cells to 1,712 storage cells. This report provides the design basis, analysis methodology, and results for the proposed spent fuel storage racks at VCSNS and is prepared to support the licensing process.

VCSNS is a single unit pressurized water reactor plant located near Jenkinsville, South Carolina. VCSNS is a Westinghouse 3-loop PWR rated for 2900 MWt. The plant has been in operation since 1982.

Reracking will replace all eleven of the existing storage racks with ten Region 2 style storage racks with a capacity of 1,512 assemblies and two Region 1 style storage racks with a capacity of 200 assemblies. The definitions of 'Region 1' and 'Region 2' racks are provided in Section 2 of this report. The proposed fuel storage rack array is shown in the plan view provided by Figure 1.1.1.

The new Spent Fuel Pool storage racks are freestanding and self-supporting. The principal construction materials for the SFP racks are SA240-Type 304L stainless steel sheet and plate stock, and SA564-630 (precipitation hardened stainless steel) for the adjustable support spindles. The only non-stainless material utilized in the rack is the neutron absorber material, which is a boron carbide and aluminum-composite sandwich available under the patented product name Boral™.

The racks are designed to the stress limits of, and analyzed in accordance with, Section III, Division 1, Subsection NF of the ASME Boiler and Pressure Vessel (B&PV) Code [1]. The material procurement, analysis, fabrication, and installation of the rack modules conform to 10CFR50 Appendix B requirements.

The rack design and analysis methodologies employed are a direct evolution of previous license applications. This report documents the design and analyses performed to demonstrate that the racks meet all governing requirements of the applicable codes and standards, in particular, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", USNRC (1978) and 1979 Addendum thereto [2].

Sections 2 and 3 of this report provide an abstract of the design and material information on the racks.

Section 4 provides a summary of the methods and results of the criticality evaluations performed for the new and spent fuel storage racks. The criticality safety analysis requires that the neutron multiplication factor for the stored fuel array be bounded by the USNRC k_{eff} limit of 0.95 under assumptions of 95% probability and 95% confidence. The criticality safety analysis sets the requirements on the Boral panel length and the amount of B^{10} per unit area (i.e., loading density) of the Boral panel for the SFP racks.

Thermal-hydraulic consideration requires that fuel cladding will not fail due to excessive thermal stress, and that the steady state pool bulk temperature will remain within the limits prescribed for the Cask Pit and Spent Fuel Pool to satisfy the pool structural strength, operational, and regulatory requirements. The thermal-hydraulic analyses carried out in support of this storage expansion effort are described in Section 5.

Rack module structural analysis requires that the primary stresses in the rack module structure will remain below the ASME B&PV Code (Subsection NF) [1] allowables. Demonstrations of seismic and structural adequacy are presented in Section 6.0. The structural qualification also requires that the subcriticality of the stored fuel will be maintained under all postulated accident scenarios. The structural consequences of these postulated accidents are evaluated and presented in Section 7 of this report.

Section 8 discusses the salient considerations in the installation of the racks.

All computer programs utilized to perform the analyses documented in this report are benchmarked and verified. These programs have been utilized by Holtec International in numerous license applications over the past decade.

The analyses presented herein clearly demonstrate that the rack module arrays possess wide margins of safety in respect to all considerations of safety specified in the OT Position Paper [2], namely, nuclear subcriticality, thermal-hydraulic safety, seismic and structural adequacy, radiological compliance, and mechanical integrity.

1.1 References

[1] American Society of Mechanical Engineers (ASME), Boiler & Pressure Vessel Code, Section III, 1989 Edition, Subsection NF, and Appendices.

[2] USNRC, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, April 14, 1978, and Addendum dated January 18, 1979.

SHADED AREAS DENOTE PROPRIETARY INFORMATION

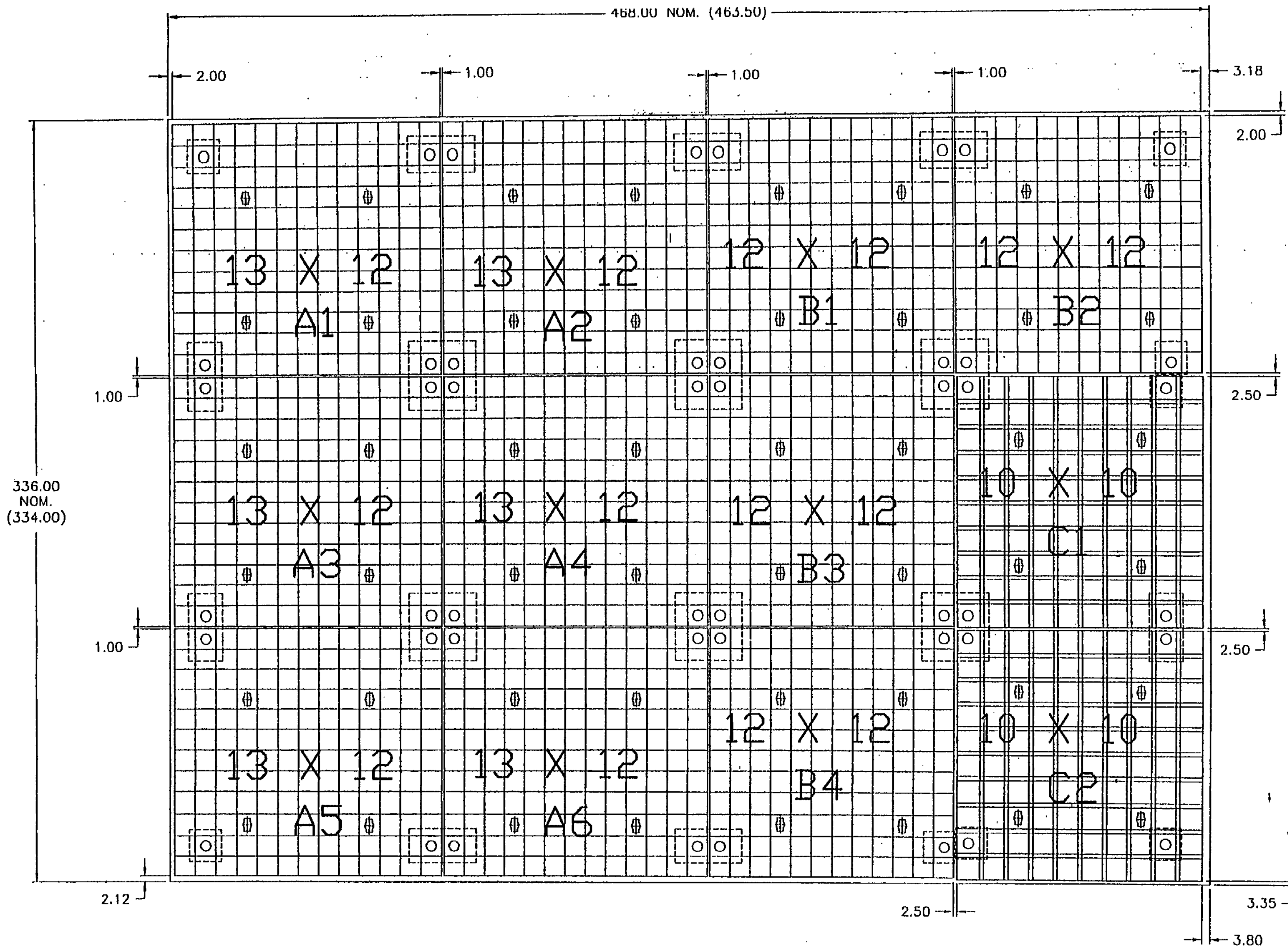

FIGURE 1.1.1 SPENT FUEL POOL RACK LAYOUT

2.0 HIGH DENSITY STORAGE RACKS

2.1 Introduction

In its fully implemented configuration, the VCSNS Spent Fuel Pool will contain twelve fuel racks with a total storage capacity of 1712 assemblies. All spent fuel storage racks will consist of freestanding modules, made primarily from Type 304L austenitic stainless steel containing honeycomb storage cells interconnected through longitudinal welds. A panel of Boral cermet containing a high areal loading of the Boron-10 (B-10) isotope provides appropriate neutron attenuation between adjacent storage cells. Figures 2.1.1 and 2.1.2 provide isometric schematics of typical Region 1 and Region 2 storage rack modules, respectively. Data on the cross sectional dimensions, weight and cell count for each rack module is presented in Table 2.1.1.

The spent fuel rack modules that do not utilize flux traps between storage cells are referred to as Region 2 style racks in wet storage technology terminology. Region 1 style racks contain a water gap (a.k.a flux traps) between storage cells to provide greater margin against reactivity, thereby allowing more reactive fuel to be stored within.

The baseplates on all spent fuel rack modules extend out beyond the rack module periphery wall such that the plate protrusions act to set a required minimum separation between the facing cells in adjacent rack modules. Each spent fuel rack module is supported by a minimum of four pedestals, which are remotely adjustable. Thus, the racks can be made vertical and the top of the racks can easily be made co-planar with each other. The rack module support pedestals are engineered to accommodate minor level variations in the pool floor flatness.

Between the rack module pedestals and the pool floor liner is a bearing pad, which serves to diffuse the dead load of the loaded racks into the reinforced concrete structure of the pool slab. Additional bearing pads, already existing on the pool floor from a previous rack installation, will also be relied upon to provide additional pedestal load distribution on the liner.

The overall design of the rack modules is similar to those presently in service in the spent fuel pools at many other nuclear plants, among them Davis-Besse of First Energy, Callaway of Union Electric, and Byron-Braidwood of Exelon. Altogether, over 50 thousand storage cells of this design have been provided by Holtec International to various nuclear plants around the world.

2.2 Summary of Principal Design Criteria

The key design criteria for the new spent fuel racks are set forth in the USNRC memorandum entitled "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", dated April 14, 1978 as modified by amendment dated January 18, 1979. The individual sections of this report expound on the specific design bases derived from the above-mentioned "OT Position Paper". The design bases for the new SFP storage racks are summarized in the following:

a. Disposition: All new rack modules are required to be free-standing.

b. Kinematic Stability: All freestanding modules must be kinematically stable (against tipping or overturning) if a seismic event is imposed on any module.

c. Structural Compliance: All primary stresses in the rack modules must satisfy the limits postulated in Section III subsection NF of the ASME B & PV Code.

d. Thermal-Hydraulic Compliance: The spatial average bulk pool temperature is required to remain below 165ºF in the wake of a partial core offload or a full core offload with two operating cooling loops and below 170ºF subsequent to a full core offload with only one cooling loop in operation.

e. Criticality Compliance: The NFSR and SFSRs must be able to store Zircaloy clad fuel of 5.0 weight percent (w/o) maximum enrichment while maintaining the reactivity (K_{eff}) less than 0.95.

f. Bearing Pads: The bearing pad size and thickness must ensure that the pressure transferred through the liner to the concrete base slab continues to satisfy the American Concrete Institute (ACI) limits during and after a seismic event.

g. Accident Events: In the event of postulated drop events (uncontrolled lowering of a fuel assembly, for instance), it is necessary to demonstrate that the subcritical geometry of the rack structure is not compromised.

The foregoing design bases are further articulated in Sections 4 through 7 of this licensing report.

2.3 Applicable Codes and Standards

The following codes, standards and practices are used as applicable for the design, construction, and assembly of the fuel storage racks. Additional specific references related to detailed analyses are given in each section.

a. Design Codes

(1) American Institute of Steel Construction (AISC) Manual of Steel Construction, 9th Edition, 1989.

(2) American National Standards Institute/ American Nuclear Society ANSI/ANS-57.2-1983, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants" (contains guidelines for fuel rack design).

(3) ASME B & PV Code Section III, 1989 Edition; ASME Section VIII, 1989 Edition; ASME Section IX, 1989 Edition.

(4) American Society for Nondestructive Testing SNT-TC-1A, June 1980, Recommended Practice for Personnel Qualifications and Certification in Non-destructive Testing.

(5) American Concrete Institute Building Code Requirements for Reinforced Concrete (ACI 318-71).

(6) Code Requirements for Nuclear Safety Related Concrete Structures, ACI 349-76/ACI 349R-76, and ACI 349.1R-80.

