ENCLOSURE 1

Proposed Technical Specification Change

Watts Bar Nuclear Plant Unit 1

Docket No. 50-390

(WBN-TS 96-010)

Spent Fuel Pool Modifications
for Increased Storage Capacity

WBN-TS-96-010
CHANGED PAGES

TECHNICAL SPECIFICATIONS

Table of Contents Pages iii
v

Pages 4.0-2
4.0-3
4.0-4
4.0-5 (paginated)
4.0-6 (paginated)
4.0-7 (Figure 4.3-1)
3.7-31
3.7-32 (Figure 3.7.15-1)
3.9-17

TECHNICAL SPECIFICATION BASES

Table of Contents Page iii

B 3.7-75
B 3.7-76
B 3.7-77
B 3.9-33
B 3.9-35
B.3.9-36

TECHNICAL REQUIREMENTS MANUAL

Table of Contents Page vi

3.9-5
3.9-6
3.9-7 (Figure 3.9-1)

TECHNICAL REQUIREMENTS BASES

B 3.9-8
B 3.9-9
B 3.9-10

(continued)

LIST OF TABLES

TABLE OF CONTENTS (continued)

LIST OF FIGURES

LIST OF MISCELLANEOUS REPORTS AND PROGRAMS

4.0 DESIGN FEATURES

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks (shown in Figure 4.3-1) are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which, includes an allowance for uncertainties as described in Sections 4.3.2.7 and 9.1 of the FSAR;

c. Distances between fuel assemblies as follows:

1. A nominal 10.375 inch center-to-center spacing in the twenty-four flux trap rack modules (Region 1).

2. A nominal 8.972 inch center-to-center spacing in the ten burnup credit rack modules peripherally located adjacent to the south and west pool walls (Region 2); and

3. A nominal 8.972 inch center-to-center spacing in the single 15 x 15 burnup credit rack module in the fuel cask loading area of the cask pit (Region 2).

d. Spent fuel assemblies with a burnup in the "acceptable burnup domain" of Figure 3.7.15-1 may be allowed unrestricted storage in either type of fuel storage rack.

e. New or partially spent fuel assemblies with a burnup in the "unacceptable burnup domain" of Figure 3.7.15-1 will be stored in compliance with the following configuration:

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

1. In the flux trap rack modules (Region 1), fuel assemblies with enrichments less than or equal to 3.80 weight percent U-235 are allowed unrestricted storage. Fuel assemblies with enrichment greater than 3.80 weight percent U-235 and burnup less than 6.750 megawattday/kilogram uranium (MWD/KgU) shall be placed in storage cells that face adjacent cells in the flux trap modules containing either water or fuel assemblies with accumulated burnup of at least 20 MWD/KgU.

2. Storage in any burnup credit rack modules (Region 2) located in the pool as well as in the fuel cask loading area is restricted to fuel of 4.95 ± 0.05 weight percent initial enrichment burned to at least 41 MWD/KgU (assembly average), or fuel of other enrichments with a burnup yielding an equivalent reactivity in the fuel racks. The minimum required assembly average burnup is given by Y(MWD/KgU) where $Y = 0.0666E^3 - 1.3933E^2 + 18.7600E - 25.7425$, where E is the initial enrichment in the axial zone of highest enrichment. Figure 3.7.15-1 illustrates the burnup enrichment equation in graphical form.

3. New fuel with enrichment up to 4.95 ± 0.05 weight percent U-235 may be placed in the burnup credit rack (Region 2) in the cask pit rack location with face adjacent storage cells containing water.

A water cell is less reactive than any cell containing fuel and therefore a water cell may be used at any location in the loading arrangements.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

a. Fuel assemblies having a maximum U-235 enrichment of 4.3 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR; and

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pool and cask pit area are designed and shall be maintained to prevent inadvertent draining of the pool below Elevation 747 feet - 1 1/2 inches.

4.3.3 Capacity

The total spent fuel storage capacity is 1835 fuel assemblies.

4.3.3.1 The primary portion of the spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1386 fuel assemblies in 24 flux trap rack modules.

4.3.3.2 No more than 224 fuel assemblies will be stored in ten smaller burnup credit rack modules peripherally located adjacent to the south and west walls of the pool.

4.3.3.3 In addition, no more than 225 fuel assemblies will be stored in a single 15 x 15 burnup credit rack module in the cask loading area of the cask pit.

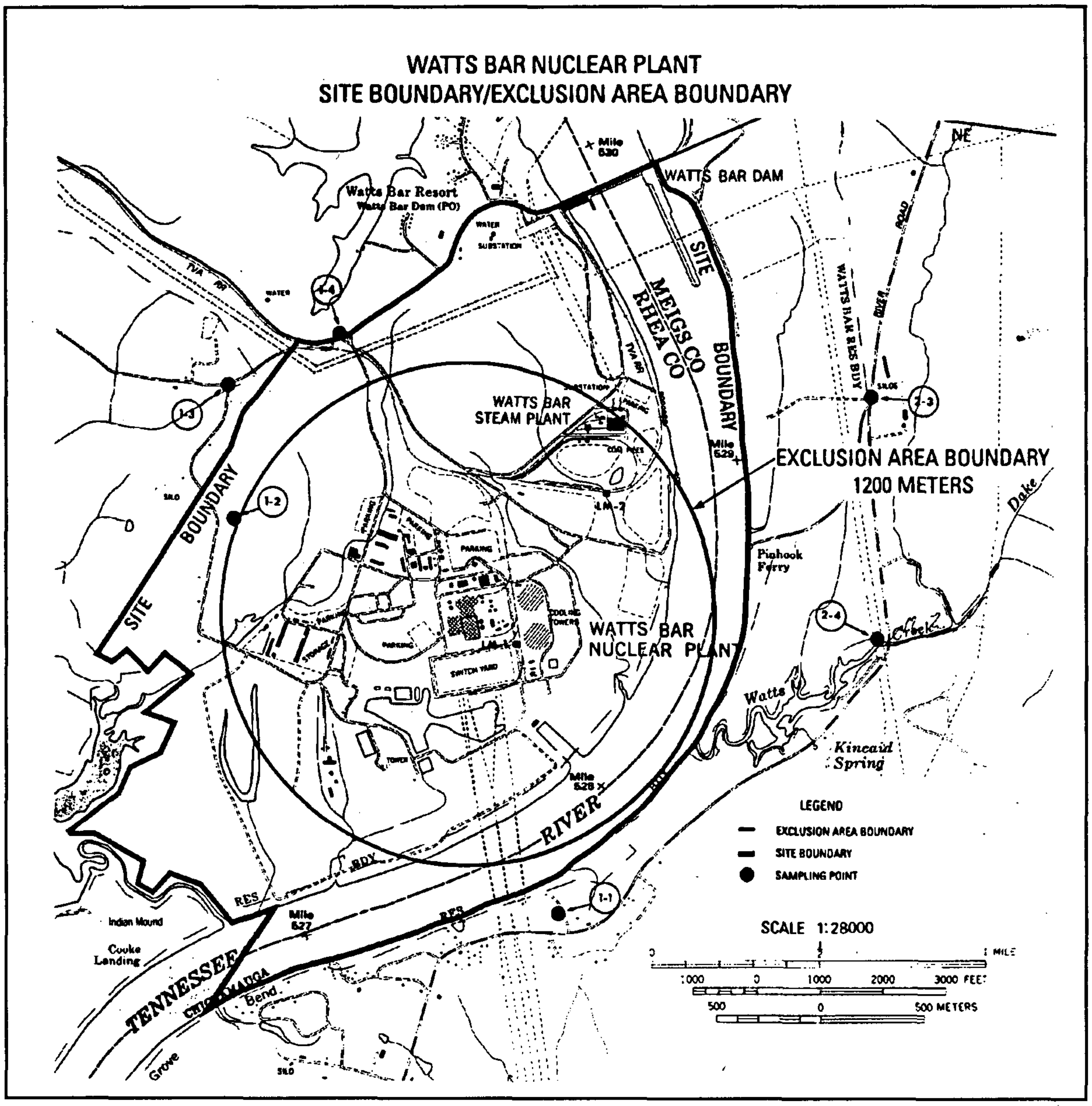


Figure 4.1-1 (page 1 of 1)
Site and Exclusion Area Boundaries

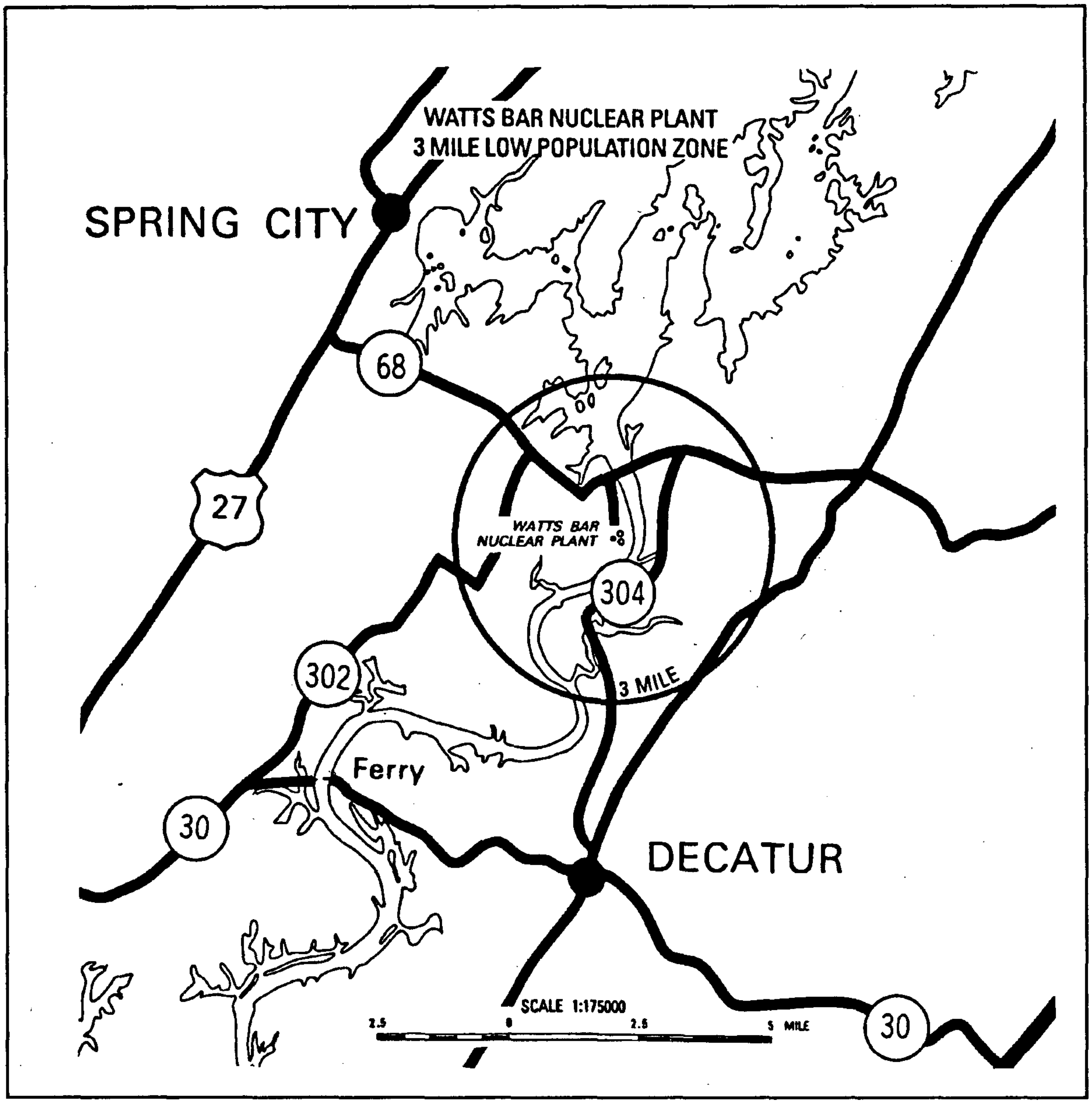
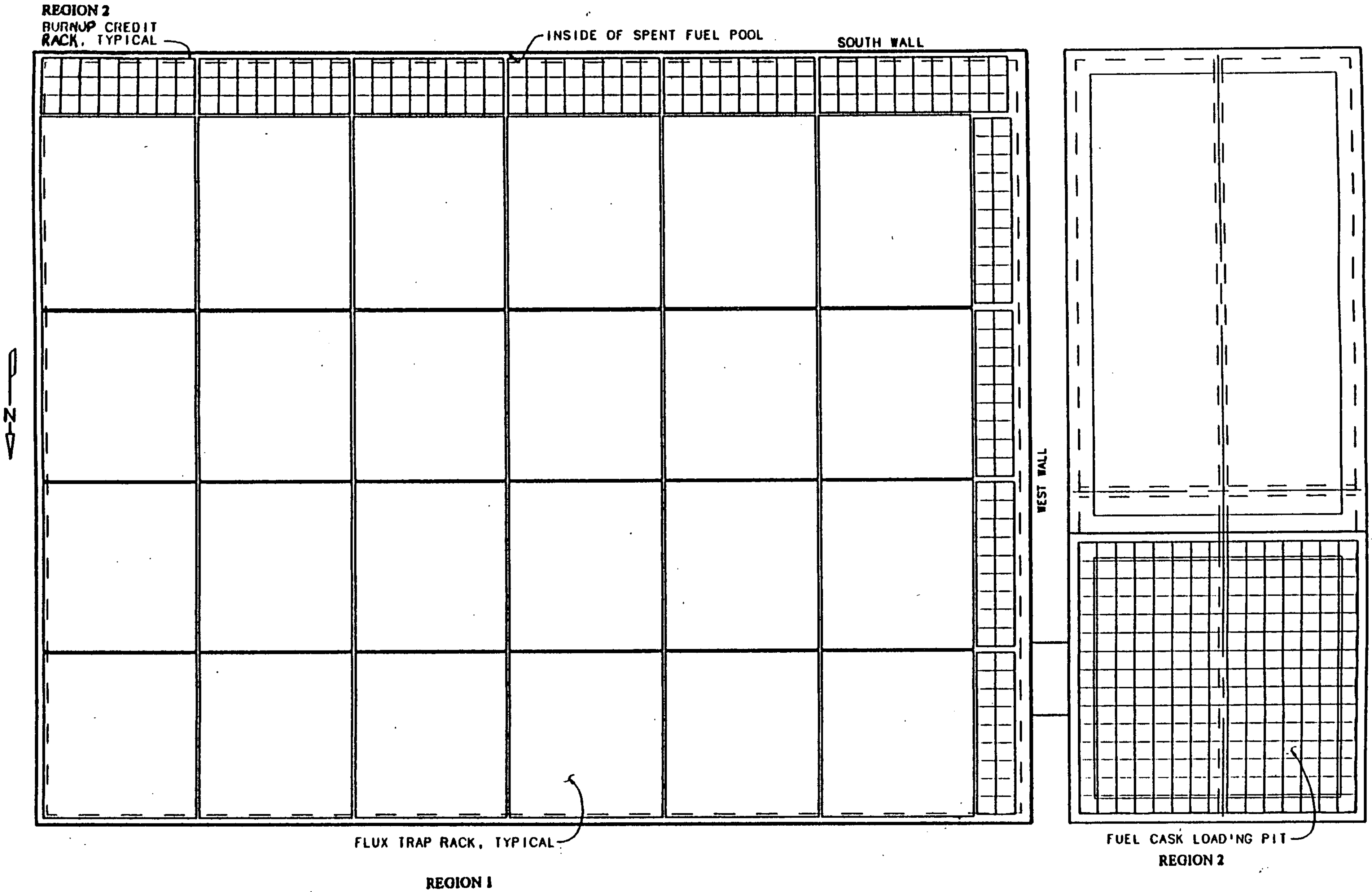


Figure 4.1-2 (page 1 of 1)
Low Population Zone



PLAN
SPENT FUEL POOL

FIGURE 4.3-1
SPENT FUEL STORAGE RACKS

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Assembly Storage

LCO 3.7.15 The combination of initial enrichment and burnup of each spent fuel assembly stored in Region 1 or Region 2 shall be within the Acceptable Burnup Domain of Figure 3.7.15-1 or in accordance with Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel storage pool or cask loading area.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 --------NOTE-------- LCO 3.0.3 is not applicable. -------------------- Initiate action to move the noncomplying fuel assembly.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.15.1 Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.15-1 or Specification 4.3.1.1.	Prior to storing the fuel assembly.

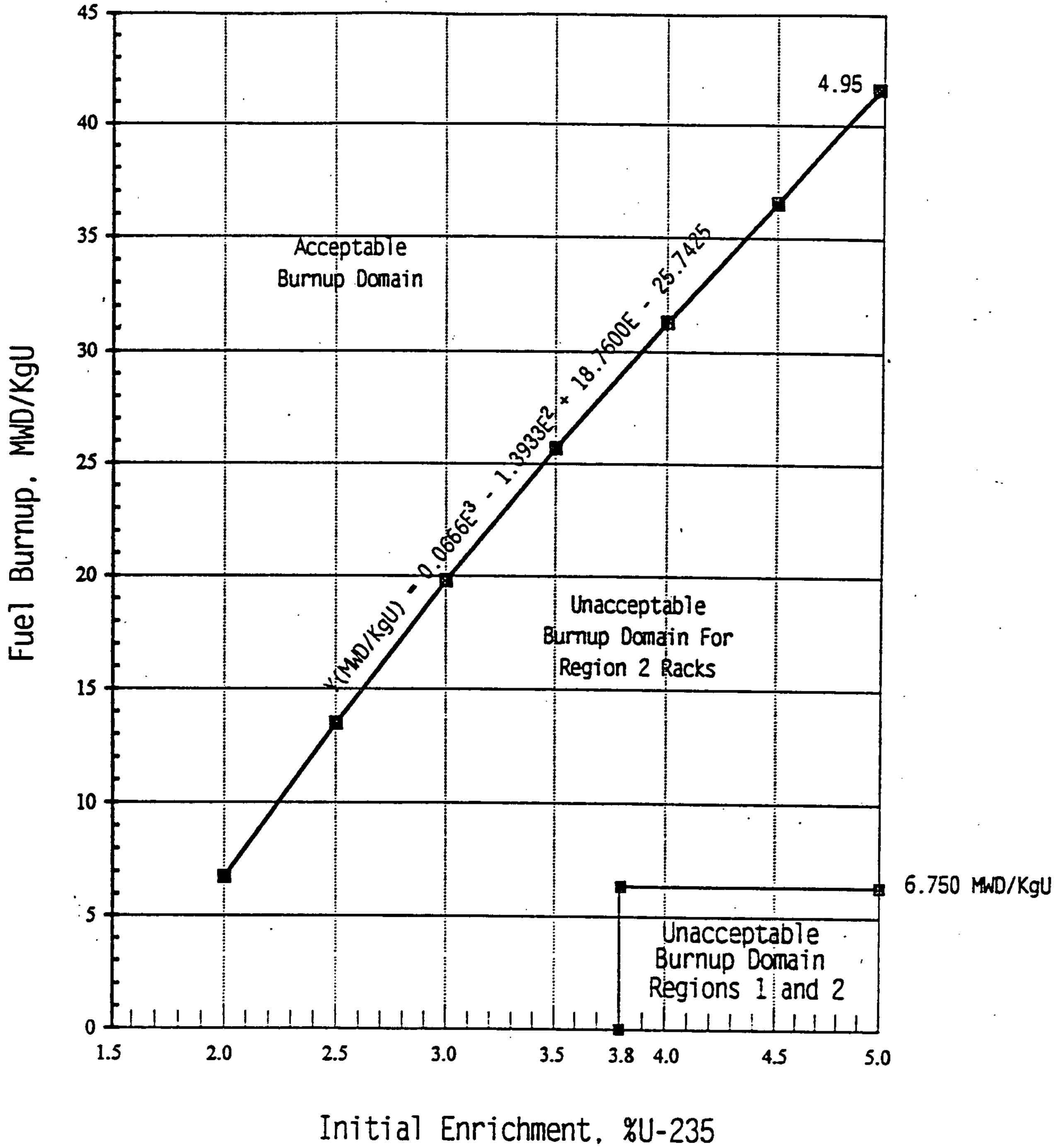


Figure 3.7.15-1 Acceptable Burnup Domain - Watts Bar Spent Fuel Storage Racks

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Assembly Storage

BASES

BACKGROUND

In the spent fuel storage design (References 1 and 2), the spent fuel pool area is divided into two separate and distinct regions for criticality considerations. Region 1, flux trap modules with 1386 storage positions, is designed to accommodate fuel with enrichment as high as 3.8 weight percent U-235 without restrictions. Storage of fuel assemblies with enrichment between 3.8 and 5.0 weight percent in Region 1 requires either fuel burnup of ≥6.750 MWD/KgU in accordance with Figure 3.7.15-1, or placement in storage locations which have face adjacent storage cells containing either water or fuel assemblies with accumulated burnup of at least 20.0 MWD/KgU in accordance with paragraph 4.3.1.1.

Region 2 burnup credit rack modules, with 449 storage positions, is designed to accommodate fuel with 4.95 ± 0.05 weight percent initial enrichment burned to at least 41 MWD/KgU or, in accordance with Figure 3.7.15-1, fuel of lower enrichment which yields an equivalent reactivity. In addition, the Region 2 rack (15 x 15 cell array) in the cask loading area of the cask pit is designed to store up to 4.95 ± 0.05 weight percent initial enrichment new fuel when placed in storage locations which have empty face-adjacent storage cells.

The water in the spent fuel storage pool and cask loading area normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975, and the April 1978 NRC letter (Reference 3) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, an abnormal scenario

(continued)

BASES

BACKGROUND (continued)	could be associated with the improper movement of a relatively high enrichment, low exposure fuel assembly from Region 1 to Region 2, or the misloading of a fuel assembly in either region. This could potentially increase the criticality of the storage regions. To mitigate these postulated criticality-related events, boron is dissolved in the pool water. Safe operation of the spent fuel storage design with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with the accompanying LCO. Prior to movement of an assembly in the pool or cask loading area, it is necessary to perform SR 3.9.9.1 and SR 3.9.10.1, respectively.
APPLICABLE SAFETY ANALYSES	The hypothetical events can only take place during or as a result of the movement of an assembly. For these occurrences, the presence of soluble boron in the spent fuel storage pool and cask loading area, (controlled by LCO 3.9.9, "Spent Fuel Pool Boron Concentration," and LCO 3.9.10, "Cask Pit Boron Concentration,") prevents criticality in both storage rack regions. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential occurrences may be limited to a small fraction of the total operating time. During the remaining time period with no potential for such events, the operation may be under the auspices of the accompanying LCO. The configuration of fuel assemblies in the fuel storage pool and cask loading area satisfies Criterion 2 of the NRC Policy Statement.
LCO	The restrictions on the placement of fuel assemblies within the spent fuel pool and cask loading area in accordance with Figure 3.7.15-1, in the accompanying LCO, ensures the k_{eff} will always remain ≤ 0.95, assuming the pool and cask loading area to be flooded with unborated water. Fuel assemblies not meeting the criteria of Figure 3.7.15-1 shall be stored in accordance with Specification 4.3.1.1 in Section 4.3.
APPLICABILITY	This LCO applies whenever any fuel assembly is stored in the spent fuel storage pool or cask loading area.

(continued)

BASES (continued)

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

If unable to move irradiated fuel assemblies while in Mode 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in Mode 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

When the configuration of fuel assemblies stored in the spent fuel storage pool or the cask loading area is not in accordance with Figure 3.7.15-1, or paragraph 4.3.1.1, the immediate action is to initiate action to make the necessary fuel assembly movements to bring the configuration into compliance with Figure 3.7.15-1 or Specification 4.3.1.1.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.15-1 in the accompanying LCO. For fuel assemblies in the unacceptable range of Figure 3.7.15-1, performance of this SR will ensure compliance with Specification 4.3.1.1.

REFERENCES

1. Watts Bar FSAR, Sections 4.3.2.7 and 9.1.2.

2. Spent Fuel Pool Modification for Increased Storage Capacity, (Chapter 4), Watts Bar Unit 1, submitted by TVA letter dated October 23, 1996.

3. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978, NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

B 3.9 REFUELING OPERATIONS

B 3.9.9 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND	The spent fuel storage rack criticality analysis assumes 2000 ppm soluble boron in the fuel pool during a dropped/misplaced fuel assembly event.
APPLICABLE SAFETY ANALYSES	This requirement ensures the presence of at least 2000 ppm soluble boron in the spent fuel pool water as assumed in the spent fuel rack criticality analysis for dropped/misplaced fuel assembly event. The RCS boron concentration satisfies Criterion 2 of the NRC Policy Statement.
LCO	The LCO requires that the boron concentration in the spent fuel pool be greater than or equal to 2000 ppm during fuel movement.
APPLICABILITY	This LCO is applicable when the spent fuel pool is flooded and fuel is being moved. Once fuel movement begins, the movement is considered in progress until the configuration of the assemblies in the storage racks is verified to comply with the criticality loading criteria specified in Figure 3.7.15-1 and Specification 4.3.1.1.
ACTIONS	<u>A.1</u> If the spent fuel pool boron concentration does not meet the above requirements, fuel handling in the spent fuel pool must be suspended immediately. This action precludes a fuel handling accident, when conditions are outside those assumed in the accident analysis. Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

(continued)

3.9 REFUELING OPERATIONS

3.9.10 Cask Pit Boron Concentration

LCO 3.9.10 Boron concentration of the cask pit shall be ≥ 2000 ppm.

APPLICABILITY: During fuel movement in the flooded cask pit.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend fuel movement.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.10.1 Verify boron concentration in the cask pit is ≥ 2000 ppm.	Prior to movement of fuel in the cask pit <u>AND</u> 72 hours thereafter

B 3.9 REFUELING OPERATIONS

B 3.9.10 Cask Pit Boron Concentration

BASES

BACKGROUND	The spent fuel storage rack criticality analysis assumes 2000 ppm soluble boron in the cask pit during a dropped/misplaced fuel assembly event.
APPLICABLE SAFETY ANALYSES	This requirement ensures the presence of at least 2000 ppm soluble boron in the cask pit water as assumed in the spent fuel rack criticality analysis for dropped/misplaced fuel assembly event. The RCS boron concentration satisfies Criterion 2 of the NRC Policy Statement.
LCO	The LCO requires that the boron concentration in the cask pit be greater than or equal to 2000 ppm during fuel movement.
APPLICABILITY	This LCO is applicable when the cask pit is flooded and fuel is being moved in the cask pit. Once fuel movement begins, the movement is considered in progress until the configuration of the assemblies in the storage racks is verified to comply with the criticality loading criteria specified in Figure 3.7.15-1 and Specification 4.3.1.1.
ACTIONS	<u>A.1</u> If the cask pit boron concentration does not meet the above requirements, fuel handling in the cask pit must be suspended immediately. This action precludes a fuel handling accident, when conditions are outside those assumed in the accident analysis. Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS	<u>SR 3.9.10.1</u> This SR requires that the cask pit boron concentration be verified greater than or equal to 2000 ppm. This surveillance is to be performed prior to movement of fuel in the cask pit and at least once every 72 hours thereafter during the movement of fuel in the cask pit. The Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of the sample. The Frequency is based on operating experience, which has shown 72 hours to be adequate.
REFERENCES	1. Watts Bar FSAR, Chapter 15, "Accident Analysis."

TR 3.9 REFUELING OPERATIONS

TR 3.9.4 Auxiliary Building Crane Travel - Auxiliary Building

TR 3.9.4 Loads traveling over fuel assemblies in the spent fuel storage pool area shall be restricted as follows:

a. Loads > 2059 pounds shall be prohibited from travel over fuel assemblies in the spent fuel pool.

---------------------------NOTES----------------------

1. The spent fuel pool transfer canal gate and the spent fuel pool cask pit gate may travel over fuel assemblies in the spent fuel pool.

2. The crane interlocks and physical stops may be defeated for activities associated with the WBN Rerack Project including installation of the burnup credit racks.

----

b. Cask loading area of the cask pit:

1. Loads which meet the weight, cross-sectional impact area, and allowable travel height criteria of Figure 3.9-1 may be carried over fuel assemblies stored in the cask loading area of the cask pit if the impact shield is in place over the cask loading area.

2. Loads which do not meet the weight, cross-sectional impact area, and allowable travel height criteria of Figure 3.9-1 shall be prohibited from travel over the cask loading area of the cask pit when fuel is stored in this area.

APPLICABILITY: With fuel assemblies in the spent fuel pool or in the cask loading area of the cask pit.

-------------------------NOTE----------------------------

TR 3.0.3 is not applicable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Technical Requirement not met.	A.1 Place the crane load in a safe condition.	Immediately

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.9.4.1	Demonstrate crane interlocks and physical stops which prevent crane travel over fuel assemblies to be OPERABLE. -------------------NOTE---------------- The crane interlocks and physical stops may be defeated for activities associated with the WBN Rerack Project, including installation of the burnup credit racks. ---------------------------------------	Within 7 days prior to crane use. AND At least once per 7 days thereafter during crane operation.
TSR 3.9.4.2	Verify administrative requirements concerning the impact shield are met.	Each time before an allowable load exceeding 2059 lbs is moved across fuel stored in the cask pit area.

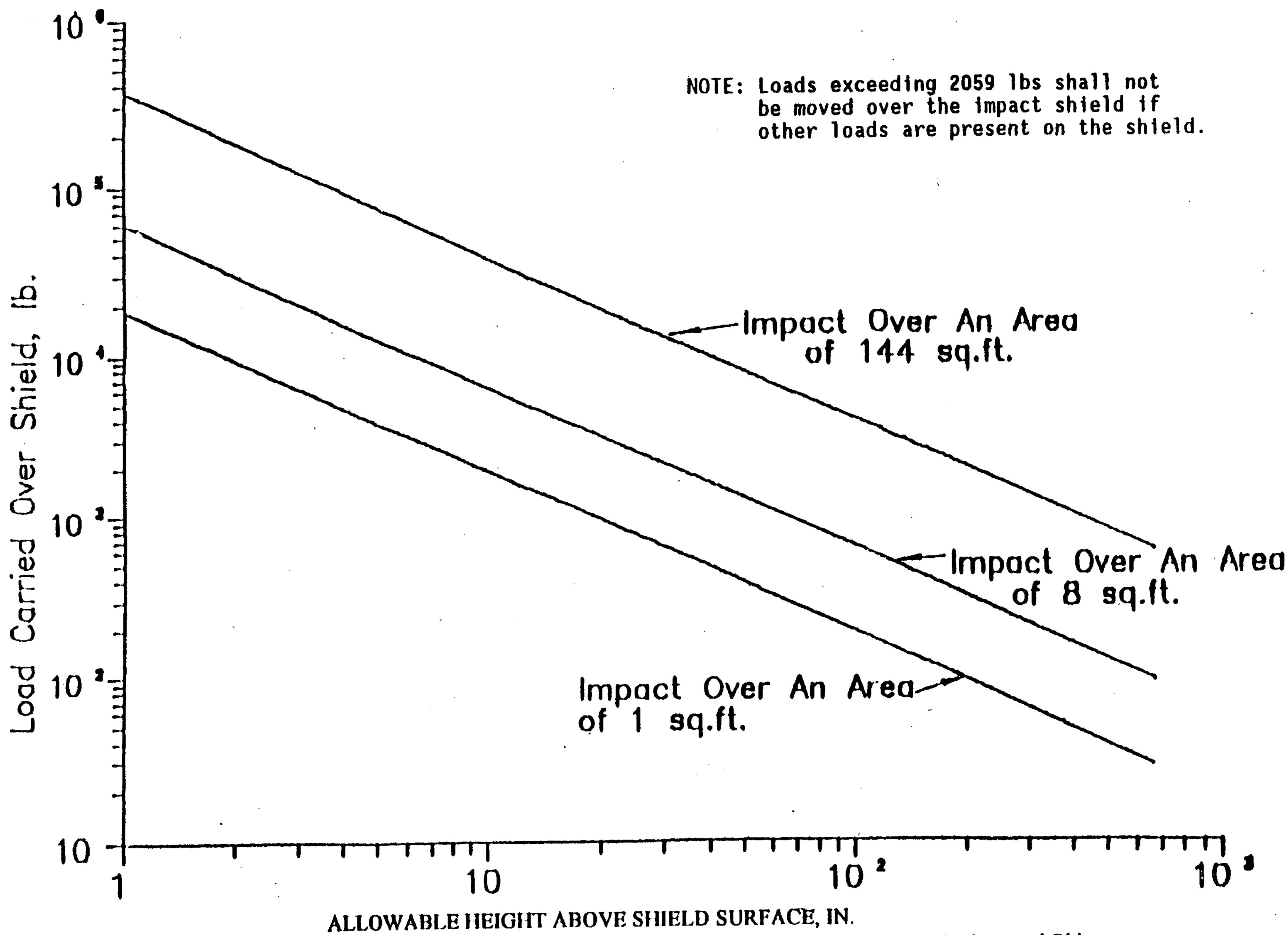


FIGURE 3.9-1 Relationship Between Load, Allowable Height and Impact Area for Dropped Object

B 3.9 REFUELING OPERATIONS

B 3.9.4 Auxiliary Building Crane Travel - Auxiliary Building

BASES

BACKGROUND

The spent fuel pool and cask pit area are reinforced concrete structures with a stainless steel liner for leak tightness. The spent fuel storage racks consist of stainless steel structures with receptacles for nuclear fuel assemblies as they are used in a reactor. Design of these storage racks is in accordance with Reference 1.

The racks can withstand the drop of a fuel assembly from its maximum supported height, drop of the transfer canal gate or cask pit gate from a height of eight feet above the top of the racks, and the drop of tools used in the pool area. Crane travel over the spent fuel storage pool is limited through electrical and mechanical stops which prevent the movement of heavy objects, including shipping casks, over the spent fuel pool. The movement of casks is restricted to the cask loading area (when no fuel is stored in this area) and areas away from the pool (Reference 2).

APPLICABLE SAFETY ANALYSES

The release of radioactive material from fuel may occur during the refueling process, and at other times, as a result of fuel-cladding failures or mechanical damage caused by the dropping of fuel elements or the dropping of objects onto fuel elements (Reference 1). The restriction on the movement of loads in excess of the nominal weight of a fuel and control rod assembly and the associated handling tools over other fuel assemblies in the storage pool areas, and the allowance of gate movement over fuel assemblies in the storage pool, ensures that, in the event these loads are dropped, the activity release will be limited to that contained in a single fuel assembly, and that any possible distortion of fuel in the storage racks will not result in a critical array. Fuel assembly drops are design basis type accidents that have not been significant to risk when analyzed in environmental reports (Reference 3).

(continued)

BASES (continued)

TR

TR 3.9.4 requires that loads greater than 2059 pounds, other than the transfer canal and cask pit gates (3820 pounds) (Note 1), shall be prohibited from travel over fuel assemblies in the spent fuel pool. This ensures that objects traversing the pool are within the design basis and will not cause an unsafe condition if accidentally dropped. The evaluation of dropped gates or loads associated with rack installation under the WBN Rerack Project is provided in Reference 4. Assurance against load drops over fuel stored in the cask loading area of the cask pit is also evaluated in Reference 4 and is obtained by requiring conformance to calculated load criteria which will prevent penetration of the impact shield in the event of a load drop. NOTE 2 has been added which allows defeating the crane interlocks and physical stops when necessary to allow rack installation activities associated with the WBN Rerack Project including installation of the burnup credit racks as discussed in Reference 4. Such activities may involve movement of loads over portions of the fuel pool not containing fuel.

APPLICABILITY

TR 3.9.4 is applicable only when fuel assemblies are in the spent fuel pool or cask loading area of the cask pit. If there are no fuel assemblies in the pool or cask pit, there is no danger of damaging a fuel assembly with a dropped load, therefore, the TR does not apply. The Applicability has been modified by a Note stating that the provisions of TR 3.0.3 do not apply.

ACTIONS

A.1

If a load in excess of 2059 pounds, except the gates, is allowed to traverse fuel assemblies in the spent fuel pool, the load must immediately be placed in a safe condition. The same action is required for loads moved over fuel in the cask loading area when the impact shield is not in place or when the loads do not meet the weight, cross-sectional impact area, and allowable travel height criteria of Figure 3.9-1. These actions require moving the load to a position which is not over the spent fuel pool or cask loading area.

(continued)

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.9.4.1

TSR 3.9.4.1 requires that the crane interlocks and physical stops, which prevent crane travel over fuel assemblies, are demonstrated to be OPERABLE. This surveillance must be performed within 7 days prior to using the crane and atleast once per 7 days thereafter during crane operation. The Frequency of 7 days corresponds to ANSI B30.2, "Frequent Inspection for Heavy to Severe Service."

The surveillance is modified by a NOTE which allows defeating the crane interlocks and physical stops when necessary to allow rack installation activities associated with the WBN Rerack Project including installation of the burnup credit racks as discussed in Reference 4. Such activities may involve movement of loads over portions of the fuel pool not containing fuel.

TSR 3.9.4.2

TSR 3.9.4.2 requires prior verification that the impact shield is properly in place and that each load exceeding 2059 lbs carried over fuel in the cask loading area of the cask pit meet the weight, cross-sectional impact area, and allowable travel height criteria shown in Figure 3.9-1.

REFERENCES

1. Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis."
2. Watts Bar FSAR, Section 9.1.2, "Spent Fuel Storage."
3. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.
4. "Spent Fuel Pool Modification for Increased Storage Capacity," Watts Bar Nuclear Plant Unit 1, report submitted by Tennessee Valley Authority letter dated October 23, 1996.

MARKUP OF
DRAFT TECHNICAL SPECIFICATIONS

(WBN-TS-96-010)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

(shown in Figure 4.3-1)

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of ~~3.50~~ 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Sections 4.3.2.7 and 9.1 of the FSAR;

INSERT 2

~~c. A nominal 10.72 inch center to center distance between fuel assemblies placed in the high density fuel storage racks; and~~

~~d. With the spent fuel storage pool flooded, fuel assemblies must not be loaded in cells on the pool periphery or loaded in a configuration which would contain fuel assemblies in cells which are face adjacent.~~

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.3 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR; and

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

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Page 4.0-2
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c. Distances between fuel assemblies as follows:

1. A nominal 10.375 inch center-to-center spacing in the 24 flux trap rack modules (Region 1).

2. A nominal 8.972 inch center-to-center spacing in the ten burnup credit rack modules peripherally located adjacent to the south and west pool walls (Region 2); and

3. A nominal 8.972 inch center-to-center spacing in the single 15 x 15 burnup credit rack module in the fuel cask loading area of the cask pit (Region 2).

d. Spent fuel assemblies with a burnup in the "acceptable burnup domain" of Figure 3.7.15-1 may be allowed unrestricted storage in either in either type of fuel storage rack.

e. New or partially spent fuel assemblies with a burnup in the "unacceptable burnup domain" of Figure 3.7.15-1 will be stored in compliance with the following configuration:

1. In the flux trap rack modules (Region 1), fuel assemblies with enrichment greater than 3.80 weight percent U-235 and burnup less than 6.750 megawattday/kilogram uranium (MWD/KgU) shall be placed in storage cells that face adjacent cells in the flux trap modules containing either water or fuel assemblies with accumulated burnup of at least 20 MWD/KgU.

2. Storage in any burnup credit rack modules (Region 2) located in the pool as well as in the fuel cask loading area is restricted to fuel of 4.95 ± 0.05 weight percent initial enrichment burned to at least 41 MWD/KgU (assembly average), or fuel of other enrichments with a burnup yielding an equivalent reactivity in the fuel racks. The minimum required assembly average burnup is given by Y(MWD/KgU) where $Y= 0.0666E^3 - 1.3933E^2 + 18.7600E - 25.7425$, where E is the initial enrichment in the axial zone of highest enrichment. Figure 3.7.15-1 illustrates the burnup enrichment equation in graphical form.

3. New fuel with enrichment up to 4.95 ± 0.05 weight percent U-235 may be placed in the burnup credit rack (Region 2) in the cask pit rack location with face adjacent storage cells containing water.

A water cell is less reactive than any cell containing fuel and therefore a water cell may be used at any location in the loading arrangements.

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

4.3.2 Drainage

The spent fuel storage pool ~~is~~ and cask pit area are designed and shall be maintained to prevent inadvertent draining of the pool below elevation 747' - 1 1/2".

4.3.3 Capacity

The total spent fuel storage ~~pool is designed and shall be maintained with a storage~~ capacity ~~limited to no more than 484~~ is 1835 fuel assemblies.

INSERT

INSERT Above

4.3.3.1 The primary portion of the spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1386 fuel assemblies in twenty-four flux trap rack modules.

4.3.3.2 No more than 224 fuel assemblies will be stored in ten smaller burnup credit rack modules peripherally located adjacent to the south and west walls of the pool.

4.3.3.3 In addition, no more than 225 fuel assemblies will be stored in a single 15 x 15 burnup credit rack module in the cask loading area of the cask pit.

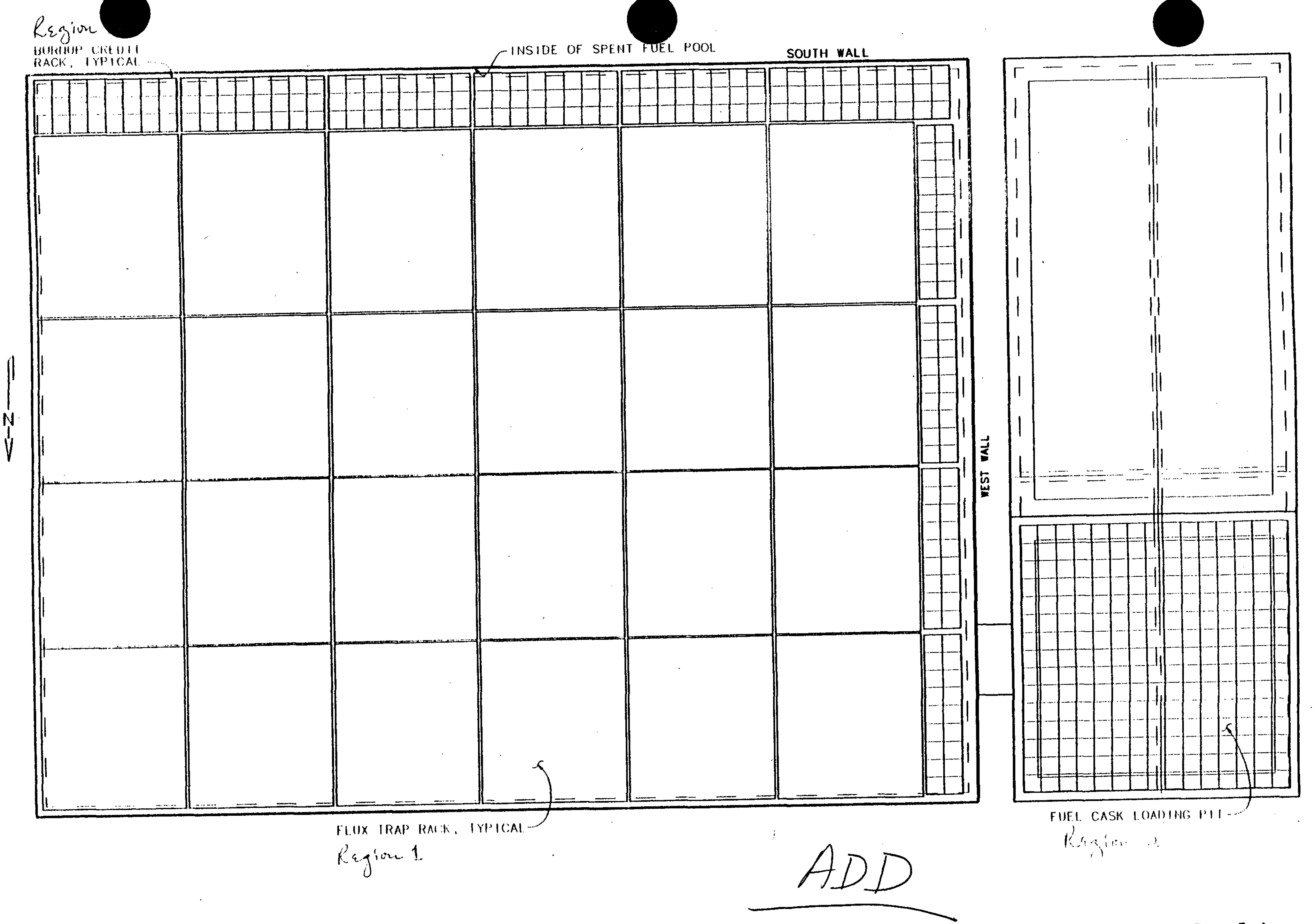


FIGURE 4.3-1
SPENT FUEL STORAGE RACKS

3.7 PLANT SYSTEMS

3.7.~~17~~ 15 Spent Fuel Assembly Storage

LCO 3.7.~~17~~ 15 The combination of initial enrichment and burnup of each spent fuel assembly stored in Region 1 or Region 2 ~~[Region 2]~~ shall be within the Acceptable {Burnup Domain} of Figure 3.7.~~17~~ 15-1 or in accordance with Specification 4.3.1.1.

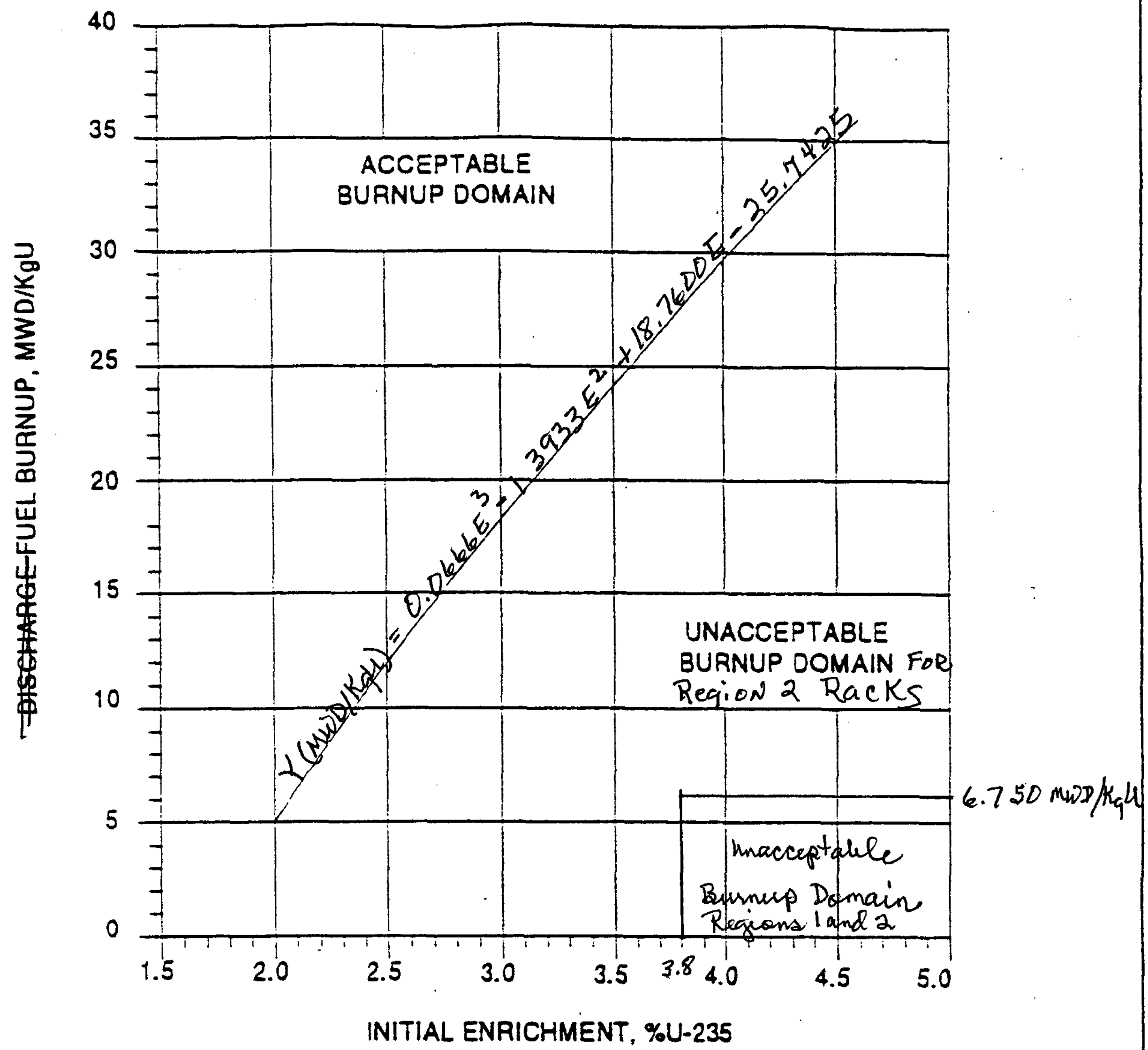
APPLICABILITY: Whenever any fuel assembly is stored in ~~[Region 2] of~~ the spent fuel storage pool or cask loading area.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ---------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Initiate action to move the noncomplying fuel assembly. ~~from [Region 2].~~	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.~~17~~ 15.1 Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.~~17~~ 15-1 or Specification 4.3.1.1.	Prior to storing the fuel assembly. ~~in [Region 2]~~



~~Not to be used for Operation.~~
~~For illustration purposes only.~~

Figure 3.7.~~17~~ 15-1 ~~(page 1 of 1)~~
~~Fuel Assembly Burnup Limits in Region 2~~
Acceptable Burnup Domain - Watts Bar
Spent Fuel Storage Racks

Add Figure

B 3.7 PLANT SYSTEMS

B 3.7.~~17~~ 15 Spent Fuel Assembly Storage

BASES

BACKGROUND

INSERT 1

~~In the Maximum Density Rack (MDR) [(Refs. 1 and 2)] design, the spent fuel storage pool is divided into two separate and distinct regions which, for the purpose of criticality considerations, are considered as separate pools. [Region 1], with [336] storage positions, is designed to accommodate new fuel with a maximum enrichment of [4.65] wt% U-235, or spent fuel regardless of the discharge fuel burnup. [Region 2], with [2670] storage positions, is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure 3.7.17-1, in the accompanying LCO. Fuel assemblies not meeting the criteria of Figure [3.7.17-1] shall be stored in accordance with paragraph 4.3.1.1 in Section 4.3, Fuel Storage.~~

The water in the spent fuel storage pool and cask loading area normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 3) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, ~~the most severe accident scenario is associated with the movement of fuel from [Region 1 to Region 2], and accidental misloading of a fuel assembly in [Region 2]. This could potentially increase the criticality of [Region 2].~~ INSERT 2 To mitigate these postulated criticality related ~~accidents~~ events, boron is dissolved in the pool water. Safe operation of the ~~MDR~~ spent fuel storage design with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with the accompanying LCO. Prior to movement of an assembly in the pool or cask loading area, it is necessary to perform SR ~~3.7.16.1.~~ 3.9.9.1 and 3.9.10.1, respectively.