(7) ASME Y14.5M, Dimensioning and Tolerancing

(8) ASME B & PV Code, Section II-Parts A and C, 1989 Edition.

(9) ASME B & PV Code NCA3800 - Metallic Material Organization's Quality System Program.

b. Standards of American Society for Testing and Materials (ASTM)

(1) ASTM E165 - Standard Test Method for Liquid Penetrant Examination.

(2) ASTM A240 - Standard Specification for Heat-Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet and Strip for Pressure Vessels.

(3) ASTM A262 - Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steel.

(4) ASTM A276 - Standard Specification for Stainless Steel Bars and Shapes.

(5) ASTM A479 - Standard Specification for Stainless Steel Bars and Shapes for use in Boilers and other Pressure Vessels.

(6) ASTM A564 - Standard Specification for Hot-Rolled and Cold-Finished Age-Hardening Stainless Steel Bars and Shapes.

(7) ASTM C750 - Standard Specification for Nuclear-Grade Boron Carbide Powder.

(8) ASTM A380 - Standard Practice for Cleaning, Descaling, and Passivation of Stainless Steel Parts, Equipment and Systems.

(9) ASTM C992 - Standard Specification for Boron-Based Neutron Absorbing Material Systems for Use in Nuclear Spent Fuel Storage Racks.

(10) ASTM E3 - Standard Practice for Preparation of Metallographic Specimens.

(11) ASTM E190 - Standard Test Method for Guided Bend Test for Ductility of Welds.

c. Welding Code:

ASME B & PV Code, Section IX - Welding and Brazing Qualifications, 1989.

d. Quality Assurance, Cleanliness, Packaging, Shipping, Receiving, Storage, and Handling

(1) ANSI N45.2.1 - Cleaning of Fluid Systems and Associated Components during Construction Phase of Nuclear Power Plants - 1973 (R.G. 1.37).

(2) ANSI N45.2.2 - Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Power Plants - 1972 (R.G. 1.38).

(3) ANSI N45.2.6 - Qualifications of Inspection, Examination, and Testing Personnel for the Construction Phase of Nuclear Power Plants - 1978. (R.G. 1.58).

(4) ANSI N45.2.8 - Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Plants - 1975 (R.G. 1.116).

(5) ANSI N45.2.11 - Quality Assurance Requirements for the Design of Nuclear Power Plants - 1974 (R.G. 1.64).

(6) ANSI N45.2.12 - Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants - 1977 (R.G. 1.144).

(7) ANSI N45.2.13 - Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants - 1976 (R. G. 1.123).

(8) ANSI N45.2.23 - Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants - 1978 (R.G. 1.146).

(9) ASME B & PV Code, Section V, Nondestructive Examination, 1992 Edition.

(10) ANSI N16.9-75 - Validation of Calculation Methods for Nuclear Criticality Safety.

(11) ASME NQA-1 – Quality Assurance Program Requirements for Nuclear Facilities.

(12) ASME NQA-2 – Quality Assurance Requirements for Nuclear Power Plants.

e. USNRC Documents

(1) "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, and the modifications to this document of January 18, 1979.

(2) NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants", USNRC, Washington, D.C., July, 1980.

f. Other ANSI Standards (not listed in the preceding)

(1) ANSI/ANS 8.1 - Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.

(2) ANSI/ANS 8.17 - Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors.

(3) ANSI N45.2 - Quality Assurance Program Requirements for Nuclear Power Plants - 1977.

(4) ANSI N45.2.9 - Requirements for Collection, Storage and Maintenance of Quality Assurance Records for Nuclear Power Plants - 1974.

(5) ANSI N45.2.10 - Quality Assurance Terms and Definitions - 1973.

(6) ANSI N14.6 - American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4500 kg) or more for Nuclear Materials - 1993.

(7) ANSI/ASME N626-3 - Qualification and Duties of Specialized Professional Engineers.

(8) ANSI/ANS- 57.3 – Design Requirements for New Fuel Storage Facilities at Light Water Reactor Plants.

g. Code-of-Federal Regulations (CFR)

(1) 10CFR20 - Standards for Protection Against Radiation.

(2) 10CFR21 - Reporting of Defects and Non-compliance.

(3) 10CFR50 Appendix A - General Design Criteria for Nuclear Power Plants.

(4) 10CFR50 Appendix B - Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.

(5) 10CFR61 - Licensing Requirements for Land Disposal of Radioactive Waste.

(6) 10CFR71 - Packaging and Transportation of Radioactive Material.

(7) 10CFR100 – Reactor Site Criteria

h. Regulatory Guides (RG)

(1) RG 1.13 - Spent Fuel Storage Facility Design Basis (Revision 2 Proposed).

(2) RG 1.25 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors, Rev. 0 - March, 1972.

(3) RG 1.28 - Quality Assurance Program Requirements - Design and Construction, Rev. 2 - February, 1979 (endorses ANSI N45.2).

(4) RG 1.33 – Quality Assurance Program Requirements.

(5) RG 1.29 - Seismic Design Classification, Rev. 2 - February, 1976.

(6) RG 1.31 - Control of Ferrite Content in Stainless Steel Weld Metal.

(7) RG 1.38 - Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants, Rev. 2 - May, 1977 (endorses ANSI N45.2.2).

(8) RG 1.44 - Control of the Use of Sensitized Stainless Steel.

(9) RG 1.58 - Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel, Rev. 1 - September 1980 (endorses ANSI N45.2.6).

(10) RG 1.60 – Design Response Spectra for Seismic Design of Nuclear Power Plants.

(11) RG 1.61 - Damping Values for Seismic Design of Nuclear Power Plants, Rev. 0, 1973.

(12) RG 1.64 - Quality Assurance Requirements for the Design of Nuclear Power Plants, Rev. 2 - June, 1976 (endorses ANSI N45.2.11).

(13) RG 1.71 - Welder Qualifications for Areas of Limited Accessibility.

(14) RG 1.74 - Quality Assurance Terms and Definitions, Rev. 2 - February, 1974 (endorses ANSI N45.2.10).

(15) RG 1.85 - Materials Code Case Acceptability - ASME Section III, Division 1.

(16) RG 1.88 - Collection, Storage and Maintenance of Nuclear Power Plant Quality Assurance Records, Rev. 2 - October, 1976 (endorses ANSI N45.2.9).

(17) RG 1.92 - Combining Modal Responses and Spatial Components in Seismic Response Analysis, Rev. 1 - February, 1976.

(18) RG 1.116 - Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems, Rev. 0-R - May,1977 (endorses ANSI N45.2.8-1975)

(19) RG 1.123 - Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants, Rev. 1 - July, 1977 (endorses ANSI N45.2.13).

(20) RG 1.124 - Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports, Revision 1, January,1978.

(21) RG 1.144 - Auditing of Quality Assurance Programs for Nuclear Power Plants, Rev.1 - September, 1980 (endorses ANSI N45.2.12-1977)

(22) RG 3.4 - Nuclear Criticality Safety in Operations with Fissionable Materials at Fuels and Materials Facilities.

(23) RG 8.8 - Information Relative to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as Reasonably Achievable (ALARA).

(24) IE Information Notice 83-29 - Fuel Binding Caused by Fuel Rack Deformation.

(25) RG 8.38 - Control of Access to High and Very High Radiation Areas in Nuclear Power Plants, June, 1993.

i. <u>Branch Technical Position</u>

(1) CPB 9.1-1 - Criticality in Fuel Storage Facilities.

(2) APCSB 9-2 - Residual Decay Energy for Light-Water Reactors for Long-Term Cooling - November, 1975.

j. American Welding Society (AWS) Standards

(1) AWS D1.1 - Structural Welding Code - Steel.

(2) AWS D1.3 - Structure Welding Code - Sheet Steel.

(3) AWS D9.1 - Sheet Metal Welding Code.

(4) AWS A2.4 - Standard Symbols for Welding, Brazing and Nondestructive Examination.

(5) AWS A3.0 - Standard Welding Terms and Definitions.

(6) AWS A5.12 - Specification for Tungsten and Tungsten Alloy Electrodes for Arc-Welding and Cutting

(7) AWS QC1 - Standard for AWS Certification of Welding Inspectors.

(8) AWS 5.4 – Specification for Stainless Steel Electrodes for Shielded Metal Arc Welding.

(9) AWS 5.9 – Specification for Bare Stainless Steel Welding Electrodes and Rods.

2.4 Quality Assurance Program

The governing quality assurance requirements for design and fabrication of the spent fuel racks are stated in 10CFR50 Appendix B. Holtec's Nuclear Quality Assurance program complies with this regulation and is designed to provide a system for the design, analysis and licensing of customized components in accordance with various codes, specifications, and regulatory requirements.

The manufacturing of the racks will be carried out by Holtec's designated manufacturer, U.S. Tool & Die, Inc. (UST&D). The Quality Assurance system enforced on the manufacturer's shop floor shall provide for all controls necessary to fulfill all quality assurance requirements. UST&D has manufactured high-density racks for over 60 nuclear plants around the world. UST&D has been audited by the nuclear industry group Nuclear Procurement Issues Committee (NUPIC), and the Quality Assurance branch of the USNRC Office of Nuclear Material Safety and Safeguards (NMSS) with satisfactory results.

The Quality Assurance System that will be used by Holtec to install the racks is also controlled by the Holtec Nuclear Quality Assurance Manual and by the VCSNS site-specific requirements.

2.5 Mechanical Design

The VCSNS rack modules are designed as cellular structures such that each fuel assembly has a square opening with conforming lateral support and a flat horizontal-bearing surface. All of the storage locations are constructed with multiple cooling flow holes to ensure that redundant flow paths for the coolant are available. The basic characteristics of the spent fuel racks are summarized in Table 2.5.1.

A central objective in the design of the new rack modules is to maximize structural strength while minimizing inertial mass and dynamic response. Accordingly, the rack modules have been designed to simulate multi-flange beam structures resulting in excellent de-tuning characteristics with respect to the applicable seismic events. The next subsection presents an item-by-item description of the rack modules in the context of the fabrication methodology.

2.6 Rack Fabrication

The object of this section is to provide a brief description of the rack module construction activities, which enable an independent appraisal of the adequacy of design. The pertinent methods used in manufacturing the new and spent fuel storage racks may be stated as follows:

1. The rack modules are fabricated in such a manner that the storage cell surfaces, which would come in contact with the fuel assembly, will be free of harmful chemicals and projections (e.g., weld splatter).

2. The component connection sequence and welding processes are selected to reduce fabrication distortions.

3. The fabrication process involves operational sequences that permit immediate accessibility for verification by the inspection staff.

4. The racks are fabricated per the UST&D Appendix B Quality Assurance program, which ensures, and documents, that the fabricated rack modules meet all of the requirements of the design and fabrication documents.

5. The storage cells are connected to each other by austenitic stainless steel corner welds which leads to a honeycomb lattice construction. The extent of welding is selected to "detune" the racks from the seismic input motion

2.6.1 Rack Module for Region 1

This section describes the constituent elements of the VCSNS Region 1 rack modules in the fabrication sequence. Figure 2.1.1 provides a schematic view of a typical Region 1 rack.

The rack module manufacturing begins with fabrication of the "box". The boxes are fabricated from two precision formed channels by seam welding in a machine equipped with copper chill bars and pneumatic clamps to minimize distortion due to welding heat input. Figure 2.6.1 shows the box. The minimum weld seam penetration is 80% of the box metal gage, which is 0.075 inch (14 gage).

A die is used to flare out one end of the box to provide the tapered lead-in (Figure 2.6.2). One-inch diameter holes are punched on at least two sides near the other end of the box to provide the requisite auxiliary flow holes.

Each box constitutes a storage location. Each external box side is equipped with a stainless steel sheathing, which holds one integral Boral sheet (poison material). The design objective calls for attaching Boral tightly on the box surface. This is accomplished by die forming the internal and external box sheathings, as shown in Figure 2.6.3. The flanges of the sheathing are attached to the box using skip welds and spot welds. The sheathings serve to locate and position the poison sheet accurately, and to preclude its movement under seismic conditions.

Having fabricated the required number of composite box assemblies, they are joined together in a fixture using connector elements in the manner shown in Figure 2.6.4. Figure 2.6.5 shows an elevation view of two storage cells of a Region 1 rack module. A representative connector element is also shown in the figure. Joining the cells by the connector elements results in a well-defined shear flow path, and essentially makes the box assemblage into a multi-flanged beam-type structure. The "baseplate" is attached to the bottom edge of the boxes. The baseplate is a 0.75 inch thick austenitic stainless steel plate stock which has 5-1/4 inch diameter holes (except lift locations, which are rectangular) cut out in a pitch identical to the box pitch. The baseplate is attached to the cell assemblage by fillet welding the box edge to the plate.

In the final step, adjustable leg supports (shown in Figure 2.6.6) are welded to the underside of the baseplate. The adjustable legs provide a ± 1/2-inch vertical height adjustment at each leg location.

Appropriate NDE (nondestructive examination) occurs on all welds including visual examination of sheathing welds, box longitudinal seam welds, box-to-baseplate welds, and box-to-box connection welds; and liquid penetrant examination of support leg welds, in accordance with the design drawings.

2.6.3 Rack Module for Region 2

Region 2 storage cell locations have a single poison panel between adjacent cell boxes on the wall surfaces separating them. The significant components (discussed below) of the Region 2 racks are: (1) the storage box subassembly (2) the baseplate, (3) the neutron absorber material, (4) the sheathing, and (5) the support legs.

1. Storage cell box subassembly: As described for Region 1, the boxes are fabricated from two precision formed channels by seam welding in a machine equipped with copper chill bars and pneumatic clamps to minimize distortion due to welding heat input. Figure 2.6.1 shows the box.

   Each box has two lateral holes punched near its bottom edge to provide auxiliary flow holes. As shown in Figure 2.6.3, sheathing is attached to each side of the box with the poison material installed in the sheathing cavity (per design drawings, box walls which form the external wall of the fuel rack may not have sheathing attached). The edges of the sheathing and the box are welded together to form a smooth edge. The box, with integrally connected sheathing, is referred to as the "composite box".

   The composite boxes are arranged in a checkerboard array to form an assemblage of storage cell locations (Figure 2.6.7). Filler panels and corner angles are welded to the edges of boxes at the outside boundary of the rack to make the peripheral formed cells. The inter-box welding and pitch adjustment are accomplished by small longitudinal connectors. This assemblage of box assemblies is welded edge-to-edge as shown in Figure 2.6.7, resulting in a honeycomb structure with axial, flexural and torsional rigidity depending on the extent of intercell welding provided. It can be seen from Figure 2.6.7 that two edges of each interior box are connected to the contiguous boxes resulting in a well-defined path for "shear flow".

2. Baseplate: The baseplate provides a continuous horizontal surface for supporting the fuel assemblies. The baseplate has a 5-1/4 inch diameter hole (except lift locations which are rectangular) in each cell location as described in the preceding section. The baseplate is attached to the cell assemblage by fillet welds.

3. The neutron absorber material: As mentioned in the preceding section, Boral is used as the neutron absorber material.

4. Sheathing: As described earlier, the sheathing serves as the locator and retainer of the poison material.

5. Support legs: As stated earlier, all support legs are the adjustable type (Figure 2.6.6). The top position is made of austenitic steel material. The bottom part is made of 17:4 Ph series stainless steel to avoid galling problems.

   Each support leg is equipped with a readily accessible socket to enable remote leveling of the rack after its placement in the pool.

An elevation view of three contiguous Region 2 cells is shown in Figure 2.6.8.

TABLE 2.1.1: GEOMETRIC AND PHYSICAL DATA FOR SPENT FUEL RACKS							
MODULE I.D.	RACK TYPE	NO. OF CELLS		MODULE ENVELOPE SIZE		WEIGHT (lbs)	NO. OF CELLS PER RACK
		N-S Direction	E-W Direction	N-S	E-W		
A1	Region 2	12	13	109.295”	118.365”	23,314	156
A2	Region 2	12	13	109.295”	118.365”	23,314	156
A3	Region 2	12	13	109.295”	118.365”	23,321	156
A4	Region 2	12	13	109.295”	118.365”	23,321	156
A5	Region 2	12	13	109.295”	118.365”	22,836	156
A6	Region 2	12	13	109.295”	118.365”	22,836	156
B1	Region 2	12	12	109.295”	109.295”	21,601	144
B2	Region 2	12	12	109.295”	109.295”	21,153	144
B3	Region 2	12	12	109.295”	109.295”	21,608	144
B4	Region 2	12	12	109.295”	109.295”	21,162	144
C1	Region 1	10	10	107.178”	107.178”	25,279	100
C2	Region 1	10	10	107.178”	107.178”	24,895	100

Note: Variations in weights of racks of similar size are due to slight variations in baseplate extensions beyond the periphery of the rack cells.

Table 2.5.1

MODULE DATA FOR REGION 1 SPENT FUEL RACKS †

Storage cell inside nominal dimension	8.85 in.
Cell pitch	10.867 in.
Storage cell height (above the plate)	167.0 in.
Baseplate hole size (except for lift location)	5.25 in.
Baseplate thickness	0.75 in.
Support pedestal height	4.25 in.
Support pedestal type	Remotely adjustable pedestals
Number of support pedestals per rack	4
Number of cell walls containing 1" diameter flow holes at base of cell wall	All Cell Walls
Remote lifting and handling provisions	Yes
Poison material	Boral
Poison length	145 in.
Poison width	7.5 in.

† All dimensions indicate nominal values

Table 2.5.2 MODULE DATA FOR REGION 2 SPENT FUEL RACKS †	
Storage cell inside nominal dimension	8.85 in.
Cell pitch	9.07 in.
Storage cell height (above the plate)	167.0 in.
Baseplate hole size (except for lift location)	5.25 in.
Baseplate thickness	0.75 in.
Support pedestal height	4.25 in.
Support pedestal type	Remotely adjustable pedestals
Number of support pedestals per rack	4
Minimum number of cell walls containing 1" diameter supplemental flow holes at base of each cell located away from pedestals	2
Number of cell walls containing 1" diameter flow holes at base of each cell located above a pedestal	4
Remote lifting and handling provisions	Yes
Poison material	Boral
Poison length	145 in.
Poison width	7.5 in.