(continued)

Draft page 3.7-85

INSERT 1 In the spent fuel storage design (References 1 and 2), the spent fuel pool area is divided into two separate and distinct regions for criticality considerations. Region 1, flux trap modules, with 1386 storage positions, is designed to accommodate fuel with enrichment as high as 3.8 weight percent without restrictions. Storage of fuel assemblies with enrichment between 3.8 and 5.0 weight percent in Region 1 requires either fuel burnup of ≥6.750 MWD/KgU in accordance with Figure 3.7.15-1, or placement in storage locations which have face adjacent storage cells containing either water or fuel assemblies with accumulated burnup of at least 20.0 MWD/KgU in accordance with paragraph 4.3.1.1.

Region 2, burnup credit racks, with 449 storage positions, is designed to accommodate fuel with 4.95 ±0.05 weight percent initial enrichment burned to at least 41 MWD/KgU or, in accordance with Figure 3.7.15-1, fuel of lower enrichment which yields an equivalent reactivity. In addition, the Region 2 rack (15 x 15 cell array) in the cask loading area of the cask pit is designed to store up to 4.95 ±0.05 weight percent initial enrichment new fuel when placed in storage locations which have empty face adjacent storage cells.

INSERT 2 an abnormal scenario could be associated with the improper movement of a relatively high enrichment, low exposure fuel assembly from Region 1 to Region 2, or the misloading of a fuel assembly in either region. This could potentially increase the criticality of the storage regions.

BASES (continued) and cask loading area

APPLICABLE SAFETY ANALYSES

The hypothetical events ~~accidents~~ can only take place during or as a result of the movement of an assembly (Ref. 4). For these accident occurrences, the presence of soluble boron in the spent fuel storage pool (controlled by LCO 3.9.9 ~~3.7.16~~, "Spent ~~Fuel Storage~~ Pool Boron Concentration," LCO 3.9.10, "Cask Pit Boron Concentration,") prevents criticality in both storage rack regions. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for such events, ~~accidents,~~ the operation may be under the auspices of the accompanying LCO.

The configuration of fuel assemblies in the fuel storage pool and cask loading area satisfies Criterion 2 of the NRC Policy Statement.

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with Figure 3.7.15 ~~17~~-1, in the accompanying LCO, ensures the k_{eff} ~~of the spent fuel storage pool~~ will always remain < 0.95, assuming the pool and cask loading area to be flooded with unborated water. Fuel assemblies not meeting the criteria of Figure [3.7.15 ~~17~~-1] shall be stored in accordance with Specification 4.3.1.1 in Section 4.3.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in ~~[Region 2] of~~ the fuel storage pool or cask loading area.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in ~~[Region 2]~~ the spent fuel storage pool or the cask loading area is not in accordance with Figure 3.7.15 ~~17~~-1, or paragraph 4.3.1.1, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure 3.7.15 ~~17~~-1 or Specification 4.3.1.1.

(continued)

BASES

ACTIONS

A.1 (continued)

If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.~~17~~15.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure [3.7.~~17~~15-1] in the accompanying LCO. For fuel assemblies in the unacceptable range of Figure 3.7.~~17~~15-1, performance of this SR will ensure compliance with Specification 4.3.1.1.

REFERENCES

1. Watts Bar FSAR, Sections 4.3.2.7 and 9.1.2 ~~Callaway FSAR, Appendix 9.1A, "The Maximum Density Rack (MDR) Design Concept."~~

2. "Spent Fuel Pool Modification for Increased Storage Capacity," ~~Description and Evaluation for Proposed Changes to Facility Operating Licenses DPR-39 and DPR-48 (Zion Power Station).~~ (Chapter 4), Watts Bar Unit 1, submitted by TVA letter dated October __, 1996.

3. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

~~4. FSAR, Section [15.7.4].~~

B 3.9 REFUELING OPERATIONS

B 3.9.9 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

The spent fuel storage rack criticality analysis assumes 2000 ppm soluble boron in the fuel pool during a dropped/misplaced fuel assembly event.

APPLICABLE SAFETY ANALYSES

This requirement ensures the presence of at least 2000 ppm soluble boron in the spent fuel pool water as assumed in the spent fuel rack criticality analysis for dropped/misplaced fuel assembly event.

The RCS boron concentration satisfies Criterion 2 of the NRC Policy Statement.

LCO

The LCO requires that the boron concentration in the spent fuel pool be greater than or equal to 2000 ppm during fuel movement.

APPLICABILITY

This LCO is applicable when the spent fuel pool is flooded and fuel is being moved. Once fuel movement begins, ~~the~~ movement is considered in progress ~~will continue~~ until the configuration of the assemblies in the storage racks is verified to comply with the criticality loading criteria specified in Figure 3.7.15-1 and Specification 4.3.1.1~~.d.~~

ACTIONS

A.1

If the spent fuel pool boron concentration does not meet the above requirements, fuel handling in the spent fuel pool must be suspended immediately. This action precludes a fuel handling accident, when conditions are outside those assumed in the accident analysis.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

(continued)

3.9 REFUELING OPERATIONS

3.9.~~9~~ 10 ~~Spent Fuel Pool~~ Cask Pit Boron Concentration

LCO 3.9.~~9~~ 10 Boron concentration of the ~~spent fuel pool~~ cask pit shall be ≥ 2000 ppm.

APPLICABILITY: During fuel movement in the flooded ~~spent fuel pool~~ cask pit.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend fuel movement.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.9.1 Verify boron concentration in the ~~spent fuel pool~~ cask pit is ≥ 2000 ppm.	Prior to movement of fuel in the cask pit ~~spent fuel pool~~ AND 72 hours thereafter

B 3.9 REFUELING OPERATIONS

B 3.9.~~9~~10 ~~Spent Fuel Pool~~ Cask Pit Boron Concentration

BASES

BACKGROUND

The spent fuel storage rack criticality analysis assumes 2000 ppm soluble boron in the fuel ~~pool~~ cask pit during a dropped/misplaced fuel assembly event.

APPLICABLE SAFETY ANALYSES

This requirement ensures the presence of at least 2000 ppm soluble boron in the spent fuel ~~pool~~ cask pit water as assumed in the spent fuel rack criticality analysis for dropped/misplaced fuel assembly event.

The RCS boron concentration satisfies Criterion 2 of the NRC Policy Statement.

LCO

The cask pit LCO requires that the boron concentration in the spent fuel ~~pool~~ be greater than or equal to 2000 ppm during fuel movement.

APPLICABILITY

This LCO is applicable in the cask pit, when the spent fuel ~~pool~~ cask pit is flooded and fuel is being moved. Once fuel movement begins, the movement is considered in progress ~~will continue~~ until the configuration of the assemblies in the storage racks is verified to comply with the criticality loading criteria specified in Figure 3.7.15-1 and Specification 4.3.1.1.~~d~~.

ACTIONS

A.1

If the spent fuel ~~pool~~ cask pit boron concentration does not meet the above requirements, fuel handling in the spent fuel ~~pool~~ cask pit must be suspended immediately. This action precludes a fuel handling accident, when conditions are outside those assumed in the accident analysis.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.~~8~~ 10.1

This SR requires that the spent fuel ~~pool~~ cask pit boron concentration be verified greater than or equal to 2000 ppm. This surveillance is to be performed prior to movement of fuel in the spent fuel pool and at least once ever 72 hours thereafter during the movement of fuel in the spent fuel ~~pool.~~ cask pit.

The Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of the sample. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. Watts Bar FSAR, Section 15, "Accident Analysis."

TR 3.9 REFUELING OPERATIONS

TR 3.9.4 Auxiliary Building Crane Travel - ~~Spent Fuel Storage Pool~~ Auxiliary Building

TR 3.9.4 Loads > 2059 pounds shall be prohibited from travel over fuel assemblies in the spent fuel pool.

INSERT 1

APPLICABILITY: With fuel assemblies in the spent fuel pool or in the cask loading area of the cask pit,

--------------------------NOTE--------------------------
TR 3.0.3 is not applicable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Technical Requirement not met.	A.1 Place the crane load in a safe condition.	Immediately

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.9.4.1	Demonstrate crane interlocks and physical stops which prevent crane travel over fuel assemblies to be OPERABLE. ——— NOTE ——— The crane interlocks and physical stops may be defeated for activities associated with the WBN Rerack Project including installation of the burnup credit racks, INSERT 2	Within 7 days prior to crane use. AND At least once per 7 days thereafter during crane operation.

Draft TR 3.9.4, page 3.9-1

INSERT 1 --NOTES---

1. The spent fuel pool transfer canal gate and the spent fuel pool cask pit gate may travel over fuel assemblies in the spent fuel pool.

2. The crane interlocks and physical stops may be defeated for activities associated with the WBN Rerack Project including installation of the burnup credit racks.

----

b. Cask loading area of the cask pit:

1. Loads which meet the weight, cross-sectional impact area, and allowable travel height criteria of Figure 3.9-1 may be carried over fuel assemblies stored in the cask loading area of the cask pit if the impact shield is in place over the cask loading area.

2. Loads which do not meet the weight, cross-sectional impact area, and allowable travel height criteria of Figure 3.9-1 shall be prohibited from travel over the cask loading area of the cask pit when fuel is stored in this area.

INSERT 2

	SURVEILLANCE	FREQUENCY
TSR 3.9.4.2	Verify administrative requirements concerning the impact shield are met.	Each time before an allowable load exceeding 2059 lbs. is moved across fuel stored in the cask pit area.

B 3.9 REFUELING OPERATIONS

Auxiliary Building Auxiliary

B 3.9.4 Crane Travel - ~~Spent Fuel Storage Pool~~ Building

BASES

BACKGROUND

The spent fuel pool and cask pit area are ~~is~~ a reinforced concrete structure with a stainless steel liner for leak tightness. The spent fuel storage racks consist of stainless steel structures with receptacles for nuclear fuel assemblies as they are used in a reactor. ~~, receptacles for neutron poison assemblies, and a supporting structure.~~ Design of these storage racks is in accordance with Reference 1.

of the transfer canal gate or cask pit gate from a height of eight feet above the top of the racks, and the drop

The racks can withstand the drop of a fuel assembly from its maximum supported height and the drop of tools used in the pool area. Crane travel over ~~in~~ the spent fuel storage pool ~~building~~ is limited through electrical and mechanical stops which prevent the movement of heavy objects, including shipping casks, over the spent fuel pool. The movement of casks is restricted to the cask loading area and areas away from the pool (Ref. 2).

(when no fuel is stored in this area)

APPLICABLE SAFETY ANALYSES

The release of radioactive material from fuel may occur during the refueling process, and at other times, as a result of fuel-cladding failures or mechanical damage caused by the dropping of fuel elements or the dropping of objects onto fuel elements (Reference 1). The restriction on the movement of loads in excess of the nominal weight of a fuel and control rod assembly and the associated handling tool over other fuel assemblies in the storage pool areas ensures that, in the event the ~~this~~ loads are ~~is~~ dropped, the activity release will be limited to that contained in a single fuel assembly, and that any possible distortion of fuel in the storage racks will not result in a critical array. ~~These~~ Fuel assembly drops are design basis type accidents that have not been significant to risk when analyzed in environmental reports (Reference 3).

And the allowance of gate movement over fuel assemblies in the storage pool

TR

(other than the transfer canal and cask pit gates (3826 pounds) (NOTE 1))

TR 3.9.4 requires that loads greater than 2059 pounds shall be prohibited from travel over fuel assemblies in the spent fuel pool. This ensures that objects traversing the pool are within the design basis and will not cause an unsafe condition if accidentally dropped.

INSERT 1

(continued)

BASES (continued)

APPLICABILITY

TR 3.9.4 is applicable only when fuel assemblies are in the spent fuel pool and cask loading area of the cask pit. If there are no fuel assemblies in the pool or cask pit, there is no danger of damaging a fuel assembly with a dropped load, therefore, the TR does not apply. The Applicability has been modified by a Note stating that the provisions of TR 3.0.3 do not apply.

ACTIONS

A.1

If a load in excess of 2059 pounds, except the gates, is allowed to traverse fuel assemblies in the spent fuel pool, the load must immediately be placed in a safe condition. INSERT 2 ~~This entails~~ These actions require moving the load to a position which is not over the spent fuel pool or cask loading area.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.9.4.1

TSR 3.9.4.1 requires that the crane interlocks and physical stops, which prevent crane travel over fuel assemblies, are demonstrated to be OPERABLE. This surveillance must be performed within 7 days prior to using the crane and at least once per 7 days thereafter during crane operation. The Frequency of 7 days corresponds to ANSI B30.2, "Frequent Inspection for Heavy to Severe Service." INSERT 3

REFERENCES

1. Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis."
2. Watts Bar FSAR, Section 9.1.2, "Spent Fuel Storage."
3. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.
4. "Spent Fuel Pool Modification for Increased Storage Capacity," Watts Bar Nuclear Plant Unit 1, report submitted by Tennessee Valley Authority letter dated October _, 1996.

Draft B3.9.4, Page B 3.9-8

INSERT 1 - TR
The evaluation of dropped gates or loads associated with rack installation under the WBN Rerack Project is provided in Reference 4. Assurance against load drops over fuel stored in the cask loading area of the cask pit is also evaluated to calculated load criteria which will prevent penetration of the impact shield in the event of a load drop. NOTE 2 has been added which allows defeating the crane interlocks and physical stops when necessary to allow rack installation activities associated with the WBN Rerack Project including installation of the burnup credit racks as discussed in Reference 4. Such activities may involve movement of loads over portions of the fuel pool not containing fuel.

Draft B 3.9.4, Page B 3.9-9

INSERT 2 - ACTIONS
The same action is required for loads moved over fuel in the cask loading area when the impact shield is not in place or when the loads do not meet the weight, cross-sectional impact area, and allowable travel height criteria of Figure 3.9-1.

Draft B 3.9.4. page B 3.9-9

INSERT 3 - TECHNICAL SURVEILLANCE REQUIREMENTS

The surveillance is modified by a NOTE which allows defeating the crane interlocks and physical stops when necessary to allow rack installation activities associated with the WBN Rerack Project including installation of the burnup credit racks as discussed in Reference 4. Such activities may involve movement of loads over portions of the fuel pool not containing fuel.

ENCLOSURE 2

REPORT
SPENT FUEL POOL MODIFICATION
FOR
INCREASED STORAGE CAPACITY

WATTS BAR NUCLEAR PLANT UNIT 1

DOCKET NO. 50-390

(WBN-TS 96-010)

REPORT

SPENT FUEL POOL MODIFICATION

FOR

INCREASED STORAGE CAPACITY

WATTS BAR UNIT 1
DOCKET NO. 50-390

OCTOBER 1996

Tennessee Valley Authority

TABLE OF CONTENTS

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (continued)

CHAPTER 1

INTRODUCTION

1.0 INTRODUCTION

The Watts Bar Nuclear Plant (WBN) is a pressurized water nuclear power reactor installation owned and operated by the Tennessee Valley Authority (TVA). The facility is located approximately 50 miles northeast of Chattanooga in Rhea County, Tennessee. WBN Unit 1 went into commercial operation on May 27, 1996. The WBN fuel storage system includes a fuel pool 474 inches long and 380.5 inches wide with an adjacent cask pit area for cask setdown and loading. The pool contains 1312 spent fuel storage locations of which 484 are currently usable. The existing racks were manufactured by Wachter Associates in Pittsburgh, Pennsylvania. Unit 1 is presently in its first operating cycle and no fuel assemblies are stored in the pool. Since the full core has 193 fuel assemblies, maintaining a full core offload capability implies that 291 storage cells (484 minus 193) are available for normal offload storage. This capacity for discharge fuel storage will be adequate for either 3 or 4 cycles of operation depending on reload batch sizes; thus, WBN could lose full core discharge capability as early as the year 2001 if operating capacity factors are relatively high.

The relatively near-term projected loss of full core discharge capability and the desirability of replacing the Boraflex neutron absorber material which is contained in the existing storage rack modules, prompted the present undertaking to rerack the WBN pool before the end of WBN Unit 1's first operating cycle. Replacing the existing racks before initial fuel is discharged reduces nuclear waste because the racks removed will be uncontaminated. Removal before the end of cycle one will also minimize personnel exposures, reduce the risks associated with reracking with spent fuel in the pool, avoid moving heavy loads near spent fuel, and result in significantly lower installation costs.

The purpose of this submittal is to request authorization to rerack the WBN pool area by equipping it with storage racks containing up to 1835 storage cells. Subject to fuel placement controls and burnup requirements, the racks will be capable of storing assemblies with up to 5.0 weight percent (wt%) Uranium-235 (U-235) initial enrichment. The replacement storage cells have Boral rather than Boraflex neutron absorber material for criticality control. TVA entered into a contract with Holtec International in January 1996 for engineering analysis assistance. This licensing document has been prepared by TVA with support from Holtec.

Twenty-four racks containing 1386 storage cells with Boral neutron absorber material will be positioned in the spent fuel pool in a free-standing four-by-six module array. These replacement storage racks were designed and manufactured by Programmed and Remote System Corporation (PaR) of St. Paul, Minnesota, in 1979 for TVA's Sequoyah Nuclear Plant (SQN) and were licensed and used successfully in service at that facility for about 13 years until their replacement in 1995 with higher density racks which increased SQN's pool storage capacity. The SQN and WBN pools are essentially identical, as is the fuel used at both plants.

The PaR storage cells are identical and are referred to as flux trap cells meaning there is a water gap between adjacent storage cells such that the neutrons emanating from a fuel assembly are thermalized before reaching an adjacent assembly and the Boral neutron absorber panel in its storage cell. Boral is a more effective neutron absorber for thermalized or low energy neutrons. The placement of fuel with enrichments greater than 3.80 wt% U-235 and burnups less that 6.75 Megawatt Days per Kilogram of Uranium (MWD/KgU) in the PaR flux trap cells is administratively controlled. No credit is taken for soluble boron in normal refueling and full core offload storage situations.

In addition to the PaR racks previously described, TVA plans to install smaller burnup-credit rack modules, or "baby" racks, peripheral to the PaR racks along the south and west walls of the pool. The cells in these ten peripheral modules (224 cells) and a 15 x 15 module (225 cells) planned for the cask loading area of the cask pit are identical in design and construction to the Holtec rack cells presently installed and in use at SQN. These racks will extend WBN's spent fuel storage capacity for the Unit 1 lifetime. The peripheral "baby" racks and the cask pit rack will not be installed in the 1997 WBN rerack activity, but utilized at a later date when the increase in the fuel pool inventory, or other factors, warrant their deployment.

The planned pool layout is shown in Figure 1.1. Cell design and construction information is provided in Chapter 3 of this document. Table 1 provides key comparison data for the existing and proposed WBN rack modules.

The replacement and new spent fuel storage racks are free-standing and self-supporting. The principal construction materials for the PaR flux trap racks are 304 stainless steel canisters, CF-3M grid castings, and pedestals from age hardened 17-4 PH. The Holtec burnup credit racks will utilize 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 racks is the neutron absorber material, Boral, in sheets consisting of boron carbide in an aluminum composite matrix which is clad with rolled 1100 aluminum. The racks are designed and analyzed in accordance with Section III, Subsection NF of the ASME Boiler and Pressure Vessel (B & PV) Code. The material procurement, analysis, and fabrication of the rack modules conform to 10 CFR 50, Appendix B requirements.

This licensing modification report documents the design and analyses performed to demonstrate that the spent fuel racks satisfy the 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. The safety assessment of the proposed rack modules involved demonstration of their thermal-hydraulic, criticality and structural adequacy. Hydrothermal adequacy requires that fuel cladding will not fail due to excessive thermal stress, and that the steady state pool bulk temperature remains within the limits prescribed for the spent fuel pool to satisfy the pool structural strength constraints. Demonstration of structural adequacy primarily involves analyses showing that the free-standing rack modules will not impact with each other in the cellular region or with the pool walls under the postulated Design Basis Earthquake (DBE) and Safe Shutdown Earthquake (SSE) events, and that the primary stresses in the rack module structure will remain below the ASME B&PV Code (subsection NF) allowable. The structural qualification also includes analytical demonstration that the subcriticality of the stored fuel will be maintained under accident scenarios such as fuel assembly drop and drop of a gate. The structural consequences of these postulated accidents are evaluated and presented in Chapter 7 of this report. The criticality safety analysis shows that the neutron multiplication factor for the stored fuel array is bounded by the NRC limit of 0.95 (OT Position Paper) under assumptions of 95% probability and 95% confidence.

This licensing modification report contains documentation of the analyses performed to demonstrate the margins of safety with respect to NRC's specified criteria. This report also contains the results of the analysis performed to demonstrate the integrity of the fuel pool reinforced concrete structure and an appraisal of radiological considerations. A summary of the cost/benefit consideration demonstrating reracking as the most cost effective approach to increase the onsite storage capacity of the WBN spent fuel storage pool is also included in this report.

Computer programs utilized in performing the analyses documented in this report are identified in the appropriate sections. Contractor computer codes are benchmarked and verified in accordance with Holtec International's Nuclear Quality Program.

The analyses presented herein clearly demonstrate that the rack module arrays possess acceptable margins of safety from all four vantage points: thermal-hydraulic, criticality, structural and radiological. The no significant-hazards considerations determination for the Technical Specification amendment, along with this licensing modification report is based on the descriptions and analyses synopsized in the subsequent sections of this report.

TABLE 1

RACK MODULE DATA, EXISTING AND PROPOSED RACKS

ITEM	EXISTING RACKS Wachter	PROPOSED RACKS PaR	Holtec
Number of Cells	1312 (484 usable)	1386	449
Number of Modules	16	24	11
Neutron Absorber	Boraflex	Boral	Boral
Nominal Cell Pitch, (inches)	10.72	10.375	8.972
Nominal Cell Envelope (inches)	8.99 x 8.99	8.75 x 8.75	8.75 x 8.75
Maximum Initial Enrichments (Weight % U-235)	3.50	5.0[1]	5.0[1]

(1) With burnup credit and administrative placement restrictions.

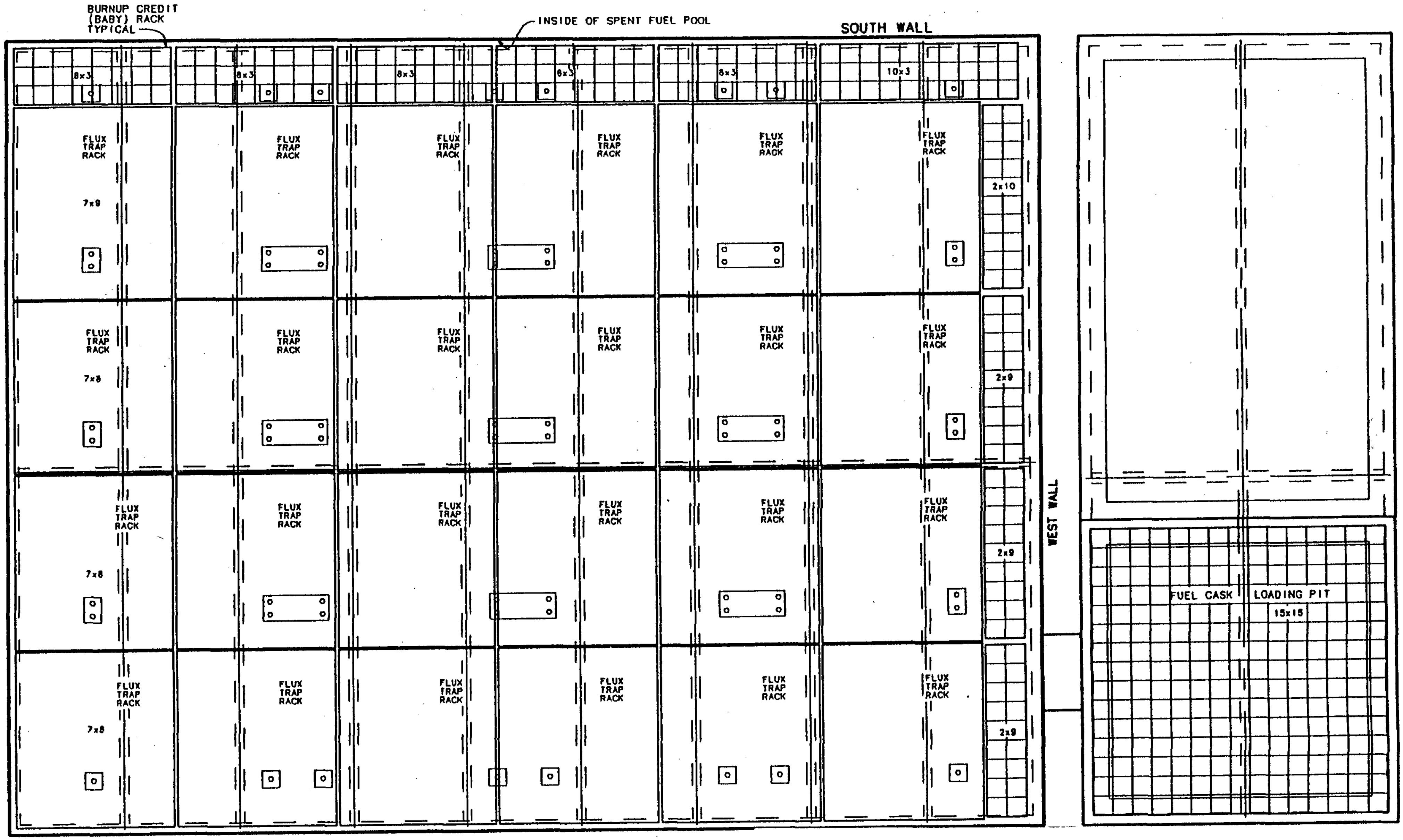


Figure 1.1 **Module Layout in the Watts Bar Spent Fuel Pool and Cask Pit**

CHAPTER 2

MODULE DATA AND INSTALLATION INFORMATION

2.0 MODULE DATA AND INSTALLATION INFORMATION

2.1 SYNOPSIS OF THE MODULES

The WBN spent fuel storage pool consists of a 474 inch x 380.5 inch (nominal) rectangular pit and a separate 144 inch x 380.5 inch (nominal) pit next to the fuel pool pit for cask handling operations. The fuel pit is connected to the cask handling area and the fuel transfer canal through weir gates in the west wall between the two pits and the south wall, respectively. Figure 2.1.1 shows a planar section of the WBN spent fuel pool region.

At the present time, the WBN pool contains medium density racks with a 10.72 inch nominal storage cell center-to-center pitch. There is a total of 1312 storage cells in the pool with 484 of these usable at the present time. There are twelve 8x10 modules (80 cells each) and four 8x11 modules (88 cells each).

Figure 2.1.2 shows the module layout for the WBN pool after the proposed reracking campaign assuming nominal pool dimensions. Final design checks and dimensional rollups may produce very slight changes to the design dimensions shown. The final as-built layout will be based on the prismatic envelope dimension of the spent fuel pool at the elevation of the racks. Based on investigative measurements made in June 1996, the actual rack-to-wall gaps at certain locations in the pool differ only slightly from the nominal gaps indicated in Figure 2.1.2. Based on the margins evident from the dynamic impact evaluation described in Chapter 6 of this report, there will be no adverse effects from these differences. There are to be 35 rack modules containing a total of 1835 storage cells in the reracked configuration of the WBN pool as shown in Figure 2.1.2 and as compiled in Table 2.1.1. The essential cell data for the replacement and new storage cells is given in Table 2.1.2. The physical size and weight data on the modules is found in Table 2.1.3.

The high density spent fuel storage racks for the WBN pool and cask pit provide storage locations for up to 1835 fuel assemblies and are designed to maintain stored fuel, having an initial enrichment of up to 5 wt% U-235, in a safe, coolable, and sub-critical configuration during normal discharge and full core offload situations and postulated accident conditions. Appropriate restrictions are placed on the enrichment/burnup parameters and storage locations of the fuel. These restrictions are presented in Chapter 4, Criticality Safety Analyses.

The rack modules for the WBN spent fuel pool are the "free-standing" type inasmuch as the modules are not attached to the pool floor and do not require any lateral braces or restraints to the pool walls. The “baby” racks are connected to a larger flux trap “mother” rack at both pedestal and top region locations. The rack modules will be placed in the pool in the designated locations using specially designed lifting devices. The rack modules are supported by four pedestal legs which are remotely adjustable using telescopic handling tools. Leveling operations are performed with the support legs lifted off the floor. Racks with such adjustable pedestals can be easily leveled and made coplanar with each other and can accommodate variations in the flatness of the pool floor. The support legs also provide an under-rack plenum for natural circulation of water through the storage cells. The placement of the rack pedestals in the spent fuel pool has been designed to preclude any support legs from being located over existing recesses in the pool floor which are associated with the existing racks.

The WBN racks are subjected to designated seismic loadings. Two sets of orthogonal, statistically independent, artificial time histories are generated and each satisfies both the reference response spectrum and power spectral density enveloping criteria. The governing time history is then established as the one determined to produce the most severe response for the safe shutdown earthquake (SSE) event. Analyses undertaken to confirm structural integrity of the rack modules as described in Chapter 6 of this report include:

- Three-dimensional transient analyses of the spent fuel racks individually and as an assemblage acting as free-standing submerged bodies subjected to seismic excitations (i.e., the synthetic acceleration time histories).

- Evaluation of the primary stresses in the rack structure to establish compliance with ASME stress limits.

- Evaluation of secondary and peak stress amplitudes in the most critically loaded rack sections to ensure that failure from cyclic fatigue will not occur.

Under the seismic events, the rack modules have four designated locations of potential impact:

(i) Support leg to bearing pad
(ii) Storage cell to fuel assembly contact surfaces
(iii) Baseplate edges
(iv) Rack top corners

The support leg to pool slab bearing pad impact would occur whenever the rack support leg lifts off the pool floor during a seismic event. The "rattling" of the fuel assemblies in the storage cell is a natural phenomenon associated with seismic conditions. The baseplate and rack top corner impacts would occur if the rack modules tend to slide or tilt towards each other during the postulated seismic events. Chapter 6 of this report presents the analysis methodology and results for the four locations of potential impact and establishes the structural integrity of the racks under the postulated load combinations.

A bearing pad, made of austenitic stainless steel, is interposed between the rack support leg and the pool liner such that the loads transmitted to the slab by the rack module under steady state, as well as seismic conditions, are diffused into the pool slab, and allowable local concrete surface pressures are not exceeded. The cask pit rack is to rest on a support frame[1,2] rather than a bearing pad because the liner in the cask loading area is at a lower elevation than the pool liner. The stainless steel support frame maintains the cask pit rack at essentially the same elevation as the pool racks; therefore, the same fuel and component handling tools can be used. Chapter 8 of this report presents the details of pool structure analyses performed in support of this licensing application.

2.2 MATERIAL CONSIDERATIONS

2.2.1 Introduction

Safe storage of nuclear fuel in the WBN spent fuel pool requires that the materials utilized in the fabrication of racks be of proven durability and be compatible with the pool water environment. This section provides necessary information on this subject.

2.2.2 Structural Materials

The following structural materials are utilized in the fabrication of the burnup credit racks to be provided by Holtec International.

a. ASME SA240-304L for sheet metal stock for the storage cell structures.

b. Internally threaded support legs: ASME SA240-304L.

c. Externally threaded support spindle: ASME SA564-630 precipitation hardened stainless steel.

d. Weld material per the following ASME specification: SFA 5.9 ER308L.

The PaR flux trap racks being transferred from SQN contain the following proven materials:

a. Poison can inner and outer tubes: 304 stainless steel, ASTM Standard A-666-72 Grade B.

b. Top and bottom grid casting: CF-3M, ASTM Standard A-296-77.

c. Threaded pedestal foot: 17-4 PH, ASTM Standard A-564-66.

2.2.3 Poison Material

In addition to the structural and non-structural stainless material, the racks employ Boral, a patented product of AAR Brooks & Perkins, as the thermal neutron absorber material. Boral is a thermal neutron absorbing material consisting of finely divided particles of boron carbide (B_4C) uniformly distributed in type 1100 aluminum, pressed and sintered in a hot rolling process. Boron carbide is a compound having a high boron content in a physically stable and chemically inert form. The 1100 alloy aluminum is a light-weight 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 spent fuel pool.

The selection of Boral for use in the spent fuel pool as the neutron absorbing material can be attributed to the following reasons:

(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.

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

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

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

Boral has garnered an excellent record of application in light water reactor fuel pools and there has been no reported degradation of Boral used in spent fuel pool applications for at least the last ten years. Tests simulating the radiological, thermal, and chemical environment of the spent fuel pool have demonstrated the stability and chemical inertness of Boral.[3,4,5] The accumulated dose to the Boral over the expected rack lifetime is estimated to be about 3×10^{10} to 1×10^{11} rads depending upon how the racks are used and the number of full-core off-loads that may be necessary. As indicated in the aforementioned references, the laboratory and test reactor data have confirmed the ability of this material to withstand equivalent gamma dosages which are an order of magnitude higher than those expected in the spent fuel pools.

Based upon accelerated test programs, Boral is considered a satisfactory material for reactivity control in spent fuel storage racks and is fully expected to fulfill its design function over the lifetime of the racks. As can be inferred from Table 2.2.1, Boral's significant in-pool experience is over twenty years, and cumulative underwater experience is in excess of 200 pool years. The PaR flux trap racks being transferred to WBN were used successfully at SQN for 13 years and contained approximately 900 spent fuel assemblies when replaced in 1995 to increase the SQN pool storage capacity.

Boral is manufactured by AAR Brooks & Perkins under the control and surveillance of a computer-aided Quality Assurance/Quality Control Program that conforms to the requirements of 10CFR50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." As indicated in Table 2.2.1, Boral has been licensed by the NRC for use in numerous BWR and PWR spent fuel storage racks and has also been extensively used in overseas nuclear installations.

2.2.3.1 <u>Boral Material Characteristics</u>

Aluminum: Aluminum is a silvery-white, ductile metallic element that is abundant in the earth's crust. 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 environments. Aluminum has atomic number of 13, atomic weight of 26.98, specific gravity of 2.69 and valence of 3. The physical/ mechanical properties and chemical composition of the 1100 alloy aluminum are listed in Tables 2.2.2 and 2.2.3.

The excellent corrosion resistance of the 1100 alloy aluminum is provided by the protective oxide film that develops on its surface from exposure to the atmosphere or water. This film prevents the loss of metal from general corrosion or pitting corrosion and the film remains stable between a pH range of 4.5 to 8.5.[6]

Boron Carbide: The boron carbide contained in Boral is a fine granulated powder that conforms to ASTM C-750-80 nuclear grade Type III for the Holtec burnup credit racks and ASTM C-750-74, Type II for the PaR flux trap racks. The particles range in size between 60 and 200 mesh and the material conforms to the chemical composition and properties listed in Table 2.2.4.

2.2.4 Compatibility with Coolant

Materials used in the construction of the WBN racks have a successful history of in-pool usage. Their physical, chemical and radiological compatibility with the pool environment is an established fact at this time. Austenitic stainless steel (304L) is widely used in nuclear power plants and, as noted in Table 2.2.1, Boral has been widely used in spent fuel storage pools for many years.

2.3 EXISTING RACK MODULES AND PROPOSED RERACKING OPERATIONS

The WBN fuel pool currently has medium density rack modules containing a total of 1312 storage cells in sixteen modules. For various reasons, only 484 of the cells are presently usable. When the PaR flux trap rack modules are installed, there will be no spent fuel stored in the pool. Remotely engagable lift rigs, designed to meet the criteria of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," will be available to lift existing, replacement, and new rack modules. The Auxiliary Building crane will be used for this purpose. A module change out/addition plan and procedure will be developed which is consistent with reducing the travel distance of empty modules over the refueling floor.

The Auxiliary Building has one overhead crane which rides on rails that traverse the entire fuel handling area of the building. The crane has a main hook designed and rated at 125 tons. In addition, there is an auxiliary hoist on the overhead crane rated at 10 tons.

Pursuant to the defense-in-depth approach of NUREG-0612, the following additional measures of safety will be undertaken for the reracking operation.

(i) The crane and hoist will be given a preventive maintenance checkup and inspection within 3 months of the beginning of the reracking operation.

(ii) The main hoist will be used to lift no more than 20% of its rated capacity of 125 tons at any time during the reracking operation. (The maximum weight of any module and its associated handling tool is less than 20 tons).

(iii) The existing fuel racks will be lifted no more than 6 inches above the pool floor and held in that elevation for approximately 10 minutes before beginning the vertical lift.

(iv) The rate of vertical movement will not exceed 6 feet per minute.

(v) The rate of horizontal movement will not exceed 6 feet per minute when over the spent fuel pool.

(vi) Safe load paths will be developed. Neither the existing, replacement, or new racks will be carried over any region of the pool containing fuel. Rack height above the refuel floor will be limited to approximately one foot unless necessary to clear permanent obstructions.

(vii) The rack upending and laying down operations will be carried out in areas which do not encroach on any space ascribed to safety related equipment.

(viii) Crew members involved in the reracking operation will be given training in the use of the lifting and upending equipment. Every crew member will be required to pass a written examination in the use of lifting and upending apparatus.

(ix) In addition to the in-class training, the rack installation crew will be given "hands-on" rack handling experience prior to executing any handling operation over the fuel pool. The unloading, rigging, upending, and staging of the racks, upon their arrival from the SQN site, will be carried out by members of the installation crew. As a result, these crew members will acquire considerable handling "feel" of the racks before bringing the hardware to the refueling floor level of the Auxiliary Building.

The racks will be transported into and out the Auxiliary Building through the truck bay access door which is at ground level. This direct access to the building greatly facilitates the rack removal and installation effort. Groups of the replacement PaR flux trap racks will normally be staged on the refuel floor south of the pool. It is planned to normally transport the existing Wachter rack modules directly to the truck bay upon removal from the pool. Heavy loads will be lifted in such a manner that the center-of-gravity of the lift point is aligned with the center-of-gravity of the load being lifted. Turnbuckles will be utilized to "fine tune" the verticality of the rack being lifted.

For the later addition of new Holtec burnup credit "baby" racks to the pool, a fuel reshuffle plan and load handling plan will be developed which assures that no heavy load (rack or lifting rig) with a potential to drop on a storage module has less than three feet lateral free zone clearance from active fuel. This would be accomplished by shuffling stored fuel away from the south and west pool walls (refer to Figure 2.1.2) as necessary. The relatively small burnup credit rack modules would be brought up from the truck bay through the refueling floor hatches and moved over the cask pit area to their pool insertion locations along the west and south walls, or alternatively, to a staging area. When the larger 15 x 15 burnup credit rack is needed for interim use in the cask loading area, it also would be brought to the refuel floor via the truck bay and inserted without passing over stored fuel.

Although spent fuel will be stored in the pool during the installation of the burnup credit racks, postulated load drops during this operation will not damage the pool liner and structure to such an extent that the stored fuel would become uncovered. The reracking activity will be conducted in accordance with written procedures which will be reviewed and approved. NUREG-0612,

Appendix A, mandates that the structural analysis of the load be predicated on the following bases/limitations.

(I) The load drop orientation is the most adverse which would result in the most severe consequences.

(II) The fuel has decayed for at least 100 hours before movement from the Reactor Vessel.

(III) The true stress-strain relationship of the deforming structure is employed.

Postulating accidental drop of a rack along with its lifting attachment from the maximum possible height, and further postulating the most adverse physically admissible drop profile, results in a primary impact of the rack with the pool slab. Recognizing that the pool slab is located over a rock subgrade and buttressed with approximately 25 feet of reinforced concrete, the postulated rack drop is incapable of actuating a gross structural failure. This conclusion is substantiated by prior analyses performed for recent rerack licensing submittals such as the one for Three Mile Island Unit 1. The physical integrity of the pool and its function as a container of cooling and shielding water is, therefore, unimpaired. While localized damage to the liner can be hypothesized, the associated leakage would be minor and contained within the relatively small volume of the leak chase system, and therefore, does not constitute a loss of water from the spent fuel pool system. The leak rate would be small compared to available installed makeup capacity from the Refueling Water Storage Tank (RWST). The reactor unit has an RWST with a volume greater than 370,000 gallons and a boron concentration maintained greater than 2000 ppm as required by the WBN Technical Specifications. Borated water may be supplied from the RWST via a refueling water purification pump, which has a 200 gpm design flow. Two such pumps are available. Alternatively, a temporary line can be run from the boric acid blender, located in the Chemical and Volume Control System, directly into the spent fuel pool. In summary, the cooling and shielding of spent fuel in the pool would remain unaffected by a postulated heavy load accident during the rerack operation.

Compliance with the objectives of NUREG-0612 will follow the guidelines contained in Chapter 5 of that document. The guidelines of NUREG-0612 call 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 WBN rerack project ensures maximum emphasis to mitigate the potential load drop accidents by implementing measures to eliminate shortcomings in each aspect of the operation, including the four aforementioned areas. A summary of the measures specifically planned to deal with the major causes is provided below.

<u>OPERATOR ERRORS</u>: As mentioned above, TVA plans to provide comprehensive training to the installation crew including "hands-on" rack handling experience.

RIGGING FAILURE: The lifting devices designed for handling and installation of the old and new racks in the WBN fuel pool will have redundancies in the lift legs and lift eyes such that there are independent load paths. Failure of any one load bearing member would not lead to uncontrolled lowering of the load. The rigs will comply with the provisions of ANSI 14.6 - 1986, including compliance with the primary stress criteria, load testing at 150% of maximum lift load, and dye examination of critical welds. The rigs to be used for installing the WBN racks are similar to those rigs used in the rerack of numerous other plants.

LACK OF ADEQUATE INSPECTION: A set of inspection points which have proven to eliminate any incidence of rework or erroneous installation in numerous prior rerack projects will be developed for the racks.

INADEQUATE PROCEDURES: Several installation procedures will be prepared, reviewed, and integrated with the work orders to cover operations such as handling and shipping, receipt inspection, removal and installation, drag testing, and reworking. Distinct procedures for existing and replacement racks will be developed. The series of installation procedures planned for the WBN reracking are the successors of procedures implemented successfully in other past industry rerack projects.

The conceptual initial phase of the rack change out plan is illustrated in Figures 2.3.1 and 2.3.2. The sequential steps to accomplish the reracking are also noted on those figures. The final version of the rack handling and staging sequence operations may be a slightly optimized version of the one presented herein. In addition to removal of the existing racks, the reracking installation operation involves removal, relocation, or modification of other storage cell access obstructions such as the coolant diffuser/sparger piping at the southwest corner of the spent fuel pool. Also, the pool underwater lighting system is to be replaced with Remote Ocean Systems (ROS) Components including HPS 1000 bulbs which have been used in numerous other PWR spent fuel pool and reactor vessel containment areas.

Table 2.3.1 provides a synopsis of the requirements delineated in NUREG-0612 and intended compliance. In summary, the measures to be implemented for the WBN reracking are identical to those utilized in recent successful rerack projects.

2.4 IMPACT SHIELD FOR CASK LOADING AREA

Pursuant to the defense-in-depth concept, an impact shield[7,8] has been designed and is to be placed over the cask loading area of the cask pit when fuel is stored in the 15 x 15 Holtec burnup credit rack planned for installation in that area. This will provide additional protection against accidental drop of a heavy load on fuel stored for an interim period in that area.

This shield, shown in Figure 2.4.1, is designed to withstand a total uniform load of 144 tons which envelopes expected heavy loads by a large margin. In Chapter 7, a parametric chart correlating the cross-section, drop height, and allowable heavy load mass is provided for implementing administrative control on heavy load movements over the impact shield. Procedural controls will be established for loads carried over the cask loading area of the cask pit when the impact shield is in place and to ensure that no loads are carried over the cask loading area if fuel is present and the impact shield is not in place. One of these impact shields has been built for and used at TVA's SQN when fuel was stored in the cask loading area.

Fuel stored in the cask pit is not placed in jeopardy by an uncontrolled vertical movement of the impact shield during its installation or removal. The impact shield is not moved over any portion of the main pool, but is moved into its final position by moving directly over the cask pit. The impact shield supports will be over the cask pit concrete walls, the shield itself will be parallel with the horizontal plane, and the height of travel of the shield above the top of the cask pit surrounding walls will be minimized. Because of these factors, there is no credible scenario by which the impact shield could drop into the cask pit, rather, any accident during movement would simply bring the shield supports onto the top of the supporting walls.

2.5 REFERENCES

1. Calculation Package for Analysis of Cask Pit Platform, Holtec Report No. HI-941185, Revision 1, SQN DCN No M08736A, page 34.

2. Holtec Drawing No. 1107, Revision 6, "Adjustable Rack Support Frame for Rack Installation in the Cask Pit," SQN DCN No. M08736A, page 34a.

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

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

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

6. Brooks and Perkins Technical Bulletin No. 624, Livonia, Michigan.

7. "Design and Analysis of Impact Shield," TVA SQN Calculation No. SCG1SA69, Revision 1, Holtec Report No. HI-91712, Revision 4, SQN DCN No. M08736A, page 34.

8. Holtec Drawing No. 861, Revision 6, Cask Pit Impact Shield, SQN DCN No. M08736A, page 34a.

TABLE 2.1.1

MODULE DATA

Type	Array Cell Size	Quantity	Total Cell Count for this Module Type
Replacement Flux Trap Racks	7 x 8	18	1008
	7 x 9	6	378
New Burnup Credit "Baby" Racks	8 x 3	5	120
	10 x 3	1	30
	2 x 10	1	20
	2 x 9	3	54
New Cask Pit Burnup Credit Rack	15 x 15	1	225
		35 Modules	1835 Cells

TABLE 2.1.2

COMMON MODULE DATA

	Replacement Flux Trap Racks	New Burnup Credit "Baby" Racks
Storage cell inside dimension: (inches)	8.75	8.75 ± 0.04
Storage cell height (inches) (above the baseplate):	165.5	168 ± 1/16
Baseplate thickness (inches):	0.500 (casting minimum)	0.75 (nominal)
Support leg height (inches):	3.00 (maximum)	5.25 (nominal)
Support leg type:	Remotely Adjustable	Remotely Adjustable
Number of support legs:	4	4 (minimum)
Remote lifting and handling provision:	Yes	Yes
Poison material:	Boral	Boral
Poison length (inches):	147 ± 0.156	144 (nominal)
Poison width (inches):	8.625 (nominal)	7.5 (nominal)
Cell Pitch (inches):	10.375 (nominal)	8.972 (nominal)

TABLE 2.1.3

MODULE DATA

Type/Size	Horizontal Dimensions (inches) North - South	East - West	Weight Empty (lbs)
Flux Trap Racks[1]			
7 x 8	72.625	83.00	18,200
7 x 9	72.625	93.375	20,500
Burnup Credit "Baby" Racks[2]			
8 x 3	28.19	73.05	3430
10 x 3	28.19	90.99	4290
2 x 10	90.99	19.22	2860
2 x 9	82.02	19.22	2570
Cask Pit Burnup Credit Rack[2]			
15 x 15	135.85	135.85	32,200

[1]Horizontal dimensions shown are for the top and bottom grid castings.
[2]Horizontal dimensions shown are the baseplate dimensions.