† All dimensions indicate nominal values

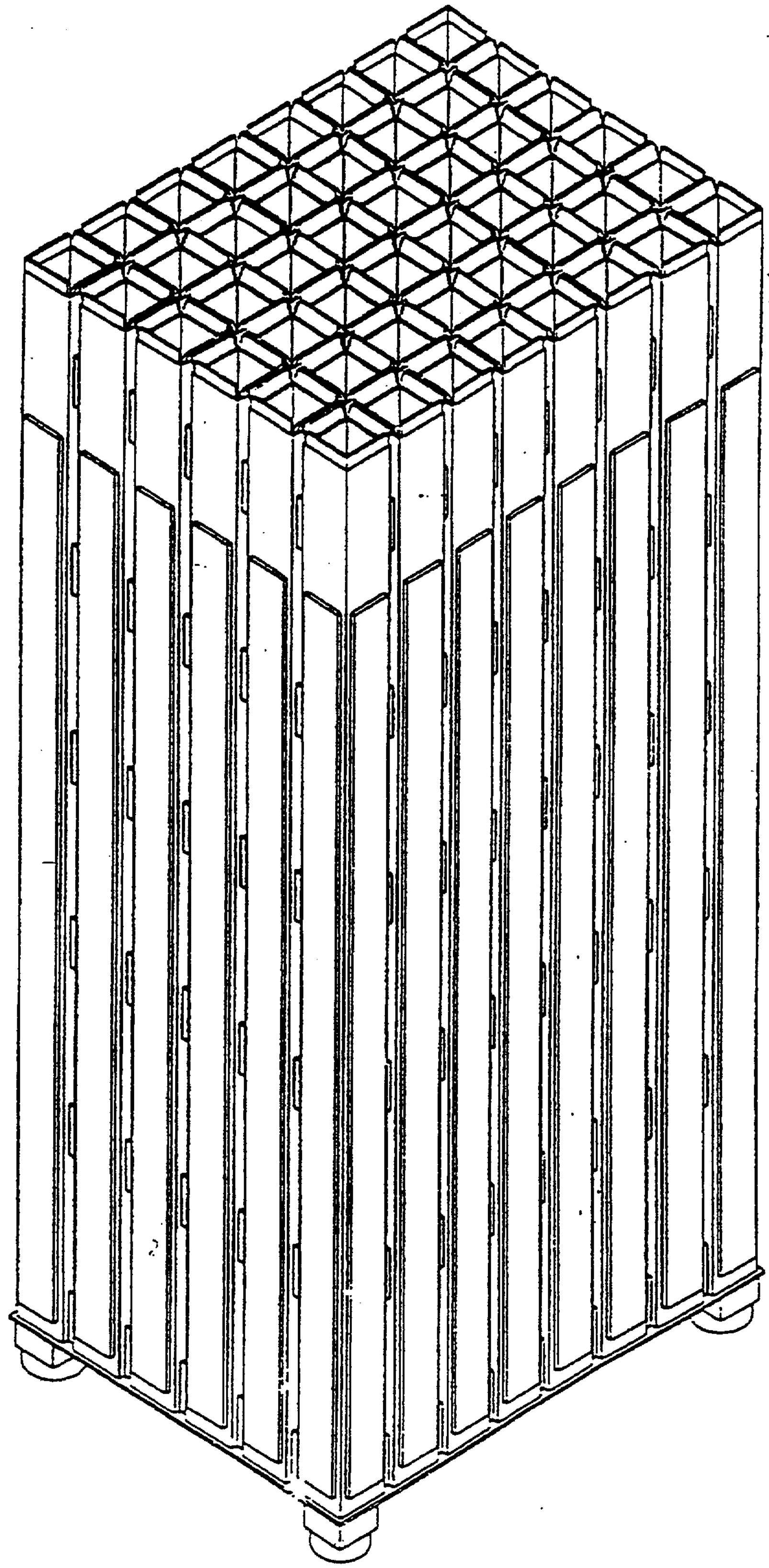

FIGURE 2.1.1 SCHEMATIC VIEW OF REGION 1 RACK STRUCTURE

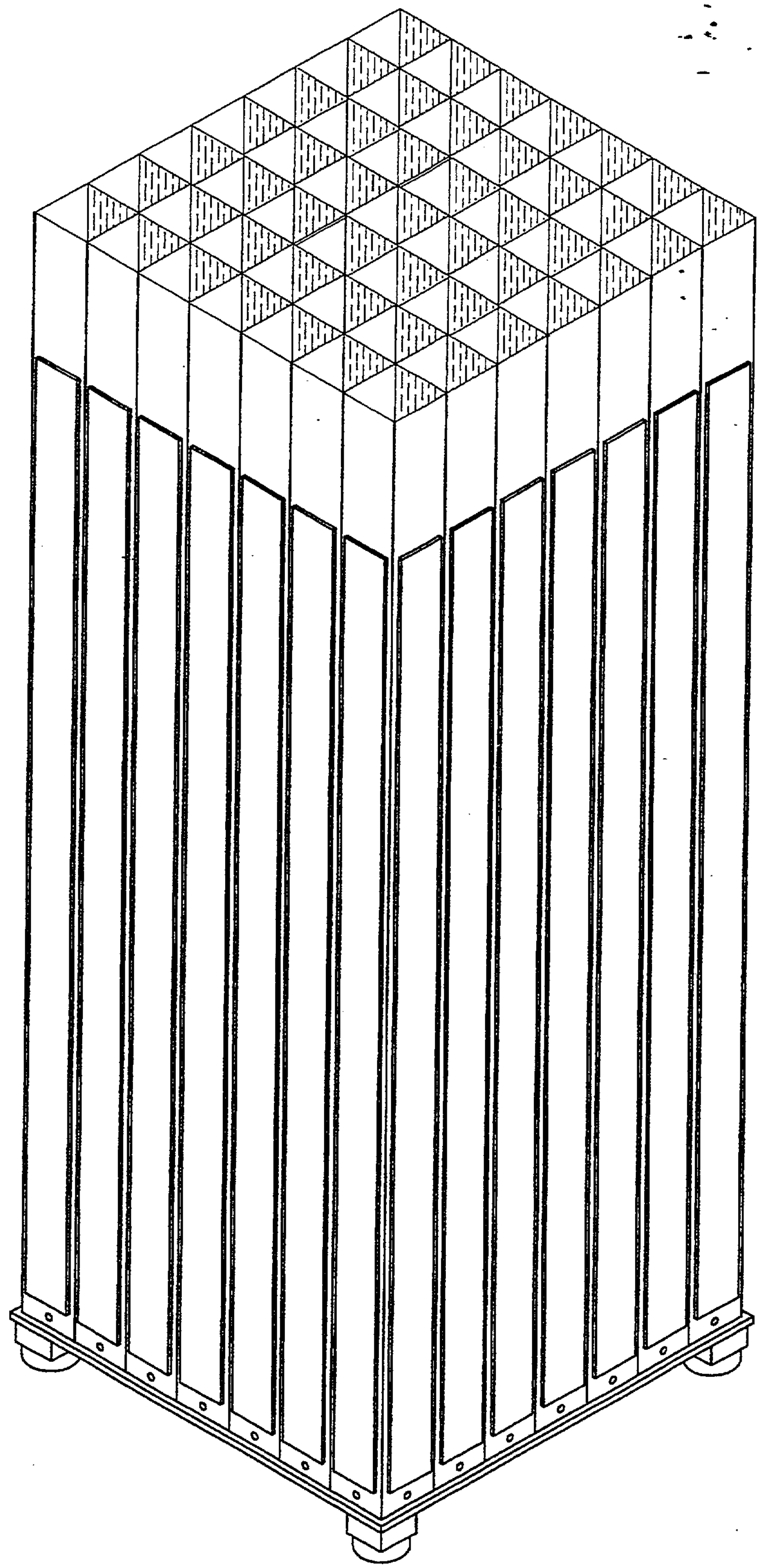

FIGURE 2.2.1; SCHEMATIC OF TYPICAL REGION 2 RACK STRUCTURE

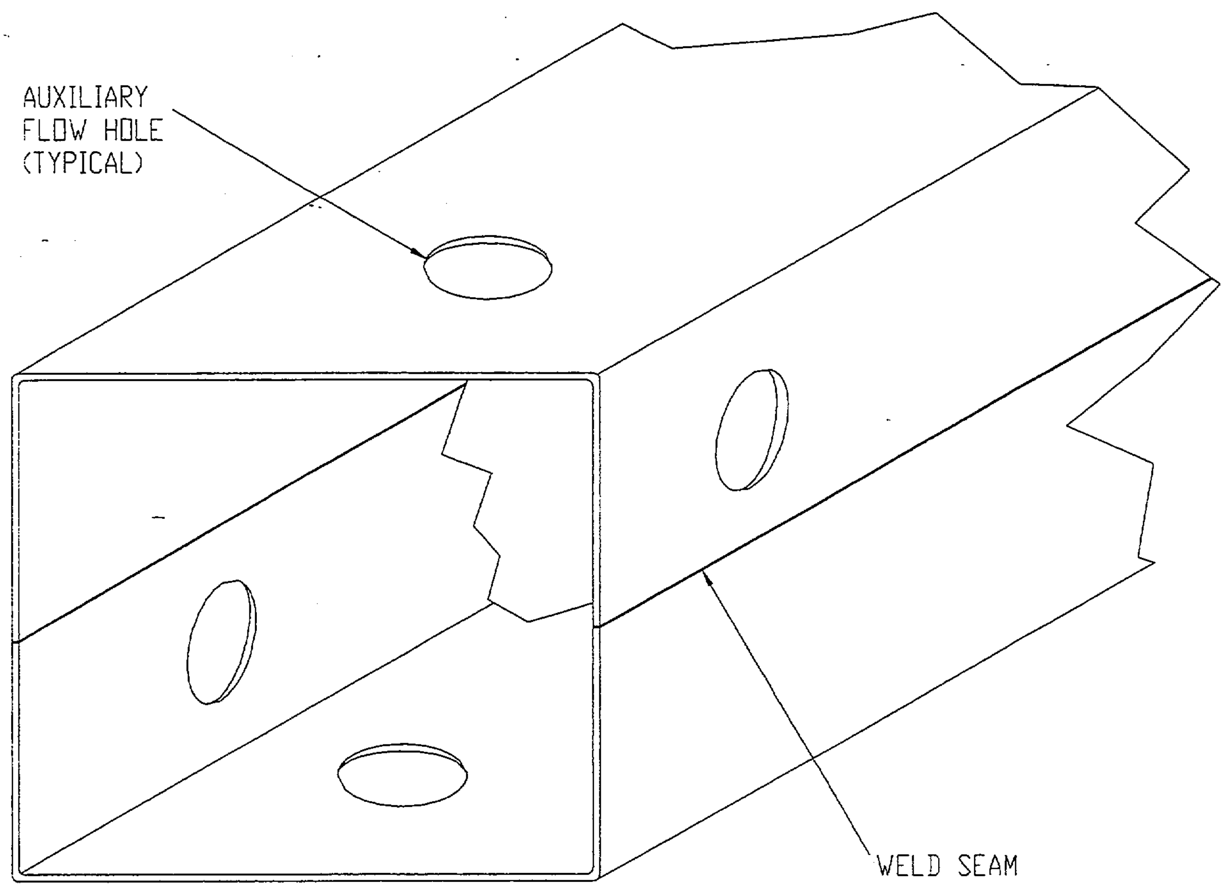

FIGURE 2.6.1j SEAM WELDED PRECISION FORMED CHANNELS

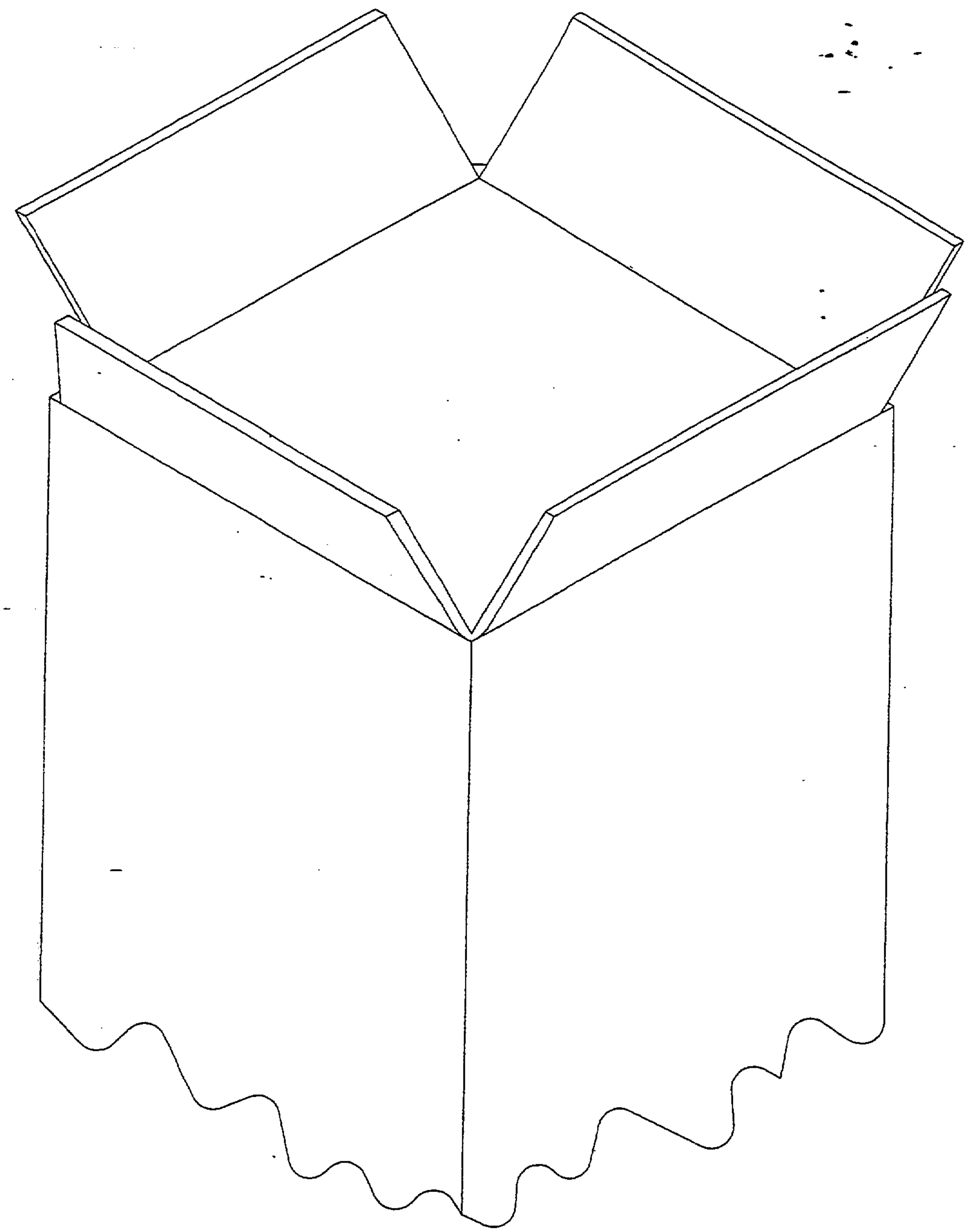

FIGURE 2.6.2; TAPERED REGION 1