TABLE 2.2.1

BORAL EXPERIENCE LIST (Domestic and Foreign)

Pressurized Water Reactors

Plant	Utility	Mfg. Year
Yankee Rowe	Yankee Atomic Electric	1964
Maine Yankee	Maine Yankee Atomic Power	1977
Donald C. Cook	Indiana & Michigan Electric	1979
Sequoyah 1,2	Tennessee Valley Authority	1979
Salem 1, 2	Public Service Elec & Gas	1980
Zion 1,2	Commonwealth Edison Co.	1980
Yankee Rowe	Yankee Atomic Power	1983
Indian Point 3	NY Power Authority	1987
Byron 1,2	Commonwealth Edison Co.	1988
Braidwood 1,2	Commonwealth Edison Co.	1988
Yankee Rowe	Yankee Atomic Electric	1988
Three Mile Island 1	GPU Nuclear	1990
Shearon Harris Pool B	Carolina Power & Light	1991
Donald C. Cook	American Electric Power	1991
Donald C. Cook	Indiana & Michigan Electric	1993
Sequoyah 1, 2	Tennessee Valley Authority	1994

Boiling Water Reactors

LaCrosse	Dairyland Power	1976
Monticello	Northern States Power	1978
Vermont Yankee	Vermont Yankee Atomic Power	1978
J.A. Fitzpatrick	NY Power Authority	1978
Pilgrim	Boston Edison	1978
Susquehanna 1,2	Pennsylvania Power & Light	1979
Cooper	Nebraska Public Power	1979
Duane Arnold	Iowa Elec. Light & Power	1979
Perry, 1,2	Cleveland Elec. Illuminating	1979
Limerick 1,2	Philadelphia Electric	1980
Peachbottom 2,3	Philadelphia Electric	1980
Browns Ferry 1,2,3	Tennessee Valley Authority	1980
Brunswick 1,2	Carolina Power & Light	1981
Clinton	Illinois Power	1981
Dresden 2,3	Commonwealth Edison Co.	1981
E.I. Hatch 1,2	Georgia Power	1981
Hope Creek	Public Service Elec & Gas	1985
Humboldt Bay	Pacific Gas & Electric	1986
Vermont Yankee	Vermont Yankee Atomic Power	1986
Hope Creek	Public Service Elec & Gas	1989

TABLE 2.2.1 (continued)

<u>Foreign Installations Using Boral</u>	
<u>France</u>	
12 PWR Plants	Electricite de France
<u>South Africa</u>	
Koeberg 1,2	ESCOM
<u>Switzerland</u>	
Beznau 1,2	Nordostschweizerische Kraftwerke AG
Gosgen	Kernkraftwerk Gosgen-Daniken AG
<u>Taiwan</u>	
Chin-Shan 1,2	Taiwan Power Company
Kuosheng 1,2	Taiwan Power Company
<u>Mexico</u>	
Laguna Verde 1,2	Comision Federal de Electricidad

TABLE 2.2.2

1100 ALLOY ALUMINUM PHYSICAL AND MECHANICAL PROPERTIES

Density	0.098 lb/cu. in. 2.713 gm/cc
Melting Range	1190-1215 deg. F 643-657 deg. C
Thermal Conductivity (77 deg. F)	128 BTU/hr/sq ft/deg. F/ft 0.53 cal/sec/sq cm/deg. C/cm
Coef. of Thermal Expansion (68-212 deg. F)	13.1 x 10^{-6}/deg. F 23.6 x 10^{-6}/deg. C
Specific heat (221 deg. F)	0.22 BTU/lb/deg. F 0.23 cal/gm/deg. C
Modulus of Elasticity	$10 x 10^6$ psi
Tensile Strength (75 deg. F)	13,000 psi annealed 18,000 psi as rolled
Yield Strength (75 deg. F)	5,000 psi annealed 17,000 psi as rolled
Elongation (75 deg. F)	35-45% annealed 9-20% as rolled
Hardness (Brinell)	23 annealed 32 as rolled
Annealing Temperature	650 deg. F 343 deg. C

TABLE 2.2.3

CHEMICAL COMPOSITION (by weight) - ALUMINUM (1100 Alloy)

99.00% min.	Aluminum
1.00% max.	Silicon and Iron
0.05-0.20% max.	Copper
.05% max.	Manganese
.10% max.	Zinc
.15% max.	others each

TABLE 2.2.4

BORON CARBIDE CHEMICAL COMPOSITION, Weight %

Total boron	70.0 min.
Boron-10 isotopic content in natural boron	18.0 to 20.75
Boric oxide	3.0 max.
Iron	1.0 max.
Total boron plus total carbon	94.0 min.

BORON CARBIDE PHYSICAL PROPERTIES

Chemical formula	B_4C
Boron content (weight)	78.28%
Carbon content (weight)	21.72%
Crystal Structure	rhombohedral
Density	2.51 gm./cc(0.0907 lb/cu in)
Melting Point	2450^0C (4442^0F)
Boiling Point	3500^0C (6332^0F)

TABLE 2.3.1

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?	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 14.6-1978?	Yes
5.	Will non-custom lifting devices be installed and used in accordance with ANSI B30.9-1971?	Yes
6.	Will the cranes be inspected and tested prior to use in rerack?	Yes
7.	Does the crane meet the intent of ANSI B30.2-1976 and CMAA-70?	Yes

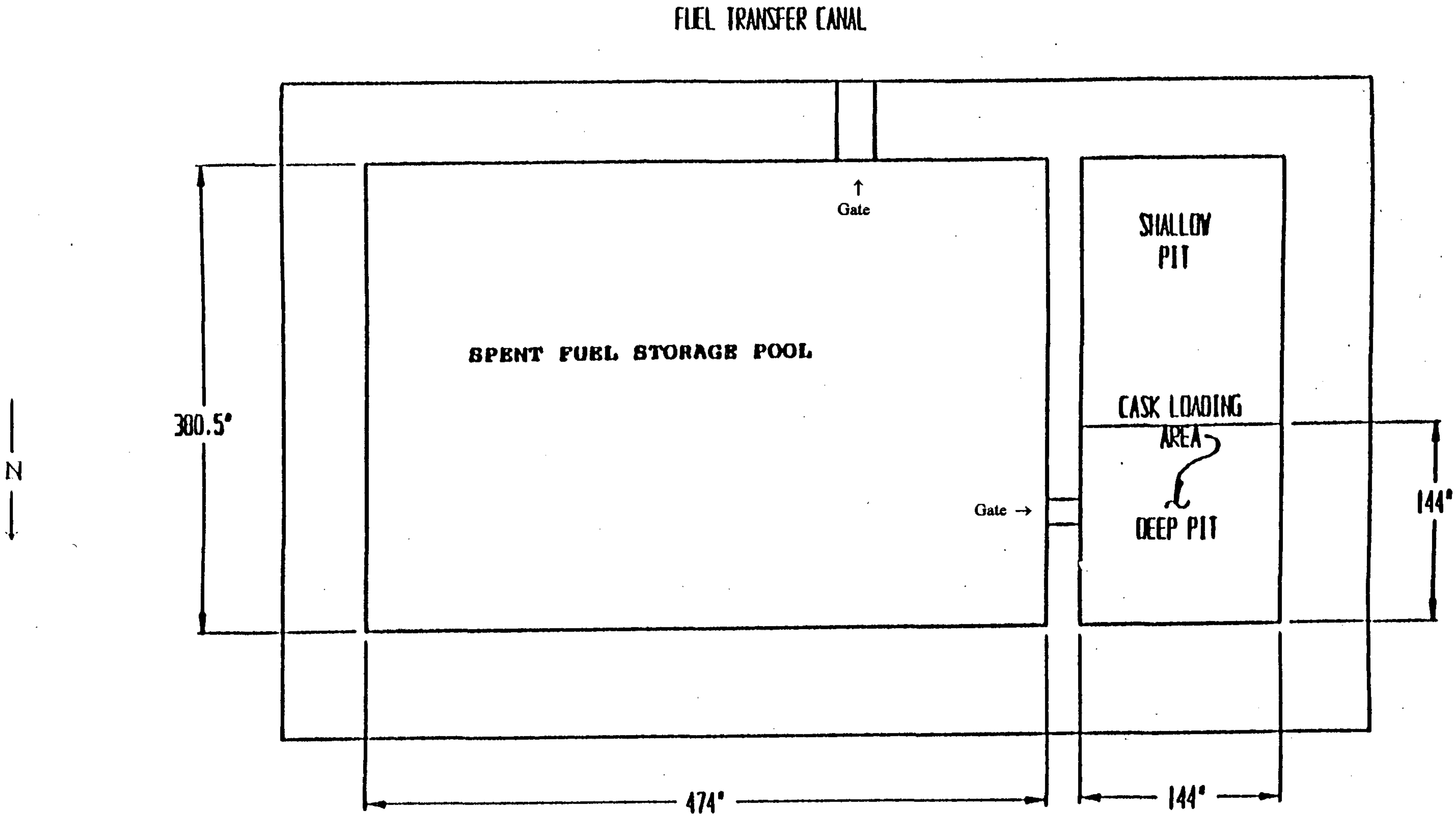
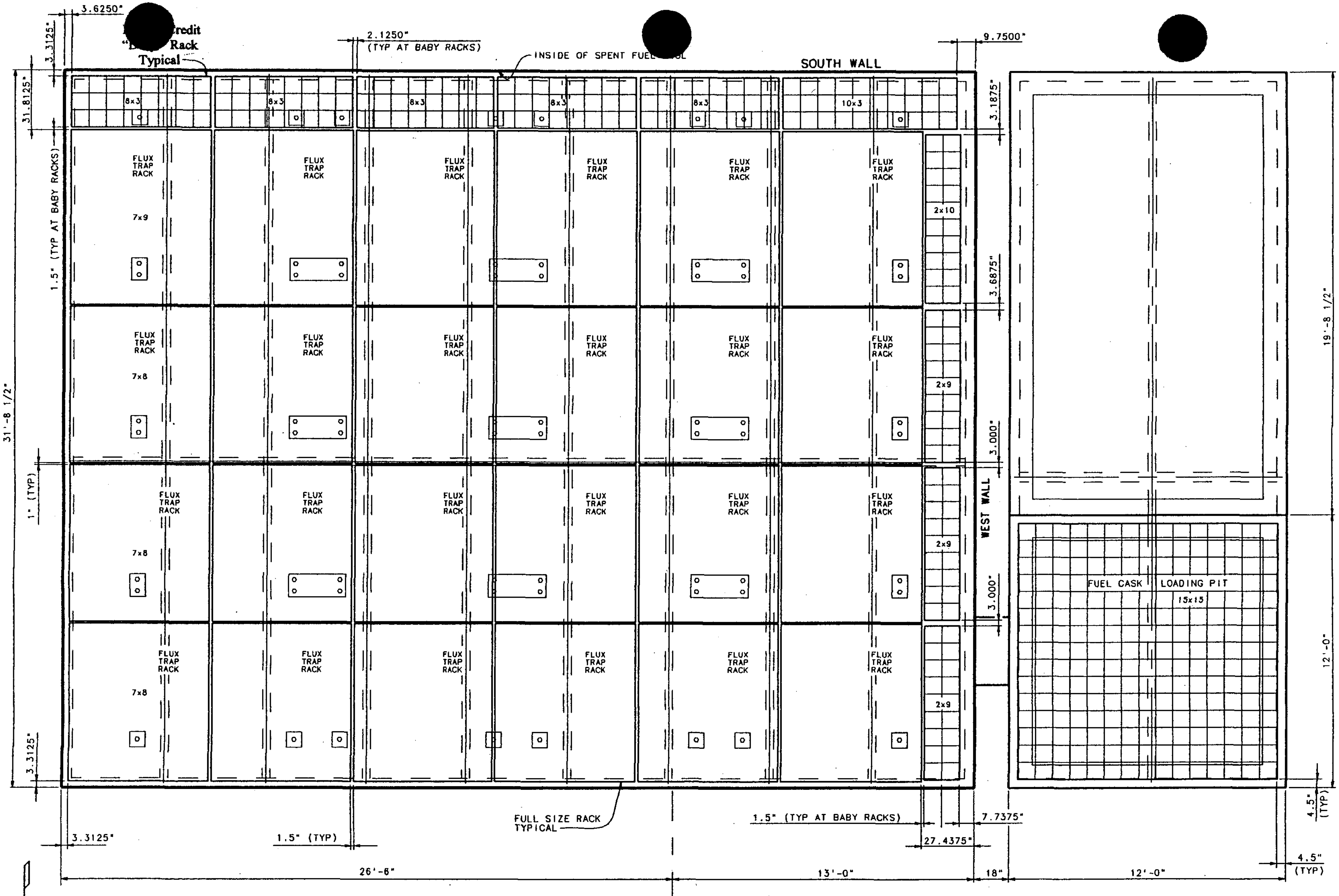


Figure 2.1.1 Spent Fuel Pit Area Of Watts Bar Nuclear Plant



PLAN
SCALE: 3/8"= 1'-0"

Figure 2.1.2 Module Layout in the Watts Bar Spent Fuel Pool and Cask Pit

EXISTING RACK REMOVAL SAFE LOAD PATHS

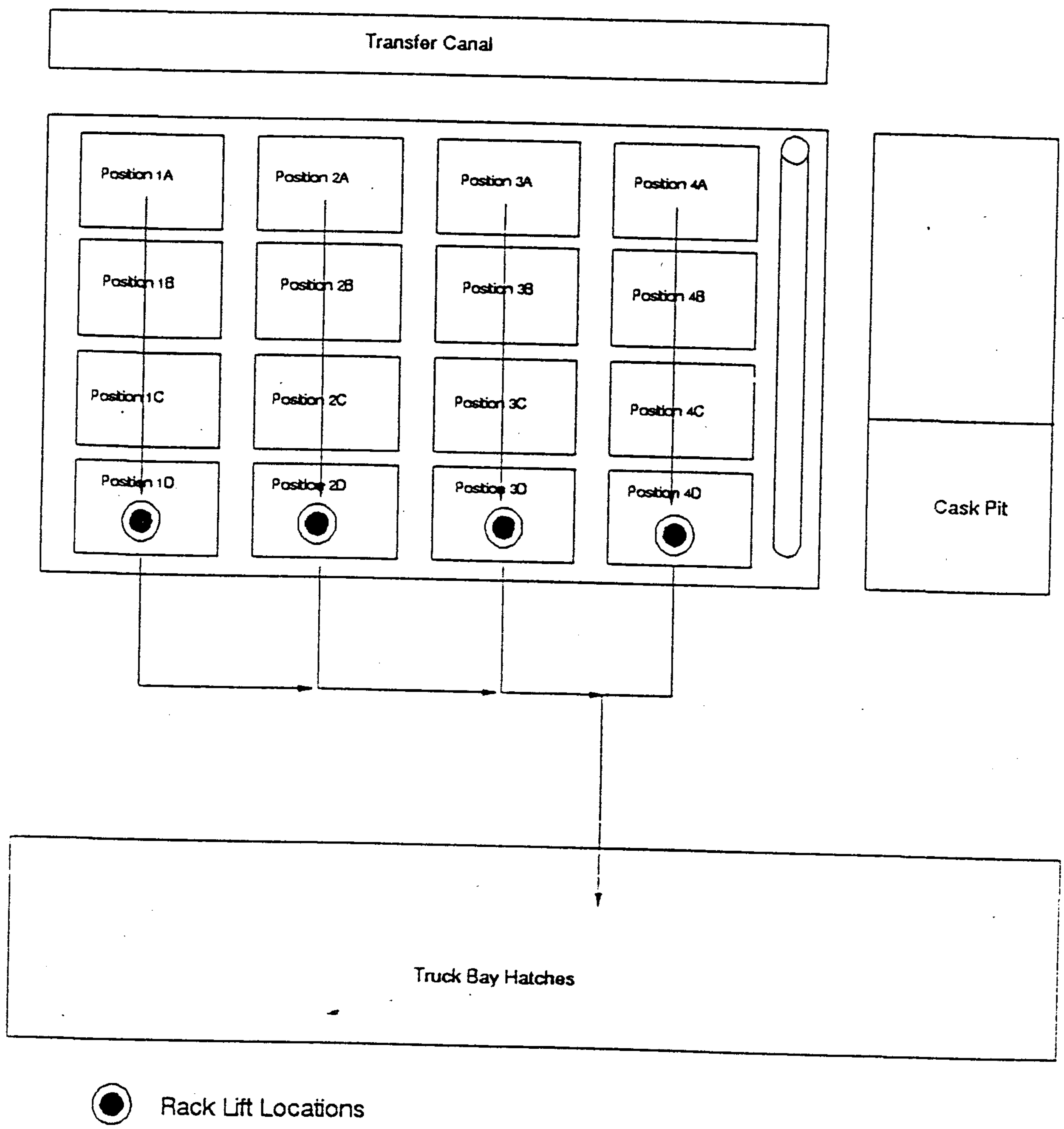
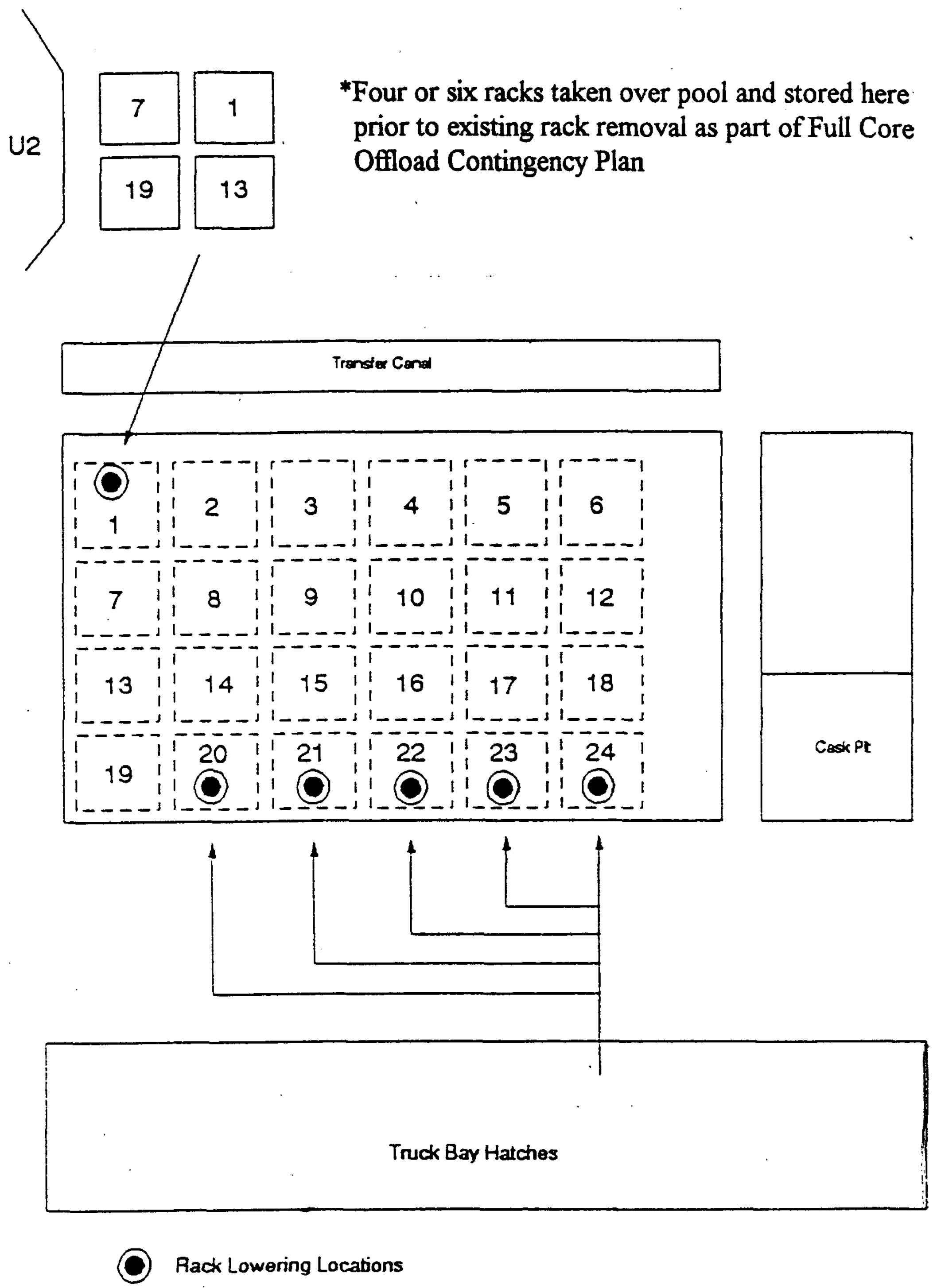


Figure 2.3.1

Rack Installation Safe Load Path



NOTE: The order of installation shall be as follows:

19, 13, 7, 1, 20, 14, 8, 2, 21, 15, 9, 3, 22, 16,
10, 4, 23, 17, 11, 5, 24, 18, 12, 6.

Figure 2.3.2

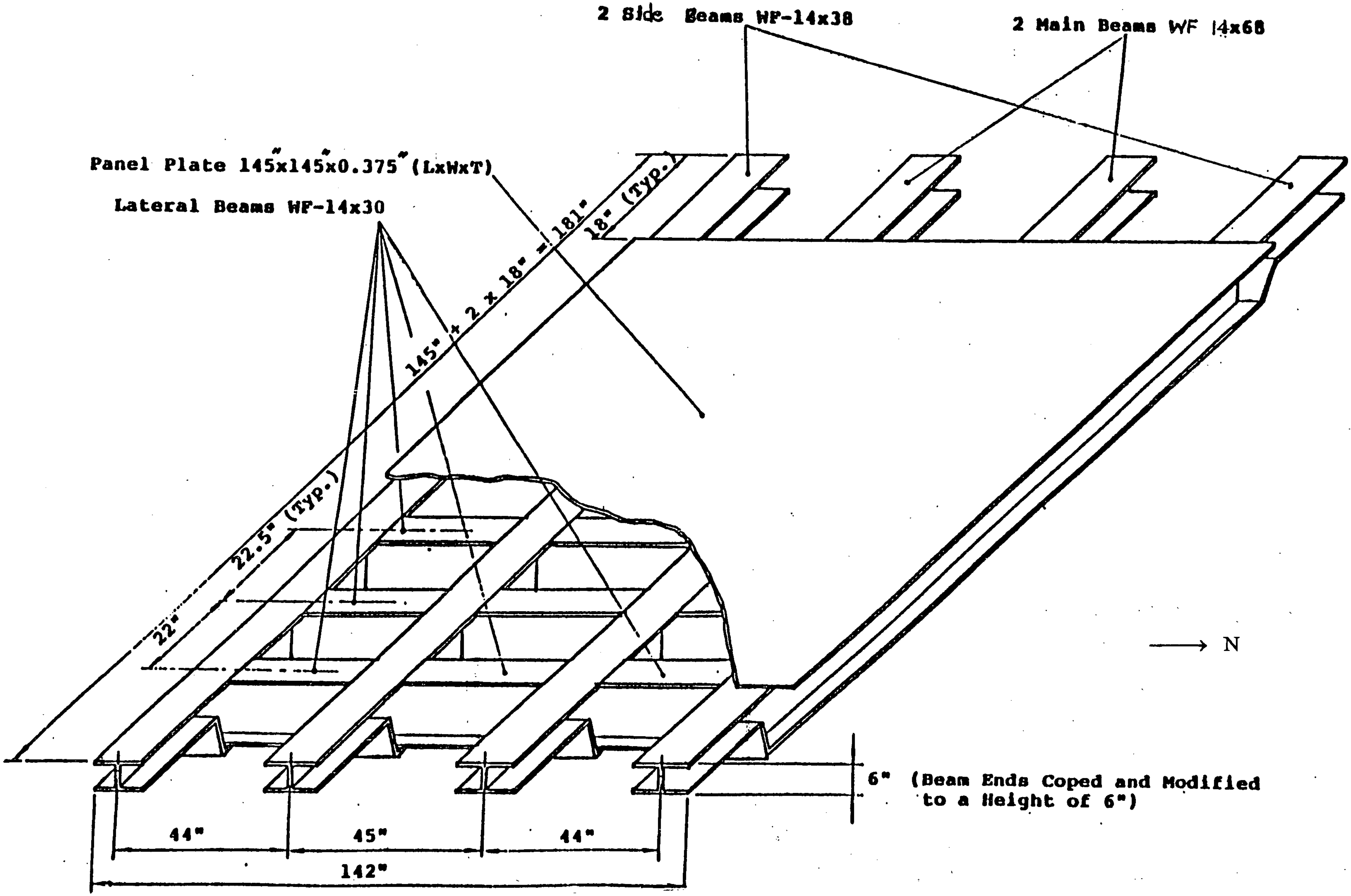


Figure 2.4.1 Cask Pit Impact Shield

CHAPTER 3

CONSTRUCTION OF RACK MODULE

3.0 CONSTRUCTION OF RACK MODULES

The objective of this section is to provide a description of the rack module construction for both the PaR flux trap racks and the Holtec burnup credit modules. The PaR racks have been used previously at TVA's SQN Plant. The spent fuel storage system presently in service at SQN incorporates a high density cell design identical to the Holtec design now being proposed to meet a portion of WBN 's storage capacity needs. Also included in this chapter is a description of the attachments which will connect the smaller Holtec burnup credit "baby" racks to the peripheral PaR modules.

3.1 PaR FLUX TRAP RACK MODULES

3.1.1 Summary Description

An isometric drawing of one of the PaR flux trap racks being transferred from SQN is shown in Figure 3.1.1. No changes are being made to the cell design for any of these 24 PaR modules. The racks performed successfully at SQN for about 13 years with no deficiencies noted. The racks were replaced at SQN to gain a 51 percent increase in storage capability for that two unit plant which resulted in a projected 8 years of additional in-pool storage before encroaching on full core offload reserve capacity. The PaR modules will be installed at WBN in a free-standing mode on bearing pads which rest directly on the pool liner. Seismic and structural analyses reported in subsequent sections of this report validate this method of installation rather than reuse of the seismic grid support structure which was utilized with the racks at SQN.

The PaR flux trap racks are of stainless steel welded construction and are designed to store fuel safely on a 10.375 inch center-to-center spacing. The racks consist of four basic components:

- Top grid casting
- Bottom grid casting
- Neutron absorber canister (poison cans)
- Adjustable foot assembly

The top and bottom grids, Type CF-3M stainless steel, locate and support the poison cans and fuel assemblies. The grid castings have pockets cast in every cavity opening into which the inner can of the neutron absorber canisters are welded. The neutron absorber canister (or poison can) consists of two concentric Type 304 stainless steel tubes with the Boral neutron absorber plates located in the annular gap. The outer can of the poison canister is folded onto the inner tube at the ends and is totally seal welded to isolate the neutron poison from the pool water. The inner tubes of the poison canisters act as structural elements between the top and bottom grids since these tubes are welded onto them. Large leveling screws, Type 17-4 PH H-1100 stainless steel, are located at the bottom grid corners to adjust for variations in pool floor level. Some of these features are illustrated in Figures 3.1.2 and 3.1.3.

3.1.2 Component Specifics

TOP CASTING

The cavity opening or envelope dimension is 8.750 inches square and the individual opening center-to center spacing is 10.375 inches. A symmetrical lead-in or guide (≤30°) is provided for each rack cavity with 400AA micro-inches smoothness.

BOTTOM CASTING

Every cavity contains a bottom seat with four 1/2 x 2.00 inch leg bosses on a 6.75 inch square pattern for fuel assembly support.

POISON CAN

The inner and outer canister wall thickness is nominally 0.09 and 0.036 inches, respectively. The nominal inside opening of the inner can is 8.750 inches with a minimum inside envelope dimension of 8.575 inches square. (Nominal fuel assembly cross-section is 8.424 inches square.) The interior surfaces of the poison can which contact the fuel assembly are free of obstructions and smoothed to 125 micro-inches. The Boral neutron poison plate overlaps the top and bottom of the active fuel length by at least an inch and is 147 inches long, 8.625 inches wide, and 0.09 inches thick.

ADJUSTMENT FOOT

The four level adjusting feet on each module are capable of a ±1.00 inch total adjustment.

RACK DESIGN

The nominal module overall dimensions and weight are as follows:

Rack Size	Height	Width (inches)	Width (inches)	Weight (lbs)
7 x 8	177.5 ± 1.00	72.625	83.000	18,200
7 x 9	177.5 ± 1.00	72.625	93.375	20,500

The clear space under the racks for water flow is 4.25 inches minimum.

3.1.3 Welding

The welding procedures, procedure qualifications, and welder qualifications are in accordance with ASME Boiler and Pressure Vessel Code, Section IX, 1977 Edition and AWS D1.1, Revision 2, for low carbon steel welding. Welding procedures include semi-automatic gas metal arc welding, automatic gas tungsten arc welding, manual gas tungsten arc welding, shielded metal arc welding, and manual shielded metal arc welding.

3.1.4 Applicable References for the WBN Flux Trap Racks

(1) United States Nuclear Regulatory Commission (USNRC)

a.	NRC Reg. Guide 1.13	Spent Fuel Storage Facility Design Basis, Revision 1, December. 1975
b.	NRC Reg. Guide 1.29	Seismic Design Class, Revision 2, February 1976
c.	NRC Reg. Guide 1.92	Combination of Modes in Seismic Analysis, Revision 1, February 1976
d.	NRC Reg. Guide 1.70	"Validation of Calculational Methods for Nuclear Criticality Safety"
e.	NRC SRP 3.8.4	Seismic Category I Structures, 1975
f.	NRC SRP 9.1.2	Spent Fuel Storage Review Responsibility, 1975
g.	NRC SRP 9.2.5	Ultimate Heat Sink, Pages 9.2.5-8 through 9.2.5-14, 1975

(2) Industry Codes and Standards

a.	ASME	ASME Boiler and Pressure Vessel Code, Section IX, 1977 Edition Boiler & Pressure Vessel Code Section III Subsection NA Appendix I, XVII, and subarticle NF-4000, 1974 Edition (American Society of Mechanical Engineers.)
b.	AISC	Steel Construction Manual AISC (7th Edition), June 1973 (American Institute of Steel Construction.)
c.	ACI 318-71	Building Code Requirements for Reinforced Concrete. (American Concrete Institute.)
d.	AA	"Aluminum Standards and Data" published by Aluminum Association 5th Edition, Jan. 1976 Aluminum Association.
e.	ASTM	ASTM Standards: A240-72b, A276-71, A312-72a, B209-73, B26-74, B211-74.
f.	ANSI N45.2	Quality Assurance Requirements of Nuclear Power Plants, 1971

g.	ANSI N45.2.2	Packaging and Shipping, Receiving Storage and Handling of Items for Nuclear Power Plants, 1972
h.	ANSI N16.9	Validation of Calculation Methods for Nuclear Criticality Safety, 1975
i.	ANSI N210	Design Objectives for Light Water Reactors Spent Fuel Storage Facilities at Nuclear Power Stations, 1976
j.	ANSI N45.2.10	Quality Assurance Terms and Definitions, 1973
k.	ANSI N18.2	Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactors Plants, 1973
l.	AWS	Specification D1.1, Revision 2-77, Structural Welding Code

(3) Federal Specifications

10 CFR 50, Appendix B	Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.
10 CFR 73.55	Requirements for physical protection of licensing activities in nuclear power reactors against industrial sabotage.
10 CFR 20	Standards for protection against radiation.

(4) Computer Programs

SAP IV	Computer Program, Static and Dynamic Analysis of Linear Structures
ANSYS	Computer Program "Engineering Analysis System," Swanson Analysis Systems, Inc.
SAGS	Static Analysis of General Structures, SDRC Version III, March 1977
DAGS	Dynamic Analysis of General Structures, SDRC Version III, May 1977
CHEETAH-CORC-PDQ-07	Computer programs for criticality analysis
KENO-IV-AMPX-(NITAWL-X5DRNPM)	Computer programs for criticality analysis

HPOOL	Computer program for spent fuel pool cooling water flow and heat transfer under normal conditions
BPOOL	Computer program for spent fuel pool cooling water flow and heat transfer under boiling conditions

3.2 HOLTEC BURNUP CREDIT RACK MODULES

The following information is offered to provide an understanding of the construction design adequacy of the 10 peripheral burnup credit rack modules as well as the 15 x 15 rack for the cask loading area. The design of the storage cells in these racks are essentially identical.

3.2.1 Fabrication Objective

The requirements for the high density, burnup credit storage racks for the WBN fuel pool may be stated in four interrelated points:

(1) The rack module will be fabricated in such a manner that there is no weld splatter on the storage cell surfaces which come in contact with the fuel assembly.

(2) The storage locations will be constructed so that redundant flow paths for the coolant are available.

(3) The fabrication process involves operational sequences which permit immediate verification by the inspection staff.

(4) The storage cells are connected to each other by austenitic stainless steel welds which leads to a honeycomb lattice construction. The extent of welding is selected to "harden" the racks from the seismic input motion of the postulated safe shutdown and operating basis earthquakes.

3.2.2 Burnup Credit Storage Modules

Burnup credit rack modules for the WBN spent fuel pool are of the so-called "non-flux trap" type. The design and construction of the 449 burnup credit storage cells are essentially identical with the exception of some cells distributed along the periphery of the modules which are 8 inches shorter than the normal 168 inches in order to avoid transfer canal and cask area gate guide obstructions on the south and west pool walls, respectively. In the non-flux trap modules, a single panel of neutron absorber or poison material is interposed between two fuel assemblies. The poison material utilized is Boral, which does not require lateral support to prevent slumping due to its inherent stiffness. However, accurate dimensional control of the poison location is essential for nuclear criticality and thermal-hydraulic considerations. The design and fabrication approach to realize this objective is presented in the next sub-section.

3.2.3 Anatomy of Rack Modules

As stated earlier, the storage cell locations have a single poison panel between adjacent austenitic stainless steel surfaces. The major components of the rack module are: (a) the storage box subassembly, (b) the baseplate, (c) the thermal neutron absorber material, and (d) support legs. A synopsis of the anatomy of the rack module is provided in the following, which explains the physical arrangement of the major constituent parts of a WBN burnup credit rack module.

(a) Storage Box Subassembly - 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 3.2.1 shows the box.

The minimum weld penetration will be 80% of the box metal gage which is 0.06 inch (16 gage). The boxes are manufactured to 8.75 inch nominal inside square dimension.

As shown in Figure 3.2.1, each box has two lateral 1-inch diameter holes punched near its bottom edge to provide auxiliary flow holes.

A double row of matching flat-faced round dimples are coined into the walls of the square storage boxes. The height of each of these local coined areas is half the thickness of the poison sheet, thus the space provided by the corresponding raised areas on adjacent box walls is the thickness of the poison sheet.

The poison sheets are axially centered on the active fuel region. These sheets are slightly longer than the active fuel length at each end to provide added assurance that there is always sufficient poison material. The sheets are scalloped along the two long edges to provide clearance for the raised, coined areas of the box walls. With the poison installed, the boxes are welded together by fusing them at the coined areas using a proprietary fusion welding processes. This process has been used to fabricate racks for more than 25,000 storage cells. The Boral panels are thus contained axially and laterally by these raised areas. Each interior poison sheet is supported axially at the bottom by a stainless steel strip of the same thickness as the poison sheet, which is welded to the wall of one of the two adjacent boxes. On the outside wall of the racks, the poison is mounted under a thin sheet of stainless steel cladding, and four edges of this stainless cladding are intermittently welded to the box wall. Figures 3.2.2, 3.2.3, and 3.2.4 show an assemblage of boxes in isometric, sectional, and elevation views, respectively.

(b) Baseplate - The baseplate provides a continuous horizontal surface for supporting the fuel assemblies.

The baseplate is attached to the box assemblage by fillet welds. In the center of each storage location, the baseplate has a 5-inch diameter flow hole. The baseplate is 3/4 inches thick to withstand accidental fuel assembly drop loads postulated and discussed in Chapter 7 of this report.

(c) Thermal Neutron Absorber Material - As mentioned in the preceding section, Boral is used as the thermal neutron absorber material.

(d) Support Legs - Adjustable support legs are shown in Figure 3.2.5. The top portion is made of austenitic steel material. The bottom part is made of SA564-630 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. Lateral holes in the support leg provide the requisite coolant flow path.

An elevation cross-section of the rack module shown in Figure 3.2.6 shows two box cells in elevation. The Boral panels and their location are also again indicated in this figure. The Boral panels are vertically positioned such that the entire enriched fuel portion of the fuel assembly is enveloped in the longitudinal direction by the thermal neutron absorber material. It is noted that the top of the boxes are flared prior to welding to provide a smooth lead-in contour for the fuel assembly.

The joint between the composite box arrays and the baseplate is made by single fillet welds which provide a minimum of 7 inches of connectivity between each cell wall and the baseplate surface.

As shown in Figure 3.2.5, the support leg is gusseted to provide an increased section for load transfer between the support legs and the cellular structure of interconnected boxes above the baseplate. Use of the gussets also minimizes heat input induced distortions of the support/baseplate contact region.

3.2.4 Welding Types and Processes

The basic types of welds are Tungsten Inert Gas (GTAW or TIG) and Metal Inert Gas (GMAW or MIG). Both fusion and filler metal added TIG arc welds and MIG welds are used. The welds are either automatic or manual, using electronically controlled welding machines.

Electronically controlled TIG arc fusion butt welding is used to fabricate each storage box from two full length channel sections. A Jetline welding machine is used. It positions and clamps the parts and provides automatic feed and the electronic weld control. Up to 100% weld penetration is achieved with a minimum of 50% being structurally adequate for this type of construction.

Electronically controlled TIG arc local fusion welds are used to fasten storage box to storage box. A special apparatus is used to provide the necessary clamping and spacing for both box-to-box and row-to-row fabrication. This is a proprietary process used to provide the honeycomb rack design which results in the smallest possible cell-to-cell pitch dimension. The welding process is verified by a test procedure implemented at the start of each manufacturing shift and any other time following a production shutdown or new set-up. Test specimens, made from box wall material, are welded by the machine and destructively tested. The destructive test assures that the process settings and welding machine operation achieve complete soundness, size, and penetration of the welds.

Manual welding with electronically controlled welding machines is performed for all other fabrication requirements. This welding is either TIG fusion or filler metal added or MIG arc welding.

3.2.5 Codes, Standards, and Practices for the WBN Burnup Credit Spent Fuel Pool Racks

The fabrication of the burnup credit modules for the WBN spent fuel pool is performed under a strict quality assurance system suitable for manufacturing and complying with the provisions of 10 CFR 50, Appendix B.

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

a. Codes and Standards for Design and Testing

(1) AISC Manual of Steel Construction, 8th Edition, 1980.

(2) ANSI N210-1976, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations."

(3) American Society of Mechanical Engineers (ASME, Boiler and Pressure Vessel Code, Section III, Subsection NF, 1989).

(4) ASNT-TC-1A, June 1980 American Society for Nondestructive Testing (Recommended Practice for Personnel Qualifications).

(5) ASME Section V - Nondestructive Examination.

(6) ASME Section IX - Welding and Brazing Qualifications.

(7) Building Code Requirements for Reinforced Concrete, ACI 318-63/ACI 318-71.

(8) Code Requirements for Nuclear Safety Related Concrete Structures, ACI 349-85 and Commentary ACI 349R-85.

(9) Reinforced Concrete Design for Thermal Effects on Nuclear Power Plant Structures, ACI 349.1R-80.

(10) ACI Detailing Manual - 1980.

(11) ASME NQA-2, Part 2.7, "Quality Assurance Requirements of Computer Software for Nuclear Facility Applications (draft).

(12) ANSI/ASME, Qualification and Duties of Personnel Engaged in ASME Boiler and Pressure Vessel Code Section III, Div. 1, Certifying Activities, N626-3-1977.

b. Material Codes

(1) American Society for Testing and Materials (ASTM) Standards - A-240.

(2) American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section II - Parts A and C, 1989.

c. Welding Codes

ASME Boiler and Pressure Vessel Code, Section IX-Welding and Brazing Qualifications (1986) or later issue accepted by USNRC.

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

(1) ANSI N45.2.2 - Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Power Plants.

(2) ANSI 45.2.1 - Cleaning of Fluid Systems and Associated Components during Construction Phase of Nuclear Power Plants.

(3) ASME Boiler and Pressure Vessel, Section V, Nondestructive Examination, 1983 Edition, including Summer and Winter Addenda, 1983.

(4) ANSI - N16.1-75 Nuclear Criticality Safety Operations with Fissionable Materials Outside Reactors.

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

(6) ANSI - N45.2.11, 1974 Quality Assurance Requirements for the Design of Nuclear Power Plants.

(7) ANSI 14.6-1978, "Special Lifting Devices for Shipping Containers weighing 10,000 lbs. or more for Nuclear Materials."

(8) ANSI N45.2.6, Qualification of Inspection and Testing Personnel.

(9) ANSI N45.2.8, Installation, Inspection.

(10) ANSI N45.2.9, Records.

(11) ANSI N45.2.10, Definitions.

(12) ANSI N45.2.12, QA Audits.

(13) ANSI N45.2.13, Procurement.

(14) ANSI 45.2.23, QA Audit Personnel.

e. Other References

(In the references below, RG is NRC Regulatory Guide)

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

(2) RG 1.123 - (endorses ANSI N45.2.13) Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants.

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

(4) RG 1.25 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility of Boiling and Pressurized Water Reactors.

(5) RG 1.28 - (endorses ANSI N45.2) - Quality Assurance Program Requirements, June 1972.

(6) RG 1.29 - Seismic Design Classification, Revision 3.

(7) RG 1.31 - Control of Ferrite Content in Stainless Steel Weld Metal, Revision 3.

(8) RG 1.38 - (endorses ANSI N45.2.2) Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants, March 1973.

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

(10) RG 1.58 - (endorses ANSI N45.2.2) Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel, Revision. 1, September 1980.

(11) RG 1.64 - (endorses ANSI N45.2.11) Quality Assurance Requirements for the Design of Nuclear Power Plants, October 1973.

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

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

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

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

(16) RG 1.92 - Combining Modal Responses and Spatial Components in Seismic Response Analysis.

(17) RG 3.41 - Validation of Calculation Methods Nuclear Criticality Safety.

(18) General Design Criteria for Nuclear Power Plants, Code of Federal Regulations, Title 10, Part 50, Appendix A (GDC Nos. 1, 2, 61, 62, and 63).

(19) NUREG-0800, Standard Review Plan, Sections 3.2.1, 3.2.2, 3.7.1, 3.7.2, 3.7.3, 3.8.4.

(20) "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. (Note: OT stands for Office of Technology).

(21) NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

(22) Regulatory Guide 8.8, "Information Relative to Ensuring that Occupational Radiation Exposure at Nuclear Power Plants will be as Low as Reasonably Achievable (ALARA)."

(23) 10 CFR 50 Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.

(24) 10 CFR 21 - Reporting of Defects and Non-Compliance.

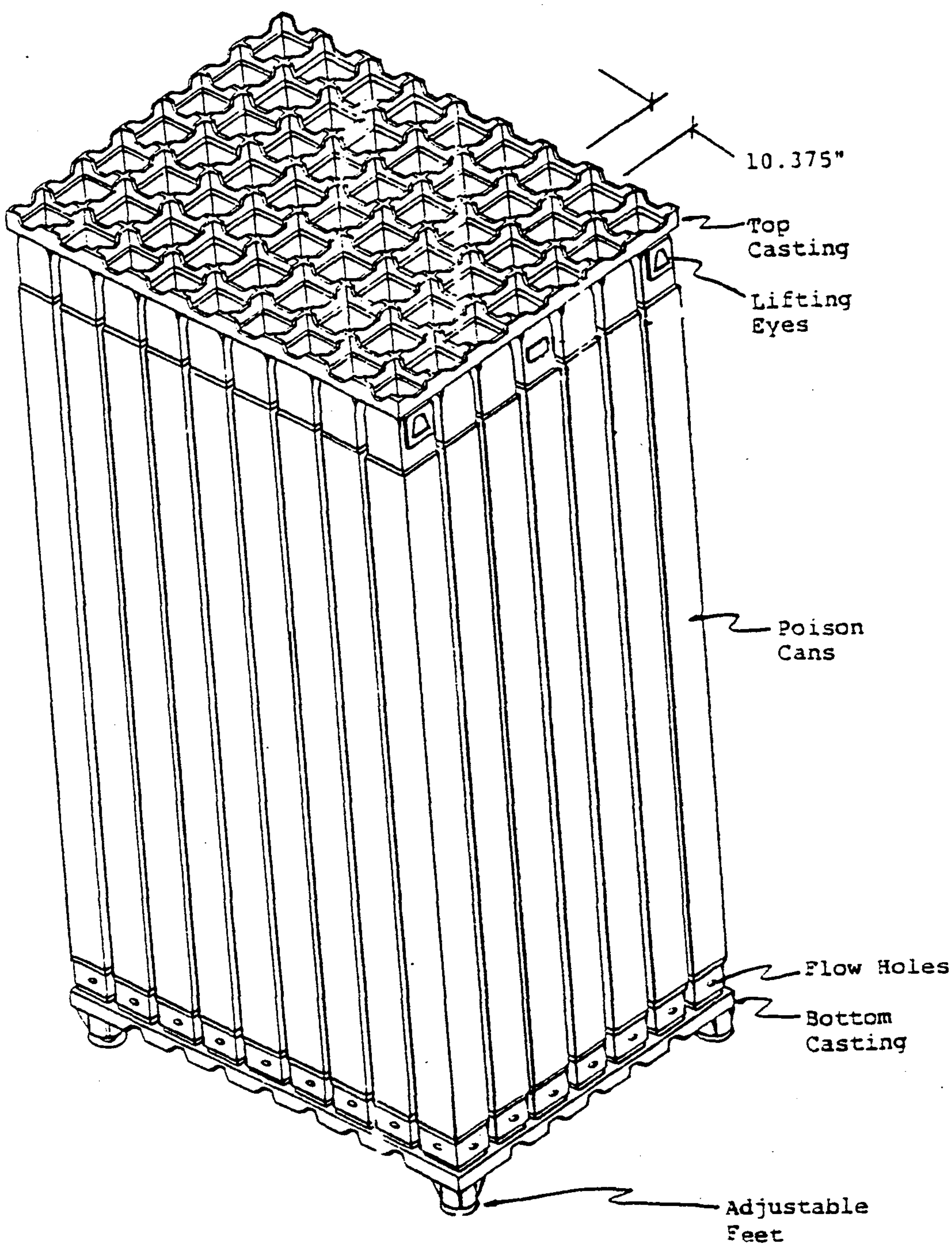
3.2.6 Materials of Construction

Storage Cell:	ASME SA240-304L
Baseplate:	ASME SA240-304L
Support Leg (female)	ASME SA240-304L
Support Leg (male)	Ferritic stainless steel (anti-galling material) ASME SA564-630
Poison:	Boral

3.3 ATTACHMENTS FOR "BABY" RACKS TO PaR RACKS

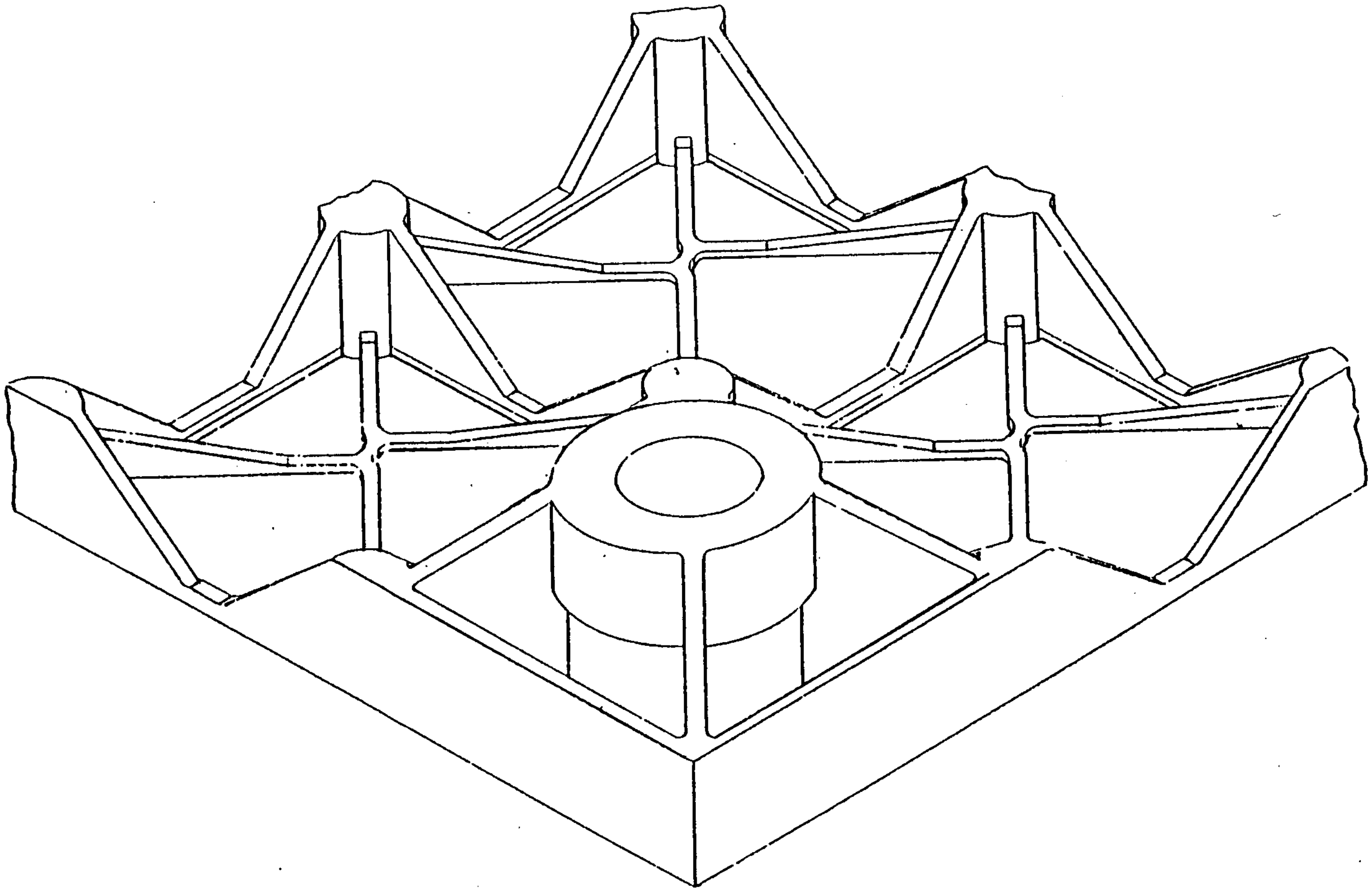
A method has been developed for future remote attachment of the peripheral, slender "baby" racks to the larger, free-standing PaR burnup credit rack modules. This will not require grinding, welding, or drilling to be performed on the PaR racks. The attachments consist of two bottom support plates that thread onto the male spindle of each PaR rack at its two exterior support locations along the south and west sides of the PaR module array. This plate also has through-holes that receive the support feet of the "baby" rack, allowing them to pass through the hole and come to rest on the bearing pads. These plates serve as a lateral restraint for these smaller modules, physically tying them to the large PaR racks. At the top of each "baby" rack, are two restraint hooks which latch into an upper support frame that rests within the PaR rack. The upper support frame consists of channel fittings to receive the hooks. The channel fittings are fastened within the PaR racks by means of threaded tie bars, end plates, and hex nuts. SA-240-304 stainless steel materials are used. Figures 3.3.1 and 3.3.2 illustrates some of the features of the "baby" rack attachments

The design fully utilizes the adjacent PaR racks as seismic restraints, and as discussed in Chapter 6 of this report, the seismic analyses performed address configurations both with and without the "baby" racks. For the case where a slender rack is fastened to a free-standing rack, there is no fluid coupling interaction between the baby and the mother rack. In such cases, including the baby rack in the seismic simulation adds to the inertia load burden without providing a concomitant hydrodynamic benefit. Therefore, the results of the three-dimensional, whole pool multi-rack analyses which included the "baby" racks bound the results for the scenario where the "baby" racks have not yet been installed.



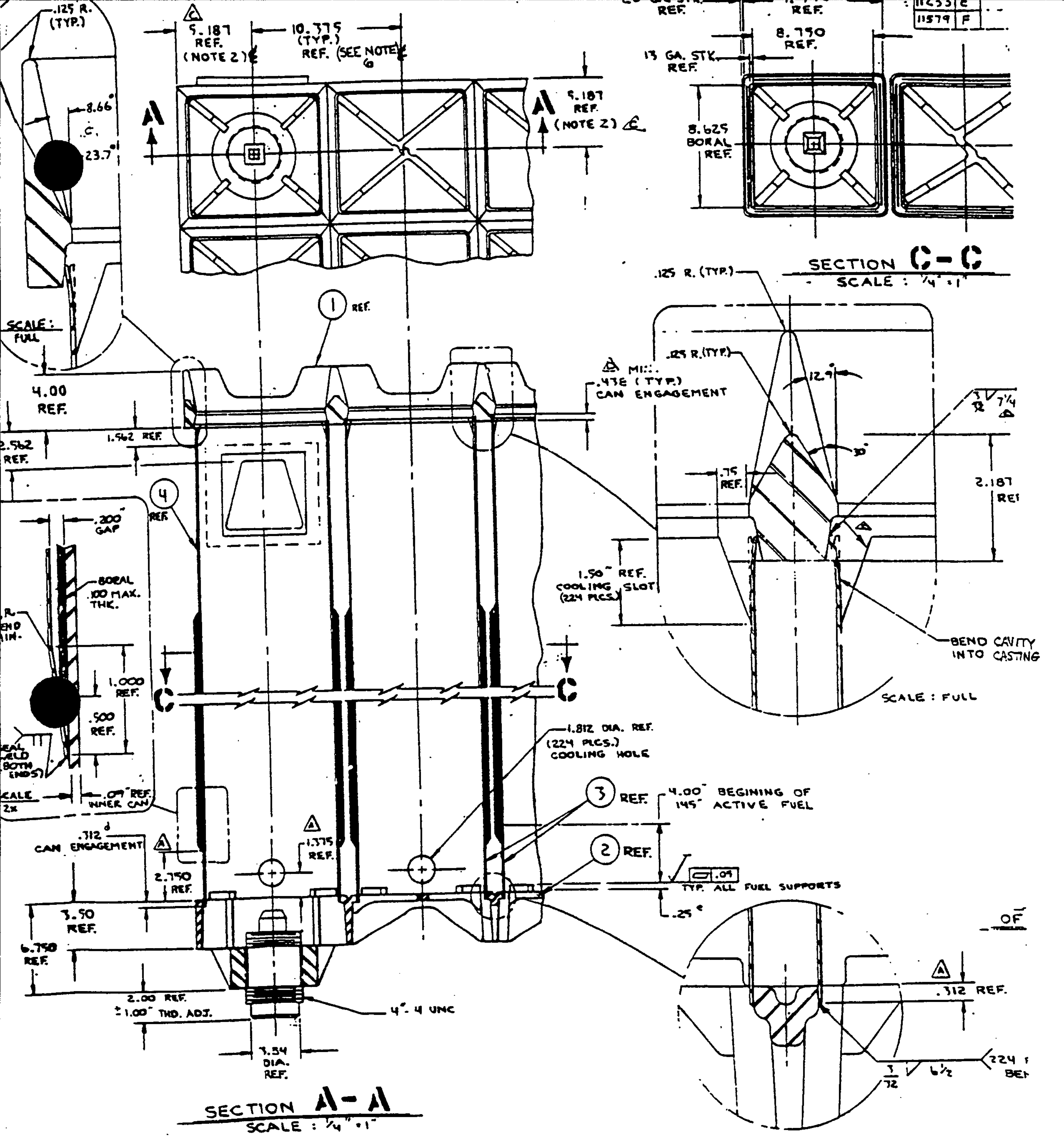
Typical Flux Trap
Fuel Rack Module

Figure 3.1.1



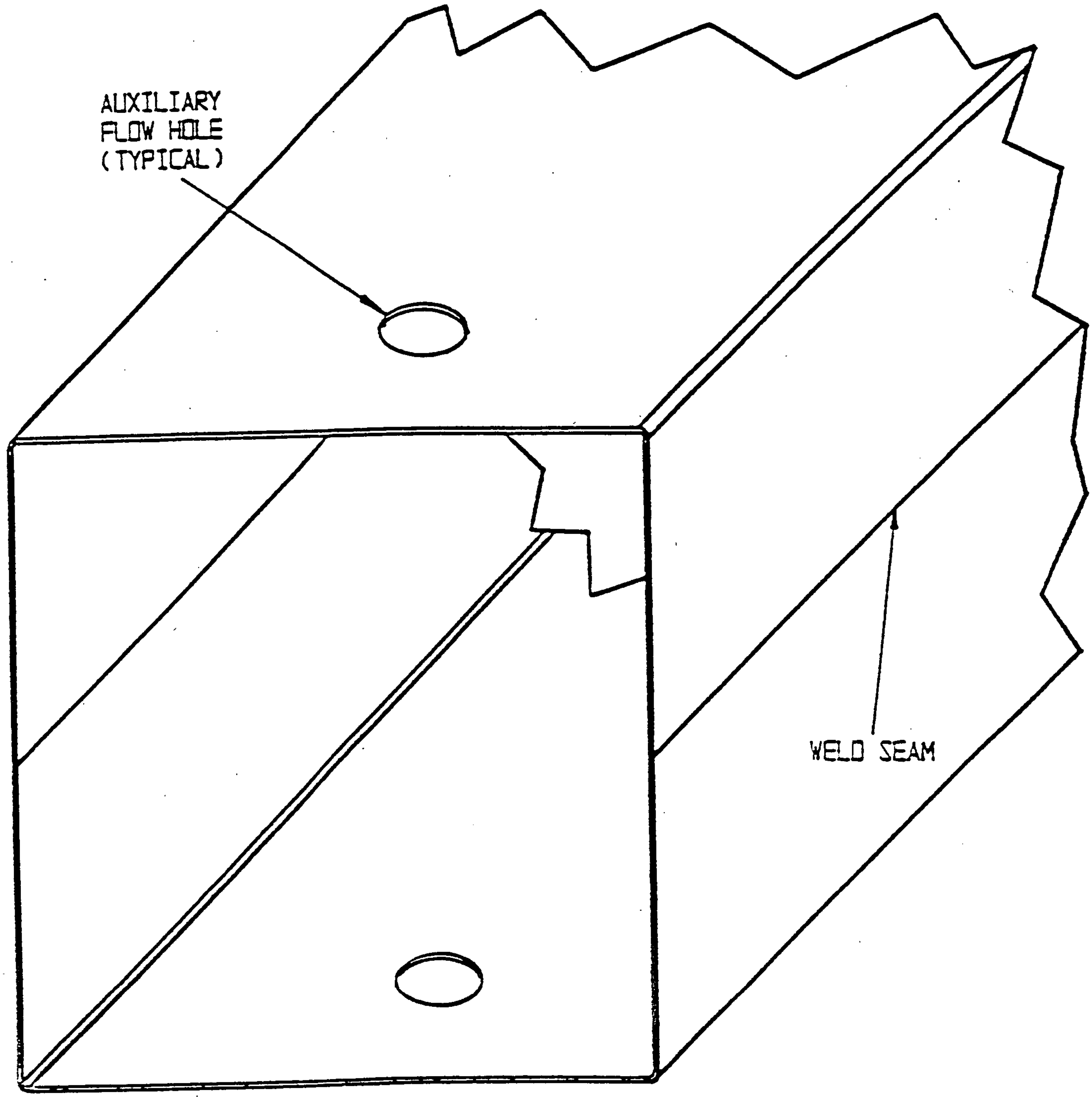
Pictorial View of Bottom
Casting Foot

Figure 3.1.2



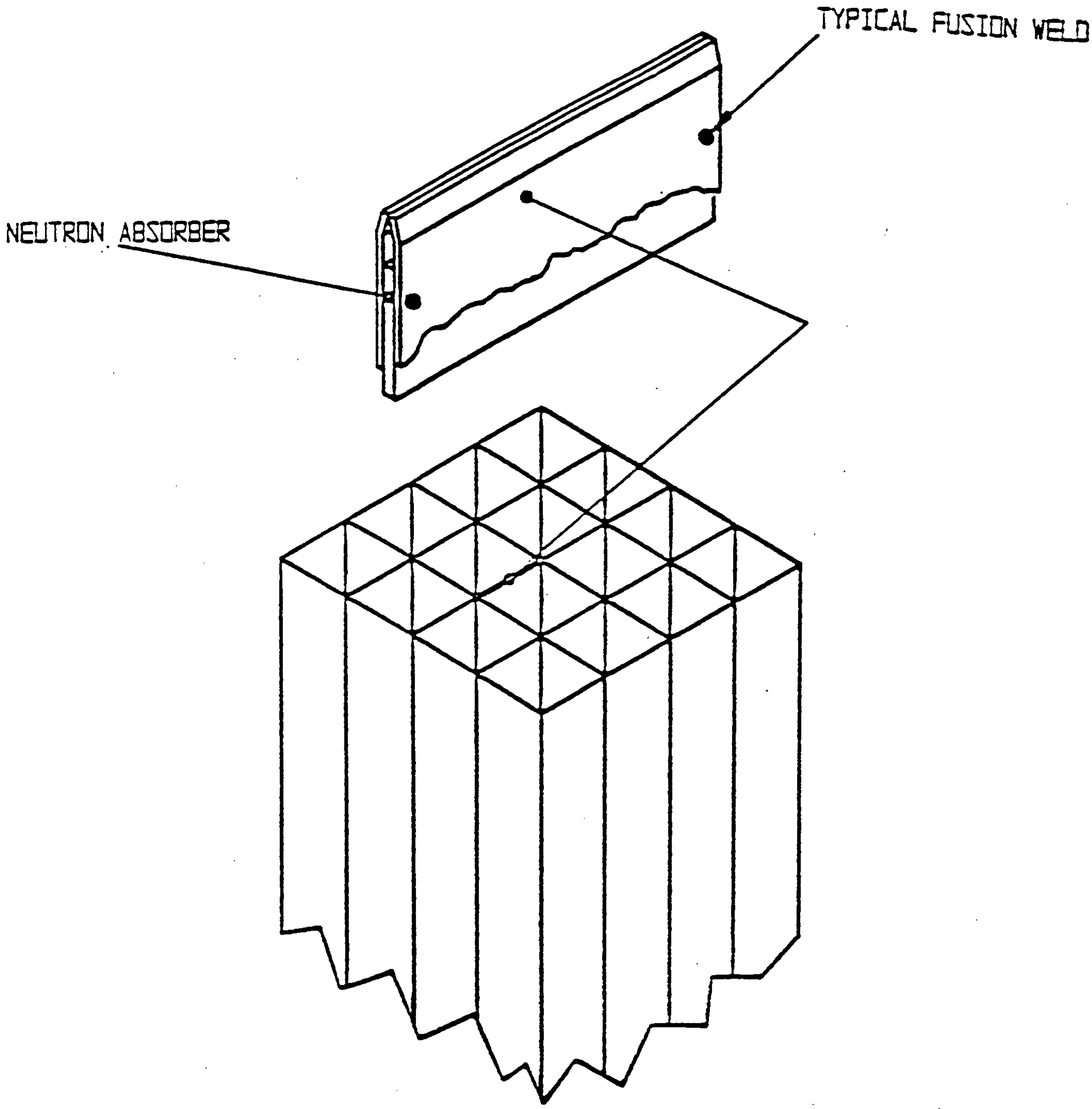
Sectional of Canister Walls
and Adjustable Foot

Figure 3.1.3



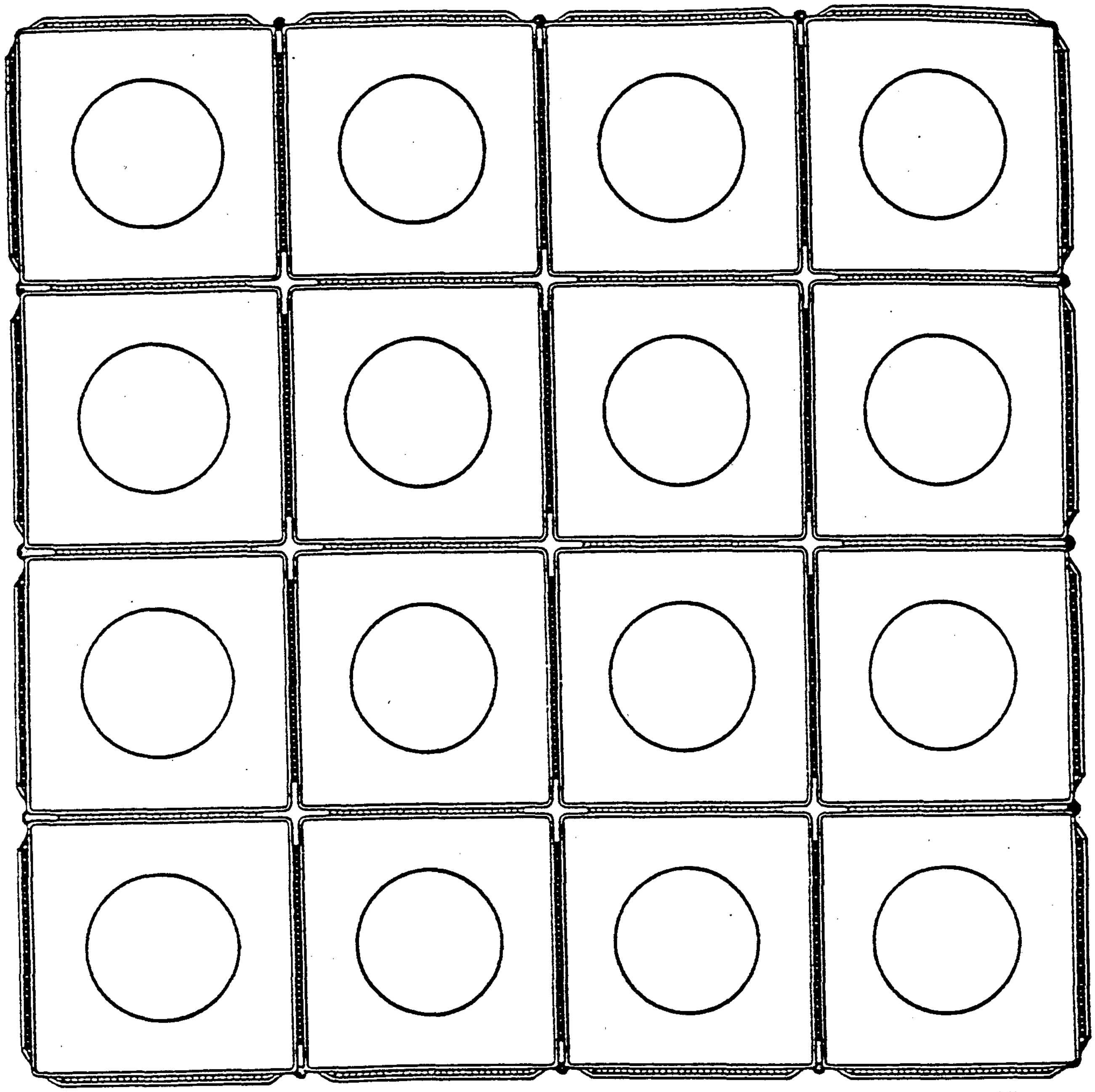
Seam Welded Box For Burnup Credit Cell

Figure 3.2.1



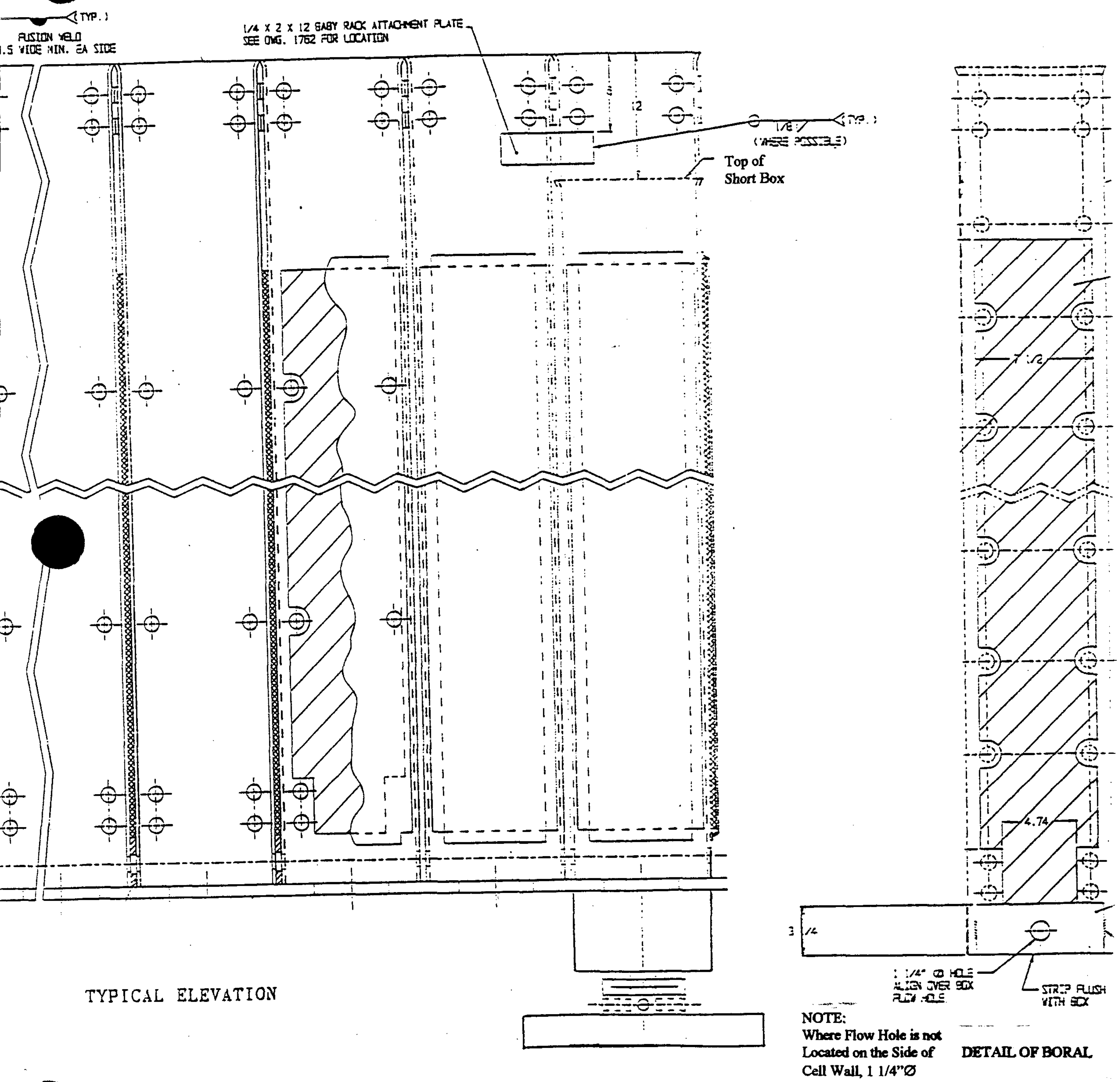
Isometric View of 4 X 4 Box Array

Figure 3.2.2



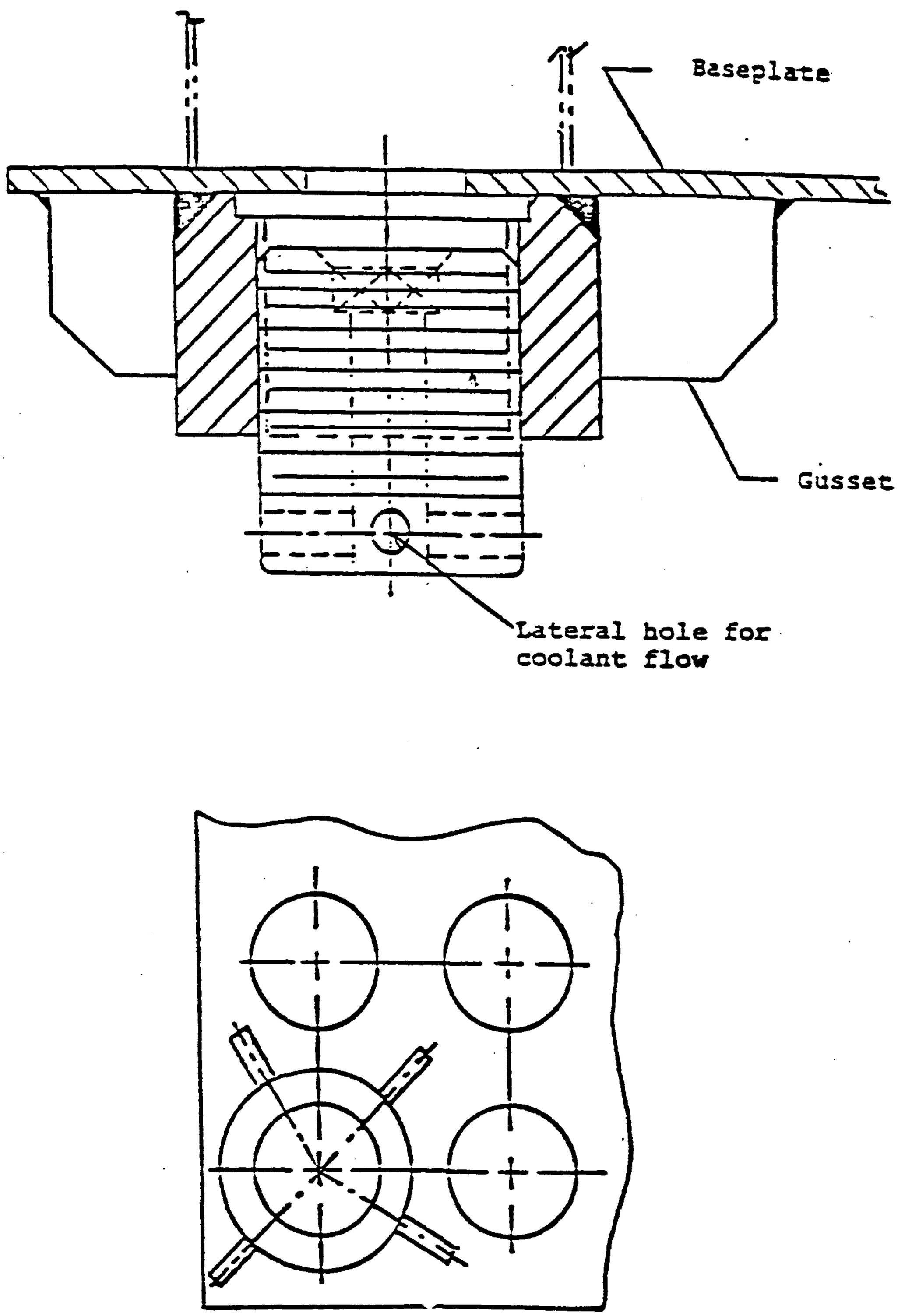
Cross Section of 4 X 4 Box Array

Figure 3.2.3



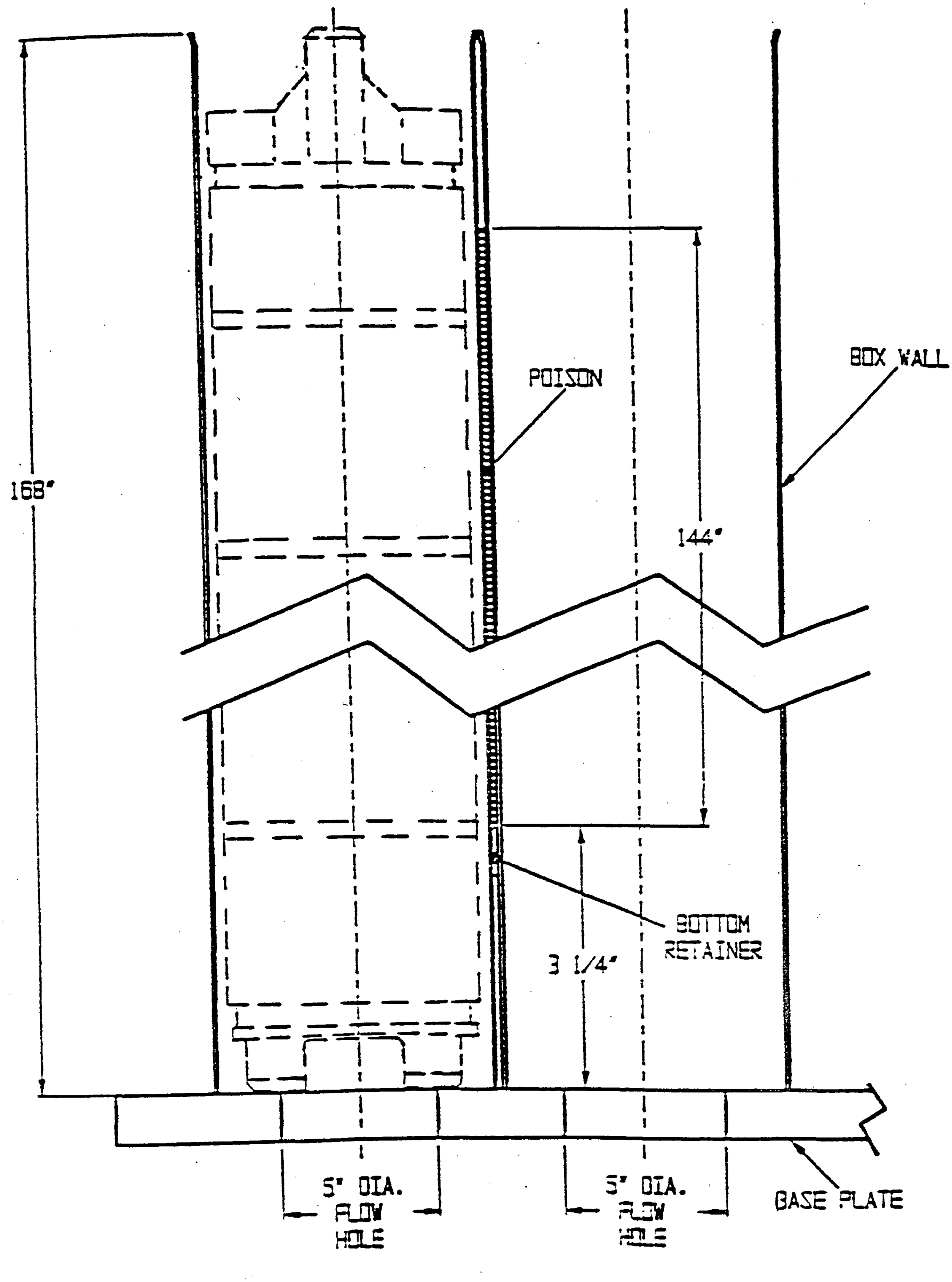
Elevation View of Box Array and Boral

Figure 3.2.4



Adjustable Support Leg

Figure 3.2.5



Elevation View of Two Contiguous Storage Cells

Figure 3.2.6

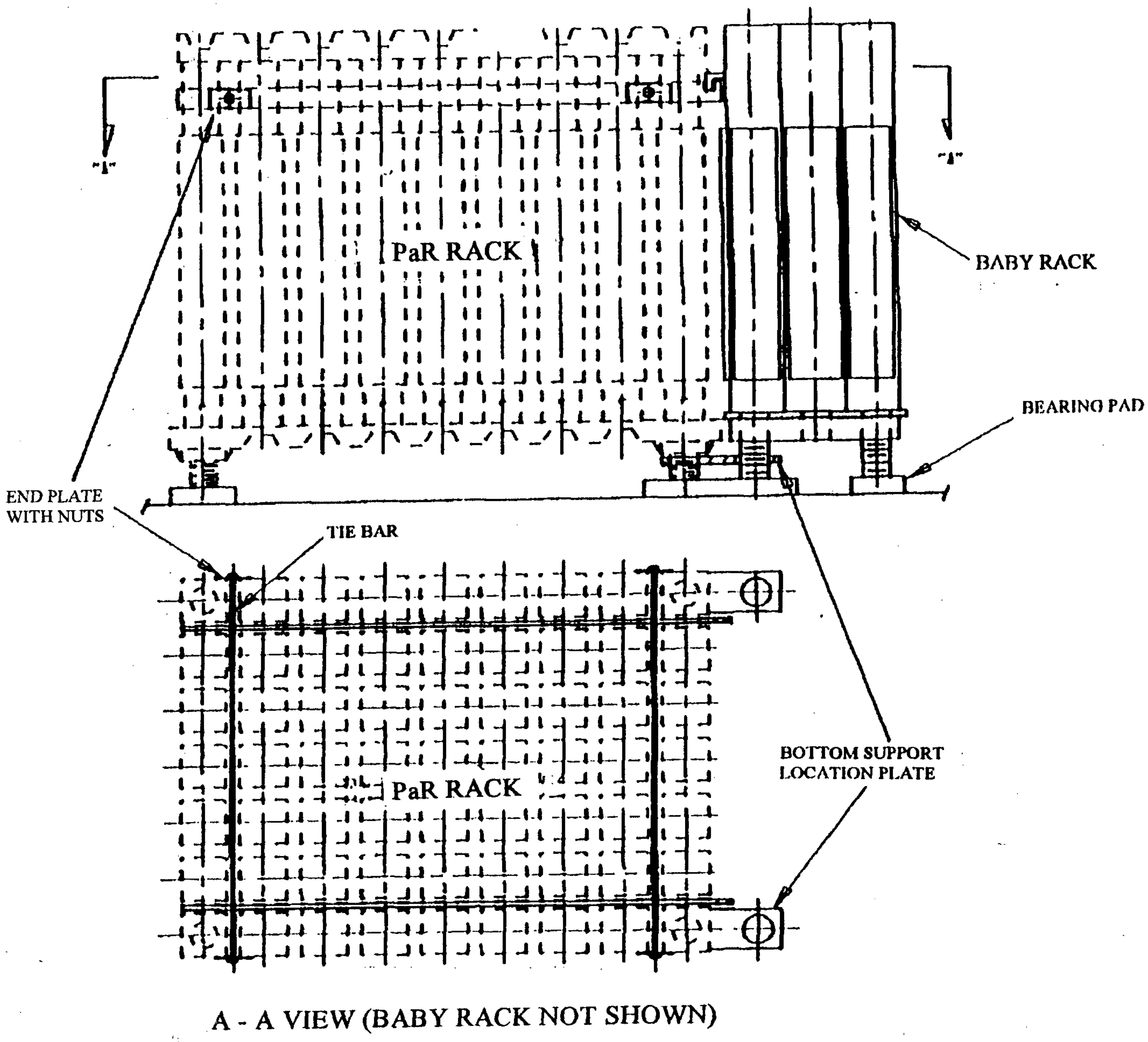
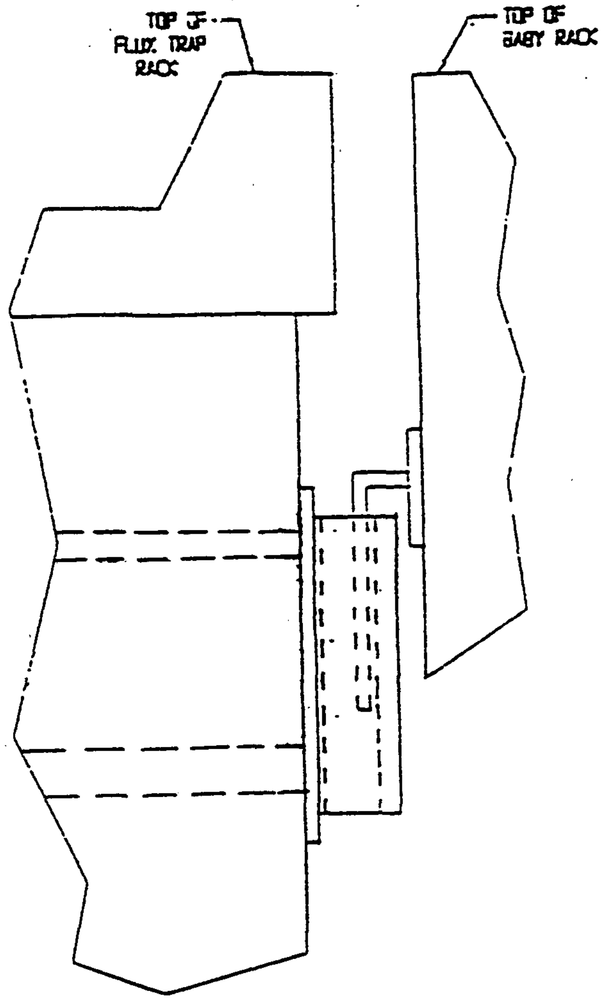
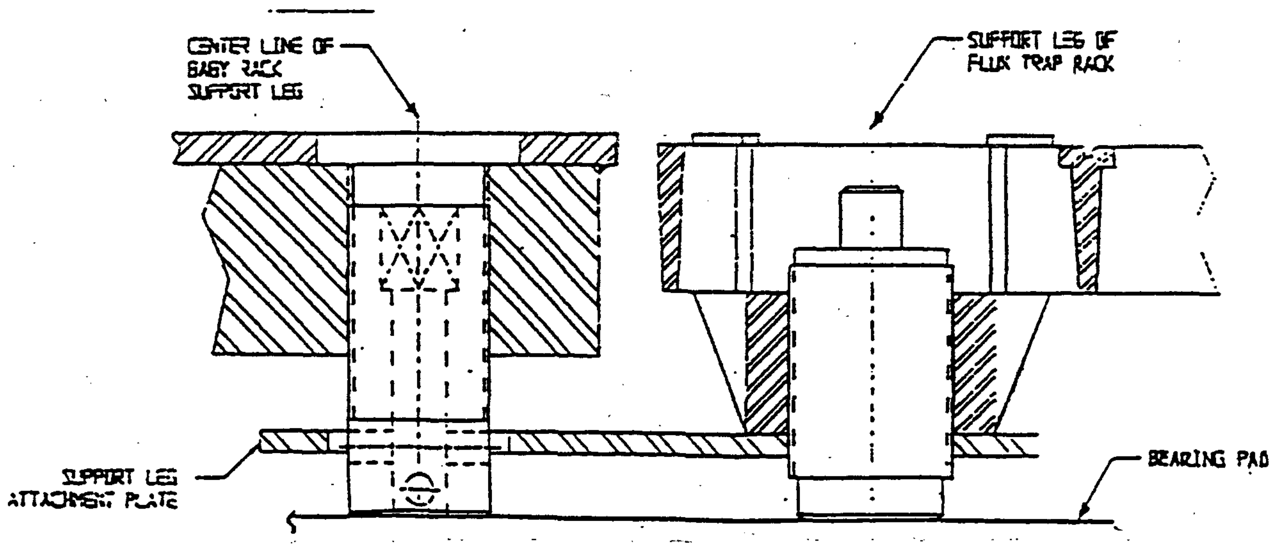


Figure 3.3.1 Attachment For Baby Rack



TOP ATTACHMENT OF BABY RACK WITH FLUX TRAP RACK



BOTTOM ATTACHMENT OF BABY RACK WITH FLUX TRAP RACK (TYPICAL)

Figure 3.3.2 Baby Rack Attachment Features

CHAPTER 4

CRITICALITY SAFETY ANALYSES

4.0 CRITICALITY SAFETY ANALYSIS

4.1 DESCRIPTION OF ANALYSIS

4.1.1 Design Basis

The high density spent fuel storage racks for WBN are designed to assure that the effective neutron multiplication factor (K_{eff}) is equal to or less than 0.95. Design calculations model the racks fully loaded with fuel of the highest anticipated reactivity, and flooded with unborated water at the temperature within the operating range corresponding to the highest reactivity. The maximum calculated reactivity includes a margin for uncertainty in reactivity calculations including mechanical tolerances. Uncertainties are statistically combined, such that the final K_{eff} will be equal to or less than 0.95 with a 95% probability at a 95% confidence level.

Applicable codes, standards, and regulations or pertinent sections thereof include the following:

- General Design Criteria 62, Prevention of Criticality in Fuel Storage and Handling.
- USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, Spent Fuel Storage, Revision 3 - July 1981.
- USNRC letter of April 14, 1978, to all Power Reactor Licensees - OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, including modification letter dated January 18, 1979.
- USNRC Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis, Revision 2 (For Comment), December 1981.
- ANSI ANS-8.17-1984, Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.

The water in the spent fuel storage pool normally contains soluble boron which would result in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting K_{eff} of 0.95 for normal storage be evaluated in the absence of soluble boron. The double contingency principle of ANSI N-16.1-1975 and of the April 1978 NRC letter allows credit for soluble boron under other abnormal or accident conditions since only a single independent accident need be considered at one time. Consequences of abnormal and accident conditions have also been evaluated, where "abnormal" refers to conditions which may reasonably be expected to occur during the lifetime of the plant and "accident" refers to conditions which are not expected to occur nevertheless must be protected against.

4.1.2 Background and Introduction

The PaR designed spent fuel storage racks previously in service at TVA's SQN are being transferred to TVA's WBN. These spent fuel racks were licensed at SQN for storage of fresh fuel with up to 5.0 (4.95 ± .05) weight percent (wt%) U-235 enrichment using administrative controls and burnup credit. The PaR-designed ("flux trap") racks (designated Region 1) will be installed at WBN with a design modification allowing future installation of a peripheral or "baby" rack (designated as Region 2) on two sides of the PaR rack configuration. In addition, a 15x15 spent fuel storage rack module has been designed for later placement in the cask loading pit. The peripheral (Region 2) and cask pit racks were designed by Holtec International and have the same cell design as the Holtec racks recently licensed for SQN. The layout of storage cells in the WBN spent fuel pool is shown in Figure 4.1.1. This report describes the criticality analysis of the WBN spent fuel pool configuration which assures that the maximum K_{eff} will be less than or equal to 0.95 with fuel up to 4.95 ± .05 wt% U-235 enrichment.

Analysis of the WBN spent fuel rack configuration[1,2] was performed using the SCALE[3,4,5] system of codes for cross section generation and reactivity calculations, and CASMO[6,7,8,9] was used for depletion calculations. The design basis fuel is a 17x17 Westinghouse VANTAGE-5H[10] assembly containing a maximum initial enrichment of 4.95 ± .05 wt% U-235. Region 2 and cask pit rack calculations were performed with a moderator temperature of 4°C. Originally, the calculations for Region 1 were performed with a moderator temperature of 68°F (20°C); and later a temperature correction was calculated allowing update of results to 4°C moderator temperature.

Interface of the PaR and Holtec rack designs, as well as effects of the configuration in the WBN spent fuel pool, were evaluated to assure reactivity impacts were assessed. Margin for uncertainty in the reactivity calculations and manufacturing tolerances were included such that the final K_{eff} for allowed storage configurations will be less than or equal to 0.95 with a 95% probability at a 95% confidence level.

In order to store fuel with U-235 enrichment as high as 4.95 ± .05 wt%, administrative controls and burnup credit must be applied. Therefore, the analysis takes credit for the reactivity decrease due to burnup of the stored fuel and for administrative controls on fuel placement. Burnup in discharged fuel was treated by an "equivalent enrichment"[11] technique described in Section 4.1.3. Two separate storage regions are provided in the spent fuel storage pool, with independent criteria defining reactivity restrictions in each of the two regions. Both regions are designed to accommodate fuel with a maximum enrichment of 4.95 ± .05 wt% U-235 with burnup credit and administrative controls. Restrictions in the two regions are:

- Fuel with enrichment as high as 3.8 wt% U-235 may be stored in Region 1 without restrictions. Storage of fuel assemblies with enrichment between 3.8 and 5.0 wt% U-235 requires either fuel burnup of at least 6.75 MWD/KgU or placement in storage locations which have face adjacent storage cells containing either water or fuel assemblies with accumulated burnup of at least 20.0 MWD/KgU.

- Storage in Region 2 is limited to fuel of 4.95 ± .05 wt% U-235 initial enrichment burned to at least 41 MWD/KgU or fuel of lower enrichments with accumulated burnup which yields an equivalent reactivity in the fuel racks.

In addition, fuel may be stored in a cask pit rack shown in Figure 4.1.1 provided one of the following conditions is maintained:

- Fuel with an initial enrichment of 4.95 ± .05 wt% U-235 must have accumulated burnup of at least 41 MWD/KgU, or fuel of lower enrichments must have an accumulated burnup which yields an equivalent reactivity in the fuel racks.

- Fresh fuel with up to 4.95 ± .05 wt% U-235 initial enrichment must be placed in storage locations with empty face adjacent storage cells (i.e., checkerboarded with empty cells).

Sections 4.2 - 4.6, provide descriptions of the methods and results which define the bases for safe storage of fuel in the WBN high density storage racks. The Methods sections describe computer codes, benchmarking, application requirements, storage conditions modeled (fuel and rack cells), as well as the analytical models used. These descriptions provide the bases for the analytical approach to the WBN specific criticality analysis as well as the method bias and uncertainties.

The Analysis Results sections provide results of the analysis along with specific studies and calculations performed to define reactivity effects and associated uncertainties and conservatism. Studies and calculations include sensitivity studies, evaluation of reactivity effects such as enrichment and burnup, postulated accidents which may impact reactivity conditions during storage and conservatism included in the analytical models. Summary and Conclusions provide a rollup of individual region analyses (including effects of Regions 1 and 2 interface) and the resultant conditions required for conservative storage.

4.1.3 Equivalent Enrichment Concept

Reactivity in a fuel assembly decreases as burnup accumulates due to the depletion of U-235 and the creation of fission product neutron poisons. Figure 4.1.2 displays the reactivity decrease with burnup of an array of Westinghouse 17x17 VANTAGE 5H fuel assemblies with an initial enrichment of 5.0 wt% in the WBN spent fuel racks.

Reactivity in a fuel assembly increases as the initial enrichment increases. Figure 4.1.3 displays the reactivity increase with enrichment at zero burnup for the same configuration described in the previous paragraph. Therefore, there exists an enrichment at zero burnup that has reactivity equivalent to a higher enrichment fuel assembly with accumulated burnup.

KENO does not have the capability to deplete fuel assemblies. In order to model burnup, CASMO-3 was used to determine the equivalent enrichment at zero burnup of a 5.0 wt% fuel assembly with accumulated burnup. KENO was then used to model a radially infinite array of high enriched fresh fuel assemblies loaded in a checkerboard configuration with the low-reactivity equivalent-enrichment fuel assemblies determined from the CASMO-3 analysis.

4.1.4 Design Basis Fuel

The VANTAGE 5H fuel design[10] was modeled as the design basis fuel for this analysis. The VANTAGE 5H design contains a smaller guide tube outer diameter and thus slightly increased neutron moderation compared with the Westinghouse Standard 17x17 fuel assembly. In addition, VANTAGE 5H fuel assemblies have zircaloy spacer grids as opposed to the more neutron-absorbing material inconel found on the Standard 17x17 fuel assembly. As a result of these differences, VANTAGE 5H fuel has a higher reactivity for a given enrichment than Standard fuel. Therefore, analysis of VANTAGE 5H fuel also covers storage of Standard 17x17 fuel. VANTAGE 5H fuel assembly data is provided in Table 4.1.1. This analysis bounds the design basis fuel assembly using the data provided in Table 4.1.1 or a more conservative value depending on the specific calculation.

4.2 METHODS: REGION 1 ANALYSIS

The criticality analysis for the WBN racks is a combination of TVA Region 1 calculations (verified by Holtec International) and Holtec analyses for Region 2, cask pit and Regions 1 and 2 interface. The Region 1 calculations performed by TVA uses CASMO-3, a two-dimensional integral transport theory code for burnup calculations, and KENO-5, a three-dimensional Monte Carlo transport theory code for reactivity calculations. The CASMO-3 cross sections are based on data from the ENDF/B-IV cross section library with microscopic cross sections tabulated in 40 energy groups. The KENO-5 analyses use a 27 energy group library also based on ENDF/B-IV cross sections and the NITAWL subroutine for U-238 resonance shielding effects.

4.2.1 Benchmarking

In order to verify the method applicability to criticality analyses, a series of calculations were performed using each of the methods. The results of these calculations established the biases and uncertainties associated with the computer codes. Application of the SCALE methodology by TVA and Holtec including models and benchmark cases vary slightly and thus result in small differences in biases and uncertainties. Both Holtec and TVA biases and uncertainties have been reviewed and determined acceptable in previous NRC approved criticality analyses[2,28]. Results of Holtec benchmark calculations are described in Section 4.4.1 and the TVA results are described below:

KENO-5

To provide a benchmark for the SCALE calculation method that is used to determine the final Region 1 reactivity, a set of 35 critical experiments was modeled to determine the methodology and cross-section bias and uncertainty in predicting K_{eff}. The criticals are representative of the WBN spent fuel pool configuration, H/U ratios, enrichments, and spent fuel storage pool materials as indicated below:

- The critical experiments described in BAW-1484[12] provide data with cylindrical and rectangular configurations, varied assembly separation distances, and varied placement of the B_4C poison.
- The critical experiments described in PNL-2438[13] use stainless steel, Boral, and Zircaloy-4 for separation. These materials are used in the WBN spent fuel pool.
- The critical experiments described in PNL-2615 [14] use an enrichment of 4.29 wt% U-235 (as compared to 2.35 wt% in PNL-2438 and 2.46 wt% in BAW-1484).