CELL END

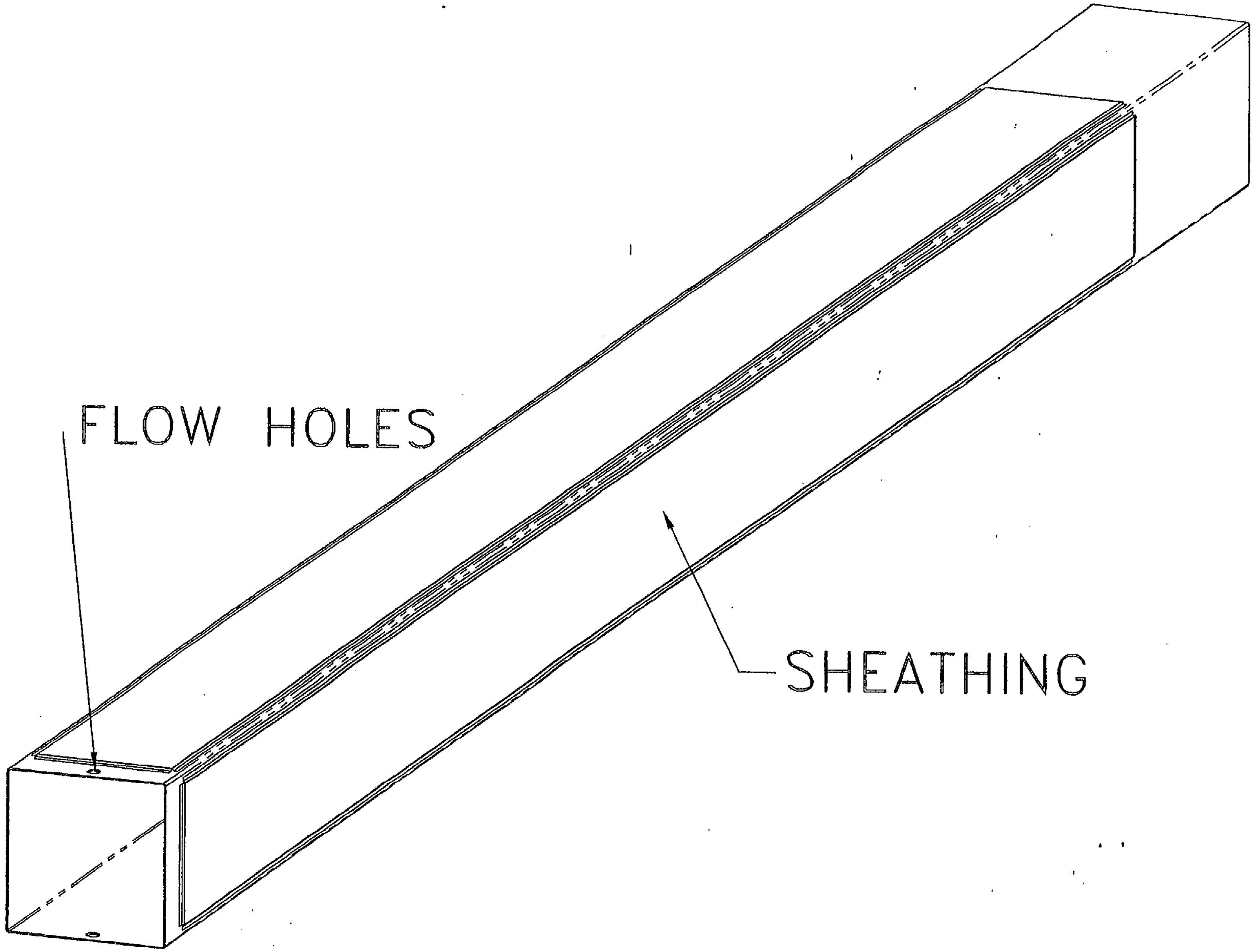

FIGURE 2.6.3; COMPOSITE BOX ASSEMBLY

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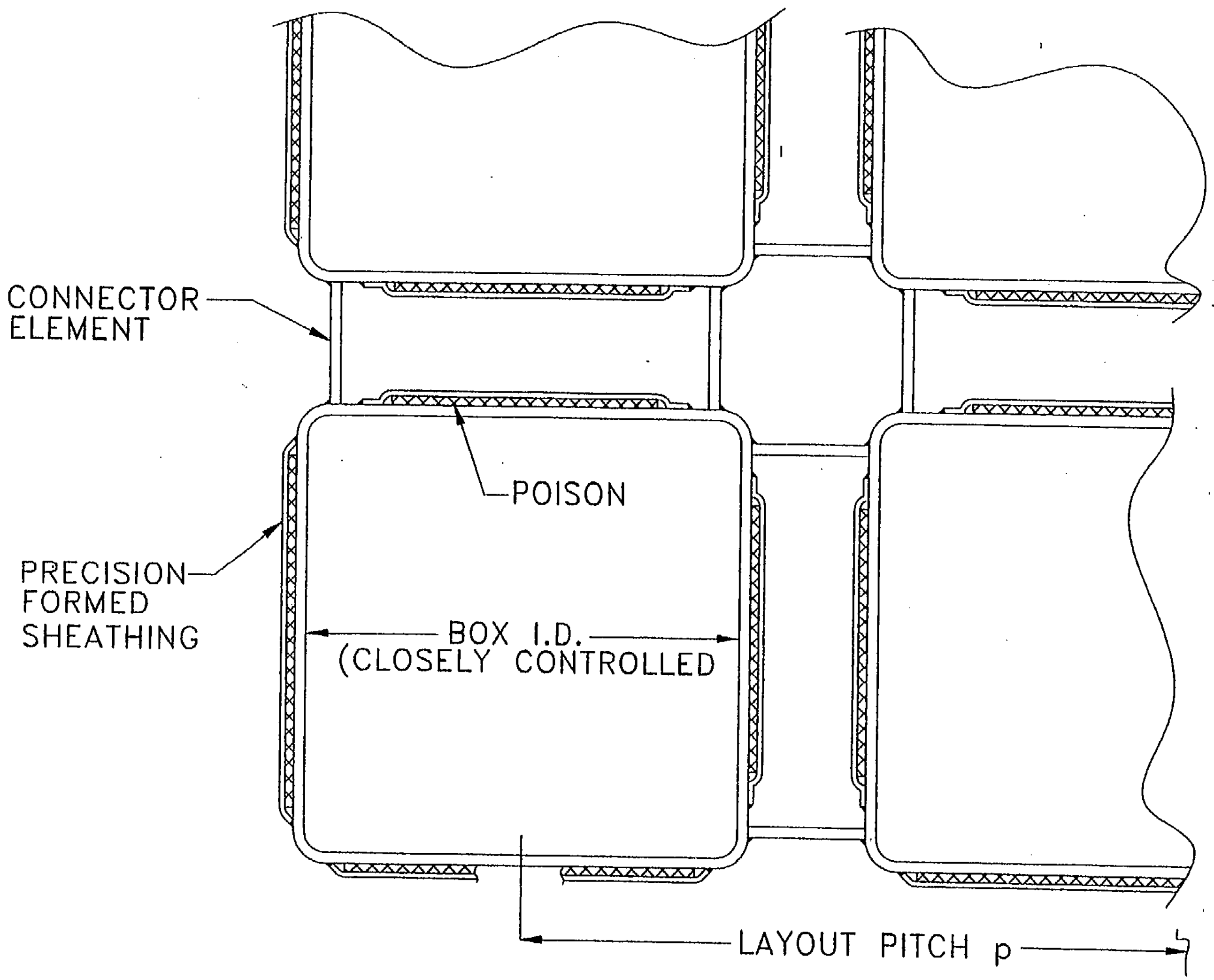

FIGURE 2.6.4; ASSEMBLAGE OF REGION 1 CELLS

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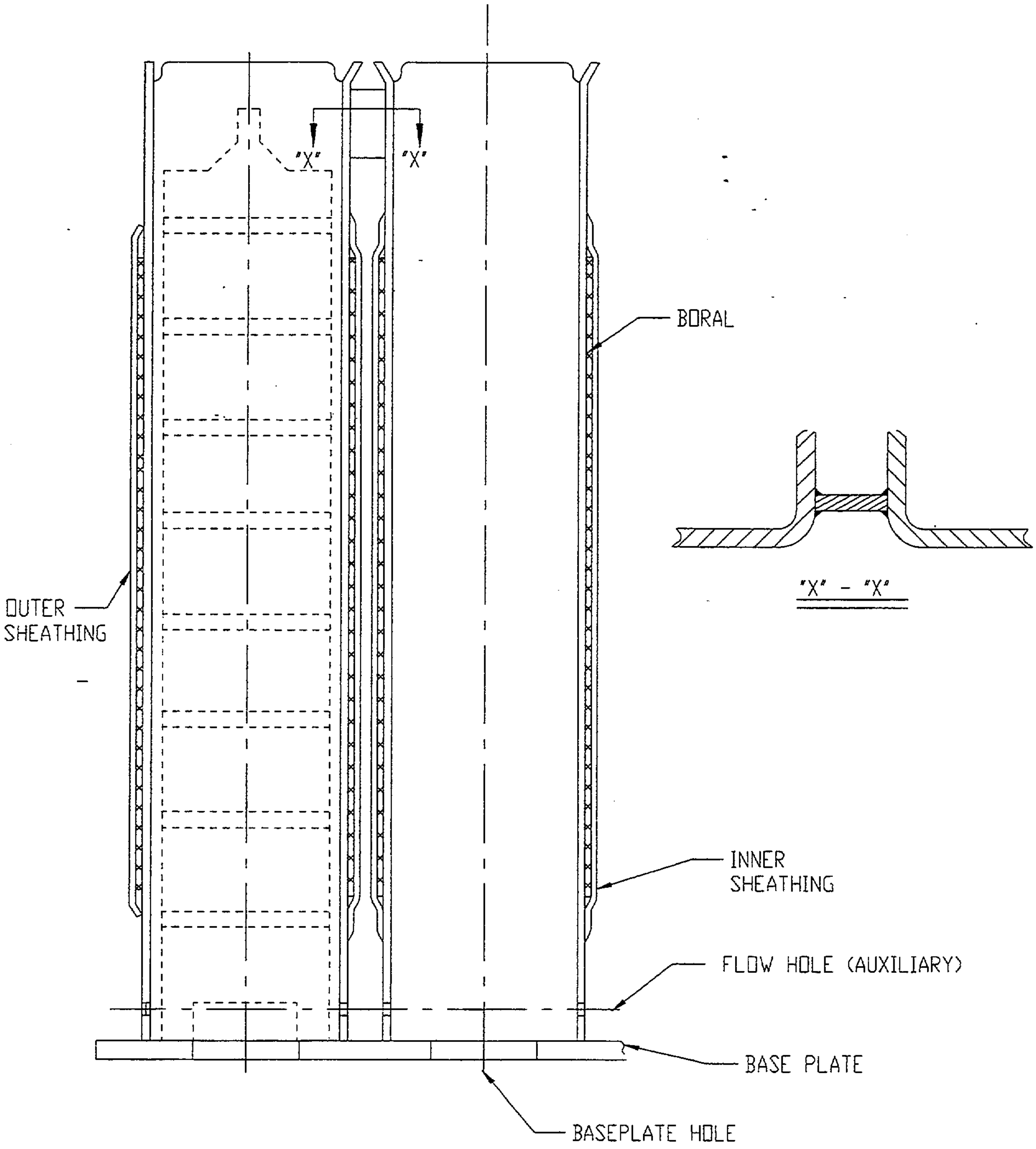
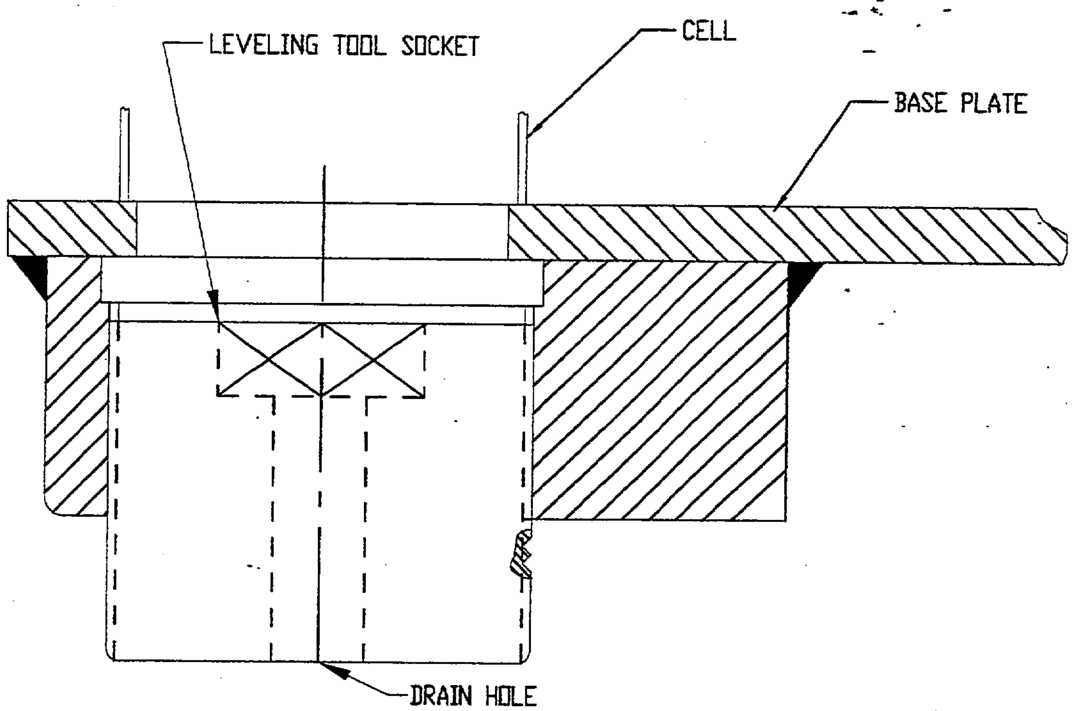
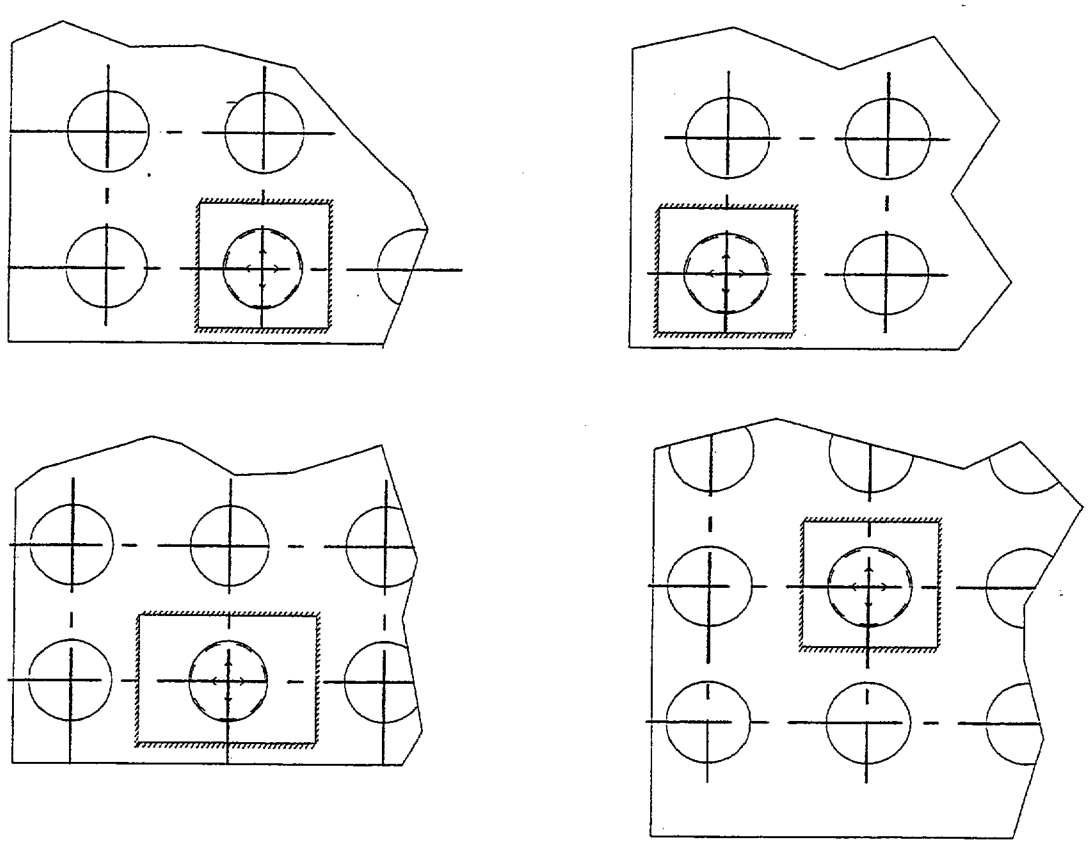