The comparison of the KENO-5 modeling with critical experiments is shown in Table 4.2.1. The criticals resulted in an average K_{eff} of 0.99273 which represents a KENO method bias of 0.00727.

The standard deviation from the average K_{eff} was calculated to be 0.00404. However, when the 'W' test[15] or the assumption of normality was applied, it was determined that the critical K_{eff} predictions are not normally distributed. Therefore, the standard deviation has been increased by a factor of 1.245 to account for an approximately normal distribution.[16] This product was further increased by the 95% probability factor (1.645), resulting in an average uncertainty of 0.00827 at the 95% probability/95% confidence level based on the comparison to the critical experiments. This uncertainty was increased further as described below.

Benchmark results were evaluated for significant trends with respect to enrichment, separation distance, H/U atom ratio, H/U-235 atom ratio, and poison material loading. No significant correlations attributed to parameter changes were observed. As shown in Table 4.2.1, variations in the results are well below the magnitude of the KENO-5 statistical uncertainty. However, this criticality analysis included 5 wt% U-235 enrichment which is greater than the maximum enrichment analyzed for the critical experiments (4.3 wt% U-235). In order to account for this extrapolation, a bias for the enrichment (i.e., H/U-235 atom ratio) change and an uncertainty for extrapolation were determined consistent with the method described in ANSI standard on Nuclear Criticality Safety.[17]

In order to determine the bias for the H/U-235 change, a linear correlation was developed for the K_{eff} versus hydrogen to U-235 atom ratio in the critical benchmark experiments. This correlation was then used to extrapolate to 5.0 wt% U-235. A bias of 0.00318 ΔK_{eff} was determined for enrichment extrapolation and is included in the final K_{eff} evaluation.

The increase in uncertainty due to extrapolation was determined by calculating the confidence interval for the extrapolated linear K_{eff} correlation. The resultant KENO-5 method uncertainty increase due to enrichment extrapolation is 0.00182 ΔK.

CASMO-3

To estimate the uncertainty for the CASMO-3[18,19] calculation method used in the determination of equivalent burnup enrichments, a comparison was made between the CASMO-3 and LATTICE[20] computer codes. LATTICE is a lattice physics computer code developed by TVA and approved for light water reactor fuel assembly analyses by the Nuclear Regulatory Commission.[20]

A PWR fuel assembly model was depleted with both CASMO-3 and LATTICE using hot-full-power conditions. The reactivity was then calculated at cold conditions and selected exposures out to 20 MWD/KgU using the restart features available in both codes.

The maximum difference in the change in reactivity with burnup between LATTICE and CASMO-3 was 0.00374 ΔK_{eff}. The CASMO-3 reactivity change was less and thus more conservative since a lower reactivity change will result in a higher equivalent enrichment. To increase the conservatism, the maximum difference was then doubled.

Therefore, the estimate of the uncertainty in CASMO-3 reactivity due to burnup is 0.00748 ΔK_{eff}. The allowance for uncertainty in burnup calculations is a conservative estimate in view of the substantial reactivity decrease as spent fuel ages (due to Pu-241 decay and Am-241 growth).

4.2.2 Computer Code Application

KENO-5 was used to establish the required equivalent enrichment at zero burnup that maintains $K_{eff} \leq 0.95$ including uncertainties and biases. CASMO-3 was then used to determine the corresponding burnup of 5.0 wt% fuel that has the same reactivity.

KENO-5

A KENO-5 model of the spent fuel racks containing fresh, low-enrichment fuel in a checkerboard configuration with fresh fuel enriched to 5.0 wt% U-235 was analyzed. The enrichment of the low-enrichment fuel was selected such that $K_{eff} \leq 0.95$ when the biases and uncertainties were taken into account.

CASMO-3

KENO-5 was used for reactivity calculations; however, as described in Section 4.1.3, KENO does not have depletion capability. Therefore, CASMO-3 was used for depletion calculations necessary to determine equivalent enrichments by determining reactivity as a function of exposure for fuel enriched to 5.0 wt% of U-235. The steps required are described below.

- An array of fuel assemblies of 5.0 wt% enrichment was depleted using a lower bound on the water density (0.60 gm/cm^3). This maximizes the reactivity addition at fuel storage conditions due to water density history considerations. The maximum number of burnable absorbers were inserted during the depletion, thus maximizing the burnable absorber history reactivity addition when the absorbers are assumed to be removed in the storage racks.

- The reactivity of the depleted fuel assemblies in the spent fuel storage racks was determined at various burnups. The burnable absorbers were removed from the assembly for this calculation. The xenon concentration was set to zero and the remaining isotopes were held constant.

- A series of KENO-5 cases were run at low enrichments (with no burnable absorbers) to calculate the reactivities.

- The reactivity (determined using KENO-5) of the fresh low-enrichment assemblies was used to select the burnup of the 5.0 wt% assembly with the same reactivity (determined using CASMO-3). The final analysis determined that for an assembly enriched to 5.0 wt% with 20.0 MWD/KgU of accumulated burnup, the equivalent fresh enrichment is 2.71 wt%. This enrichment was then used in the KENO-5 case to model fuel with 20 MWD/KgU burnup.

The models utilized in this analysis used nominal dimensions. The uncertainty resulting from dimensional tolerances was addressed using sensitivity studies.

4.2.3 Fuel Assembly

The fuel modeled in this analysis was a fresh Westinghouse VANTAGE 5H assembly without control rods. Burnable poisons were inserted during the CASMO-3 depletion but removed in the spent fuel storage rack calculations, thus accounting for burnable poison history effects which increase reactivity. The uranium in the assembly was conservatively modeled to contain only U-235 and U-238 at beginning-of-life with no fission product buildup. Depleted assemblies were modeled as previously described using the equivalent enrichment concept. CASMO-3 models the depletion of uranium, production of plutonium and creation of fission products.

Zircaloy spacer grid straps were modeled. However, the intermediate flow mixer grids were not modeled. Credit was not taken for other assembly structural material such as the top and bottom nozzles. These have been conservatively modeled as water.

Laterally, the fuel area in both the KENO-5 and CASMO-3 models were divided into a 17x17 array containing the discretely defined fuel rods and guide tubes. The fuel rod cell models the fuel pellet, gap, and clad surrounded by water . The guide tube cell model contains a water center, the guide tube and surrounding water.

CASMO-3 is a two-dimensional model and has no axial definition. The KENO-5 model contains several axial levels due to the modeling of the assembly spacer grids. Conservatively, the total active fuel length has been assumed to be the same enrichment. Natural uranium axial blankets have not been modeled. Therefore, this model is applicable to both the Standard 17x17 and VANTAGE 5H fuel assemblies since the fuel rod dimensions are the same and the zircaloy spacer in the VANTAGE 5H fuel assembly is less a neutron poison than is the inconel spacer in the Standard 17x17 fuel assembly.

4.2.4 Storage Rack

The Region 1 spent fuel storage racks modeled for this analysis are high density spent fuel storage racks manufactured by PaR. The design incorporates the use of the neutron poison material Boral for the purpose of reducing the center-to-center spacing of the storage cells.

The spent fuel storage cell consists of two concentric, square stainless steel tubes, seal welded at the ends. The Boral plate is located in the water-tight void existing between the tubes. These details were explicitly modeled in both KENO-5 and CASMO-3. Axially, the KENO-5 model describes the full Boral plate. The additional rack material has been conservatively modeled as water.

Specific spent fuel storage rack data is provided in Table 4.2.2.

4.2.5 Analytical Models

There are three analytical models used in this analysis. A half-cell model was used for CASMO-3, while KENO-5 used both a single cell and a multi-cell model. A two-dimensional CASMO-3 model was used for the burnup analysis. A single-cell, three-dimensional KENO-5 model was used for sensitivity analyses and analyses without burnup. A three-dimensional multi-cell KENO-5 model was used to determine burnup effects using the equivalent enrichment concept.

CASMO-3 MODEL

The half-cell model used by CASMO-3 consisted of a half fuel assembly and the corresponding rack and water regions in an infinite radial array. The fuel rods and one cell wall have been explicitly modeled. However, the Boral plate was homogenized with the second cell wall due to CASMO-3 code limitations. Because CASMO-3 is a two-dimensional code, the spacer grids are modeled by homogenizing the grid material with the moderator.

KENO-5 SINGLE-CELL MODEL

The KENO-5 single-cell model consists of one fuel assembly, one storage cell, and the water region surrounding the storage cell in an infinite lateral array (using reflective boundary conditions). The resulting configuration contains only one fuel enrichment.

Axially, the single-cell discretely models the fuel assembly spacer grids. Fuel assembly and storage rack material above and below the length of the Boral plate have been conservatively modeled as water.

KENO-5 MULTI-CELL MODEL

The multi-cell model consists of four quarter fuel assemblies which describes fuel of two different enrichments, four quarter storage cells, and the water region between the cells, in an infinite lateral array (using reflective boundary conditions). The infinite array results in a checkerboard configuration containing two different fuel enrichments.

Axially, the multi-cell model is identical to the single-cell model.

4.3 ANALYSIS RESULTS: REGION 1

The reactivity in Region 1 of the spent fuel pool was determined as required by ANSI/ ANS-57.2.[21] The analysis consisted of a combination of CASMO-3 and KENO-5 cases. Uncertainties associated with each calculation were included.

The CASMO-3 calculations provided the enrichment and burnup relationships required for this analysis. The KENO-5 calculations used both the single-cell and the multi-cell models. Each KENO-5 case considers 300 neutron generations and 400 neutrons per generation, resulting in 120,000 neutron histories. This number of neutron histories was determined to be adequate. KENO outputs were reviewed to determine proper convergence of the calculations, thus ensuring that each region of the model was adequately sampled. In addition, repeat runs with different random number seeds provided additional assurance that each region of the model has been sampled. The final K_{eff} is the sum of the KENO-5 K_{eff}, the method biases, and the statistical combination of the uncertainties.

As part of the design modification analysis for addition of the Region 2 ("baby") rack, the Region 1 analysis was independently verified by Holtec International.[1] The Holtec verification included review of input data, uncertainties, and evaluation results. In addition, Holtec performed independent calculations of key Region 1 cases. Table 4.3.1 provides a comparison of Holtec independent calculations with results of the TVA Region 1 analysis. Holtec concluded that the computer codes had been appropriately used, that the analyses were correct, and that the uncertainties developed are appropriate and conservative.

4.3.1 <u>Results</u>

Analysis of the WBN spent fuel racks confirmed that Region 1 (PaR) racks can safely and conservatively accommodate storage of fuel up to 5 wt% U-235 enrichment with the following storage conditions:

1. Fuel with 3.8 wt% or less U-235 enrichment may be stored in Region 1 without restriction. The maximum K_{eff} for this storage condition is 0.9456 including bias and uncertainties.

2. Fuel with enrichment greater than 3.8 wt% and up to 5.0 wt% (4.95 ± 0.05) U-235 may be stored in any Region 1 cell provided the burnup is sufficient to assure the equivalent fresh fuel enrichment is less than or equal to 3.8 wt% U-235. The required burnup for the reactivity of 5.0 wt% U-235 fuel to be equivalent to the reactivity of 3.8 wt% fresh fuel is 6.75 MWD/KgU. The maximum K_{eff} for this storage condition is 0.9477 including bias and uncertainties.

3. Fuel with enrichment greater than 3.8 wt% U-235 and burnup less than 6.75 MWD/KgU shall only be placed in Region 1 storage locations with face adjacent cells which contain either:

    - Fuel assemblies with any enrichment up to 5.0 wt% U-235 and burnup equal to or greater than 20.0 MWD/KgU (equivalent to reactivity of a fresh fuel assembly with enrichment of 2.71 wt% U-235); or,

    - Water

    The maximum K_{eff} values for 5.0 wt% U-235 fuel checkerboarded with fuel burned to 20 MWD/KgU or water are 0.9457 and 0.8673 respectively.

Accounting for biases and uncertainties, the maximum K_{eff} values for the above spent fuel storage rack conditions are less than 0.95. The maximum K_{eff} values along with storage conditions, bias and uncertainties are presented in Table 4.3.2. The maximum K_{eff} was determined as follows:

$K_{eff} = K_{eff}$ (KENO) + BIASES + UNCERTAINTIES

Biases include the CASMO and KENO method biases, a boron particle self-shielding allowance, and a bias for the extrapolation of enrichment from the critical benchmark comparisons. The uncertainties include the KENO statistical uncertainty, the KENO and CASMO method uncertainties, and the mechanical tolerance uncertainty. These values are presented in Table 4.3.2 and discussed in Sections 4.2.1 and 4.3.2.

A water gap of 1.5 inches between Region 1 and Region 2 racks, with Boral panels on both sides of the water gap (i.e., a flux trap), precludes any adverse interaction between the two regions.[1] Therefore, the results presented above are the same before and after installation of the Region 2 "baby" racks.

4.3.2 <u>Sensitivity Studies</u>

The effect of various parameters on reactivity was determined to ensure the conservatism of the analysis. This was accomplished by performing sensitivity studies on these parameters with either CASMO-3 or KENO-5. Due to the statistical variation inherent in the KENO-5 results, cases requiring direct application of the K_{eff} were run several times with different starting random number initialization values in an effort to detect any statistical aberration.

<u>AXIAL BURNUP DISTRIBUTION</u>

The effects of axial burnup distribution were analyzed using equivalent enrichment data determined by CASMO-3. The axial burnup distribution was modeled for a bottom peaked, a top peaked, and a cosine shaped distribution. These cases, when compared with an average enrichment case in which no axial variations were modeled, showed a difference of -0.00042 ΔK_{eff}. This change in reactivity is conservatively ignored. In addition, this result verifies that the axial distribution of burnup does not affect the criticality analysis. More information on axial burnup distribution is provided in Section 4.5.2.

<u>WATER TEMPERATURE/DENSITY</u>

The initial Region 1 analysis was performed at 68°F (20°C). However, industry experience indicates a potential for spent fuel pool overcooling and thus a potential for pool temperatures below 68 °F. Therefore, a temperature correction factor of +0.0015 ΔK for the reference PaR design was calculated by Holtec.[1] This correction allows updating results from 68°F (20°C) to 39.2 °F (4°C). Table 4.3.2 provides a comparison of key Region 1 results at 68 °F (20°C) and 39.2 °F (4°C).

Results of CASMO-3 sensitivity studies demonstrate that the pool water temperature coefficient of reactivity is negative (Figure 4.3.1 and Table 4.3.3); therefore, a temperature of 4 °C in the reference design assures that the true reactivity will always be lower over the expected range of water temperatures. With soluble poison present, the temperature coefficients of reactivity would differ from those inferred from the data in Figure 4.3.1. However, the reactivities would also be substantially lower at temperatures with soluble boron present, and the data in Figure 4.3.1 is pertinent to the higher-reactivity unborated case.

ASSEMBLY PLACEMENT

The analysis was performed with the fuel assembly centered inside the storage cell, since previous criticality analyses[22,23] performed on these racks determined that this was the most reactive configuration.

MECHANICAL TOLERANCES

The reactivity effect of mechanical tolerances associated with rack dimensional values has been analyzed using KENO-5. Results of sensitivity studies evaluating dimensional tolerances (cell bow, cell pitch, inner stainless steel wall, outer stainless steel wall, inner tube inside width, outer tube inside width, and poison plate width) determined that the most significant adverse reactivity effect was due to the combined impact on the flux trap. Therefore, a worst case (i.e., most positive effect on reactivity) which minimized the flux trap width was used to calculate the reactivity impact of mechanical tolerances. The minimum flux trap width is achieved by combining the maximum cell inner dimension, the maximum sheet metal thickness, and the minimum cell pitch. Results of this worst case compared to results of a KENO-5 case using nominal dimensions provided a reactivity difference of 0.00026 ΔK_{eff} with a 2σ uncertainty of 0.00585 ΔK_{eff}. Since the change in reactivity due to mechanical tolerances is less than the uncertainty in the KENO-5 run itself, the uncertainty in the KENO run was used as a conservative estimate of the reactivity uncertainty due to mechanical tolerances. Therefore, the uncertainty associated with the worst mechanical tolerance for the PaR rack calculation is 0.00585 ΔK_{eff}.

POISON LOADING

The criticality analysis was performed with a poison loading of 0.0233 gm/cm^2 of ^{10}B in the Boral. A total boron content equivalent to or greater than 0.0233 gm/cm^2 of ^{10}B was required by the specification for the Boral used in these racks.[24] Since the reactivity decreases as the poison loading increases, 0.0233 gm/cm^2 of ^{10}B is the loading assumed.

PELLET DENSITY

Calculations described in Table 4.3.2 were performed using 97% of the theoretical UO_2 density. This is greater than any fuel density projected for Westinghouse fuel in TVA reactors.[25] This approach gave conservative results when analyzed with CASMO-3.

CELL DIMENSIONS/BOW

This analysis accounts for cell bowing within the analysis for rack cell mechanical tolerances. The rack cell mechanical tolerance analysis was performed using the worst case combination of maximum cell inner dimension, maximum sheet metal thickness, and minimum cell pitch. Cell bowing causes both the cell pitch and the cell inner dimension to be altered. Therefore, the effects of cell bowing are the same as those addressed in the mechanical tolerance case and do not need to be addressed independently.

BORON PARTICLE SELF-SHIELDING EFFECT

Lower than predicted neutron transmission has been observed in strong heterogeneous absorber materials.[22] This reduction has been hypothesized to result from particle self shielding. The boron particles in the Boral were modeled as a homogenous distribution. Therefore, as a conservative measure, a bias is applied to the PaR rack calculations to assure potential effects of boron particle self-shielding are accounted.

Using calculation results,[22] the increase in K_{eff} due to particle self-shielding is 0.003 ΔK_{eff}. This value was applied to the final K_{eff} as a bias.

BORATED WATER REACTIVITY WORTH

An analysis was performed to determine the reactivity worth of the borated water in the spent fuel storage pool. Using a 2000 ppm boron concentration, both KENO-5 and CASMO-3 predicted the change in K_{eff} to be approximately 17% ΔK_{eff}. This value is not used in the determination of the system K_{eff} but is used to demonstrate that sufficient soluble boron worth is present for accident scenarios which require soluble poison credit to assure K_{eff} remains equal to or less than 0.95 (see Section 4.3.5).

WATER HOLE REACTIVITY WORTH

KENO-5 calculations were performed which demonstrate that checkerboarding fresh fuel assemblies with cells filled with unborated water does not increase the reactivity of Region 1. The maximum K_{eff} (at 4°C) for this case is 0.8673 which is less than 0.9457 where fresh fuel and burned fuel is considered. Therefore, water holes can replace the burned fuel requirement for implementing the results of this analysis.

4.3.3 Reactivity vs Enrichment

The sensitivity of the Region 1 K_{eff} to U-235 enrichment was determined using both CASMO-3 and KENO-5. The results showed that reactivity increases with increasing enrichment. This relationship is shown in Figure 4.1.3.

If a fresh fuel assembly of 5.0 wt% enrichment accumulates 6.75 MWD/KgU of burnup, it has an equivalent reactivity to a 3.8 wt% fuel assembly without burnup. Therefore, once a fresh fuel assembly accumulates at least 6.75 MWD/KgU of burnup, the restrictions on its placement in the Region 1 (PaR) spent fuel storage racks can be removed. This includes allowances for the potential effect of burnable absorber history and water density history.

Note: The uncertainty associated with burnup measurement is not included in this analysis and must be addressed in the procedural process for designation of an acceptable pool region for storage.

4.3.4 Credit For Burnup

Credit for burnup was determined using the equivalent enrichment concept. Fresh fuel assemblies at lower enrichment (2.71 wt%) were used to simulate burned fuel. These depleted assemblies were inserted in a checkerboard storage configuration, along with fuel assemblies at 5.0 wt% representing fresh fuel assemblies. The equivalent enrichment of 2.71 wt% corresponds to a 5.0 wt% fuel assembly depleted to 20.0 MWD/KgU. This infinite checkerboard array of depleted and fresh 5.0 wt% assemblies results in a Region 1 $K_{eff} \leq 0.95$.

Note: The uncertainty associated with burnup measurement is not included in this analysis and must be addressed in the procedural process for designation of an acceptable pool region for storage.

4.3.5 Special Cases and Postulated Accidents

Although credit for soluble poison normally present in the spent fuel pool water is permitted under abnormal or accident conditions (double contingency principle), most abnormal or accident conditions will not result in exceeding the limiting reactivity ($K_{eff} = 0.95$) even in the absence of soluble poison. The following discussion addresses credible abnormal occurrences and accident conditions for the spent fuel storage pool with respect to criticality.

DROPPED FUEL ASSEMBLY

No adverse reactivity impact is expected from dropping a fuel assembly on a fully loaded PaR storage rack. The dropped fuel assembly will come to rest horizontally on top of the fuel rack or, between the periphery of the rack and the spent fuel pool wall. These configurations were analyzed and determined to be acceptable (including potential deformation of the storage racks due to the accident condition) as discussed below:

- Including potential deformation of the storage rack, the separation distance for a fuel assembly laying horizontally on top of the fuel rack is sufficient to assure insignificant effect on reactivity under seismic or accident conditions. There is a 44.6 cm distance between the top of the active fuel and the top of the storage racks (or the dropped fuel assembly). Since the migration length of neutrons in water at 1 gm/cm^3 is 6.2 cm,[26] then the separation distance that exists can be assumed to be an infinite distance and the stored fuel assemblies do not interact with the dropped assembly.

  Furthermore, for accident conditions, the double contingency principle[17] can be applied when two unlikely events are required to produce a criticality accident. Therefore, credit for borated water can be applied.

- Before installation of the Region 2 racks, a fuel assembly can come to rest in the space between the racks and the spent fuel pool wall. However, for accident conditions, the double contingency principle[17] can be applied. Therefore, the analysis for this configuration takes credit for the borated water normally in the pool.

An analysis was performed with KENO-5 that demonstrated that the addition of one fuel assembly to the spent fuel rack periphery adds less positive reactivity than the negative reactivity associated with the borated water. Therefore, a dropped fuel assembly will not result in the K_{eff} limit of 0.95 being exceeded.

DROPPED CASK/HEAVY LOAD

Heavy loads are prohibited from being moved over fuel assemblies in the Region 1 spent fuel storage racks. There is no effect on the criticality of the spent fuel storage rack.

SEISMIC EVENT

The spent fuel storage racks are designed to withstand loads from a safe shutdown earthquake.[27] There is no effect on the criticality of the spent fuel storage rack.

LOSS OF SPENT FUEL POOL COOLING

A loss of spent fuel pool cooling would result in an increase in the spent fuel pool water temperature. The effects of increased pool water temperature have been addressed in Section 4.3.2 on sensitivity studies and are not a criticality concern because reactivity decreases as the water density decreases.

MISPLACED FUEL ASSEMBLIES

The implementation of this analysis requires administrative controls to be placed on Region 1 storage of fuel assemblies with enrichment greater than 3.8 wt% and burnup less than 6.75 MWD/KgU. If the fuel is inadvertently misplaced in the Region 1 spent fuel storage racks ignoring the administrative controls, then the double contingency principle can be applied (i.e., two unlikely events are required to produce a criticality accident). Therefore, the presence of boron in the spent fuel pool can be assumed.

The worth of the borated water (i.e., 2000 ppm boron) is sufficient to lower the K_{eff} of the storage racks to 0.8307 (including bias and uncertainty at 4°C) assuming that the racks are loaded with fresh 5.0 wt% fuel assemblies. Therefore, inadvertently misloading the fuel in the spent fuel storage racks would not result in a criticality accident.

4.3.6 Summary of Conservatism

These analyses do not take credit for the following phenomena which would decrease reactivity.

PRESENCE OF BORATED WATER

Technical Specification 3.9.9 requires that during fuel movement in the flooded spent fuel pool, the boron concentration must be 2000 ppm or higher. In addition, a study was performed by Holtec[1] to determine the concentration of soluble boron sufficient to permit the unrestricted storage of fresh fuel with enrichment up to 5.0 wt% U-235 in the PaR racks. Allowing for biases and uncertainties (including the uncertainty in calculating the soluble boron concentration), storage of fresh 5.0 wt% fuel requires 520 ppm soluble boron concentration to assure that the maximum K_{eff} is always less than 0.95. Except as discussed in Section 4.3.5, credit for borated water (i.e., it's negative reactivity effect) is not included in this analysis.

PRESENCE OF BURNABLE ABSORBERS

High enriched fresh fuel requires the use of burnable absorbers in order to control power peaking at full power conditions. These burnable absorbers are delivered already inserted in the fresh fuel assembly. Their effect is to lower the reactivity of the fuel assembly during its first cycle of operation until accumulated burnup effectively lowers the reactivity.

This analysis assumes that the burnable absorbers are removed from the fuel assemblies when placed in the spent fuel storage rack.

LOWER ENRICHMENT

These analyses have been completed assuming that all fresh fuel assemblies in the spent fuel racks are enriched to 5.0 wt%. Near term reloads are planned with enrichments less than 4.5 wt%. As the enrichment decreases, the reactivity of the fresh fuel assemblies decrease.

HIGHER BURNUP

Projected burnup of WBN fuel is considerably greater than the 20 MWD/KgU required by this analysis. This increase in burnup and resulting decrease in assembly reactivity is ignored beyond 20 MWD/KgU.

MISCELLANEOUS CONSERVATIVE ASSUMPTIONS

Other conservative assumptions in this analysis include:

- Ignoring radial neutron leakage from the spent fuel storage racks
- Ignoring the presence of control rods
- Ignoring the presence of spent burnable absorber assemblies
- Ignoring the higher water temperature of the spent fuel pool
- Maximizing burnable poison history effects
- Maximizing water density history effects
- Minimizing the ^{10}B content in the Boral

4.4 METHODS: REGION 2 AND CASK LOADING PIT RACK ANALYSIS

The criticality analysis for the WBN racks is a combination of TVA Region 1 calculations (verified by Holtec International) and Holtec analyses for Region 2, cask pit and regions 1 and 2 interface.[1] The Region 2 and cask pit rack criticality calculations were performed by Holtec using the KENO-5a[5] computer code package, with the 27-group cross-section library and the NITAWL[4] subroutine for U-238 resonance shielding effects (Nordheim integral treatment).

Depletion analyses and determination of equivalent enrichments were made with the two-dimensional transport theory code, CASMO-3.[7,8,9]

4.4.1 Benchmarking

In order to verify the method applicability to criticality analyses, a series of calculations were performed using each of the methods. The results of these calculations establishes the biases and uncertainties associated with the computer codes. Application of the methodology by TVA and Holtec including models and benchmark cases vary slightly and thus result in small differences in biases and uncertainties. Both Holtec and TVA biases and uncertainties have been reviewed and determined acceptable in previous NRC approved criticality analyses.[2,28] Holtec benchmark results are described below:

The uncertainty allowance for CASMO-3[1,28,29,30] depletion calculations is 5% of the reactivity decrement from beginning-of-life (K_{eff} of 1.1893 in the storage racks) to 41 MWD/KgU (K_{eff} of 0.9132) resulting in a bias of +0.0138 ΔK_{eff}. The benchmark calculations for NITAWL-KENO-5a[28] indicate a bias of 0.0113 with an uncertainty of $\pm$ 0.0017 (95%/95%). ORNL, in benchmark calculations with the 27-group SCALE library, has reported comparable results.[31]

4.4.2 Computer Code Application

KENO-5a was applied to determine Region 2 and cask loading pit rack reactivity effects, and CASMO-3 was used in burnup calculations. In addition to burnup calculations, CASMO-3 was used for evaluating the small reactivity increments (by differential calculations) associated with most manufacturing tolerances and for determining temperature effects (including the consequence of the library inadequacy in NITAWL).[28]

Because the tolerance in boron-10 loading directly affects a coupling between cells of differing reactivities, KENO-5a differential calculations were used to determine this reactivity uncertainty.

Because NITAWL-KENO-5a does not have burnup capability, burned fuel was represented by fresh fuel of equivalent enrichment as determined by CASMO-3 calculations in the storage cell (i.e., an enrichment which yields the same reactivity in the storage cell as the burned fuel).

4.4.3 Fuel Assembly

The design basis fuel assembly for Region 2 calculations is the Westinghouse VANTAGE 5H design.[10] The fuel assembly was modeled as a 17x17 array of fuel rods with 25 rods replaced by 24 control rod guide tubes and 1 instrument thimble. The spacer grids and other structural material are conservatively neglected.

4.4.4 Storage Rack

The WBN Region 2 ("baby") rack and the 15x15 cask loading pit rack shown in Figure 4.1.1 are Holtec International design racks. The nominal spent fuel storage cell used for the criticality analyses is shown in Figure 4.4.1. Each storage cell is composed of single Boral absorber panels positioned between two 8.75 inch square inside dimension, 0.060-inch thick stainless steel boxes.

Peripheral cells use a 0.060 inch stainless steel sheathing on the outside supporting the Boral panel. The fuel assemblies are normally located in the center of each storage cell on a nominal lattice spacing of 8.97 ± 0.04 inches. The Boral absorber has a thickness of 0.102 ± 0.005 inch and a nominal B-10 areal density of 0.0324 g/cm^2 (0.030 g B-10/cm^2 minimum).

4.4.5 Analytical Models

The analysis required CASMO-3 and KENO-5a models:

CASMO-3 MODEL

In the CASMO-3 geometric model (cell), each fuel rod and its cladding were described explicitly and reflecting boundary conditions (zero neutron current) were used at the centerline of the Boral and steel plates between storage cells. (CASMO-3 is a two-dimensional model.) These boundary conditions have the effect of creating an infinite array of storage cells in the X-Y plane and provide a conservative estimate of the uncertainties in reactivity attributed to manufacturing tolerances.

KENO-5a MODEL

In the KENO-5a model, each fuel rod and its cladding were described explicitly. The actual fuel assembly length in the axial direction was used, assuming thick (30 cm) water reflectors top and bottom. Monte Carlo (KENO-5a) calculations inherently include a statistical uncertainty due to the random nature of neutron tracking. To minimize the statistical uncertainty of the KENO-5a calculated reactivity, a minimum of 500,000 neutron histories in 1000 generations of 500 neutrons each were accumulated in each calculation.

4.5 ANALYSIS RESULTS: REGION 2 AND CASK LOADING PIT RACKS

Figure 4.1.1 provides the layout of the Region 2 and cask pit spent fuel racks analyzed by Holtec.[1]

4.5.1 Results

Analysis confirmed that the Region 2 ("baby") and the cask loading pit racks can safely and conservatively accommodate storage of fuel with enrichment up to 4.95 ± 0.05 wt% U-235 with the following storage conditions:

1. Fuel with 4.95 ± 0.05 wt% enrichment has accumulated burnup of at least 41.0 MWD/KgU.

2. Fuel with enrichment lower than 4.95 ± 0.05 wt% U-235 has accumulated burnup which yields an equivalent reactivity in the storage racks.

Burnup vs enrichment which yields an equivalent reactivity to a 4.95 ± 0.05 wt% U-235 assembly with burnup of 41 MWD/KgU was determined using the equivalent enrichment concept described in Section 4.1.3. Acceptable burnups as a function of enrichment are represented by the following equation and presented graphically in Figure 4.5.1.

Burnup (MWD/KgU) = $0.0666\ E^3 - 1.3933\ E^2 + 18.7600\ E - 25.7425$, where
E= enrichment (wt% U-235)

The maximum K_{eff} (in Region 2 or the cask pit rack) including calculation bias and uncertainties for the above storage conditions is 0.9444.[1] This is conservatively less than the 0.95 K_{eff} limit and would be even lower if neutron leakage from the peripheral cells were included.

In addition, the cask pit rack can accommodate fresh 4.95 ± 0.05 wt% enriched fuel provided face adjacent storage cells are empty (water).[1] The maximum K_{eff} for this storage condition is 0.9370, including bias and uncertainties.

Accounting for biases and uncertainties, the maximum K_{eff} values for the above spent fuel storage rack conditions are less than 0.95. The maximum K_{eff} values along with storage conditions, bias and uncertainties are presented in Table 4.5.1. The maximum K_{eff} was determined as follows:

$$K_{eff} = K_{eff}\ (\text{KENO}) + \text{BIASES} + \text{UNCERTAINTIES}$$

Biases include the CASMO-3 depletion allowance and the KENO-5a method bias. The uncertainties include the bias statistical uncertainty, the method uncertainties, and the mechanical tolerance uncertainty. The biases and bias statistical uncertainty were described in Section 4.4.1 and the manufacturing uncertainties are discussed in the following section.

A water gap of 1.5 inches between Region 1 and Region 2 racks, with Boral panels on both sides of the water gap (i.e., a flux trap), precludes any adverse interaction between the two regions.[1] Therefore, storage of fuel in Region 1 has no adverse impact on Region 2 reactivity.

4.5.2 Sensitivity Studies

The effect of burnup distribution, temperature and density effects, fuel positioning, and tolerances of various parameters were determined to assure conservatism of the analysis.

AXIAL BURNUP DISTRIBUTION

Initially, fuel loaded into the reactor will burn with a slightly skewed cosine power distribution. As burnup progresses, the burnup distribution will tend to flatten, becoming more highly burned in the central regions than in the upper and lower ends.[28] At high burnup, the more reactive fuel near the ends of the fuel assembly (less than average burnup) occurs in regions of lower reactivity worth due to neutron leakage. Consequently, it would be expected that over most of the burnup history, fuel assemblies with distributed burnups would exhibit a slightly lower reactivity than that calculated for the average burnup. As burnup progresses, the distribution, to some extent, tends to be self-regulating as controlled by the axial power distribution, precluding the existence of large regions of significantly reduced burnup. Generic analytic results of the axial burnup effect based upon calculated and measured axial burnup distributions have been presented.[32] These analyses confirm the minor reactivity effect of the axially distributed burnup.

Calculations were made with KENO-5a in three dimensions, based upon the typical axial burnup distribution of spent fuel. In these calculations, the axial height of the burned fuel was divided into a number of axial zones (6-inch intervals near the more significant top of the fuel), each with an enrichment equivalent to the burnup of that zone. These calculations resulted in a negligible reactivity increment for the reference design burnups. Fuel of lower initial enrichments (and lower burnup) would have a more negative reactivity effect as a result of the axial variation in burnup. These estimates are conservative since smaller axial increments in the calculations have been shown to result in lower incremental reactivities.[32]

WATER TEMPERATURE/DENSITY

As shown in Figure 4.3.1 and Table 4.3.3, the pool water temperature coefficient of reactivity is negative. Therefore, use of pool water temperature of 4°C (39°F) assures that the true reactivity will always be lower over the expected range of water temperatures. Temperature effects on reactivity have been calculated (CASMO-3) and the results are shown in Table 4.3.3. With soluble poison present, the temperature coefficients of reactivity would differ from those inferred from the data in Table 4.3.3. However, the reactivities would also be substantially lower at temperatures with soluble boron present, and the data in Table 4.3.3 is pertinent to the higher-reactivity unborated case.

ASSEMBLY PLACEMENT

The fuel assembly is assumed to be normally centered in the storage rack cell. Calculations were made using KENO-5a, assuming the fuel assemblies were located in the corner of the storage rack cell (four assembly clusters at the closest possible approach).[28] These calculations indicated that the reactivity increment due to eccentricity of assembly locations was slightly negative. Therefore, the reference case with the fuel assemblies centered is controlling, and no uncertainty for eccentricity is necessary.

MECHANICAL TOLERANCES

The statistical combination of the manufacturing tolerances described below results in a mechanical uncertainty of $\pm$ 0.0057 ΔK.[1,28]

POISON LOADING

The Boral absorber panels used in the storage cells are nominally 0.102 inch thick, 7.50 inches wide, and 144 inches long, with a nominal B-10 areal density of 0.0324 g/cm^2. The vendors manufacturing tolerance limit is $\pm$ 0.0024 g/cm^2 in B-10 content which assures that at any point the minimum B-10 areal density will not be less than 0.030 g/cm^2. Differential KENO-5a calculations for the reference design with the minimum tolerance B-10 loading result in an incremental reactivity of $\pm$ 0.0045 ΔK uncertainty.

BORAL WIDTH TOLERANCE

The reference storage cell design uses a Boral panel with an initial width of 7.50 ± 0.06 inches. For the tolerance of 0.06 inch, the differential CASMO-3 calculated reactivity uncertainty is ±0.0010 ΔK.

TOLERANCES IN CELL LATTICE SPACING

The manufacturing tolerance of the inner box dimension, which directly affects the storage cell lattice spacing between fuel assemblies, is ± 0.04 inch. This corresponds to an uncertainty in reactivity of ± 0.0016 ΔK determined by differential CASMO-3 calculations.

STAINLESS STEEL THICKNESS TOLERANCES

The nominal stainless steel thickness is 0.060 ± 0.005 inch for the stainless steel box. The maximum positive reactivity effect of the expected stainless steel thickness tolerances was calculated (CASMO-3) to be ± 0.0004 ΔK.

FUEL ENRICHMENT AND DENSITY TOLERANCES

The design maximum enrichment is 4.95 ± 0.05 wt% U-235. Separate CASMO-3 burnup calculations were made for fuel of the maximum enrichment (5.0 wt% U-235) and for the maximum UO_2 density (10.61 g/cc). Reactivities in the storage cell were then calculated using the restart capability in CASMO-3. For fresh fuel, the incremental reactivity uncertainties were ±0.0021 ΔK for the enrichment tolerance and ± 0.0013 ΔK for the tolerance in fuel density. Using equivalent enrichments determined for the reference fuel burnup of 41 MWD/KgU, a 3-dimensional KENO-5a calculation was made to confirm the CASMO-estimated uncertainties. The small incremental reactivities determined with KENO-5a were consistent with those of the CASMO calculations within the normal statistical variation in KENO results. For the tolerance on U-235 enrichment, the uncertainty is ± 0.0021 ΔK and for fuel density is ± 0.0021 ΔK.

4.5.3 Reactivity vs Enrichment

The design basis fuel enrichment and burnup for Region 2 and cask loading pit storage is 4.95 ± 0.05 wt% U-235 with 41 MWD/KgU burnup. Figure 4.5.1 provides the "acceptable domain" of enrichment vs burnup to assure that fuel with a specific enrichment (less than 4.95 ± 0.05 wt%) has equivalent reactivity to a 4.95 ± 0.05 wt% U-235 enriched fuel assembly with a burnup of 41 MWD/KgU.

Note: The uncertainty associated with burnup measurement is not included in this analysis and must be addressed in the procedural process for designation of an acceptable pool region for storage.

4.5.4 Credit For Burnup

As with Region 1, credit for burnup was determined using the equivalent enrichment concept. Since there are no critical experiment data with spent fuel for determining the uncertainty in burnup-dependent reactivity calculations, a standard allowance for uncertainty in reactivity[1] was assigned. The allowance for uncertainty in depletion calculations is assumed to be 5% of the reactivity decrement from beginning-of-life (K_{eff} of 1.1893 in the storage rack) to 41 MWD/KgU (K_{eff} of 0.9132) or +0.0138 ΔK.

The allowance for uncertainty in burnup calculations is a conservative estimate in view of the substantial reactivity decrease as spent fuel ages (due to Pu-241 decay and Am-241 growth).

Note: The uncertainty associated with burnup measurement is not included in this analysis and must be addressed in the procedural process for designation of an acceptable pool region for storage.

4.5.5 Special Cases and Postulated Accidents

Although credit for soluble poison normally present in the spent fuel pool water is permitted under abnormal or accident conditions (double contingency principle), most abnormal or accident conditions will not result in exceeding the limiting reactivity (K_{eff} = 0.95) even in the absence of soluble poison. The effects on reactivity of credible abnormal and accident conditions for Region 2 and the cask loading pit rack are discussed in the following sections. Of these abnormal or accident conditions, only abnormal location of a fuel assembly has the potential for a more than negligible positive reactivity effect. As described below, credit for soluble poison is sufficient to assure that the limiting K_{eff} of 0.95 is not exceeded.

DROPPED FUEL ASSEMBLY

For a drop on top of the Region 2 or cask loading pit racks, the fuel assembly will come to rest horizontally on top of the rack with a minimum separation distance from the fuel in the rack of more than 12 inches, including the potential deformation under seismic or accident conditions. At this separation distance, the effect on reactivity is insignificant.[28] Furthermore, soluble boron in the pool water would substantially reduce the reactivity and assure that the true reactivity is always less than the limiting value for any conceivable dropped fuel accident.

After installation of the Region 2 racks, it is not physically possible to install a fuel assembly outside and adjacent to a storage module in the spent fuel storage pool. In addition, after installation of the cask pit rack, it is not physically possible to install a fuel assembly outside and adjacent to a storage module in the cask loading pit.

DROPPED CASK/HEAVY LOAD

Heavy loads are prohibited from being moved over fuel assemblies in the Region 2 or cask pit spent fuel storage racks. There is no effect on the criticality of the spent fuel storage rack.

SEISMIC EVENT

The spent fuel storage racks are designed to withstand loads from a safe shutdown earthquake.[27] There is no effect on the criticality of the spent fuel storage rack.

LOSS OF SPENT FUEL POOL COOLING

A loss of spent fuel pool cooling would result in an increase in the spent fuel pool water temperature. The effects of increased pool water temperature have been addressed in Section 4.5.2 on sensitivity studies and are not a criticality concern because reactivity decreases as the water density decreases.

MISPLACED FUEL ASSEMBLIES

The implementation of this analysis requires administrative controls to be placed on storage of fuel assemblies in Region 2 or the cask pit rack. If the fuel is inadvertently misplaced ignoring the administrative controls, then the double contingency principle can be applied since two unlikely events are required to produce a criticality accident. Therefore, the presence of boron in the spent fuel pool or cask loading pit can be assumed.

The worth of the borated water is sufficient to lower the K_{eff} of the Region 2 or cask pit storage rack below 0.95 assuming that the rack is loaded with fresh 5.0 wt% fuel assemblies. Therefore, inadvertently misloading the fuel in the spent fuel storage racks would not result in a criticality accident.

4.5.6 Summary of Conservatism

These analyses do not take credit for the following phenomena which would decrease reactivity in Region 2 or the cask loading pit racks.

PRESENCE OF BORATED WATER

Technical Specification 3.9.9 requires that during fuel movement in the flooded spent fuel pool, the boron concentration of the spent fuel pool be 2000 ppm or higher. In addition, Technical Specification 3.9.10 will require 2000 ppm boron be maintained in the cask loading pit during fuel movement. Except as discussed in Section 4.5.5, credit for borated water (i.e., it's negative reactivity effect) is not included in this analysis.

PRESENCE OF BURNABLE ABSORBERS

High enriched fresh fuel requires the use of burnable absorbers in order to control power peaking at full power conditions. These burnable absorbers are delivered already inserted in the fresh fuel assembly. Their effect is to lower the reactivity of the fuel assembly during its first cycle of operation until accumulated burnup effectively lowers the reactivity.

This analysis assumes that the burnable absorbers are removed from the fuel assemblies when placed in the spent fuel storage rack.

MISCELLANEOUS CONSERVATIVE ASSUMPTIONS

Other conservative assumptions in this analysis include:

- Ignoring radial neutron leakage from the spent fuel storage racks
- Ignoring the presence of control rods
- Ignoring the presence of spent burnable absorber assemblies
- Ignoring the reactivity decrease due to higher water temperature of the spent fuel pool
- Ignoring fuel assembly spacer grids and structural material

4.6 SUMMARY AND CONCLUSIONS

The criticality analysis described above demonstrates that fuel with enrichment up to 4.95 ± 0.05 wt% U-235 may be safely and conservatively stored in the WBN high density spent fuel racks, provided controls are implemented to assure the following:

PaR RACKS - REGION 1

- Fuel with 3.8 wt% U-235 or less enrichment may be stored in any Region 1 storage cell without restriction.

- Fuel with enrichment between 3.8 wt% and 5.0 wt% (4.95 ± 0.05) U-235 may be stored in Region 1, provided compliance is maintained with one of the following conditions:

    1. Fuel burnup is greater than or equal to 6.75 MWD/KgU.

    2. Fuel with burnup less than 6.75 MWD/KgU is placed in storage locations with face adjacent storage cells containing either water (empty cells) or fuel assemblies with accumulated burnup of at least 20.0 MWD/KgU.

HOLTEC "BABY" RACK - REGION 2

- Fuel with enrichments up to 4.95 ± 0.05 wt% U-235 may be stored in Region 2 racks, provided compliance is maintained with one of the following conditions:

    1. Fuel with 4.95 ± 0.05 wt% U-235 enrichment has accumulated burnup of at least 41.0 MWD/KgU.

    2. Fuel with enrichment lower than 4.95 ± 0.05 wt% U-235 has accumulated burnup in the acceptable burnup domain of Figure 4.5.1 which is represented by the following equation:

        Burnup (MWD/KgU) = $0.0666\,E^3 - 1.3933\,E^2 + 18.7600\,E - 25.7425$,
        where E= enrichment (wt% U-235)

- A water gap of at least 1.5 inches must be maintained between the Region 1 and Region 2 racks (i.e., minimum water gap between Region 1 storage cells interfacing with Region 2 storage cells).

CASK LOADING PIT RACK

- Fuel with enrichments up to 4.95 ± 0.05 wt% U-235 may be stored in the cask loading pit rack, provided compliance is maintained with one of the following conditions:

  1. Fuel with 4.95 ± 0.05 wt% U-235 enrichment has accumulated burnup of at least 41.0 MWD/KgU.

  2. Fuel with enrichment lower than 4.95 ± 0.05 wt% U-235 has accumulated burnup in the acceptable burnup domain of Figure 4.5.1 which is represented by the following equation:

     Burnup (MWD/KgU) = $0.0666\ E^3 - 1.3933\ E^2 + 18.7600\ E - 25.7425$,
     where E= enrichment (wt% U-235)

  3. Fresh fuel with enrichment up to 4.95 ± 0.05 wt% U-235 is placed in cask pit rack locations with face adjacent storage cells containing water (i.e., checkerboard with empty cells).

Note: The uncertainty associated with burnup measurement is not included in this analysis and must be addressed in the procedural process for designation of an acceptable pool region for storage.

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TABLE 4.1.1

DESIGN BASIS FUEL ASSEMBLY SPECIFICATIONS

Fuel Rod Data

Outside diameter, in.	0.374
Cladding thickness, in.	0.0225
Cladding inside diameter, in.	0.329
Cladding material	Zr-4
Pellet density, %T.D.	95.0
Stack density, g UO_2/cc	10.41 ±0.20
Pellet diameter, in.	0.3225
Maximum enrichment, wt% U-235	4.95 ± 0.05

Fuel Assembly Data

Fuel rod array	17 x 17
Number of fuel rods	264
Fuel rod pitch, in.	0.496
Number of control rod guide and instrument thimbles	25
Thimble O.D., in. (nominal)	0.474
Thimble I.D., in. (nominal)	0.442

TABLE 4.2.1

RESULTS OF KENO BENCHMARKING: COMPARISON OF KENO WITH CRITICAL EXPERIMENTS

Reference	Test	Poison	Characterizing Separation (cm)	KENO K_{eff}	1-Sigma
BAW-1484	01	None	0.000	0.99250	0.00268
BAW-1484	02	None	0.000	0.98677	0.00224
BAW-1484	03	None	1.636	0.99593	0.00210
BAW-1484	04	B4C-Pins	1.636	0.98779	0.00278
BAW-1484	05	B4C-Pins	3.272	0.98254	0.00273
BAW-1484	06	B4C-Pins	3.272	0.99006	0.00388
BAW-1484	07	B4C-Pins	4.908	0.98460	0.00281
BAW-1484	08	B4C-Pins	4.908	0.99077	0.00356
BAW-1484	09	None	6.544	0.98657	0.00317
PNL-2438	15	None	11.920	0.99647	0.00238
PNL-2438	05	None	8.390	0.99288	0.00282
PNL-2438	22	None	6.390	0.99138	0.00284
PNL-2438	21	None	4.460	0.99078	0.00231
PNL-2438	34	Stainless	10.440	0.99950	0.00266
PNL-2438	35	Stainless	11.470	0.99822	0.00244
PNL-2438	26	Stainless	7.760	0.99005	0.00276
PNL-2438	27	Stainless	7.420	0.99469	0.00248
PNL-2438	20	Boral	6.340	0.99944	0.00278
PNL-2438	16	Boral	9.030	0.99157	0.00278
PNL-2438	46	Zirc-4	8.790	0.99658	0.00248
PNL-2438	47	Zirc-4	8.780	0.99217	0.00294
PNL-2615	30	Zirc-4	10.920	0.99499	0.00299
PNL-2615	29	Zirc-4	10.860	0.99618	0.00285
PNL-2615	31	Boral	6.720	0.99674	0.00321
PNL-2615	14	Stainless	8.580	0.99141	0.00302
PNL-2615	13	Stainless	9.650	0.99369	0.00290
PNL-2615	08	Stainless	9.220	0.99646	0.00313
PNL-2615	07	Stainless	9.760	0.99304	0.00303
PNL-2615	04	None	10.640	0.99449	0.00347
PNL-2615	06	Aluminum	10.720	0.99612	0.00290
PNL-2615	05	Aluminum	10.770	0.99134	0.00297
PNL-2615	10	SS 1.05%B	6.100	0.98918	0.00341
PNL-2615	09	SS 1.05%B	8.080	0.99590	0.00302
PNL-2615	12	SS 1.62%B	5.760	0.99459	0.00320
PNL-2615	11	SS 1.62%B	7.900	0.99018	0.00281
			35 points average	0.99273	±0.00404

TABLE 4.2.2

PaR SPENT FUEL STORAGE RACK DATA: COMPARISON OF ACTUAL VS. ANALYZED

	ACTUAL	ANALYZED
RACK DESCRIPTION		
Storage Cell Array	42 by 33	Infinite
Cell Pitch, in.	10.375	10.375
RACK CANISTERS		
Material	304 Stainless Steel	304 Stainless Steel
Inner Can I.D., in.	8.75	8.75
Inner Can Thickness, in.	0.090	0.090
Inner Can O.D., in.	8.93	8.93
Outer Can I.D., in.	9.33	9.33
Outer Can Thickness, in.	0.0360	0.0360
Outer Can O.D., in.	9.402	9.402
NEUTRON POISON		
Material	Boral	Boral
Total Length, in.	147	147
Total Thickness, in.	0.10	0.10
Width, in.	8.625	8.625
Sheath Material	Aluminum	Aluminum
Sheath Thickness, in.	0.01	0.01
Core Material	B4C-Al	B4C-Al
Core Thickness, in.	0.08	0.08
^{10}B Density, gm/sq cm	0.0233 (minimum)	0.0233

TABLE 4.3.1

REGION 1: COMPARISON OF HOLTEC AND TVA CALCULATIONS
(as Calculated, not including bias and uncertainties)

		HOLTEC CALC	TVA CALC
	UNBURNED FUEL		
1	Initial Enrichment, wt% U-235	3.83	3.80
2	Calculated K_{eff} (KENO-5a)	0.9214 ± 0.0022 (2σ)	0.9183 ± 0.0042 (2σ)
	5 WT% FUEL AT 6.75 MWD/KgU		
3	Equivalent Enrichment, wt% U-235	3.83	3.80
4	Calculated K_{eff} (KENO-5a)	0.9214 ± 0.0022 (2σ)	0.9183 ± 0.0042 (2σ)
	5 WT% FUEL CHECKERBOARD WITH SPENT FUEL AT 20 MWD/KgU		
5	Equivalent Enrichment of Spent Fuel	2.64	2.71
6	Calculated K_{eff} (KENO-5a)	0.9148 ± 0.0022 (2σ)	0.91617 ± 0.0044 (2σ)

TABLE 4.3.2

REGION 1: RESULTS OF REFERENCE CASES

CASE 1/1a = An infinite arrary of 3.8 wt% U-235 enriched fuel stored in PaR racks with water temperatures of 20 and 4 °C, respectively.

CASE 2.2a = An infinite array of 5.0 wt% U-235 enriched fuel burned to 6.75 MWD/KgU stored in PaR racks with water temperatures of 20 and 4 °C, respectively.

CASE 3/3a = A checkerboard array of 5.0 wt% U-235 enriched fresh fuel and 5.0 wt% fuel burned to 20 MWD/KgU stored in PaR racks with water temperatures of 20 and 4°C, respectively.

CASE 4 = A checkerboard array of 5.0 wt% U-235 enriched fuel and empty cells with water temperature of 4 °C.

	CASE 1	CASE 1a	CASE 2	CASE 2a	CASE 3	CASE 3a	CASE 4
Water Temperature (°C)	20	4	20	4	20	4	4
Base K_{eff}	0.91828	0.91828	0.91828	0.91828	0.91617	0.91617	0.83982
Biases (Delta K)							
Method	0.00727	0.00727	0.00727	0.00727	0.00727	0.00727	0.00727
Enrichment Extrap.	0.00318	0.00318	0.00318	0.00318	0.00318	0.00318	0.00318
Boron self-shielding	0.00300	0.00300	0.00300	0.00300	0.00300	0.00300	0.00300
Temp. Corr. to 4 °C	0.00000	0.00150	0.00000	0.00150	0.00000	0.00150	0.00150
Total Bias	0.01345	0.01495	0.01345	0.01495	0.01345	0.01495	0.01495
Uncertainties (Delta K)							
KENO Stat. (95%/95%)	0.00423	0.00423	0.00423	0.00423	0.00441	0.00441	0.00468
KENO Method & Enr.	0.01009	0.01009	0.01009	0.01009	0.01009	0.01009	0.01009
Mechanical Tolerance	0.00585	0.00585	0.00585	0.00585	0.00585	0.00585	0.00585
Burnup	0.00000	0.00000	0.00748	0.00748	0.00748	0.00748	0.00000
Uncertainties (stat. comb.)	0.01241	0.01241	0.01449	0.01449	0.01454	0.01454	0.01257
Maximum K_{eff}	0.9441	0.9456	0.9462	0.9477	0.9442	0.9457	0.8673

Maximum K_{eff} = Base K_{eff} + Bias + Uncertainties (Statistical combination)

Statistical combination of uncertainties is the square root of the sum of the squared uncertainties.

TABLE 4.3.3

EFFECT OF TEMPERATURE AND VOID ON CALCULATED REACTIVITY OF STORAGE RACK

CASE	REGION 1 (ΔK)	REGION 2 (ΔK)
4 °C (39°F)	+0.0015	Reference
20 °C (68 °F)	Reference	-0.0017
60 °C (140 °F)	-	-0.0073
120 °C (248 °F)	-	-0.0193
120 °C (+10% void)	-	-0.0425

TABLE 4.5.1

REGION 2 AND CASK PIT: RESULTS OF REFERENCE CASES

CASE 1 = Region 2 rack with an infinite array of 4.95 wt% fuel burned to 41 MWD/KgU.

CASE 2 = Cask pit rack with an infinite array of 4.95 wt% fuel burned to 41 MWD/KgU.

CASE 3 = Cask pit rack with checkerboard array of 4.95 wt% fresh fuel and empty cells (this case was modeled with the checkerboard array surrounded with a combination of fresh 4.95 wt% fuel and 4.95 wt% fuel burned to 50 MWD/KgU which provided additional conservatism).

	CASE 1	CASE 2	CASE 3
Water temperature (°C)	4	4	4
Base K_{eff}	0.9132	0.9132	0.9191
Biases			
Calculational bias	0.0113	0.0113	0.0113
Depletion allowance	0.0138	0.0138	0.0000
Total Bias	0.0251	0.0251	0.0113
Uncertainties			
Bias Statistics (95%/95%)	0.0017	0.0017	0.0017
KENO Statistics (95%/95%)	0.0012	0.0012	0.0014
Manufacturing Tolerance	0.0057	0.0057	0.0057
Burnup (50 MWD/KgU - Case 3)	0.0000	0.0000	0.0024
Uncertainties (stat. comb.)	0.0061	0.0061	0.0066
Maximum K_{eff}	0.9444	0.9444	0.9370

Maximum K_{eff} = Base K_{eff} + Bias + Uncertainties (Statistical combination)

Statistical combination of uncertainties is the square root of the sum of the squared uncertainties.

WBN SPENT FUEL STORAGE RACKS

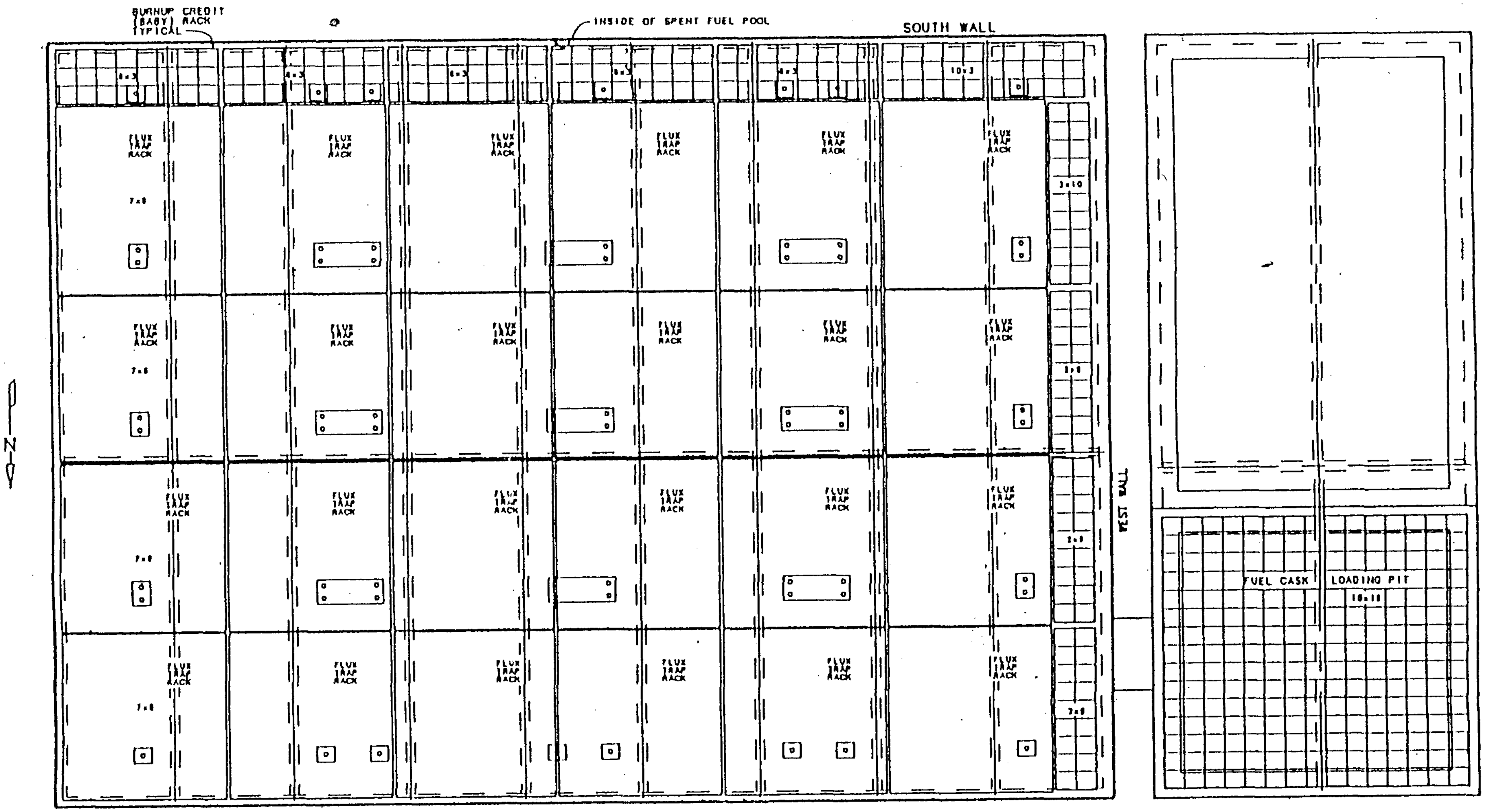


FIGURE 4.1.1

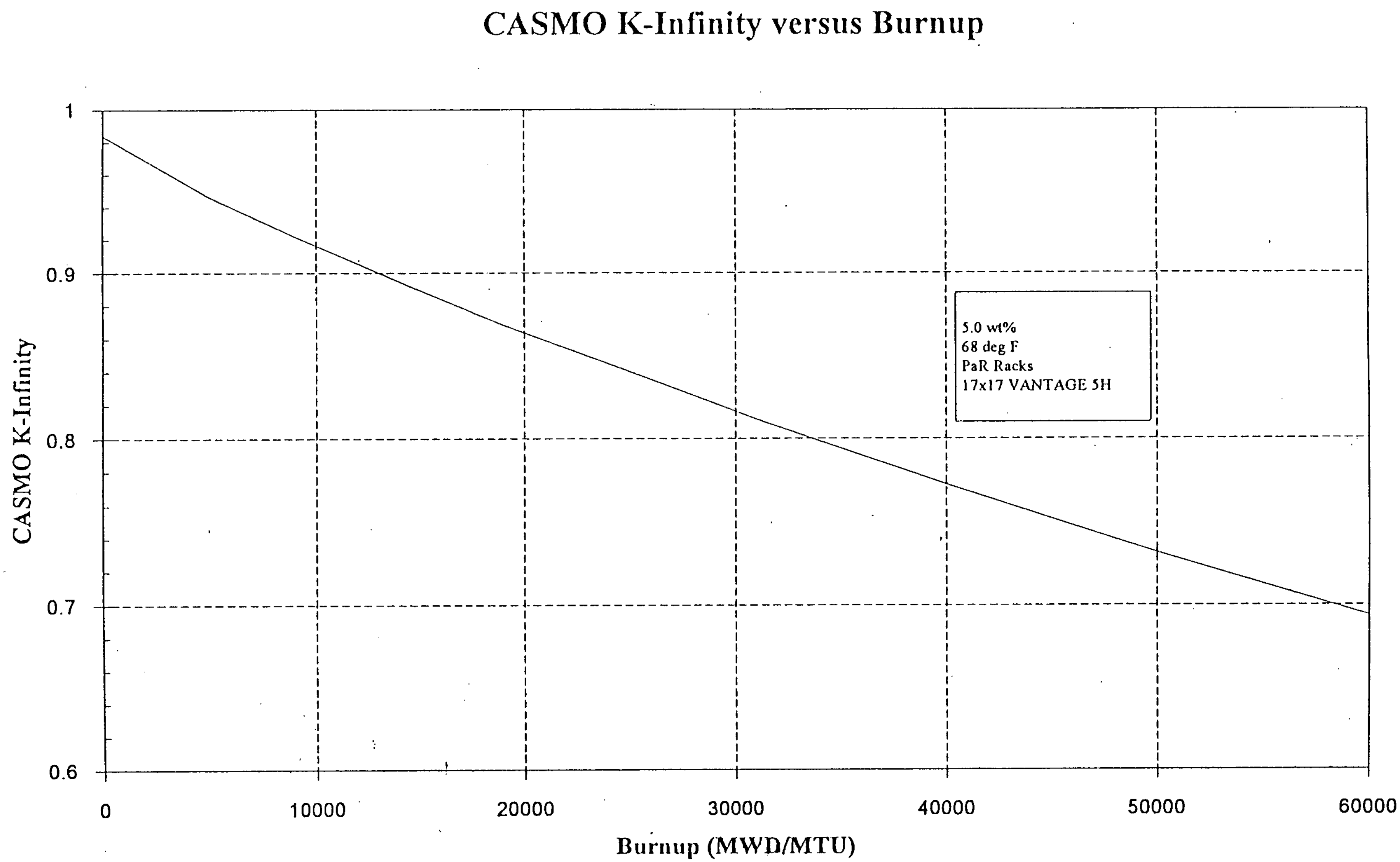

CASMO K-Infinity versus Burnup
CASMO K-Infinity
1
0.9
0.8
0.7
0.6
0
10000
20000
30000
40000
50000
60000
Burnup (MWD/MTU)
5.0 wt%
68 deg F
PaR Racks
17x17 VANTAGE 5H

FIGURE 4.1.2

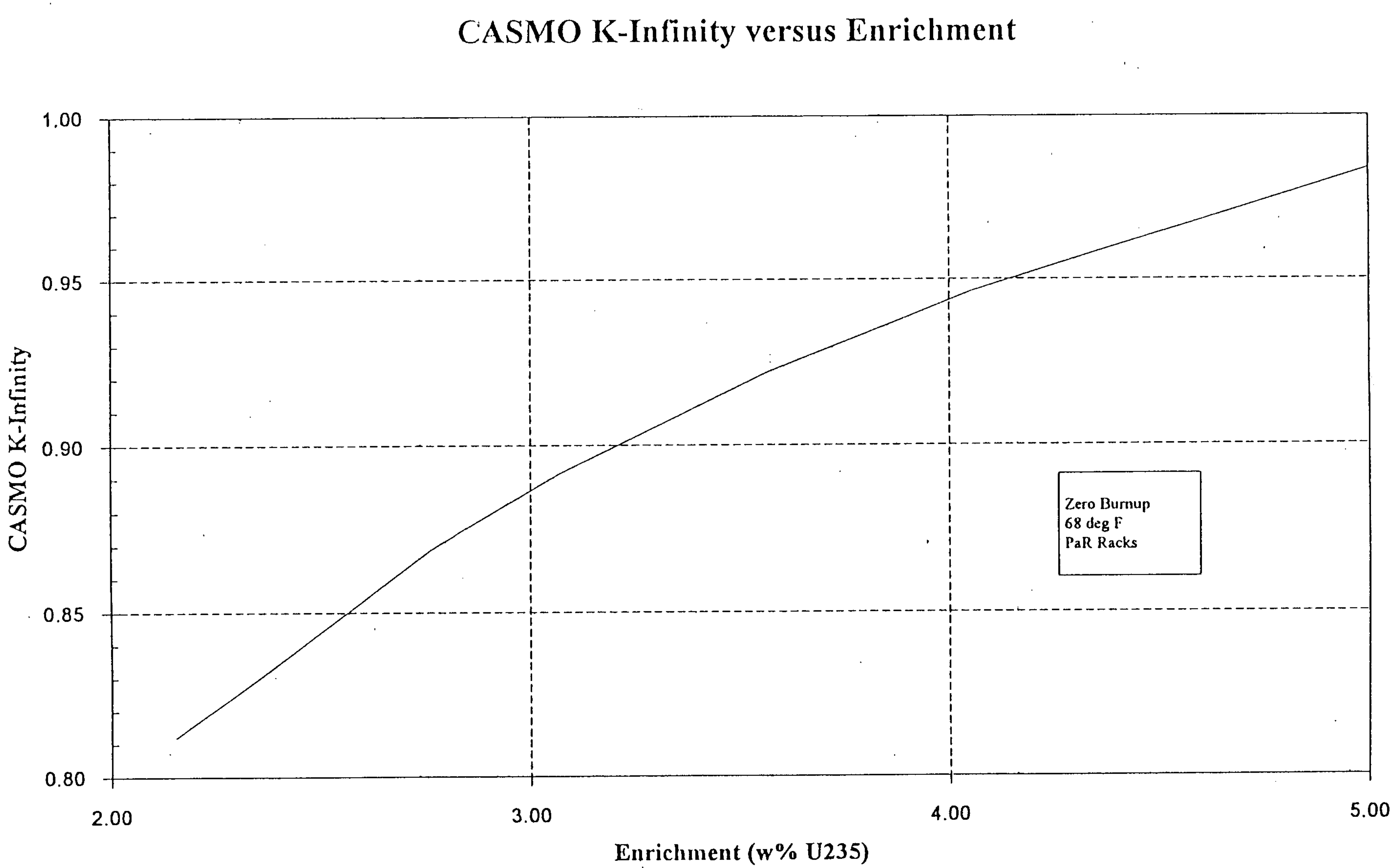

CASMO K-Infinity versus Enrichment
1.00
0.95
0.90
0.85
0.80
CASMO K-Infinity
2.00
3.00
4.00
5.00
Enrichment (w% U235)
Zero Burnup
68 deg F
PaR Racks

FIGURE 4.1.3

REGION 1: REACTIVITY vs WATER DENSITY

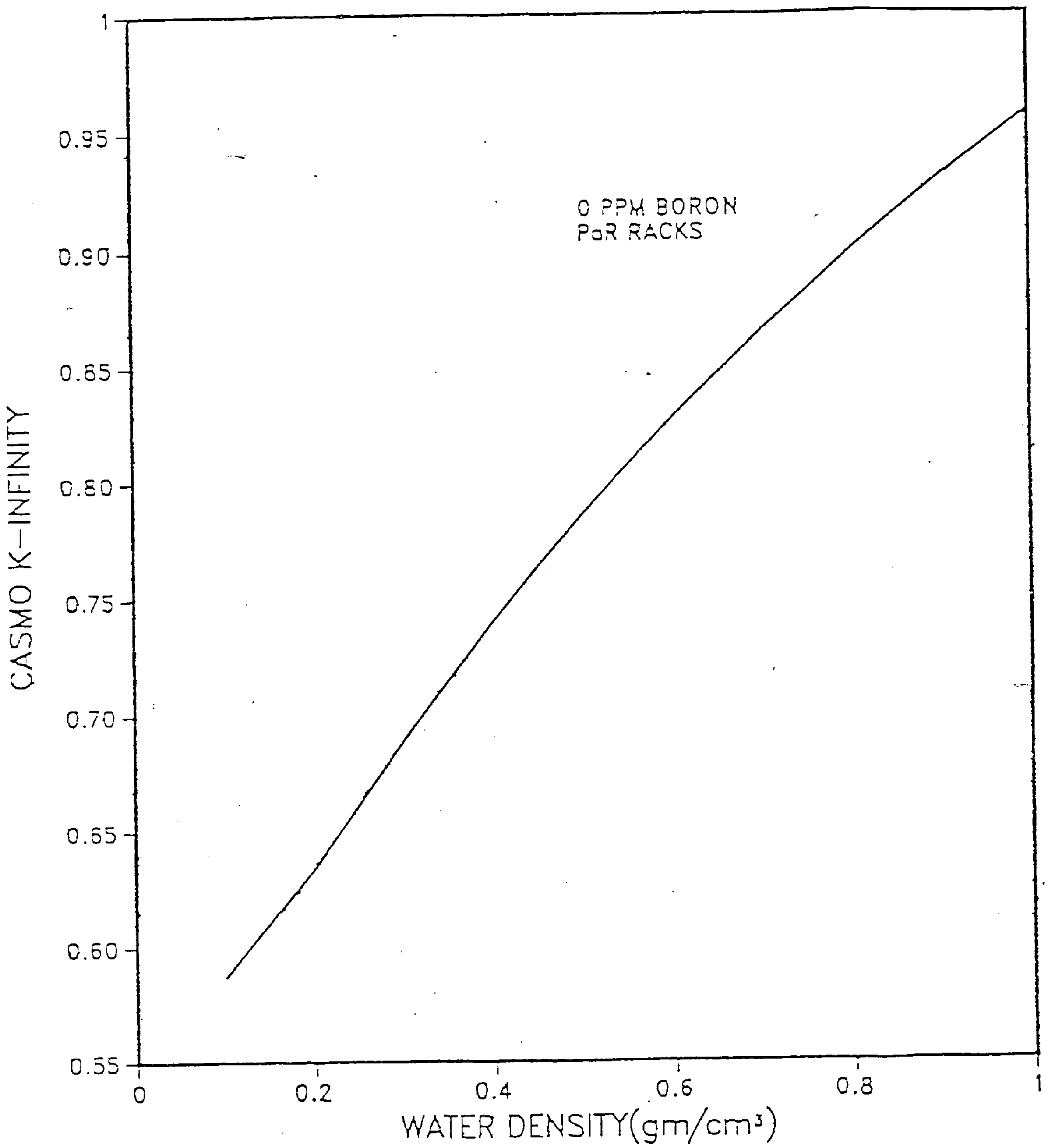


FIGURE 4.3.1

HOLTEC FUEL STORAGE CELL CROSS SECTION

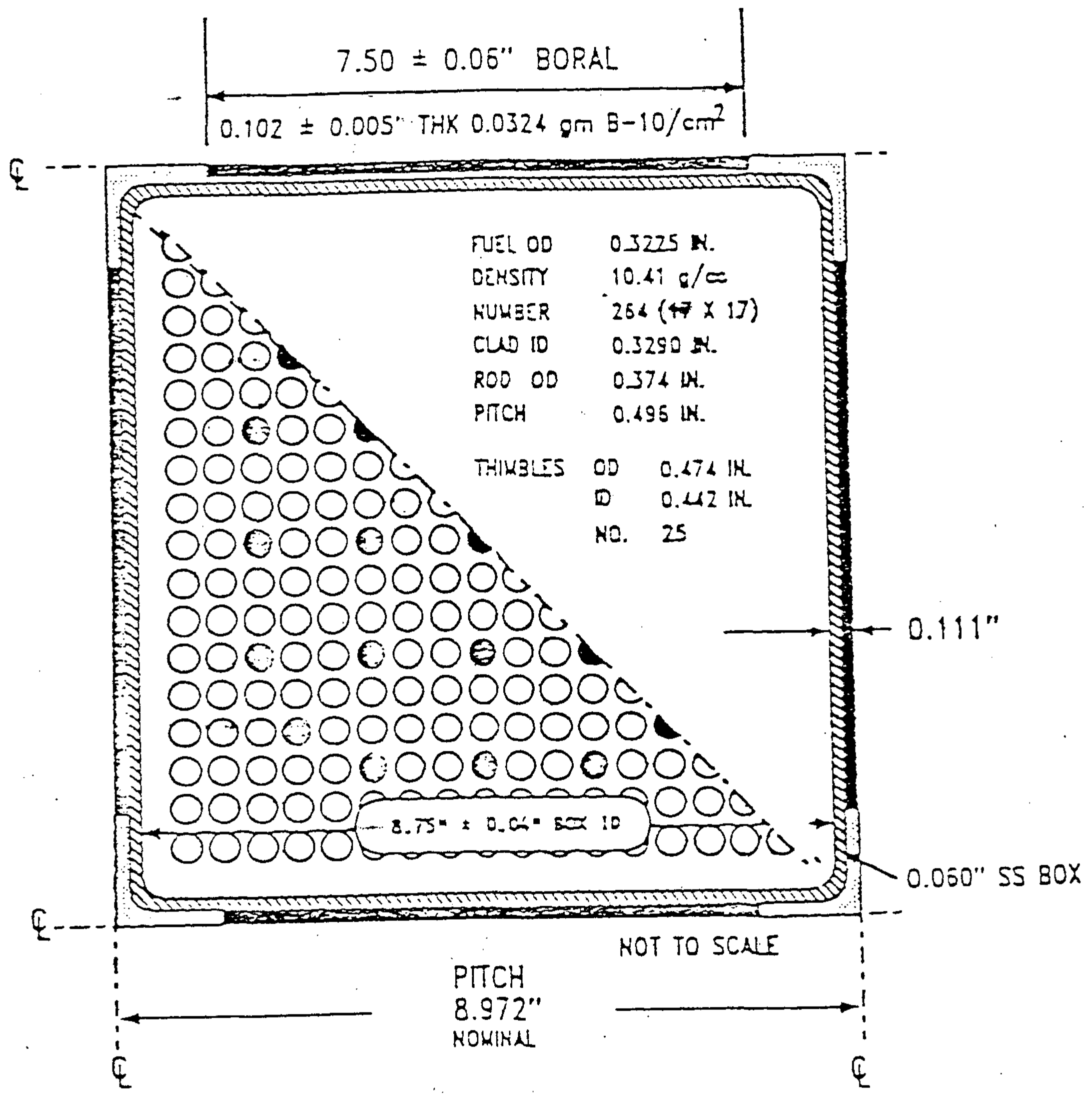


FIGURE 4.4.1

ACCEPTABLE BURNUP DOMAIN IN WATTS BAR (HOLTEC) SPENT FUEL STORAGE RACKS

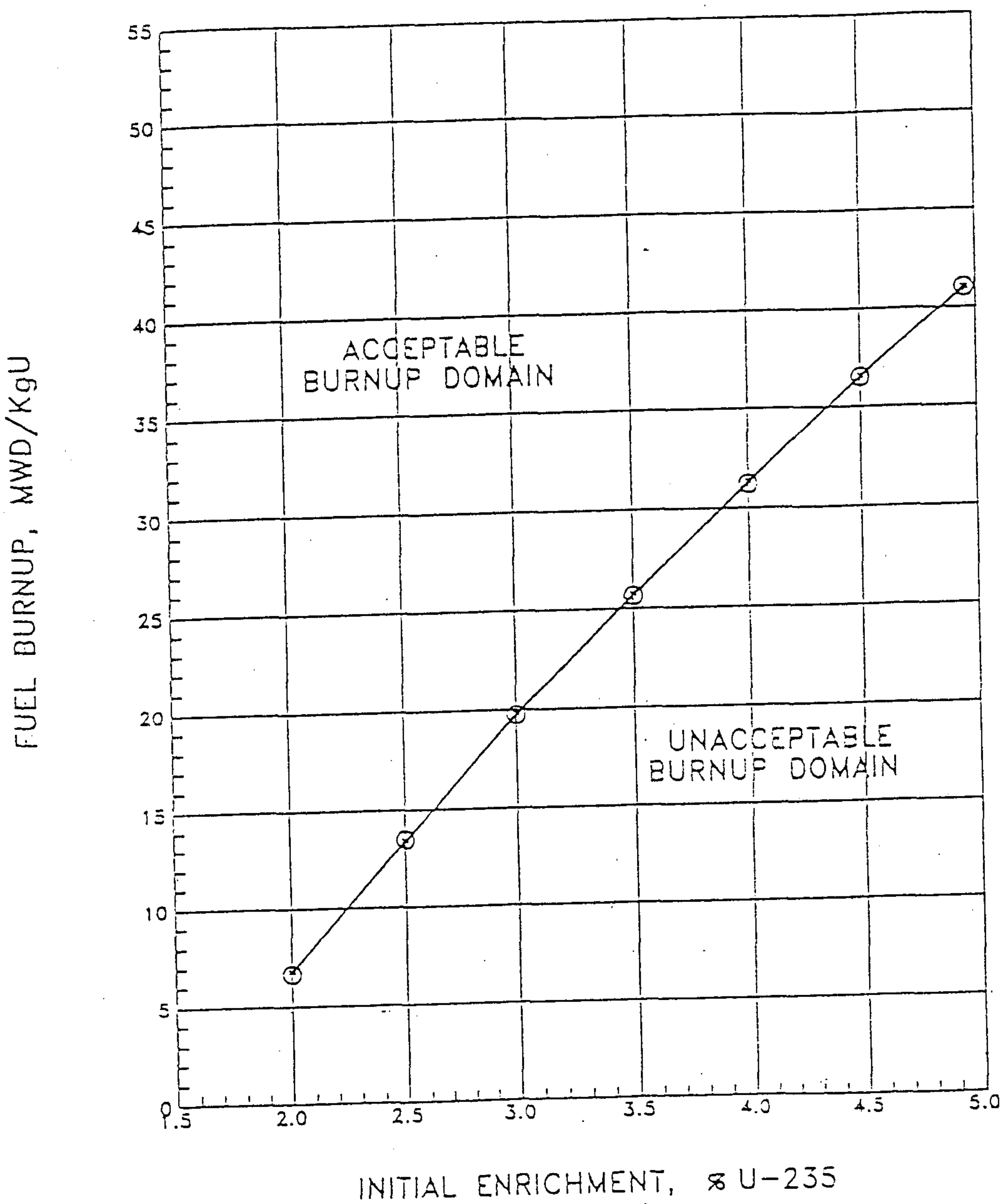


FIGURE 4.5.1

CHAPTER 5

THERMAL-HYDRAULIC CONSIDERATIONS

5.0 THERMAL-HYDRAULIC CONSIDERATIONS

5.1 INTRODUCTION

This section provides a summary of the methods, models, analyses and numerical results to demonstrate the compliance of the reracked WBN spent fuel pool with the provisions of Section III of the USNRC "OT Position Paper for Review and Acceptance of Spent Fuel Storage and Handling Applications," (April 14, 1978).

Similar methods of thermal-hydraulic analysis have been used in previous licensing efforts on high density spent fuel racks for Sequoyah (Docket 50-327 and 50-328), Fermi 2 (Docket 50-341), Quad Cities 1 and 2 (Dockets 50-254 and 50-265), Rancho Seco (Docket 50-312), Grand Gulf Unit 1 (Docket 50-416), Oyster Creek (Docket 50-219), Virgil C. Summer (Docket 50-395), Diablo Canyon 1 and 2 (Docket Nos. 50-275 and 50-323), Byron Units 1 and 2 (Docket 50-454, 455), St. Lucie Unit One (Docket 50-335), Millstone I (50-245), Vogtle Unit 2 (50-425), Kuosheng Units 1 & 2 (Taiwan Power Company), Ulchin Unit 2 (Korea Electric Power Company), J. A. FitzPatrick (New York Power Authority), TMI Unit 1 (GPU Nuclear), Donald C. Cook Units 1 & 2 (Indiana & Michigan Electric Company), and Zion Units 1 & 2 (Commonwealth Edison Company).

The analyses to be carried out for the thermal-hydraulic qualification of the rack array may be broken down into the following categories:

(i) Pool decay heat evaluation and pool bulk temperature variation with time.

(ii) Determination of the maximum pool local temperature at the instant when the bulk temperature reaches its maximum value.

(iii) Evaluation of the maximum fuel cladding temperature to establish that bulk nucleate boiling at any location in the vicinity of the fuel assembly does not occur.

(iv) Evaluation of the time-to-boil if all heat rejection paths through the cooling and cleanup system are lost.

(v) Compute the effect of a blocked fuel cell opening and removal of sections of the existing sparger piping on the local water and maximum cladding temperature.

The following sections present a brief outline of the cooling system, a synopsis of the methods employed to perform the thermal-hydraulic analyses, and a final summary of the results.

5.2 SPENT FUEL POOL COOLING AND CLEANING SYSTEM

The Spent Fuel Pool Cooling and Cleaning System (SFPCCS) is designed to remove from the Spent Fuel Pool (SFP) water the decay heat generated by stored spent fuel assemblies. Additional functions of the SFPCCS are to clarify and purify the water in the SFP, transfer canal, and refueling water storage tank. The SFPCCS is described in Section 9.1.3 of the FSAR.

System piping is arranged so that failure of any pipeline cannot drain the SFP below a water level of ten feet or more above the top of the stored spent fuel assemblies.

The system's demineralizer and filter are designed to provide adequate purification to permit unrestricted access to the spent fuel storage area for plant personnel and maintain optical clarity of the SFP water. The optical clarity of the SFP water surface is maintained by use of the system's skimmers, strainer, and skimmer filter.

5.2.1 System Description

The SFPCCS consists of six pumps (three in the cooling loop, two in the refueling water purification loop, and one in the skimmer cleaning loop), two heat exchangers, four filters, (one in the SFP cleaning loop, two in the refueling water purification loop, and one in the skimmer cleaning loop), one spent fuel pit demineralizer, three strainers (two in the cooling loop and one in the skimmer loop), piping and associated valves and instrumentation necessary for safe operation.

When the SFPCCS is in operation, water flows from the SFP to both SFP pump suctions, is pumped through the tubeside of the heat exchangers, and is returned to the pit. Each pump's suction line, which is protected by a strainer, is located at an elevation four feet below the normal SFP water level, while the return line contains an anti-siphon hole near the surface of the water to prevent gravity drainage of the pit.

While the heat removal operation is in process, a portion of the SFP water may be diverted through a demineralizer and a filter to maintain SFP water clarity and purity. This purification loop is sufficient for removing fission products and other contaminants which may be introduced if a fuel assembly with defective cladding is transferred to the SFP.

The SFP demineralizer and filter can be isolated manually from the heat removal portion of the SFPCCS. The demineralizer is used to clean and purify the refueling water while SFP heat removal operations continue. Connections are provided in the isolated loop such that the refueling water can be pumped either from the transfer canal, the refueling cavities, or the RWST through the demineralizer and purification filters. Refueling water is discharged into the transfer canal, the refueling cavities, or the RWST.

To further assist in maintaining SFP water clarity, the water surface is cleaned by a skimmer loop. Water is removed from the surface by the skimmers, pumped through a strainer and filter, and returned to the pool surface at three locations remote from the skimmers.

The SFP is initially filled with water that is at the same boron concentration as that in the refueling water storage tank. Borated water may be supplied from the refueling water storage tank via the refueling water purification pump connection, or by running a temporary line from the boric acid blender, located in the chemical and volume control system directly into the pit. Demineralized water can also be added for makeup purposes (i.e., to replace evaporative losses) through a connection in the recirculation return line.

The SFP water may be separated from the water in the transfer canal by a gate. The gate is installed so that the transfer canal may be drained to allow maintenance of the fuel transfer equipment. The water in the transfer canal is pumped, via a refueling water purification pump, into a holdup tank in the chemical and volume control system. When maintenance on the fuel transfer equipment is completed, the water is returned to the transfer canal by the holdup tank recirculation pump.

5.2.2 Component Description

SPENT FUEL POOL PUMPS

The pumps are horizontal, centrifugal units. They circulate SFP water through the heat exchangers, demineralizer, and filter. The pumps are controlled manually from a local station. A third pump is available to serve as a backup to either of the two pumps normally used for cooling the SFP water.

SPENT FUEL POOL SKIMMER PUMP

This horizontal, centrifugal pump circulates surface water through a strainer and a filter and returns it to the pool.

REFUELING WATER PURIFICATION PUMPS

These horizontal, centrifugal pumps are used to circulate water from the transfer canal, the refueling cavity and the refueling water storage tank through the SFP demineralizer and a refueling water purification filter. The pumps are operated manually from a local station.

SPENT FUEL POOL HEAT EXCHANGERS

The SFP heat exchangers are of the shell and U-tube type with the tubes welded to the tubesheet. Component cooling water circulates through the shell, and SFP water circulates through the tubes.

SPENT FUEL POOL DEMINERALIZER

This flushable, mixed-bed demineralizer is designed to provide adequate fuel pool water purity for unrestricted access by plant personnel to the SFP working area while maintaining visual clarity.

SPENT FUEL POOL FILTER

The SFP filter is designed to improve the pool water clarity by removing particles which obscure visibility.

SPENT FUEL POOL SKIMMER FILTER

The SFP skimmer filter is used to remove particles which are not removed by the skimmer strainer.

REFUELING WATER PURIFICATION FILTERS

The refueling water purification filters are designed to improve the clarity of the refueling water in the refueling canal or in the RWST by removing particles which obscure visibility.

SPENT FUEL POOL STRAINER

A strainer is located in each SFP pump suction line for removal of relatively large particles which might otherwise clog the SFP demineralizer or damage the SFP pumps.

SPENT FUEL POOL SKIMMER STRAINER

The SFP skimmer strainer is designed to remove debris from the skimmer process flow.

SPENT FUEL POOL SKIMMERS

Two SFP skimmers are provided to remove water from the SFP water surface in order to remove floating debris.

VALVES

Manual stop valves are used to isolate equipment and manual throttle valves provide flow control. Valves in contact with SFP water are austenitic stainless steel or equivalent corrosion resistant material.

PIPING

Piping in contact with SFP water is austenitic stainless steel. The piping is welded except where flanged connections are used to facilitate maintenance.

5.2.3 System Redundancy

Suitable redundancy of components is provided to assure that the SFPCCS can perform its safety functions following the failure of any active component. The SFP is served by two heat removal trains. The SFPCCS contains three identical pumps (A-A, B-B, and C-S) with motors to circulate pool cooling water. One pump is required for normal operation with a second pump and cooling train available to reduce temperatures if needed. Pumps A-A and B-B are trained. A third pump (C-S) is a spare and can be aligned to either cooling train and can be powered from either train. Electrical power is supplied from emergency power buses to each of the SFP pumps. The emergency power buses are powered from separate diesel generator sets, should offsite power be lost.

Heat from the SFP heat exchangers is removed by the component cooling system (CCS). The CCS is designed such that any failure in one CCS train will not affect the capability of the other train to provide the necessary cooling to safety equipment in that train. There are five CCS pumps located in the Auxiliary Building. Pumps 1A-A and 2A-A are Train A powered and are aligned to Unit 1 and Unit 2 (non-safety related equipment), respectively. Pumps 1B-B and 2B-B are electrically powered from Train B, but are aligned to the Train A headers to supply additional capacity for nonessential loads when the residual heat removal (RHR) heat exchangers are in service. However, they can be realigned to Train B headers if needed. Pump C-S, which is usually aligned to Train B headers, is a swing pump. It can be powered from Train B or A and can be valved to any of the three supply headers. An SI signal will start the 1A-A and 1B-B pumps. Pump C-S also starts on an SI signal.

5.3 DECAY HEAT LOAD CALCULATIONS

The decay heat loads were calculated using the ORIGEN-2 computer code,[1] from the Radiation Shielding Information Center at the Oak Ridge National Laboratory. The ORIGEN-2 code is used extensively in the United States and international nuclear power industries to perform both radiation and thermal power calculations. Version 1.0 of the computer program LONGOR,[2] incorporating the ORIGEN-2 code, was used to perform the analysis. For normal discharge scenarios a total of 1,680 assemblies are stored in the SFP (21 batches x 80 assemblies per batch) at the start of the final discharge. One additional full core discharge (193 assemblies) would result in a total SFP inventory of 1,873 assemblies, which slightly exceeds the capacity of the pool (1,835 assemblies). For the unplanned discharge scenarios a total of 1,600 assemblies are stored in the SFP (20 batches x 80 assemblies per batch) at the start of the 21st reactor discharge. One additional 80 assembly discharge, followed by a full core discharge (193 assemblies) would result in a total SFP inventory of 1,873 assemblies, which slightly exceeds the capacity of the pool. A burnup of 48,000 MWD/MTU is used for previously discharged fuel assemblies. The cumulative decay heat load is computed for the instant corresponding to the beginning of fuel transfer for discharge number 21. The ratio of the cumulative decay heat load due to the inventory of previously stored fuel to the average assembly operating power, β is calculated to be 0.094 for a full core discharge and 0.092 for the unplanned discharge case (Table 5.3.1).

In the interest of conservatism, this decay heat load from previously discharged fuel is assumed to remain constant for the duration of the pool temperature evaluations performed in the wake of normal and unplanned full core offloads discussed below.

5.4 DISCHARGE SCENARIOS

The following discharge scenarios were examined to establish a conservative design basis heat load for the SFP:

Case 1: Normal Full Core Discharge

As shown in Figure 5.4.1, the entire core (193 fuel assemblies) from Unit 1 is transferred to the pool after twelve days of decay in the reactor. The total fuel transfer rate is assumed to be four assemblies per hour (48.25 hours for 193 bundles). The analysis does not extend beyond 600 hours (25 days) for the normal discharge scenarios, therefore, fuel reload is not explicitly modeled. However, Figure 5.4.1 depicts 113 previously discharged assemblies plus 80 new assemblies being reloaded into the reactor approximately 30 days after off load at a rate of four per hour. The normal discharge scenarios consist of a full 193 assembly core discharge with the following burnup distribution.

48,000 MWD/MTU	65 assemblies
32,000 MWD/MTU	64 assemblies
16,000 MWD/MTU	64 assemblies

Two discrete analyses have been performed for this case assuming two cooling trains in operation and one cooling train in operation, respectively. We denote these two evaluations as Case 1A and Case 1B.

Case 2: Unplanned Full Core Offload

The unplanned discharge scenarios consist of an 80 assembly discharge batch with an average burnup of 48,000 MWD/MTU followed by a full core (193) assembly discharge with the above presented burnup distribution. The start of the second discharge is assumed to be 36 days (864 hours) after the start of the first discharge (see Figure 5.4.2). The analysis extends for approximately 67 days (1600 hours) for the unplanned discharge cases. Two analyses were also performed for the unplanned full core offload assuming two cooling trains in operation and one cooling train in operation. These two evaluations are denoted as Case 2A and Case 2B, respectively.

Detailed data for the two foregoing discharge scenarios are given in Tables 5.4.1 and 5.4.2.

5.5 BULK POOL TEMPERATURE

In order to perform the analysis conservatively, the heat exchangers are assumed to be fouled to their design maximum and to have had 5 percent of their tubes plugged. Thus, the temperature effectiveness, p, for the heat exchanger utilized in the analysis is the lowest postulated value calculated from heat exchanger thermal hydraulic codes. The temperature effectiveness is given by the equation:

$$p = \frac{T_{c,0} - T_{c,i}}{T_{h,i} - T_{c,i}} \qquad (5-1)$$

where:

$T_{c,i}$ is the coolant (shellside) inlet temperature, °F.
$T_{c,o}$ is the coolant (shellside) outlet temperature, °F.
$T_{h,i}$ is the SFP water (tubeside) inlet temperature, °F.

It is seen that at a constant collant temperature the thermal performance of the heat exchanger is a function of the SFP water temperature. Version 5.03 of the Q.A. validated shell-and-tube heat exchanger rating program STER[6] is used to determine the terminal temperatures of the SFPCCS heat exchangers at SFP bulk water temperatures of 100°F, 120°F, 140°F, and 160°F. The calculated terminal temperatures are then used in Equation 5-1 to determine the corresponding heat exchanger temperature effectiveness. This temperature effectiveness gradient becomes an input for subsequent analyses.

The mathematical formulation can be explained with reference to the simplified heat exchanger alignment of Figure 5.5.1. Referring to the spent fuel pool cooling system, the governing differential equation can be written by utilizing conservation of energy:

$$C \frac{dT}{d\tau} = Q_{cons} + Q(\tau) - Q_{EV}(T, t_a) - Q_{HX} \qquad (5\text{-}2)$$

where:

C	Thermal capacity of the pool (net water volume times water density and times heat capacity), Btu/°F.
Q_{cons}	Heat generation from previously discharged fuel Btu/hr.
$Q(\tau)$	Decay heat generation rate from recently discharged fuel, which is a specified function of time, τ, Btu/hr.
τ	Time after reactor shutdown, hrs.
Q_{HX}	Heat removal rate by the heat exchanger, Btu/hr.
$Q_{EV}(T, t_a)$	Heat loss to the surroundings, which is a function of pool temperature T and the building ambient air temperature t_a, Btu/hr.
T	SFP bulk water temperature, °F

Q_{HX} can be written in terms of effectiveness p as follows:

$$Q_{HX} = W_t C_t p (t_{h,i} - t_{c,i}) \qquad (5\text{-}3)$$

where:

W_t	Coolant flow rate, lb/hr
C_t	Coolant specific heat, BTU/(lb - °F).
p	Temperature effectiveness of heat exchanger.
$t_{h,i}$	SFP water temperature at the heat exchanger inlet, °F.

$t_{c,i}$ Coolant water inlet temperature, °F.

$Q(\tau)$ is the total heat generation rate from the newly discharged fuel assemblies in the pool. $Q(\tau)$ increases as additional assemblies are transferred to the pool and reaches its maximum value at the instant when the last bundle is transferred. After that, $Q(\tau)$ decreases monotonically with time. $Q(\tau)$ is determined using the ORIGEN-2 computer code as incorporated into version 2.0 of the BULKTEM[4] computer code.

Q_{EV} is a non-linear function of pool temperature and ambient air temperature. Q_{EV} includes the heat evaporation loss through the pool surface, natural convection from the pool surface and heat conduction through the pool walls and slab. Experiments show that the heat conduction takes only about 4 percent of the total heat loss,[3] and therefore, can be conservatively neglected. The evaporation heat and natural convection heat loss can be expressed as:

$$Q_{EV} = m\ \Gamma\ A_s + h_c\ A_s \theta \qquad (5\text{-}4)$$

where:

m	Mass evaporation rate, lb/(hr - ft^2)
Γ	Latent heat of pool water, Btu/lb
A_s	SFP surface area, ft^2
h_c	Convection heat transfer coefficient at SFP surface, Btu/(ft^2 - hr - °F)
$\theta = T - t_a$	The temperature difference between pool water and ambient air, °F

The mass evaporation rate, m, can be obtained as a non-linear function of θ. Therefore,

$$m = h_D(\theta)\,(W_{ps} - W_{as}) \qquad (5\text{-}5)$$

where:

W_{ps}	Humidity ratio of saturated moist air at the SFP surface temperature T.
W_{as}	Humidity ratio of saturated moist air at ambient temperature t_a
$h_D(\theta)$	Mass transfer coefficient at the SFP surface, lb/(hr - ft^2 - °F)

The non-linear single order differential equation (5-2) is solved using Holtec's Q.A. validated numerical integration code BULKTEM.

The initial temperature of the pool water is assumed to correspond to the equilibrium bulk temperature which will exist in the pool in the absence of the newly discharged batch and neglecting evaporation. As is obvious from heuristic reasoning, numerical computations show that the calculated maximum pool water temperature is rather insensitive to the initial pool water temperature value utilized. Therefore, a rigorously accurate value of the initial pool water temperature is not necessary for this analysis.

Figures 5.5.2 through 5.5.5 provide the bulk pool temperature profiles for the discharge scenarios described in Section 5.4. Figures 5.5.6 through 5.5.9 show the transient heat load of Cases 1 and 2. Table 5.5.1 gives the peak water temperature, coincident time, and coincident heat load to the cooler and coincident heat loss to the ambient for these cases.

The next step in the analysis is to determine the temperature rise profile of the pool water if all forced indirect cooling modes are suddenly lost and make-up water is provided with a fire hose.

Clearly, the most critical instant of loss-of-cooling is when pool water temperature has reached its maximum value. It is assumed that cooling water is added through a fire hose at the rate of 55 gallons per minute (gpm) initiated ten hours after loss of cooling. The cooling water is at temperature, t_{cool}. The governing enthalpy balance equation for this condition can be written as

$$[C + G(C_t)(\tau - \tau_o)] \frac{dT}{d\tau} = P_{cons} + Q(\tau+\tau_{ins}) + G(C_t)(t_{cool}-T) - Q_{EV} \qquad (5\text{-}6)$$

where water is assumed to have a specific heat of unity, and the time coordinate τ is measured from the instant maximum pool water temperature is reached.