FIGURE 2.6.5: ELEVATION VIEW OF REGION 1 RACK

NOTE: DEPICTION OF STORED FUEL ASSEMBLY IS NOT INTENDED TO BE ACCURATE.





TYPICAL PLAN VIEWS OF RACK BASEPLATE CORNER

FIGURE 2.6.6; SUPPORT PEDESTALS FOR HOLTEC PWR RACKS

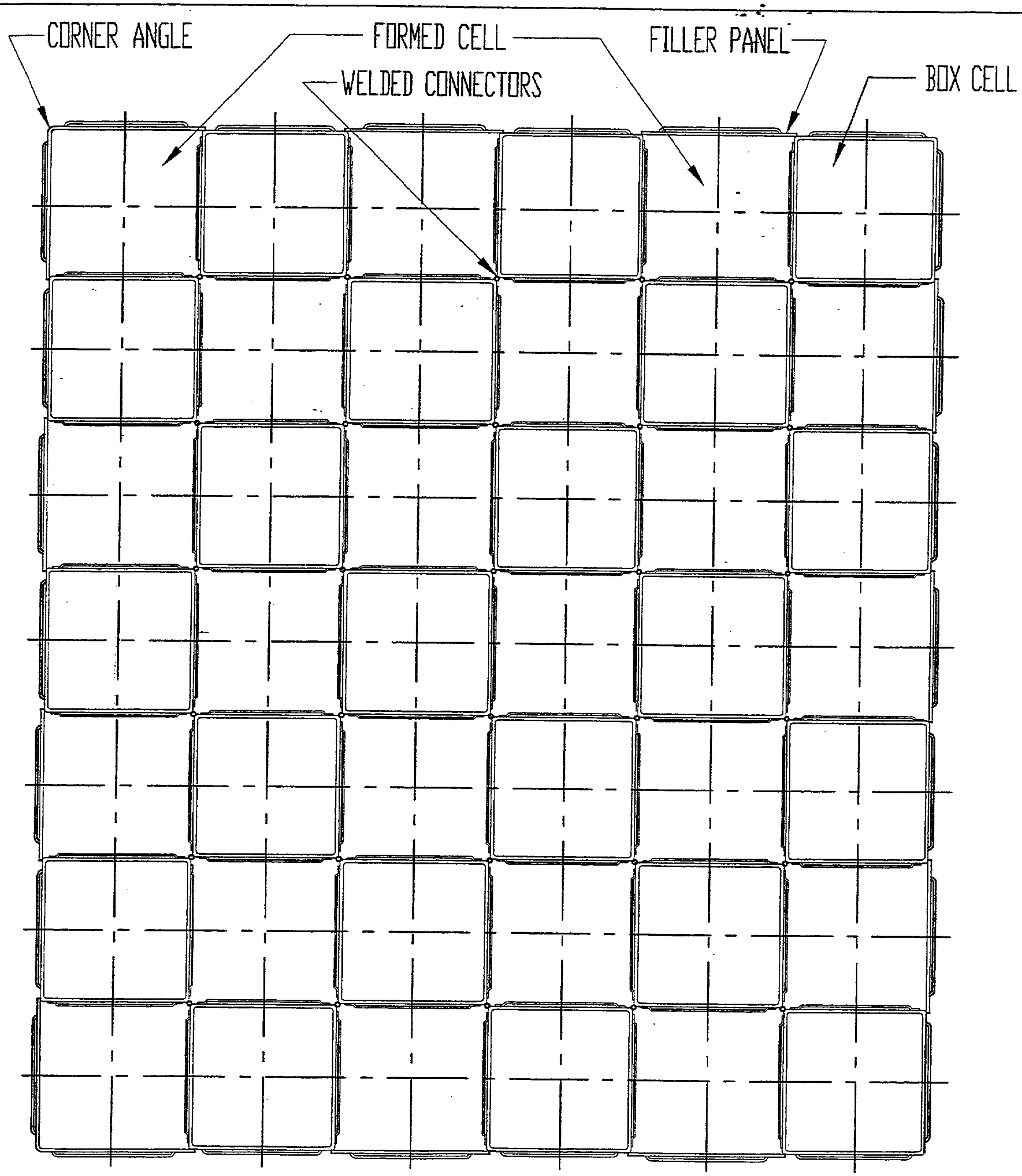


FIGURE 2.6.7j TYPICAL ARRAY OF STORAGE CELLS (NON-FLUX TRAP CONSTRUCTION)

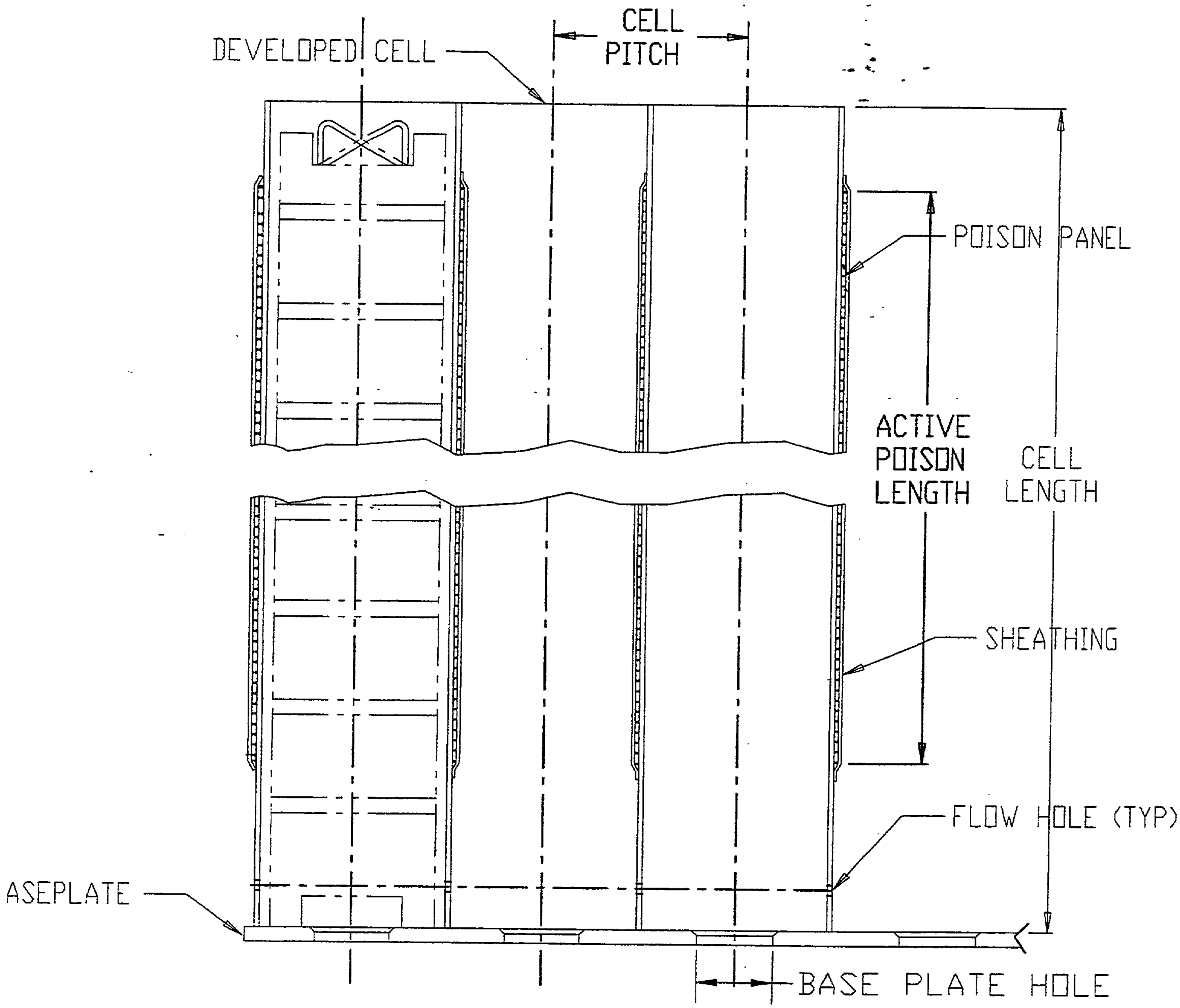


FIGURE 2.6.8; ELEVATION VIEW OF STORAGE RACK MODULE

NOTE: DEPICTION OF STORED FUEL ASSEMBLY IS NOT INTENDED TO BE ACCURATE.

3.0 MATERIAL AND HEAVY LOAD CONSIDERATIONS

3.1 Introduction

Safe storage of nuclear fuel in the pool requires that the materials utilized in the rack fabrication be of proven durability and compatible with the pool water environment. This section provides a synopsis of the considerations with regard to long-term design service life of 60 years.

3.2 Structural Materials

The following structural materials are utilized in the fabrication of the fuel racks:

a. ASTM A240-304L for all sheet metal stock and baseplate

b. Internally threaded support legs: ASTM A240-304L

c. Externally threaded support spindle: ASTM A564-630 precipitation hardened stainless steel (heat treated to 1100°F)

d. Weld material - ASTM Type 308

3.3 Neutron Absorbing Material

In addition to the structural and non-structural stainless material, the racks employ Boral™, a patented product of AAR Manufacturing, as the neutron absorber material. A brief description of Boral, and its pool experience list follows.

Boral is a thermal neutron poison material composed of boron carbide and 1100 alloy aluminum. Boron carbide is a compound having a high boron content in a physically stable and chemically inert form. The 1100 alloy aluminum is a lightweight metal with high tensile strength which is protected from corrosion by a highly resistant oxide film. The two materials, boron carbide and aluminum, are chemically compatible and ideally suited for long-term use in the radiation, thermal and chemical environment of a nuclear reactor or a spent fuel pool. Boral has been shown [3.3.1] to be superior to alternative materials previously used as neutron absorbers in storage racks.