G is the cooling water added through a fire hose (lbm/hr).

τ_o is the time coordinate when the direct addition (fire hose) cooling water application is begun.

τ_{ins} is the time coordinate measured from the instant of reactor shutdown to when maximum pool water temperature is reached.

T is the dependent variable (pool water temperature).

For conservatism, Q_{EV} is assumed to remain constant after pool water temperature reaches and rises above 170°F.

A numerical quadrature code TBOIL is used to integrate the foregoing equation. The pool water heat up rate, time-to-boil, and subsequent water evaporation-time profile are generated and compiled for safety evaluation.

Assuming no make-up water (G = O), the time-to-boil output results are presented in Table 5.5.2. Figures 5.5.10 through 5.5.13 show the plot of the inventory of water in the pool after loss-of-coolant-to-the-pool condition begins. Figures 5.5.14 through 5.5.17 show the pool water inventory status after loss of spent fuel pool cooling with make-up water added for a normal full core discharge assuming two cooling trains in operation, a normal full core discharge assuming one cooling train in operation, an unplanned full core offload with two cooling trains operating, and an unplanned full core offload with one cooling train operating, respectively, prior to the total loss of cooling. These correspond to the cases discussed in Section 5.4. Makeup water at 100°F is assumed to be added to the pool beginning 10 hours after loss of cooling at a rate of 55 gpm.

It is seen from Table 5.5.2 that sufficient time to introduce manual cooling measures exists and the available time is consistent with other PWR installations.

The time to boil analysis concluded that when make-up water at a flow rate of 55 gpm is established ten hours after loss of cooling, the SFP water level only reaches a level approximately 21 feet above the top of the racks.

5.6 LOCAL POOL WATER AND FUEL CLADDING TEMPERATURES

In this section, a summary of the basis and calculational methodology for local pool water and fuel cladding temperatures is presented.

5.6.1 Basis

The analysis described in Section 5.5 determines the peak SFP bulk water temperature. However, local regions of elevated temperature will exist inside the rack storage cells. The maximum local water temperature and the maximum local fuel cladding temperature are determined to evaluate the possibility of nucleate boiling on the surface of the fuel assemblies. This analysis is performed to show, for any scenario with at least one SFPCCS train available, that localized boiling does not occur within the fuel storage racks.

In order to determine an upper bound on the maximum local pool water and fuel cladding temperatures, a series of conservative assumptions are made. The most important assumptions are listed below:

- No downcomer flow is assumed to exist between the rack modules.

- The Westinghouse 17x17 fuel assembly has been used in the analysis which, from the thermal-hydraulic standpoint, bounds the case of the VANTAGE 5 hrybrid fuel bundles utilized in the WBN reactor.

- No heat transfer is assumed to occur between pool water and the surroundings (wall, etc.).

5.6.2 Model Description

A two-dimensional model of the SFP was created. This model is in the East-West plane, and includes the cask pit and cask pit canal. Version 4.32 of the FLUENT[5] general purpose computational fluid dynamics (CFD) program, which has been benchmarked under Holtec's Q.A. program, is used to perform this evaluation. The FLUENT program is capable of correctly modeling the buoyancy induced thermal siphon flow, with conjugate heat transfer, which is present in the SFP and cask pit. A two-dimensional model of the SFP and cask pit which includes features of the SFP, the truncated sparger and suction piping, the fuel storage racks, the cask pit, and the cask pit canal is created and evaluated using FLUENT. Modifications to actual parameters are made where required to predicate a conservative representation of the phenomena under study.

The fuel storage racks are modeled using FLUENT's porous medium capability. Rather than modeling individual fuel storage racks and fuel assemblies, the corresponding permeability and inertial resistance factors are calculated and the racks are treated as a continuous body. Both flow area reductions and expansion/contraction losses are considered in this method. Use of this modeling method for fuel storage racks results in a smaller model without an appreciable loss of solution accuracy.

The effects of radial peaking on the Spent Nuclear Fuel (SNF) decay heat generation rates are incorporated through the creation of three heating zones as shown in Figure 5.6.1. The first zone, located closest to the cask pit, is the hotter-than-average zone. The second zone, located slightly farther away from the cask pit, is the cooler-than-average zone. The third zone, comprised of the balance of the SFP and the cask pit, is the background zone. Figure 5.6.1 is not to scale.

The hotter-than-average zone is sized to hold 65 of the 193 assemblies from the discharged full core. The decay heat generation rate in this zone is the average core heat generation rate multiplied by the radial peaking factor. The cooler-than-average zone is sized to hold the remaining 128 of the 193 assemblies from the discharged full core. The balance of the full core decay heat generation is assigned to this zone. The background zone contains the decay heat from the existing stored SNF inventory.

The cask pit canal is a narrow channel that separates the SFP from the cask pit. The canal extends from slightly above the top of the fuel storage racks to the surface of the SFP. Because the canal is not as wide as the SFP, it too is modeled using FLUENT's porous medium capability. The inertial resistance of the porous canal accounts for the reduction in flow area as well as the expansion/contraction losses.

Three local temperature calculation scenarios were evaluated. The first two scenarios evaluated the maximum local temperatures with unblocked and partially blocked cells. The third scenario evaluates the effects of off-center placement of an assembly in a rack cell. These scenarios are summarized as follows:

Case 1: No Blockage

This scenario corresponds to the condition where all cells are unblocked. The inertial resistance terms for the rack porous media regions are conservatively determined. This case corresponds to the limiting intact fuel scenario from the transient response evaluation.

Case 2: Partial Blockage

This scenario is identical to Case 1, except that the inertial resistance terms for the rack porous media are multiplied by ten, which would conservatively bound any realistic blockage scenario by a considerable margin.

Case 3: Off-Center Placement

This scenario is performed for the off-center placement of an assembly in a rack cell. The square assembly is placed in the corner of the cell, and no credit is taken for direct conduction to the cell wall. The same heat load conditions from scenarios 1 and 2 are used for this scenario. Unlike the other local temperature models, however, this model uses a chopped-cosine distribution for the axial decay heat distribution along the axis of the assembly.

5.6.3 Cladding Temperature

The maximum specific power of a fuel array q_A can be given by:

$$q_A = q\, F_{xy} \tag{5-7}$$

where:

F_{xy} = radial peaking factor
q = average fuel assembly specific power, Btu/hr

The maximum temperature rise of pool water in the disadvantageously placed fuel assembly is computed for all loading cases. Having determined the maximum local water temperature in the pool, it is now possible to determine the maximum fuel cladding temperature. A fuel rod can produce F_z times the average heat emission rate over a small length, where F_z is the axial rod peaking factor. The axial heat distribution in a rod is generally a maximum in the central region, and tapers off at its two extremities.

The power distribution corresponding to the chopped cosine power emission rate can be written in the standard form as,

$$q(x) = q_A\ F_z \quad \sin \frac{\pi\,(a + x)}{h + 2a} \tag{5-8}$$

where:

h: active fuel length, in
a: chopped length at both extremities in the power curve, in
x: axial coordinate with origin at the bottom of the active fuel region, in
F_z: axial peaking factor

The value of a is given by

$$a = \frac{h\,z}{1 - 2z} \qquad (5\text{-}9)$$

where:

$$z = \frac{1}{\pi F_z} - \left[\frac{1}{\pi^2 - F_z^2} - \frac{1}{\pi F_z} + \frac{2}{\pi^2} \right]^{1/2} \qquad (5\text{-}10)$$

For an infinitesimally small axial length section, the power distribution equation above reduces to:

$$q(x) = qA\,F_z$$

The temperature difference between the fuel cladding and the axially flowing water (ΔT) can be determined from the heat transfer relationship:

$$q(x) = h\,\Delta T$$

where h is the laminar flow heat transfer coefficient. A bounding maximum fuel cladding temperature is determined by adding the temperature difference ΔT to the maximum local water temperature. Temperatures calculated in this manner bound the temperatures that occur in the pool.

The maximum fuel cladding temperature was calculated using the results of the FLUENT CFD local water temperature analysis and principles of laminar flow heat transfer.

Table 5.6.1 provides the key input data for local temperature analysis. The results of maximum local pool water temperature and maximum local fuel cladding temperature are presented in Table 5.6.2.

The local saturation temperature at the top of the racks (240.73°F) is greater than any calculated local water temperature (see Table 5.6.2), which precludes the possibility of nucleate boiling. Additionally, the local saturation temperature is greater than any calculated fuel cladding temperature, which would preclude the possibility of film boiling at the surface of the fuel rods.

Finally, it is noted that the fuel cladding temperature is considerably lower than the temperature to which the cladding is subjected inside the reactor. Therefore, it is concluded that there is sufficient margin against fuel cladding failure in the SFP.

5.6.4 Partially Blocked Cell Analysis

The effects of partial blockage of a cell was evaluated by increasing the inertial resistance of the fuel storage racks in the FLUENT CFD analysis. Increasing the inertial resistance evaluates the effects of partially blocking all storage locations in the SFP. The effects of off-center placement of a fuel assembly were also evaluated. The corresponding maximum local pool water temperature and local fuel cladding temperature data are also presented in Table 5.6.2. It is seen while both the local pool water and fuel cladding temperature sustain a slight increase, considerable margin against localized boiling remains. Under no evaluated scenario does the condition of localized nucleate boiling of the pool water or potential for fuel cladding damage occur for the WBN pool.

5.7 REFERENCES

1. A. G. Croft, "ORNL Isotope Generation and Depletion, A User's Manual for the ORIGEN-2 Computer Code," ORNL/TM-7175, RSIC/CCC-371, Oak Ridge National Laboratory, July, 1980.

2. Holtec Report HI-951390, "QA Documentation for LONGOR," Revision 0.

3. Wang, Yu, "Heat Loss to the Ambient from Spent Fuel Pools: Correlation of Theory with Experiment," Holtec Report HI-90477, Revision. 0, April 3, 1990.

4. Holtec Report HI-951391, "QA Documentation for BULKTEM," Revision 0.

5. Holtec Report HI-961444, "QA Documentation for FLUENT," Revision 0.

6. Holtec Report HI-92776, "QA Documentation for STER," Revision 10.

TABLE 5.3.1

FUEL SPECIFIC POWER AND POOL CAPACITY DATA

Total Net Water Volume of Pool: (includes the SFP, cask pit , and cask pit canal).	49,790 ft^3
Average Operating Power per Fuel Assembly, P_0:	60.32 x 10^6 Btu/hr
Dimensionless Decay Power of "Old" Discharges Normal Refueling Outage - (Full Core Discharge), β:	0.094
Dimensionless Decay Power of "Old" Discharges (Unplanned Discharge), β:	0.092

TABLE 5.4.1

DATA FOR DISCHARGE CASES 1 AND 2

Parameter	Sub-case A	Sub-case B
Number of SFPCCS Trains	2	1
Pool Thermal Capacity (MBtu/F)	3.05	3.05
CCS Inlet Temperature (°F)	95	95
CCS Flow Rate per Train (1000 lb/hr)	1490	1490
Total SFP Water Flow Rate (gpm)	4600	2300
Temperature Effectiveness	0.620@100°F 0.628@120°F 0.634@140°F 0.640@160°F	0.310@100°F 0.314@120°F 0.317@140°F 0.320@160°F

TABLE 5.4.2

DATA FOR DISCHARGE CASES 1A THROUGH 2B

Case ID	Discharge Scenario	Number of Assemblies Recently Discharged	Transfer Start Time (hours after initial shutdown)	Fuel Transfer Time (hrs)	Maximum Burnup per Batch (MWD/MTU)
1A	Full Core	193	288	48.25	48,000
1B	Full Core	193	288	48.25	48,000
2A	Unplanned	80 + 193	288 & 1152	16 & 48.25	48,000 48,000
2B	Unplanned	80 + 193	288 & 1152	16 & 48.25	48,000 48,000

TABLE 5.5.1

POOL BULK TEMPERATURE AND HEAT LOAD DATA

Case ID	Max. Pool Bulk Temperature (°F)	Coincident Net Decay Heat Load (Mbtu/hr)	Coincident Time (hours after initial shutdown)	Coincident Evaporative Heat Loss (Mbtu/hr)
1A	124.69	27.843	345	0.256
1B	151.17	26.675	352	1.225
2A	129.30	32.240	1209	0.360
2B	159.24	30.618	1215	1.799

TABLE 5.5.2

TIME-TO-BOIL CALCULATIONS RESULTS
(with no make-up water)

Case Identifier	Time-to-Boil (hours)	Time-to-Ten Feet (hours)	Maximum Boil-Off Rate (gpm)	Average Heatup Rate (°F/hr)
1A	8.84	47.4	68.59	9.88
1B	6.27	45.0	68.32	9.70
2A	8.99	50.1	64.22	9.20
2B	5.86	47.1	64.08	9.00

TABLE 5.6.1

DATA FOR LOCAL TEMPERATURE

Type of Fuel Assembly:	PWR
Fuel Cladding OD (inches):	0.374
Fuel Cladding ID (inches):	0.329
Storage Cell ID (inches):	8.80
Active Fuel Length (inches):	144
Number of Rods per Assembly:	264
Assembly Operating Power (Mbtu/hr):	60.32
Cell Pitch (inches):	8.972
Cell Height (inches):	172
Bottom Plenum Height (inches):	5.25
Radial Bundle Peaking Factor:	1.65
Total Bundle Peaking Factor:	2.50

TABLE 5.6.2

MAXIMUM LOCAL TEMPERATURE CALCULATIONS RESULTS

Parameter	No Blockage	Partially Blocked	Off-Center
Local Maximum Water Temperature (°F)	193.7	204.1	195.2
Mid-Height Fuel Cladding Temperature (°F)	208.2	217.1	208.9
*Bounding Maximum Cladding Temperature (°F)	221.5	231.9	223.1

*NOTE: The decay heat flux of the fuel rods is greatest at the fuel mid-height. If the clad superheat at the fuel mid-height position is applied to the maximum water temperature (at the outlet of the rack cell), then this bounding maximum temperature is obtained. Using a chopped-cosine distribution for the heat flux along the fuel rods this value can never be reached.

FULL CORE DISCHARGE SCENARIO FUEL INVENTORY PROFILE

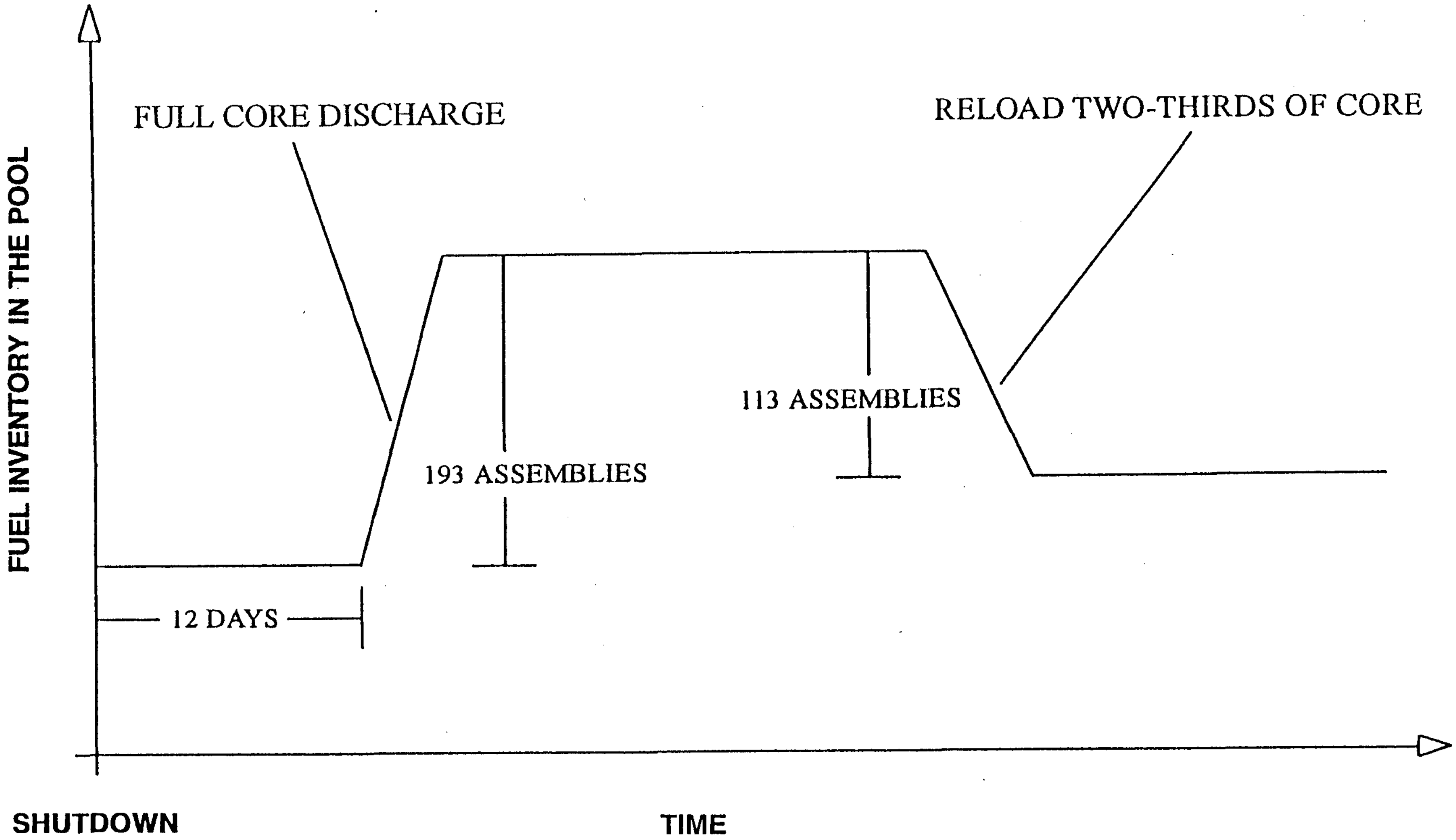


FIGURE 5.4.1

UNPLANNED DISCHARGE SCENARIO FUEL INVENTORY PROFILE

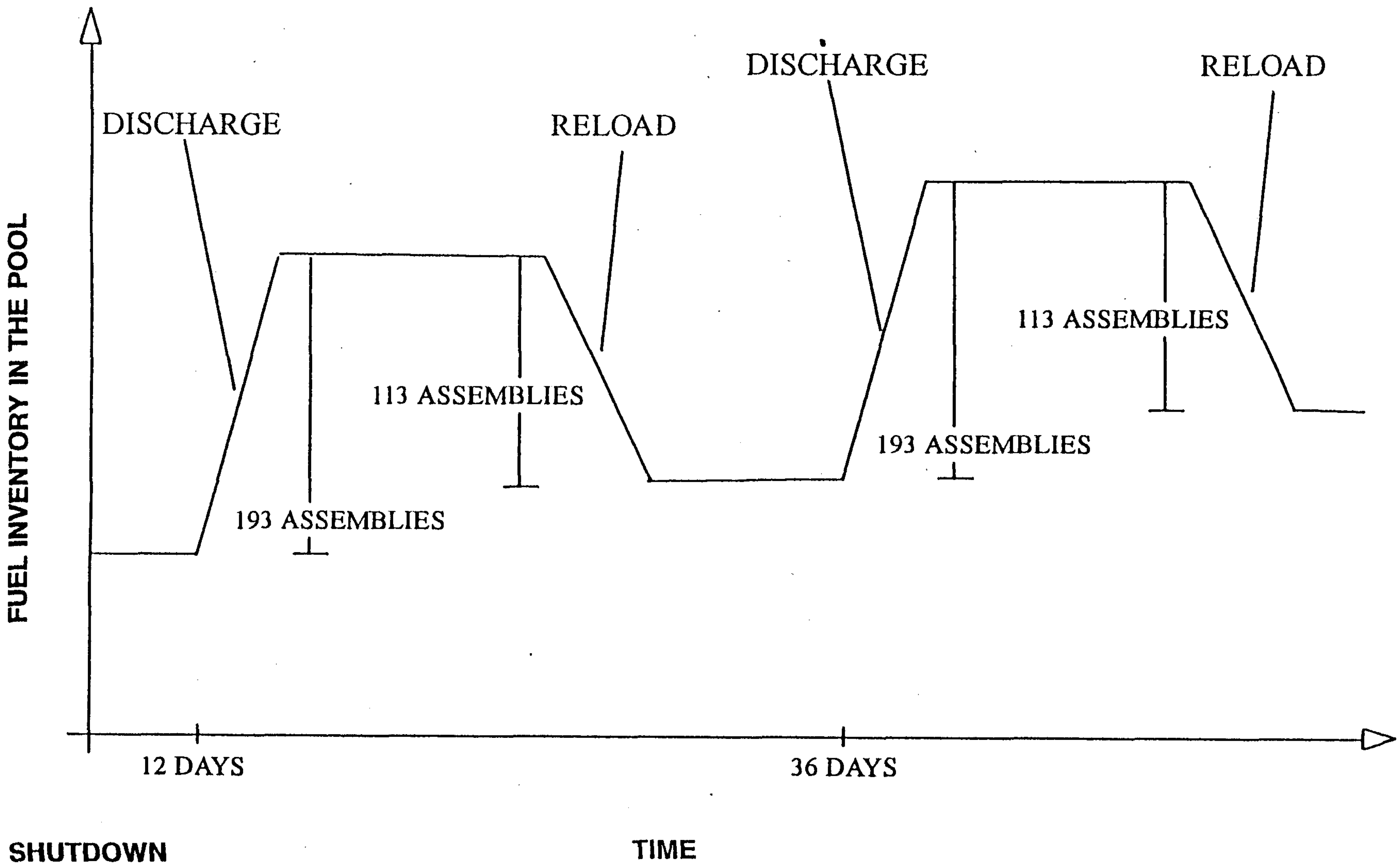


FIGURE 5.4.2

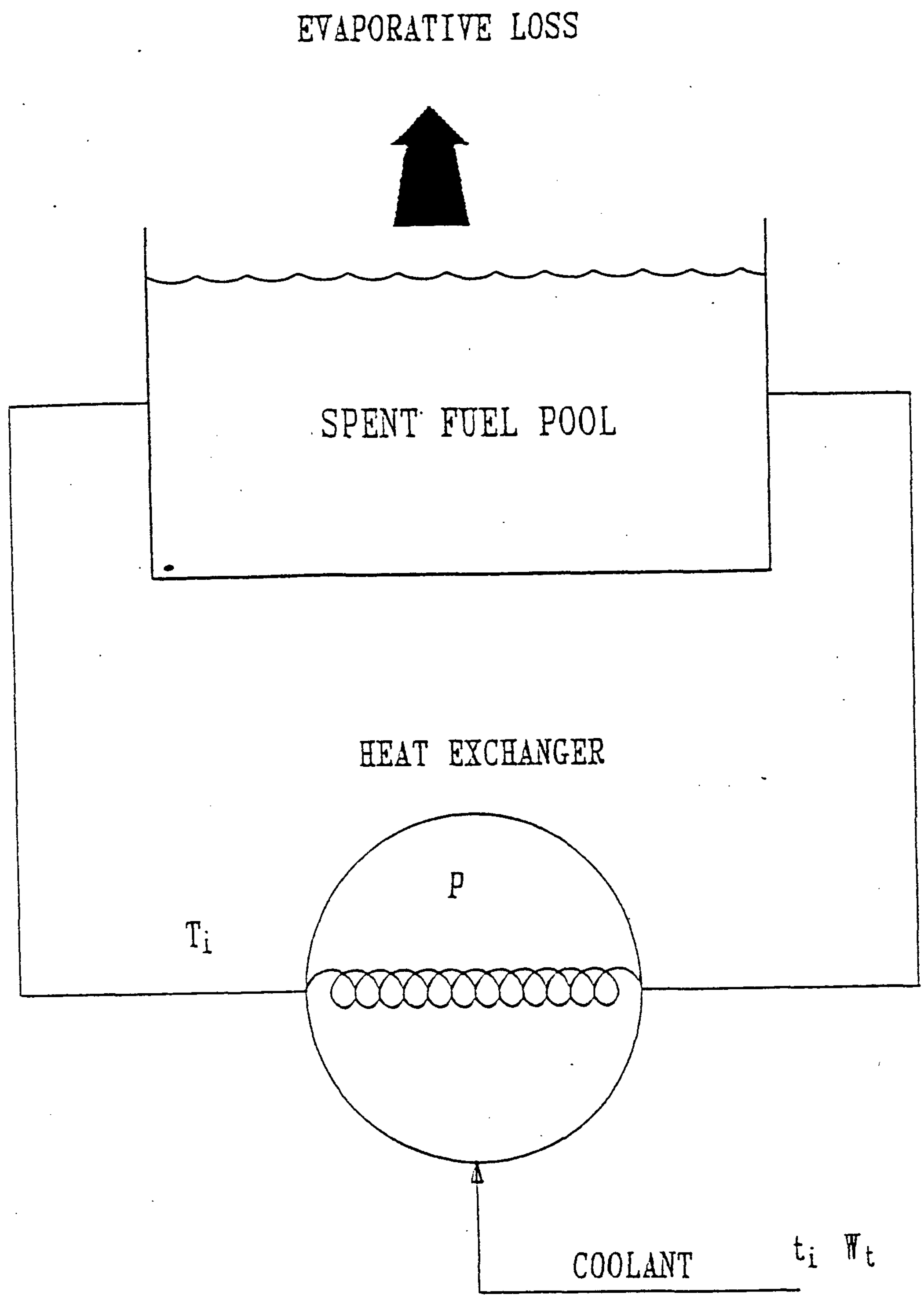


FIGURE 5.5.1: SPENT FUEL COOLING MODEL

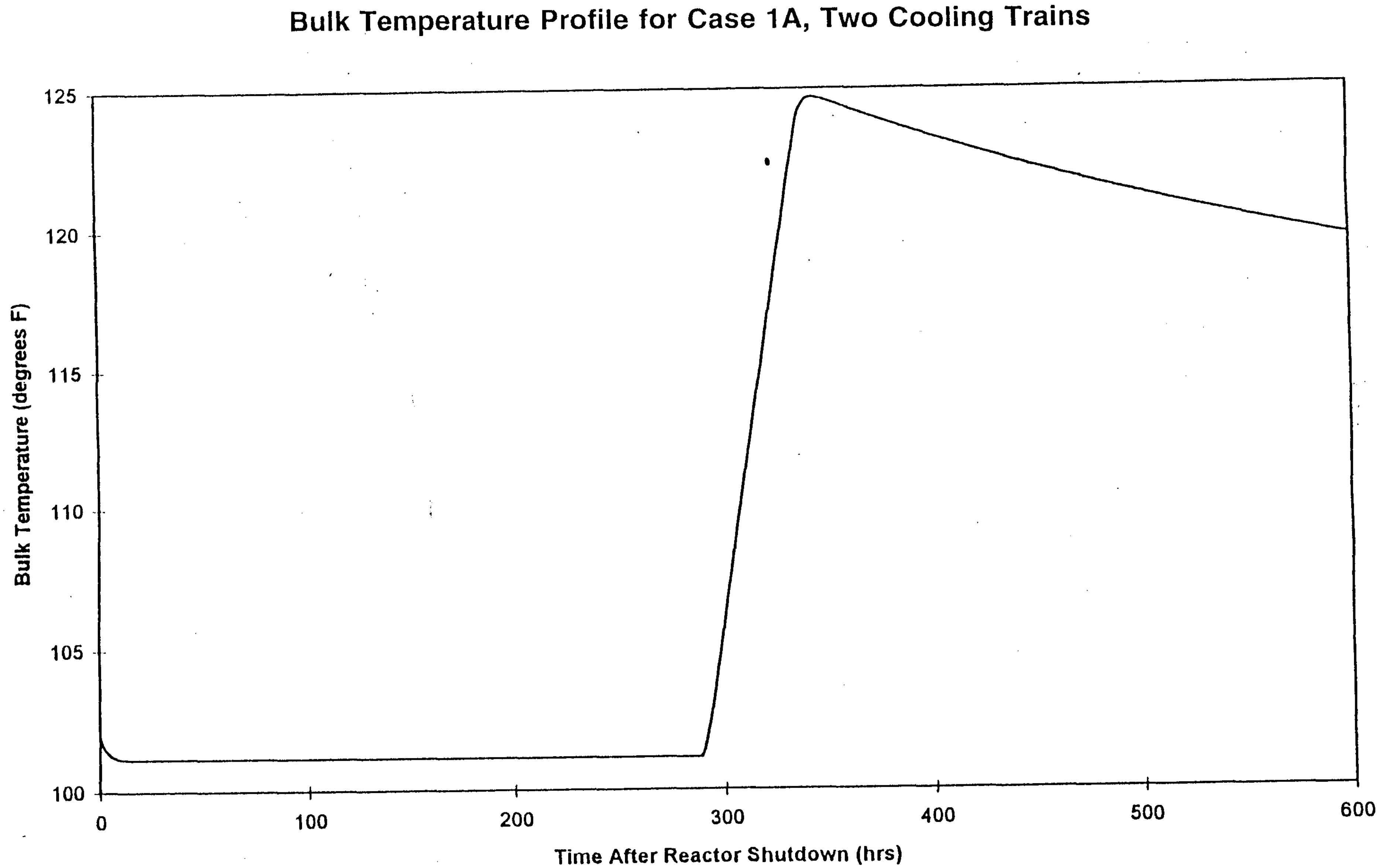

Bulk Temperature Profile for Case 1A, Two Cooling Trains
Bulk Temperature (degrees F)
125
120
115
110
105
100
0
100
200
300
400
500
600
Time After Reactor Shutdown (hrs)

FIGURE 5.5.2

Bulk Temperature Profile for Case 1B, One Cooling Train

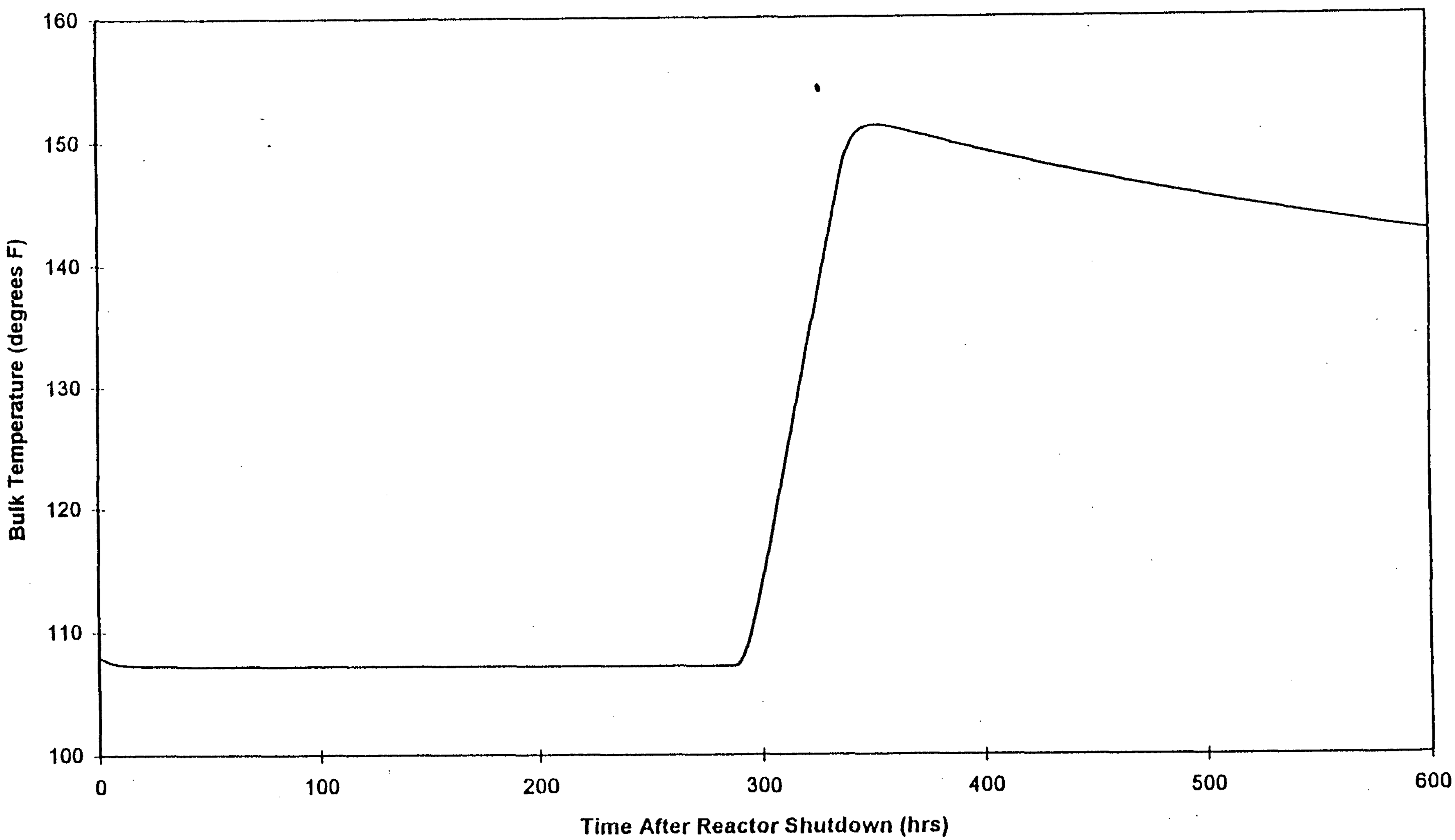


FIGURE 5.5.3

Bulk Temperature Profile for Case 2A, Two Cooling Trains

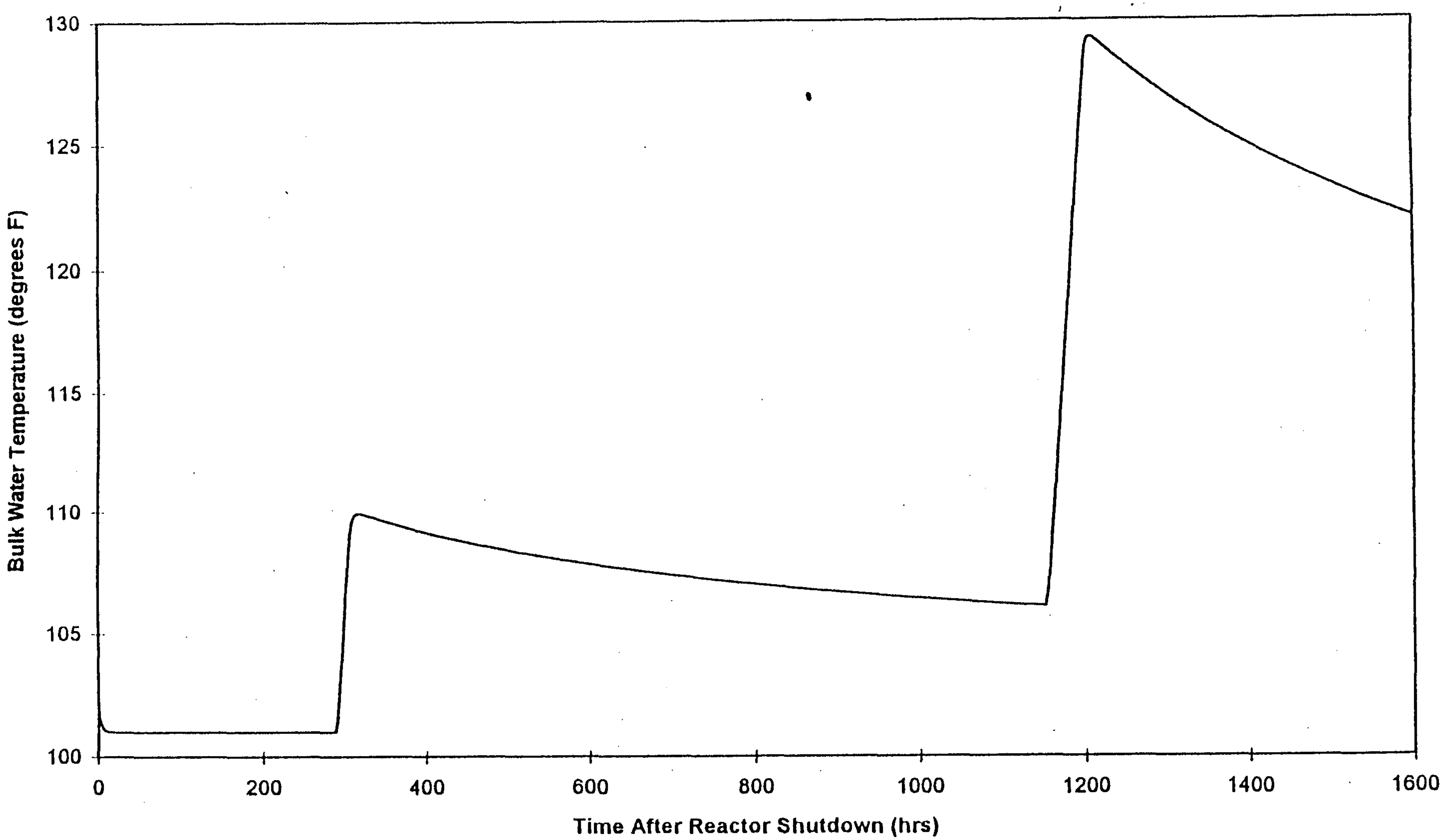


FIGURE 5.5.4

Bulk Temperature Profile for Case 2B, One Cooling Train

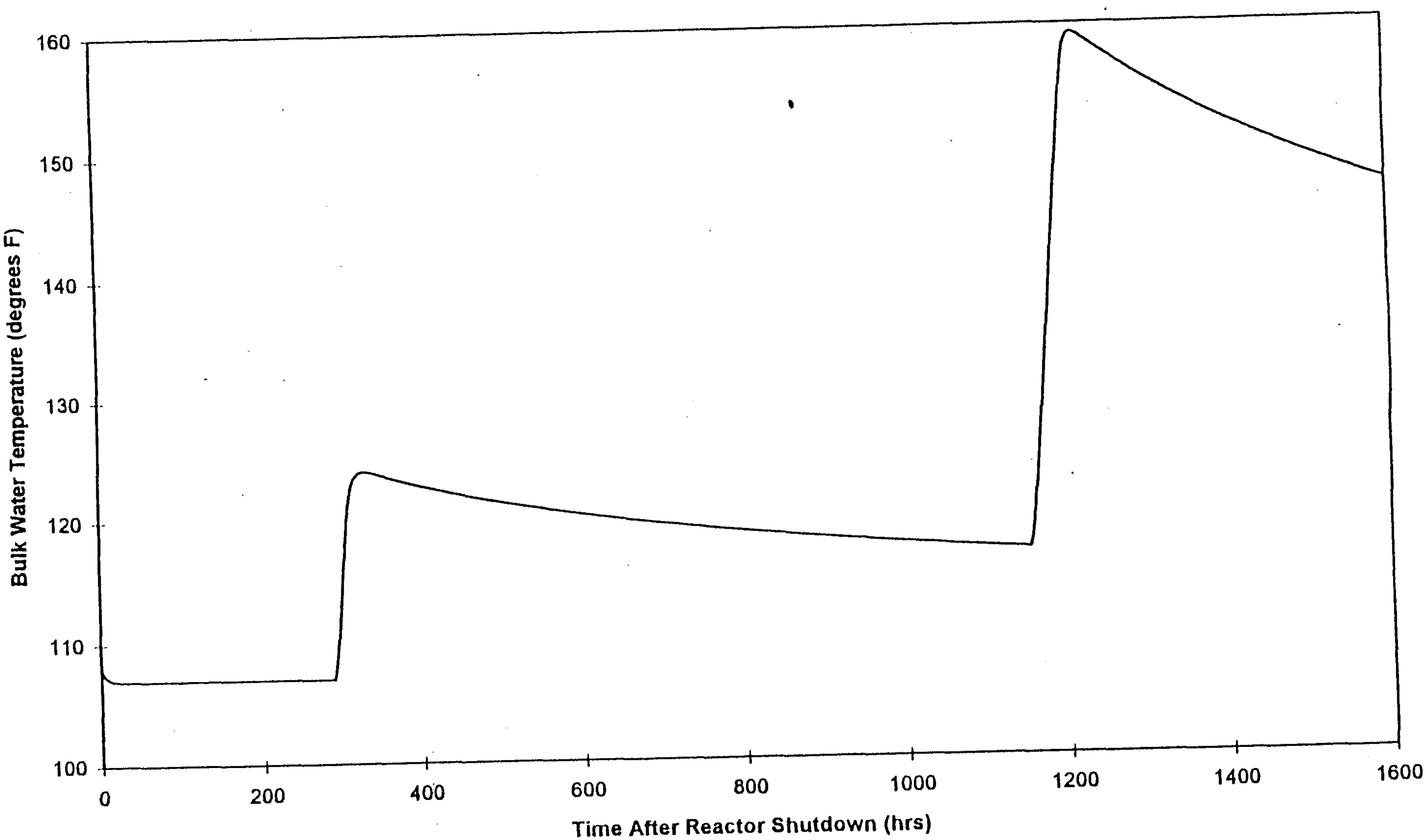


FIGURE 5.5.5

Decay Heat Load and Evaporative Heat Loss Profiles for Case 1A, Two Cooling Trains

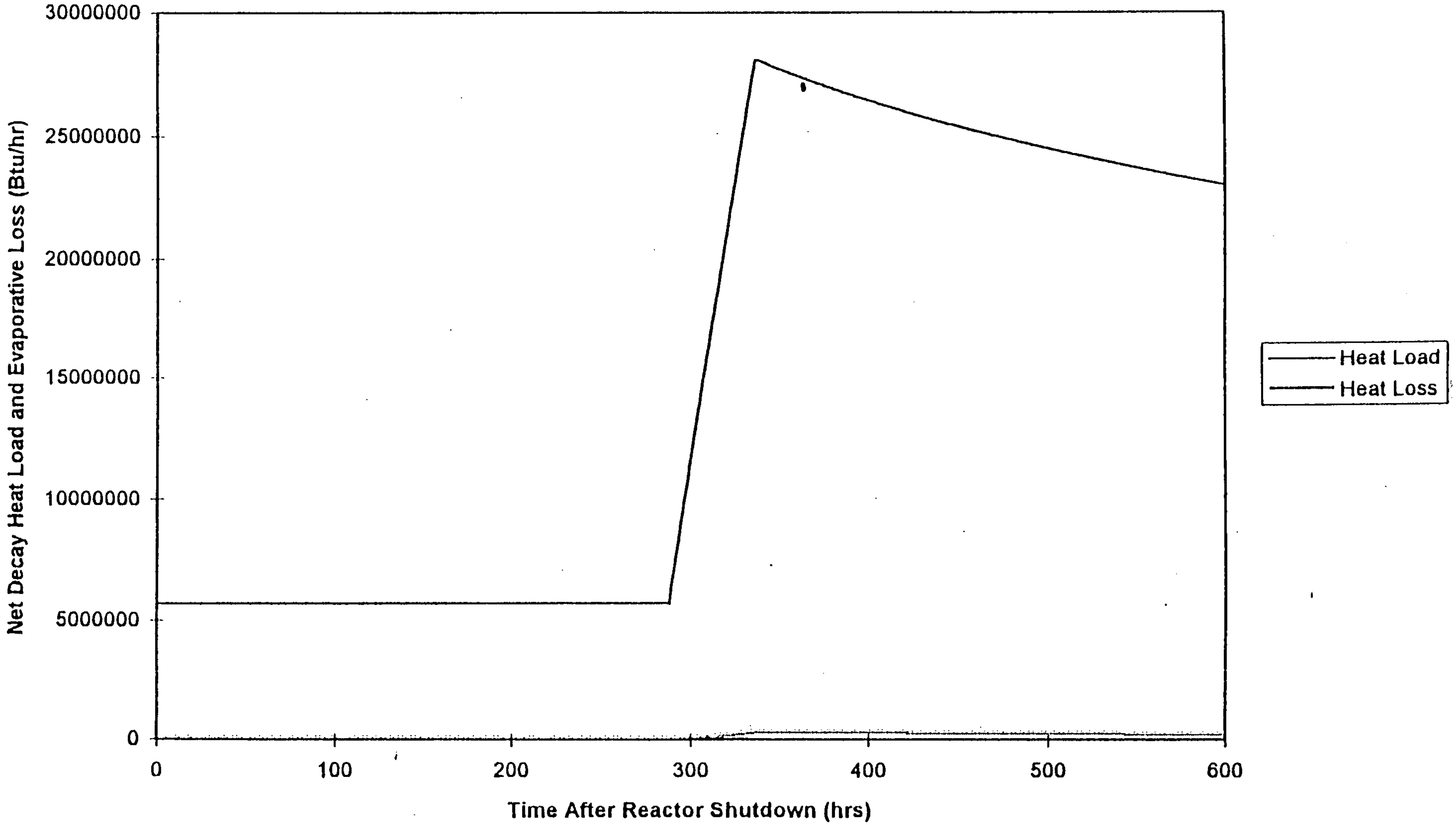


FIGURE 5.5.6

Decay Heat Load and Evaporative Heat Loss Profiles for Case 1B, One Cooling Train

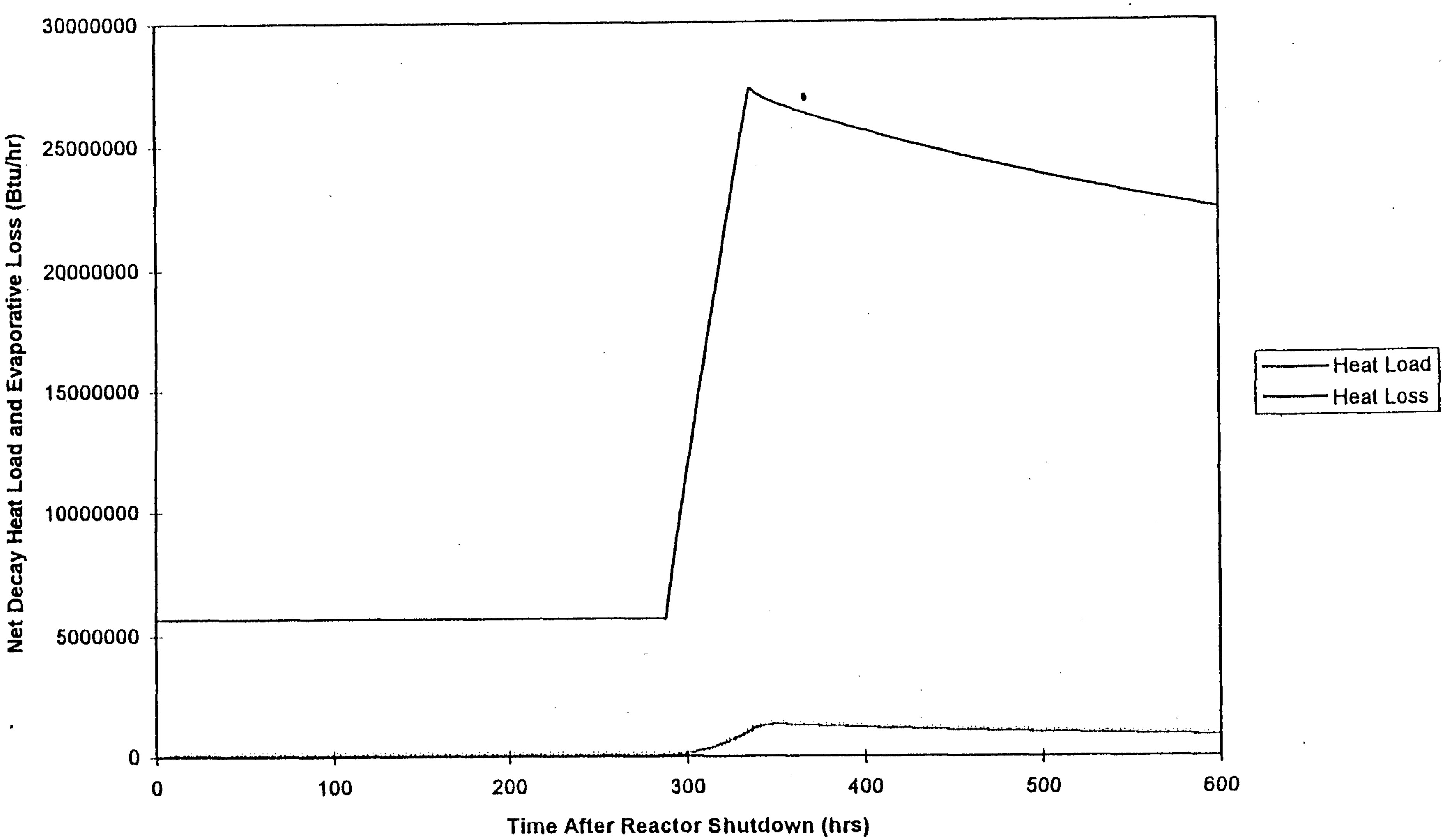


FIGURE 5.5.7

Decay Heat Load and Evaporative Heat Loss Profiles for Case 2A, Two Cooling Trains