Boral has been exclusively used in fuel rack applications in recent years. Its use in the spent fuel pools as the neutron absorbing material can be attributed to its proven performance (over 150 pool years of experience) and the following unique characteristics:

i. The content and placement of boron carbide provides a very high removal cross-section for thermal neutrons.

ii. Boron carbide, in the form of fine particles, is homogeneously dispersed throughout the central layer of the Boral panels.

iii. The boron carbide and aluminum materials in Boral do not degrade as a result of long-term exposure to radiation.

iv. The neutron absorbing central layer of Boral is clad with permanently bonded surfaces of aluminum.

v. Boral is stable, strong, durable, and corrosion resistant.

Holtec International's Q.A. program ensures that Boral is manufactured by AAR Manufacturing under the control and surveillance of a Quality Assurance/Quality Control Program that conforms to the requirements of 10CFR50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants".

As indicated in Tables 3.3.1 and 3.3.2, Boral has been licensed by the USNRC for use in numerous BWR and PWR spent fuel storage racks and has been extensively used in international nuclear installations.

3.3.1 Boral Material Characteristics

Aluminum: Aluminum is a silvery-white, ductile metallic element. The 1100 alloy aluminum is used extensively in heat exchangers, pressure and storage tanks, chemical equipment, reflectors and sheet metal work.

It has high resistance to corrosion in industrial and marine atmospheres. Aluminum has atomic number of 13, atomic weight of 26.98, specific gravity of 2.69 and valence of 3. The physical, mechanical and chemical properties of the 1100 alloy aluminum are listed in Tables 3.3.3 and 3.3.4.

The excellent corrosion resistance of the 1100 alloy aluminum is provided by the protective oxide film that quickly develops on its surface from exposure to the atmosphere or water. This film prevents the loss of metal from general corrosion or pitting corrosion.

Boron Carbide: The boron carbide contained in Boral is a fine granulated powder that conforms to ASTM C-750-80 nuclear grade Type III. The material conforms to the chemical composition and properties listed in Table 3.3.5.

References [3.3.2], [3.3.3], and [3.3.4] provide further discussion as to the suitability of these materials for use in spent fuel storage module applications.

3.4 Compatibility with Environment

All materials used in the construction of the Holtec racks have been determined to be compatible with the VCSNS Spent Fuel Pool, and have an established history of in-pool usage. As evidenced in Tables 3.3.1 and 3.3.2, Boral has been successfully used in fuel pools. Austenitic stainless steel (304L) is a widely used stainless alloy in nuclear power plants.

3.5 Heavy Load Considerations for the Proposed Rack Installations

The Fuel Handling Building Crane (FHBC) will be used for lifting the new racks onto the Fuel Building operating deck and removing the existing racks from the Fuel Building operating deck. The FHBC will also be used to assemble a temporary crane on the operating deck. The FHBC will not be used to manipulate racks or assemble the temporary crane in the vicinity of the SFP, since this crane does not have access to this area of the Fuel Building. The temporary gantry crane, having a rated lifting capacity of 37.5 metric tons (41.3 short tons), will be installed to place the new storage racks into, and remove the existing racks from, the SFP. This temporary crane is designed to meet the intent of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants".

Safe handling of heavy loads by the FHBC and the temporary crane will be ensured by following the defense in depth approach guidelines of NUREG 0612:

- Defined safe load paths in accordance with approved procedures
- Supervision of heavy load lifts by designated individuals
- Crane operator training and qualification that satisfies the requirements of ANSI/ASME B30.2-1976 [3.5.1]
- Use of lifting devices (slings) that are selected, inspected and maintained in accordance with ANSI B30.9-1971 [3.5.2]
- Inspection, testing and maintenance of cranes in accordance with ANSI/ASME B30.2-1976
- Ensuring the design of the FHBC and the temporary crane meets the requirements of CMAA-70 [3.5.3] and ANSI/ASME B30.2-1976
- Reliability of special lifting devices by application of design safety margins, and periodic inspection and examinations using approved procedures

The salient features of the lifting devices and associated procedures are described as follows:

a. Safe Load Paths and Procedures

Safe load paths will be defined for moving the new racks in the Fuel Handling Building. The racks will be lifted up through the equipment hatch from the truck bay at elevation 436'-0" using the FHBC and placed on the SFP operating deck at elevation 463'-0". A staging area will be setup on the operating deck as a laydown area for the new racks. When a new rack is to be installed into the SFP, it will be moved to the Decon Area by the FHBC. This area will be used for transfer of the heavy rack loads from the FHBC to the temporary crane. The same process will be used for the removal of the existing racks from the SFP, except in reverse. As shown in Figure 3.5.1, the SFP is located southwest of the equipment hatch. The staging area location will not require any heavy load to be lifted over the SFP or any safety-related equipment. The FHBC is not capable of travel over the SFP. Therefore, during lifts by the FHBC, the new racks will not be carried directly over any portion of the SFP. Rack lifts over the SFP will be performed using the temporary crane, which is discussed in more detail in Section 10. A spent fuel shuffling and storage rack movement plan has been developed to ensure that racks will not be carried over spent fuel. Fuel will be moved away from load paths prior to any lifted racks being carried over installed racks.

All phases of rack installation activities will be conducted in accordance with written procedures, which will be reviewed and approved by the owner.

b. Supervision of Lifts

Procedures used during the installation of the SFP racks require supervision of heavy load lifts by a designated individual who is responsible for ensuring procedure compliance and safe lifting practices.

c. Crane Operator Training

All crew members involved in the use of the lifting and upending equipment will be given training by Holtec International using a videotape-aided instruction course which has been utilized in previous rerack operations.

d. Lifting Devices Design and Reliability

The FHBC is comprised of a main hook rated for 125 tons as well as an auxiliary hook rated for 25 tons. A temporary hoist with an appropriate capacity will be attached to the temporary gantry crane hook to prevent submergence of the hook.

The following table determines the maximum lift weight during the installation of the new racks.

Item	Weight (lbs)
Rack	25,280 (max.)
Lift Rig	1,100
Rigging	500
Total Lift	26,880

The following table determines the maximum lift weight during the removal of the existing racks.

Item	Weight (lbs)
Heaviest Rack	36,300
Lift Rig	2,400
Rigging	500
Total Lift	39,200

It is clear, based on the heaviest rack weight to be lifted, that the heaviest load being lifted is well below the rating of the FHBC hook and the rating of the temporary gantry crane. The temporary gantry crane will be designed and constructed in accordance with CMAA 70 [3.5.3]. The hoist to be used in conjunction with the temporary gantry crane will be selected to provide an adequate load capacity and comply with NUREG-0612.

Remotely engaging lift rigs, meeting all requirements of NUREG-0612, will be used to lift the new rack modules. A similarly designed lift rig will be used to remove the existing racks. The new and existing rack lift rigs consist of four independently loaded traction rods in a lift configuration. The individual lift rods have a safety factor of greater than 10. If one of the rods break, the load will still be supported by at least two rods, which will have a safety factor of more than 5 against ultimate strength. Therefore, the lift rigs comply with the duality feature called for in Section 5.1.6 (3) of NUREG 0612.

The lift rigs have the following attributes:

- The traction rod is designed to prevent loss of its engagement with the rig in the locked position. Moreover, the locked configuration can be directly verified from above the pool water without the aid of an underwater camera.

- The stress analysis of the rig is carried out and the primary stress limits postulated in ANSI N14.6 [3.5.4] are met.

- The rig is load tested with 300% of the maximum weight to be lifted. The test weight is maintained in the air for 10 minutes. All critical weld joints are liquid penetrant examined to establish the soundness of all critical joints.

e. Crane Maintenance

The FHBC and the temporary gantry crane are maintained functional per the VCSNS preventative maintenance procedures.

The proposed heavy loads compliance will be in accordance with the guidelines of NUREG-0612, which calls for measures to "provide an adequate defense-in-depth for handling of heavy loads near spent fuel...". The NUREG-0612 guidelines cite four major causes of load handling accidents, namely

i. operator errors
ii. rigging failure
iii. lack of adequate inspection
iv. inadequate procedures

The re-racking program ensures maximum emphasis on mitigating the potential load drop accidents by implementing measures to eliminate shortcomings in all aspects of the operation including the four aforementioned areas. A summary of the measures specifically planned to deal with the major causes is provided below.

Operator errors: As mentioned above, comprehensive training will be provided to the installation crew. All training shall be in compliance with ANSI B30.2.

Rigging failure: The lifting device designed for handling and installation of the new racks has redundancies in the lift legs and lift eyes such that there are four independent load members in the new rack lift rig, and three independent load members in the existing rack lifting rig. Failure of any one load bearing member would not lead to uncontrolled lowering of the load. The rig complies with all provisions of ANSI 14.6-1993, including compliance with the primary stress criteria, load testing at 300% of maximum lift load, and dye examination of critical welds.

The rig designs are similar to the rigs used in the initial racking or the rerack of numerous other plants, such as Hope Creek, Millstone Unit 1, Indian Point Unit Two, Ulchin II, Laguna Verde, J.A. FitzPatrick, and Three Mile Island Unit 1.

Lack of adequate inspection: The designer of the racks has developed a set of inspection points that have been proven to eliminate any incidence of rework or erroneous installation in numerous prior rerack projects. Surveys and measurements are performed on the storage racks prior to and subsequent to placement into the pools to ensure that the as-built dimensions and installed locations are acceptable. Measurements of the pool and floor elevations are also performed to determine actual pool configuration and to allow height adjustments of the pedestals prior to rack installation. These inspections minimize rack manipulation during placement into the pool.

Inadequate procedures: Procedures will be developed to address operations pertaining to the rack installation effort, including, but not limited to, mobilization, rack handling, upending, lifting, installation, verticality, alignment, dummy gage testing, site safety, and ALARA compliance. The procedures will be the successors of the procedures successfully implemented in previous projects.

Table 3.5.1 provides a synopsis of the requirements delineated in NUREG-0612, and its intended compliance.

3.6 References

[3.3.1] "Nuclear Engineering International," July 1997 issue, pp 20-23.

[3.3.2] "Spent Fuel Storage Module Corrosion Report," Brooks & Perkins Report 554, June 1, 1977.

[3.3.3] "Suitability of Brooks & Perkins Spent Fuel Storage Module for Use in PWR Storage Pools," Brooks & Perkins Report 578, July 7, 1978.

[3.3.4] "Boral Neutron Absorbing/Shielding Material - Product Performance Report," Brooks & Perkins Report 624, July 20, 1982.

[3.5.1] ANSI/ASME B30.2, "Overhead and Gantry Cranes, (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist)," American Society of Mechanical Engineers, 1976.

[3.5.2] ANSI B30.9, "Safety Standards for Slings," 1971.

[3.5.3] CMMA Specification 70, "Electrical Overhead Traveling Cranes," Crane Manufacturers Association of America, Inc., 2000.

[3.5.4] ANSI N14.6-1993, Standard for Special Lifting Devices for Shipping Containers Weighing 10000 Pounds or more for Nuclear Materials," American National Standard Institute, Inc., 1978.

Table 3.3.1

BORAL EXPERIENCE LIST - PWRs

Plant	Utility	Docket No.	Mfg. Year
Maine Yankee	Maine Yankee Atomic Power	50-309	1977
Donald C. Cook	Indiana & Michigan Electric	50-315/316	1979
Sequoyah 1,2	Tennessee Valley Authority	50-327/328	1979
Salem 1,2	Public Service Electric & Gas	50-272/311	1980
Zion 1,2	Commonwealth Edison	50-295/304L	1980
Bellefonte 1, 2	Tennessee Valley Authority	50-438/439	1981
Yankee Rowe	Yankee Atomic Power	50-29	1964/1983
Gosgen	Kernkraftwerk Gosgen-Daniken AG (Switzerland)		1984
Koeberg 1,2	ESCOM (South Africa)		1985
Beznau 1,2	Nordostschweizerische Kraftwerke AG (Switzerland)		1985
12 various Plants	Electricite de France (France)	--	1986
Indian Point 3	NY Power Authority	50-286	1987
Byron 1,2	Commonwealth Edison	50-454/455	1988
Braidwood 1,2	Commonwealth Edison	50-456/457	1988
Yankee Rowe	Yankee Atomic Power	50-29	1988
Three Mile Island I	GPU Nuclear	50-289	1990
Sequoyah (rerack)	Tennessee Valley Authority	50-327	1992
Donald C. Cook (rerack)	American Electric Power	50-315/316	1992
Beaver Valley Unit 1	Duquesne Light Company	50-334	1993
Fort Calhoun	Omaha Public Power District	50-285	1993

Table 3.3.1

BORAL EXPERIENCE LIST - PWRs

Plant	Utility	Docket No.	Mfg. Year
Zion 1 & 2 (rerack)	Commonwealth Edison	50-295/304L	1993
Salem Units 1 & 2 (rerack)	Public Gas and Electric Company	50-272/311	1995
Ulchin Unit 1	Korea Electric Power Company (Korea)	--	1995
Haddam Neck	Connecticut Yankee Atomic Power Company	50-213	1996
Ulchin Unit 2	Korea Electric Power Company (Korea)	--	1996
Kori-4	Korea Electric Power Company (Korea)	--	1996
Yonggwang 1,2	Korea Electric Power Company (Korea)	--	1996
Sizewell B	Nuclear Electric, plc (United Kingdom)	--	1997
Angra 1	Furnas Centrais-Electricas SA (Brazil)	--	1997
Waterford 3	Entergy Operations	50-382	1997
Callaway	Union Electric	50-483	1998
Millstone 3	Union Electric	50-423	1998
Davis-Besse	First Energy	50-346	1999
Wolf Creek	Wolf Creek Nuclear Operating	50-482	1999
Harris Pool 'C'	Carolina Power & Light	50-401	1999
Yonggwang 5/6	Korea Electric Power Company (Korea	--	2001
Kewaunee	Wisconsin Public Service	50-305	2001

Table 3.3.2

BORAL EXPERIENCE LIST - BWRs

Plant	Utility	Docket No.	Mfg. Year
Cooper	Nebraska Public Power	50-298	1979
J.A. FitzPatrick	NY Power Authority	50-333	1978
Duane Arnold	Iowa Electric Light & Power	50-331	1979
Browns Ferry 1,2,3	Tennessee Valley Authority	50-259/260/296	1980
Brunswick 1,2	Carolina Power & Light	50-324/325	1981
Clinton	Illinois Power	50-461/462	1981
Dresden 2,3	Commonwealth Edison	50-237/249	1981
E.I. Hatch 1,2	Georgia Power	50-321/366	1981
Hope Creek	Public Service Electric & Gas	50-354/355	1985
Humboldt Bay	Pacific Gas & Electric Company	50-133	1985
LaCrosse	Dairyland Power	50-409	1976
Limerick 1,2	Philadelphia Electric Company	50-352/353	1980
Monticello	Northern States Power	50-263	1978
Peachbottom 2,3	Philadelphia Electric	50-277/278	1980
Perry 1,2	Cleveland Electric Illuminating	50-440/441	1979
Pilgrim	Boston Edison Company	50-293	1978
Susquehanna 1,2	Pennsylvania Power & Light	50-387,388	1979
Vermont Yankee	Vermont Yankee Atomic Power	50-271	1978/1986
Hope Creek	Public Service Electric & Gas	50-354/355	1989
Harris Pool 'B' †	Carolina Power & Light	50-401	1991
Duane Arnold	Iowa Electric Light & Power	50-331	1993
Pilgrim	Boston Edison Company	50-293	1993

Table 3.3.2 BORAL EXPERIENCE LIST - BWRs			
Plant	Utility	Docket No.	Mfg. Year
LaSalle 1	Commonwealth Edison	50-373	1992
Millstone Unit 1	Northeast Utilities	50-245	1989
James A. FitzPatrick	NY Power Authority	50-333	1990
Hope Creek	Public Service Electric & Gas Company	50-354	1991
Duane Arnold Energy Center	Iowa Electric Power Company	50-331	1994
Limerick Units 1,2	PECO Energy	50-352/50-353	1994
Harris Pool 'B' †	Carolina Power & Light Company	50-401	1996
Chinshan 1,2	Taiwan Power Company (Taiwan)	--	1986
Kuosheng 1,2	Taiwan Power Company (Taiwan)	--	1991
Laguna Verde 1,2	Comision Federal de Electricidad (Mexico)	--	1991
Harris Pool 'B' †	Carolina Power & Light Company	50-401	1996
James A. FitzPatrick	NY Power Authority	50-333	1998
Vermont Yankee	Vermont Yankee	50-271	1998
Plant Hatch	Southern Nuclear	50-321	1999
Harris Pool 'C' †	Carolina Power & Light Company	50-401	1999
Byron/Braidwood	Carolina Power & Light Company	50-401	1999
Enrico Fermi Unit 2	Detroit Edison	50-305	2000

† Fabricated racks for storage of spent fuel transhipped from Brunswick.

Table 3.3.3 1100 ALLOY ALUMINUM PHYSICAL CHARACTERISTICS	
Density	0.098 lb/in³ 2.713 g/cm³
Melting Range	1190ºF - 1215ºF 643º - 657ºC
Thermal Conductivity (77ºF)	128 BTU/hr/ft²/F/ft 0.53 cal/sec/cm²/ºC/cm
Coefficient of Thermal Expansion (68ºF - 212ºF)	13.1×10^{-6} in/in-ºF 23.6×10^{-6} cm/cm-ºC
Specific Heat (221ºF)	0.22 BTU/lb/ºF 0.23 cal/g/ºC
Modulus of Elasticity	10×10^{6} psi
Tensile Strength (75ºF)	13,000 psi (annealed) 18,000 psi (as rolled)
Yield Strength (75ºF)	5,000 psi (annealed) 17,000 psi (as rolled)
Elongation (75ºF)	35-45% (annealed) 9-20% (as rolled)
Hardness (Brinell)	23 (annealed) 32 (as rolled)
Annealing Temperature	650ºF 343ºC

Table 3.3.4 CHEMICAL COMPOSITION - ALUMINUM (1100 ALLOY)	
99.00% min.	Aluminum
1.00% max.	Silicone and Iron
0.05-0.20% max.	Copper
0.05% max.	Manganese
0.10% max.	Zinc
0.15% max.	Other

Table 3.3.5 CHEMICAL COMPOSITION AND PHYSICAL PROPERTIES OF BORON CARBIDE	
CHEMICAL COMPOSITION (WEIGHT PERCENT)	
Total boron	70.0 min.
B^{10} isotopic content in natural boron	18.0
Boric oxide	3.0 max.
Iron	2.0 max.
Total boron plus total carbon	94.0 min.
PHYSICAL PROPERTIES	
Chemical formula	B_4C
Boron content (weight percent)	78.28%
Carbon content (weight percent)	21.72%
Crystal structure	rhombohedral
Density	0.0907 lb/in^3 2.51 g/cm^3
Melting Point	4442ºF 2450ºC
Boiling Point	6332ºF 3500ºC
Boral Loading (minimum grams B^{10} per cm^2)	0.030

Table 3.5.1

HEAVY LOAD HANDLING COMPLIANCE MATRIX (NUREG-0612)

Criterion		Compliance
1.	Are safe load paths defined for the movement of heavy loads to minimize the potential of impact, if dropped, on irradiated fuel?	Yes
2.	Will procedures be developed to cover: identification of required equipment, inspection and acceptance criteria required before movement of load, steps and proper sequence for handling the load, defining the safe load paths, and special precautions?	Yes
3.	Will crane operators be trained and qualified?	Yes
4.	Will special lifting devices meet the guidelines of ANSI 14.6-1993?	Yes
5.	Will non-custom lifting devices be installed and used in accordance with ANSI B30.20 [3.5.5], latest edition?	Yes
6.	Will the cranes be inspected and tested prior to use in rack installation?	Yes
7.	Does the crane meet the requirements of ANSI B30.2-1976 and CMMA-70?	Yes

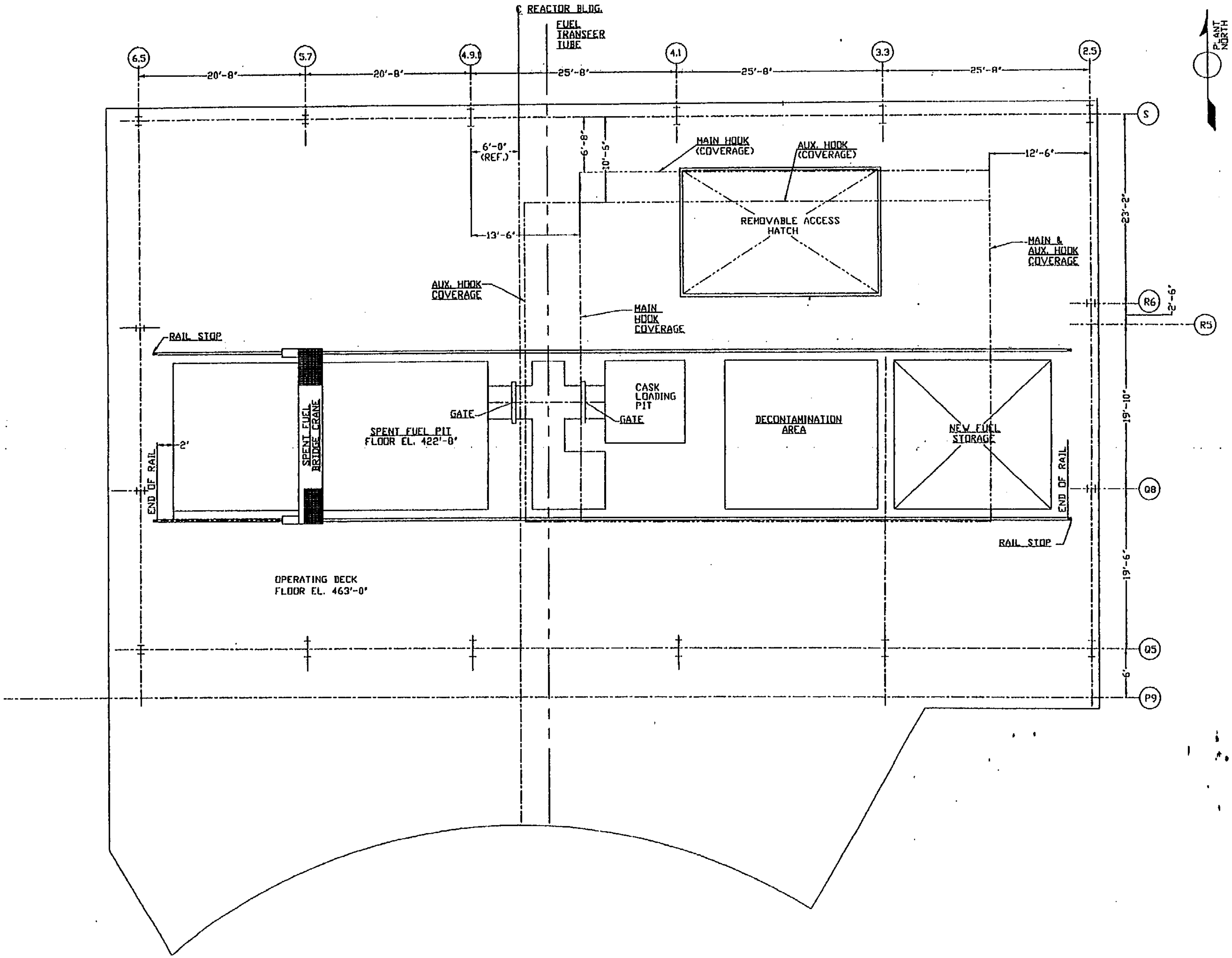


FIGURE 3.5.1 SPENT FUEL POOL OPERATING DECK LAYOUT PLAN