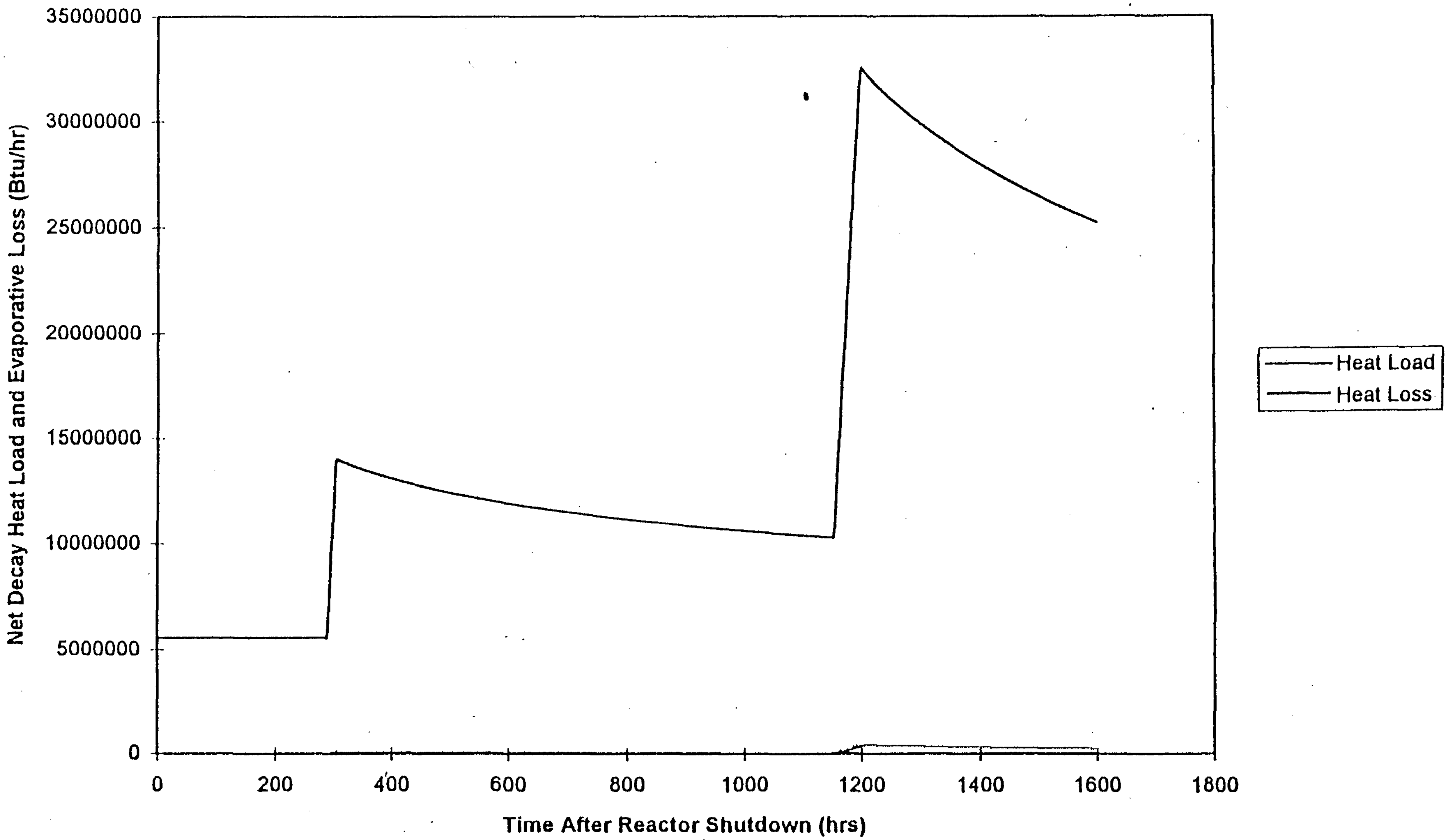


FIGURE 5.5.8

Decay Heat Load and Evaporative Heat Loss Profiles for Case 2B, One Cooling Train

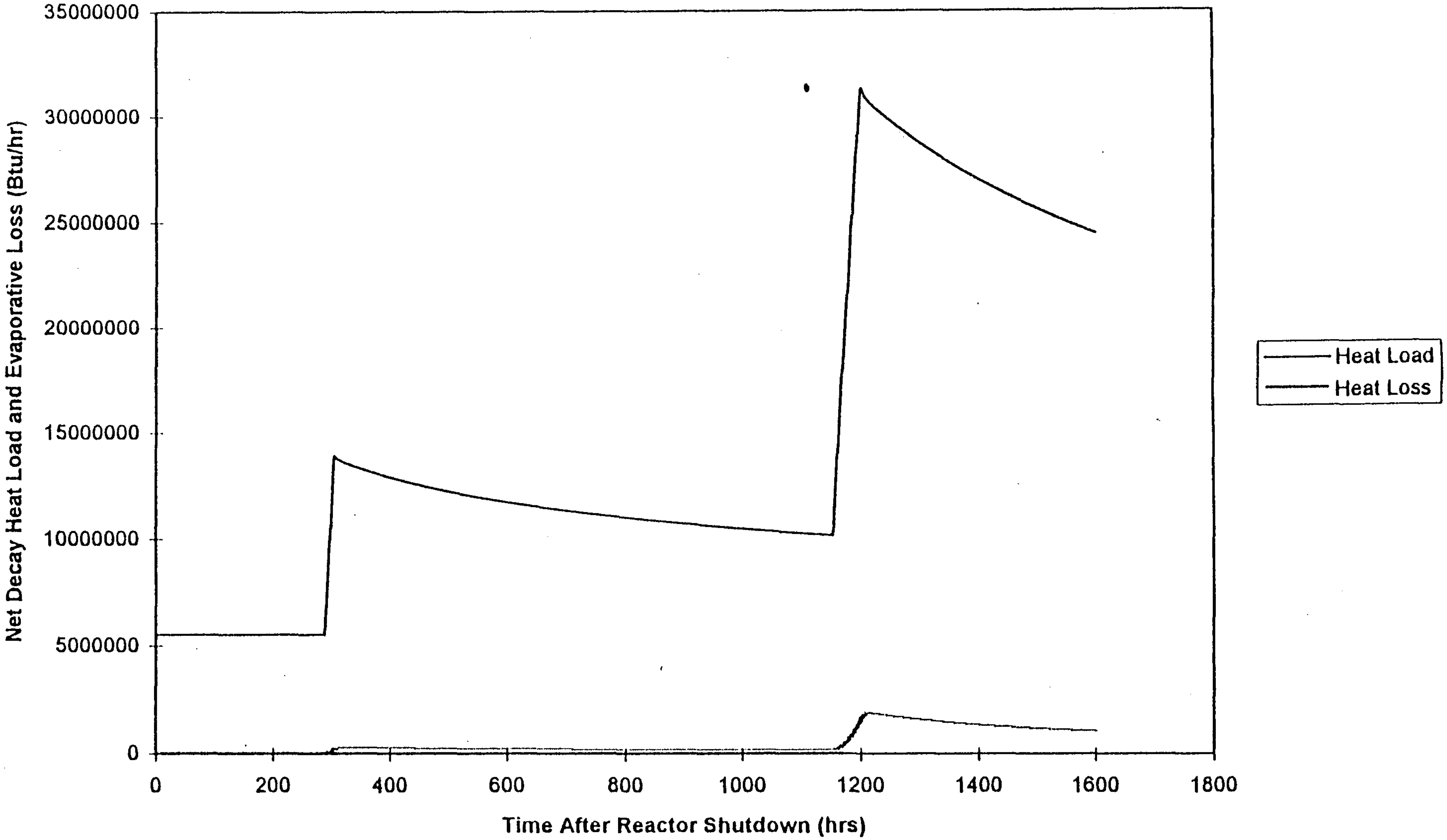


FIGURE 5.5.9

Watts Bar Loss of Cooling - Pool Volume Profile - Case 1A

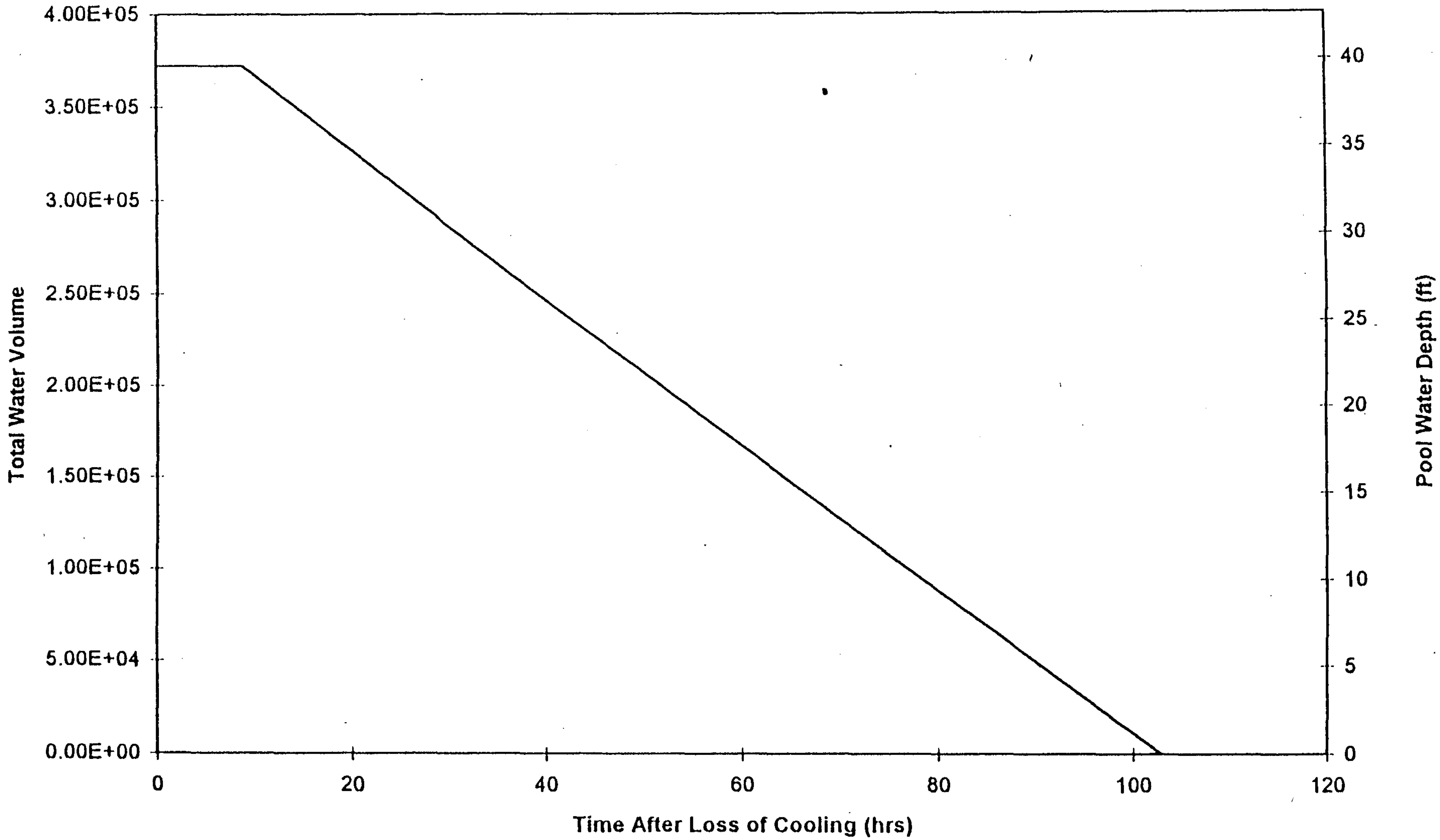


FIGURE 5.5.10

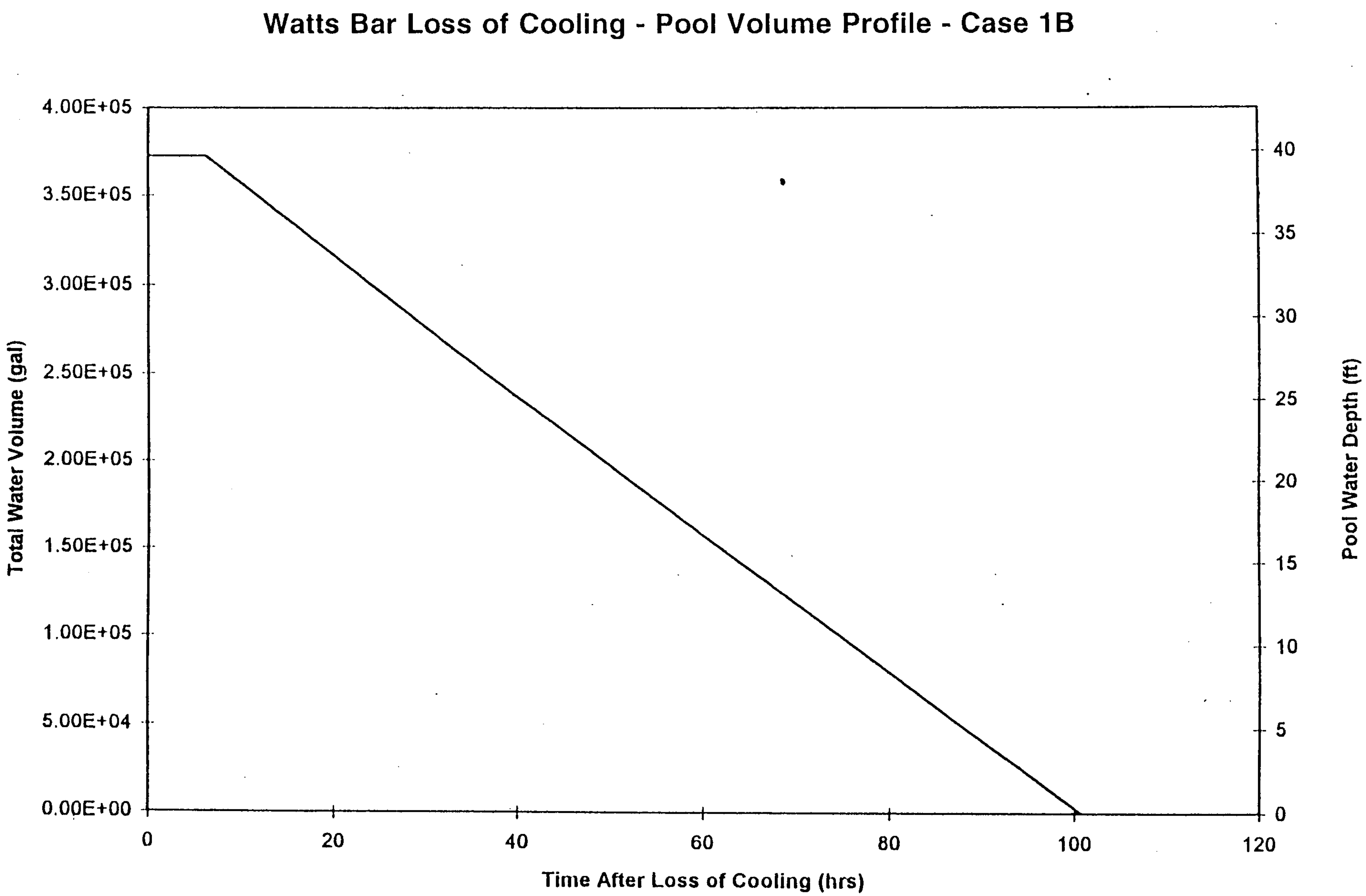

Watts Bar Loss of Cooling - Pool Volume Profile - Case 1B
Total Water Volume (gal)
4.00E+05
3.50E+05
3.00E+05
2.50E+05
2.00E+05
1.50E+05
1.00E+05
5.00E+04
0.00E+00
0
20
40
60
80
100
120
Time After Loss of Cooling (hrs)
Pool Water Depth (ft)
40
35
30
25
20
15
10
5
0

FIGURE 5.5.11

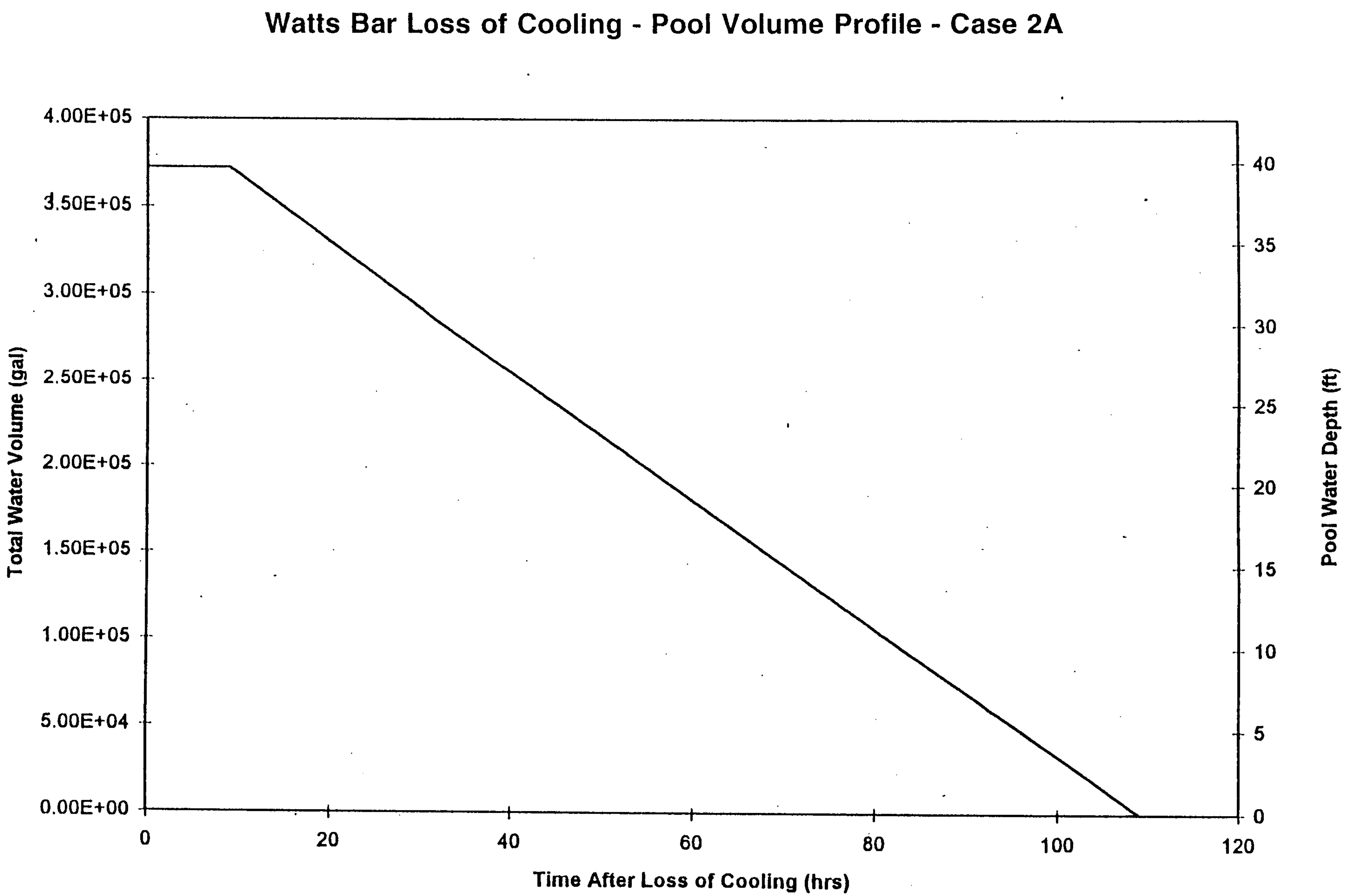

Watts Bar Loss of Cooling - Pool Volume Profile - Case 2A
4.00E+05
3.50E+05
3.00E+05
2.50E+05
2.00E+05
1.50E+05
1.00E+05
5.00E+04
0.00E+00
Total Water Volume (gal)
40
35
30
25
20
15
10
5
0
Pool Water Depth (ft)
0
20
40
60
80
100
120
Time After Loss of Cooling (hrs)

FIGURE 5.5.12

Watts Bar Loss of Cooling - Pool Volume Profile - Case 2B

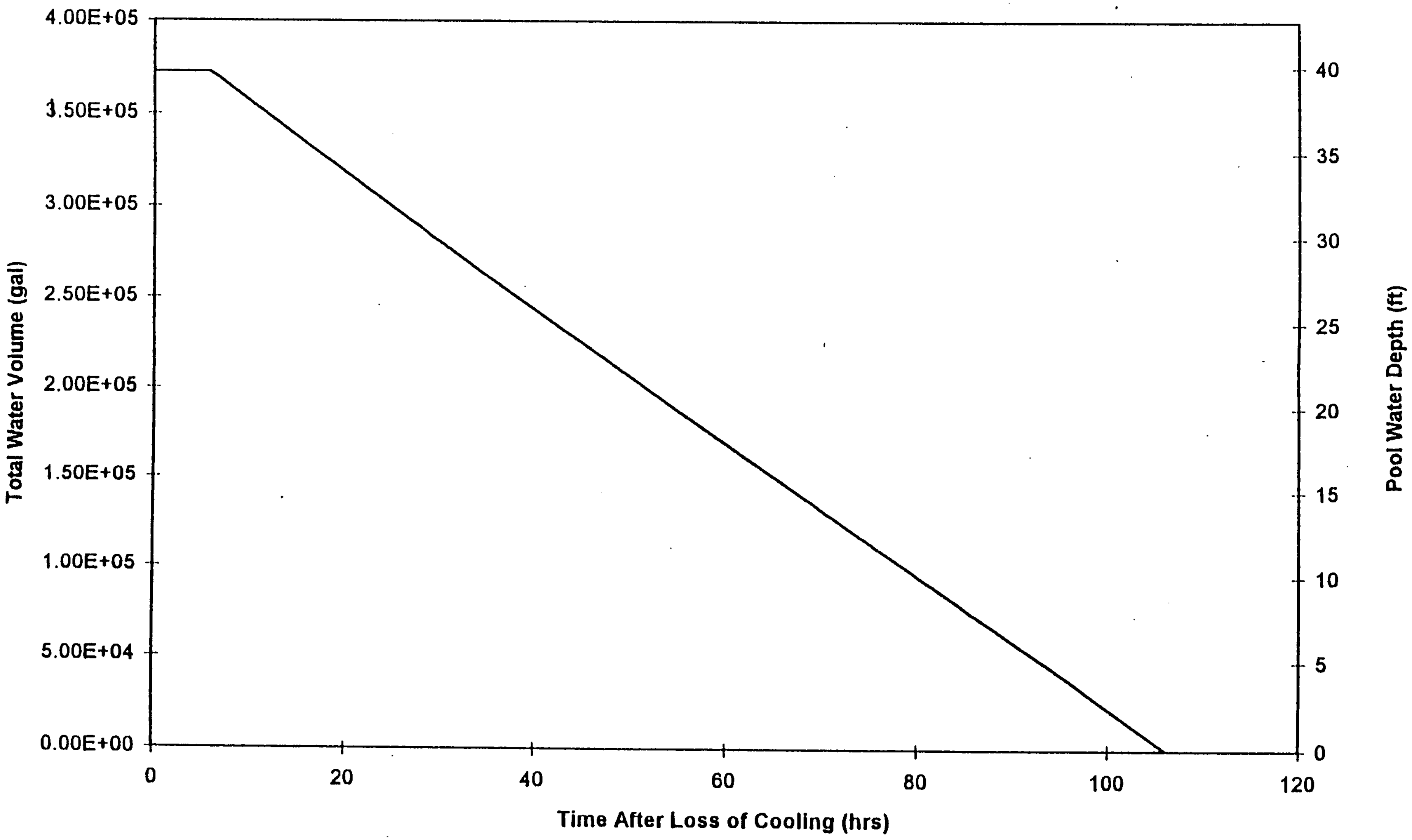
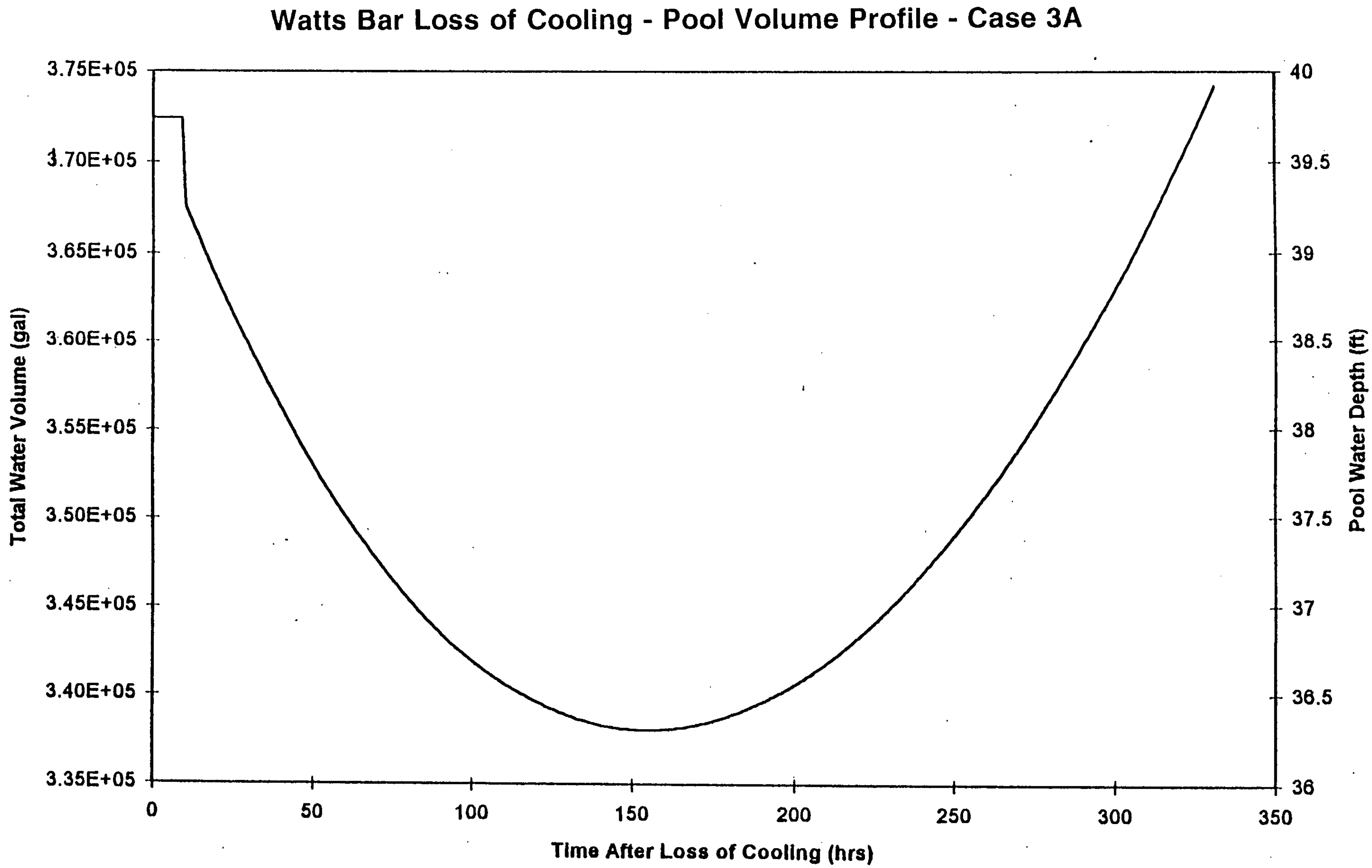
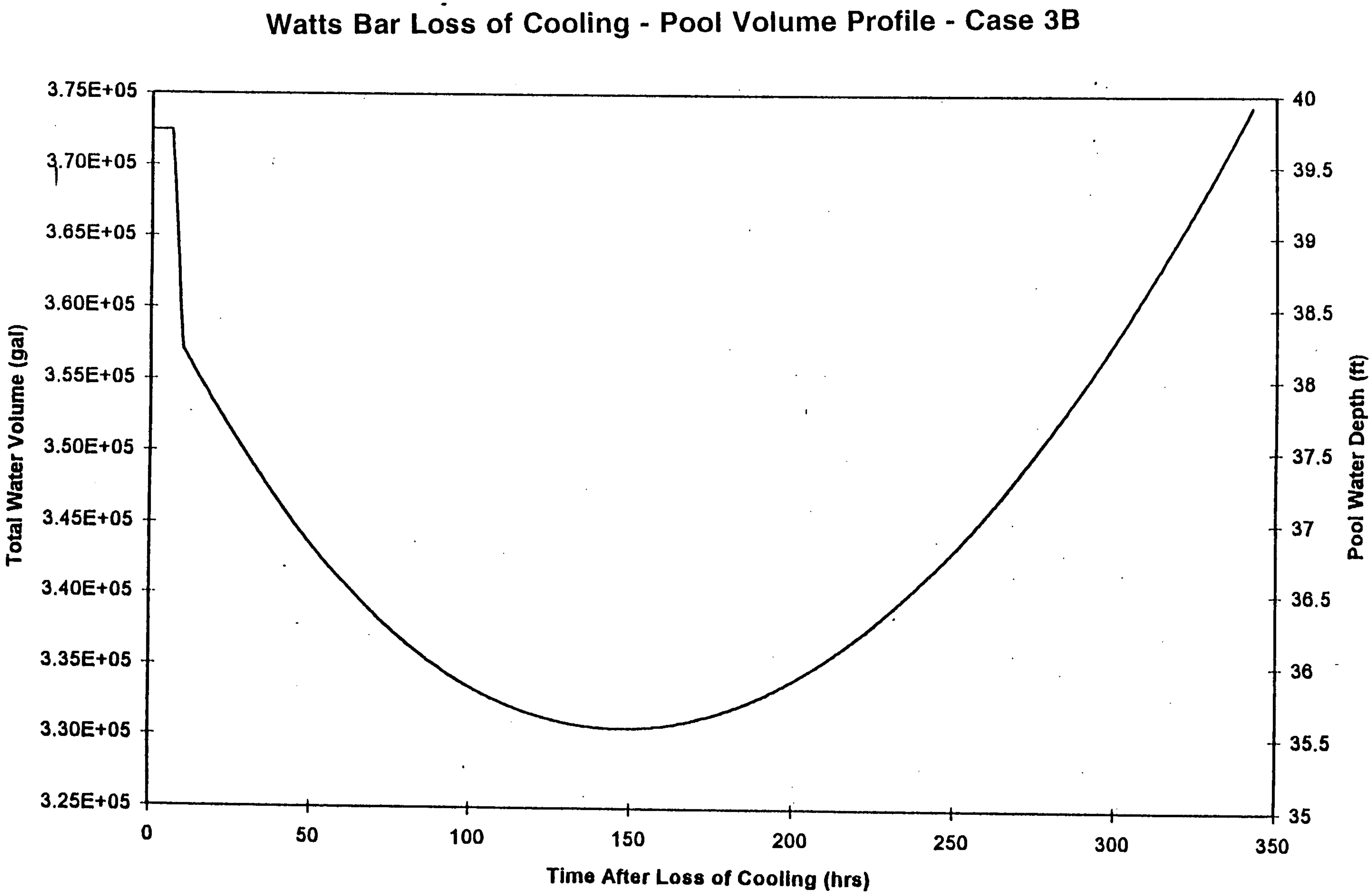


FIGURE 5.5.13



Watts Bar Loss of Cooling - Pool Volume Profile - Case 3A
3.75E+05
3.70E+05
3.65E+05
3.60E+05
3.55E+05
3.50E+05
3.45E+05
3.40E+05
3.35E+05
Total Water Volume (gal)
40
39.5
39
38.5
38
37.5
37
36.5
36
Pool Water Depth (ft)
0
50
100
150
200
250
300
350
Time After Loss of Cooling (hrs)

FIGURE 5.5.14



Watts Bar Loss of Cooling - Pool Volume Profile - Case 3B
Total Water Volume (gal)
3.75E+05
3.70E+05
3.65E+05
3.60E+05
3.55E+05
3.50E+05
3.45E+05
3.40E+05
3.35E+05
3.30E+05
3.25E+05
Pool Water Depth (ft)
40
39.5
39
38.5
38
37.5
37
36.5
36
35.5
35
0
50
100
150
200
250
300
350
Time After Loss of Cooling (hrs)

FIGURE 5.5.15

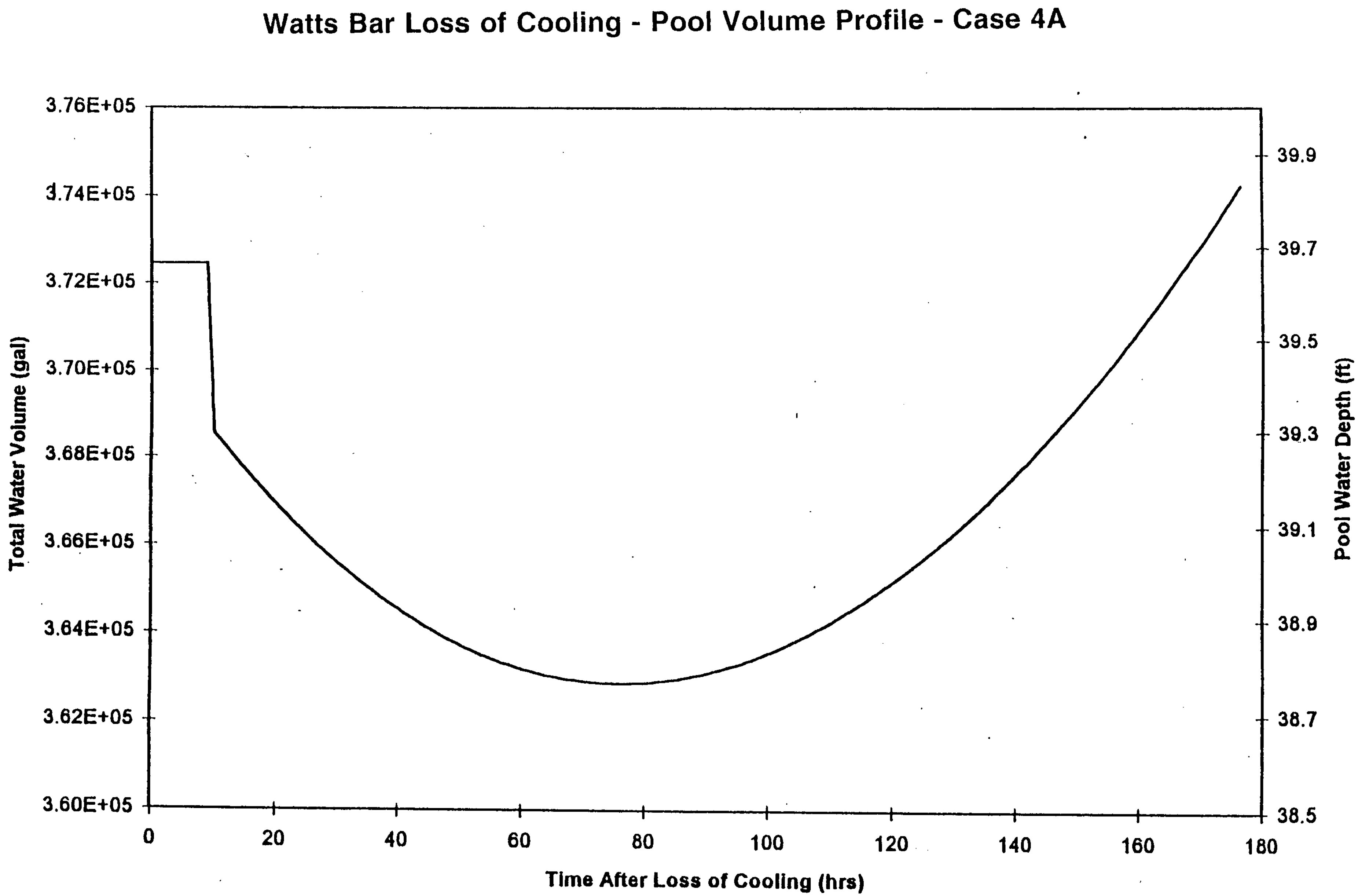

Watts Bar Loss of Cooling - Pool Volume Profile - Case 4A
3.76E+05
3.74E+05
3.72E+05
3.70E+05
3.68E+05
3.66E+05
3.64E+05
3.62E+05
3.60E+05
Total Water Volume (gal)
39.9
39.7
39.5
39.3
39.1
38.9
38.7
38.5
Pool Water Depth (ft)
0
20
40
60
80
100
120
140
160
180
Time After Loss of Cooling (hrs)

FIGURE 5.5.16

Watts Bar Loss of Cooling - Pool Volume Profile - Case 4B

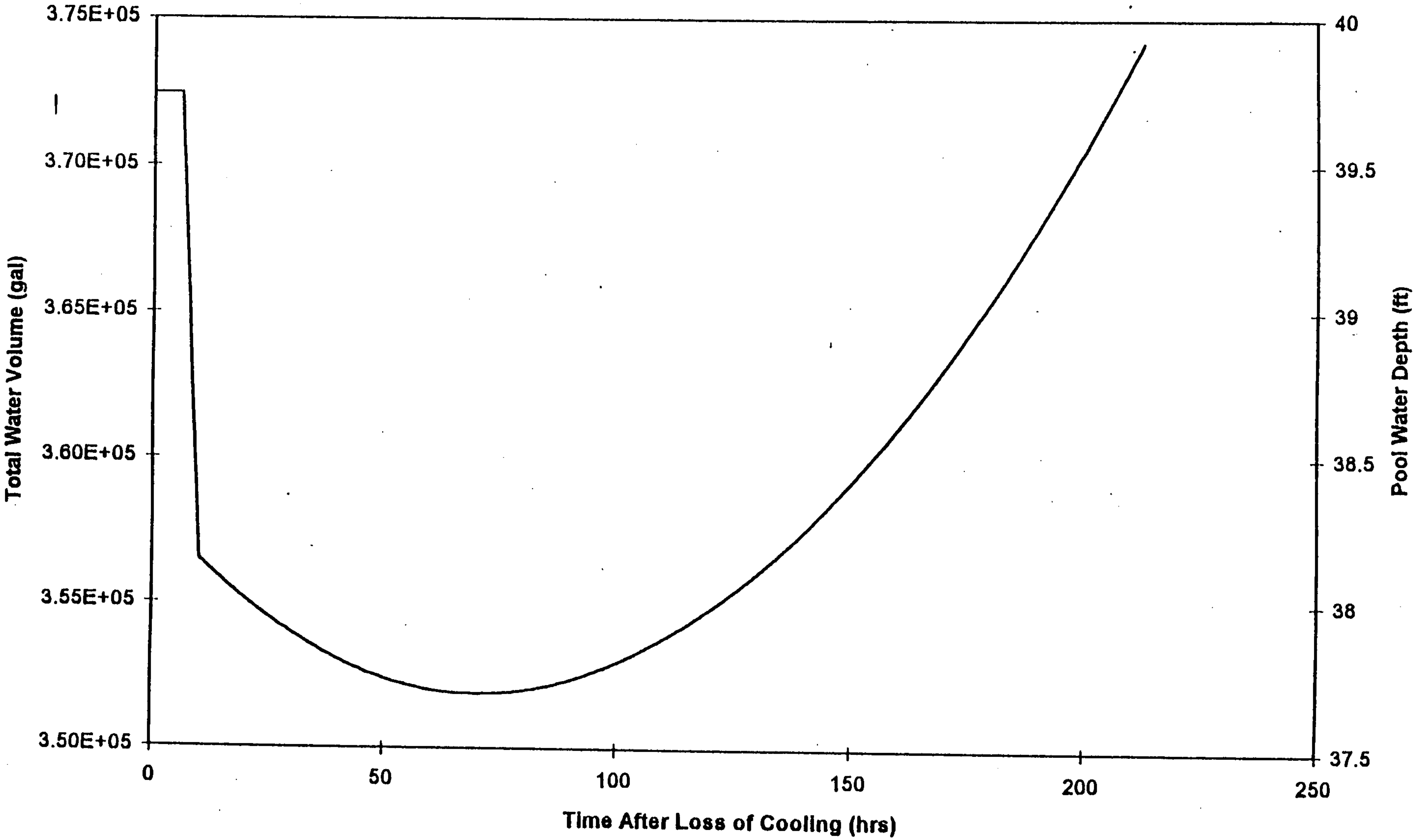


FIGURE 5.5.17

SFP/Cask Pit CFD Model Decay Heat Distribution

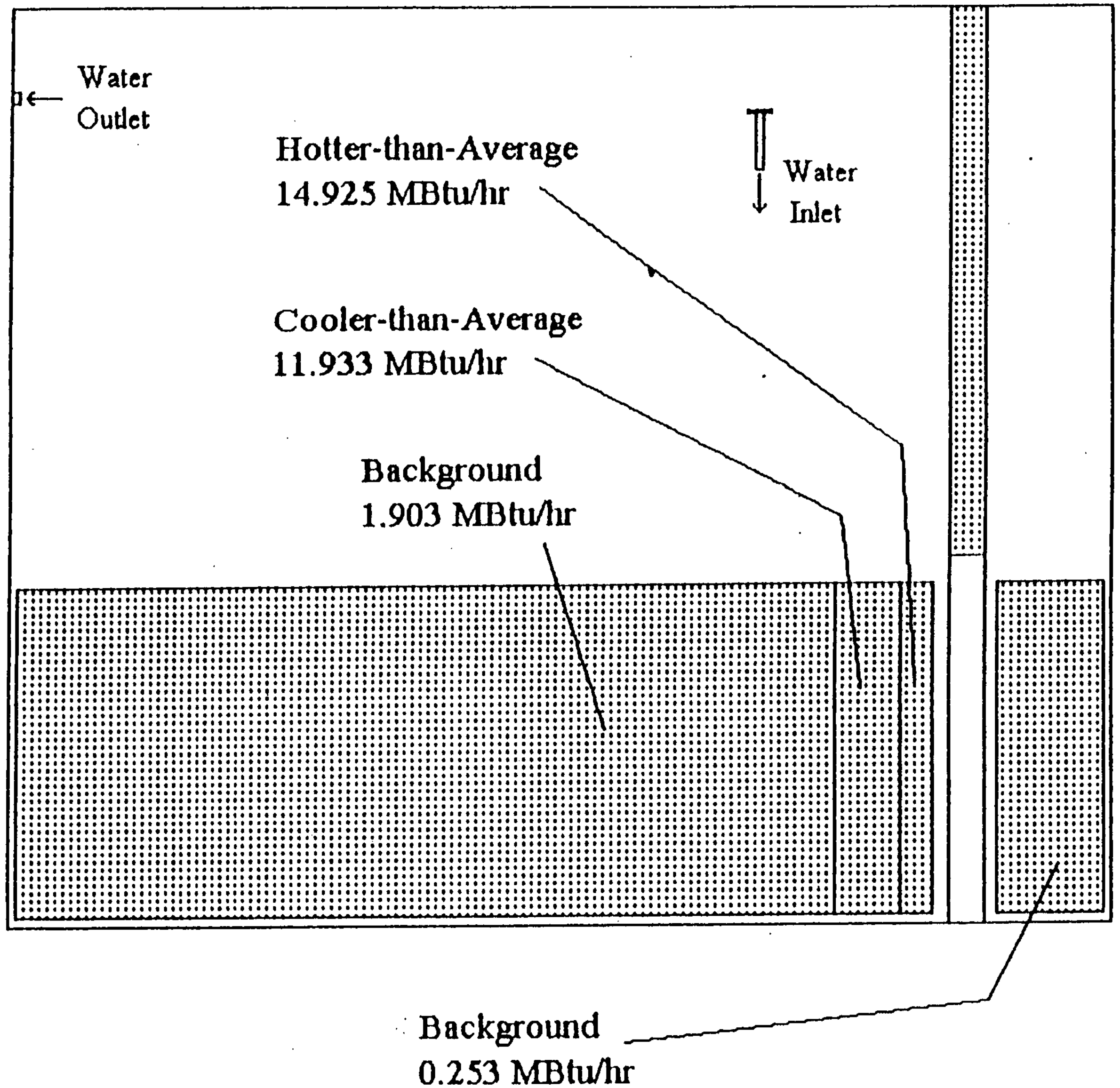


FIGURE 5.6.1

CHAPTER 6

STRUCTURAL/SEISMIC CONSIDERATIONS

6.0 STRUCTURAL/SEISMIC CONSIDERATIONS

6.1 INTRODUCTION

The structural adequacy of the maximum density spent fuel racks is considered in this section. The analyses undertaken to confirm the structural integrity of the racks to demonstrate compliance with the USNRC Standard Review Plan (SRP)[1] and the Office of Technology (O.T.) Position Paper[2] are as follows:

Determination of seismic time-histories for SSE and OBE events.

3-D transient analyses of the spent fuel racks individually and as an assemblage acting as free-standing submerged bodies subjected to seismic excitations applied as synthetic acceleration time-histories.

Evaluation of the primary stresses in the rack structure to establish compliance with the stress limits for ASME Section III Subsection NF.[3]

Evaluation of the secondary and peak stress amplitudes in the most critically loaded rack sections to ensure that failure from cyclic fatigue will not occur.

For each of the analyses undertaken, an abstract of the methodology, modeling assumptions, key results, and summary of parametric evaluations are presented.

6.2 ACCEPTANCE CRITERIA

The time-histories developed to support the analyses are in compliance with the NRC SRP.[1] The spent fuel rack analyses and evaluations are in compliance with the requirements of the OT Position Paper, Section IV,[2] and are in compliance with the stress and displacement limits of the relevant ASME Code.[3] The interfacial pressure between the rack pedestals and the pool liner is shown to comply with the provisions of the American Concrete Institute.[4] Further delineation of the relevant criteria are discussed in the text associated with each analysis.

6.3 LOADS AND LOAD COMBINATIONS

The principal loadings considered in the mechanical integrity evaluation are the following:

a. dead weight of the rack submerged in a pool of water

b. seismic excitation loads for the SSE and OBE events

c. fluid coupling loads arising from the relative motion of the racks with respect to each other and with respect to the pool walls

d. fuel assembly-to-cell impact loads

e. dynamic coupling loads due to rattling of the fuel in the storage cells arising from the seismic inputs to the free-standing racks

f. rack pedestal/liner friction forces which counteract other horizontal loadings during seismic events

g. mechanical loads arising from abnormal events such as a fuel handling accident

The mandated loads and load combinations are obtained from the references cited in the foregoing.

6.4 STRUCTURAL EVALUATION OF RACKS

6.4.1 Overview

The WBN spent fuel racks are designed as freestanding and are qualified as Seismic Category I structures.[5]

The response of a free-standing rack module to seismic inputs is highly nonlinear and involves a complex combination of motions (sliding, rocking, twisting, and turning), which could result in impacts and friction effects. Linear methods, such as modal analysis and response spectrum techniques, cannot accurately simulate the structural response of such a highly nonlinear structure to seismic excitation. An accurate simulation is obtained only by direct integration of the nonlinear equations of motion using actual pool slab acceleration time-histories as the forcing function. Therefore, the initial step in spent fuel rack qualification is to develop synthetic time-histories for three orthogonal directions which comply with the guidelines of the NRC's SRP.[1] In particular, the synthetic time-histories must meet the criteria of statistical independence and enveloping of the design response spectra.

Having obtained an admissible set of input excitations, the next step in the analysis process is to develop a suitable dynamic model. Reliable assessment of the stress field and kinematic behavior of the rack modules calls for a conservative dynamic model incorporating the key attributes of the actual structure. This means that the model must feature the ability to execute concurrent sliding, rocking, bending, twisting and other motion forms compatible with the free-standing installation of the modules. Furthermore, the model must possess the capability to effect momentum transfers which occur due to rattling of fuel assemblies inside storage cells and the capability to simulate lift-off and subsequent impact of support pedestals with the pool liner (or bearing pad). The contribution of the water mass in the interstitial spaces around the rack modules and within the storage cells must be modeled in an accurate manner since erring in quantification of fluid coupling on either side of the actual value is no guarantee of conservatism. The Coulomb friction coefficient at the pedestal-to-pool liner (or bearing pad) interface may lie in a rather wide range and a conservative value of friction cannot be deduced. In fact, a perusal of results of rack dynamic analyses in numerous dockets (Table 6.4.1) indicate that an upper bound value of the coefficient of friction, μ, often maximizes the computed rack displacements as well as the equivalent elastostatic stresses. Finally, the analysis must consider that a rack module may be fully or partially loaded with fuel assemblies or may be entirely empty. The pattern of loading in a partially loaded rack may also have innumerable combinations. In short, there are a large number of parameters with potential influence on the rack motion. The comprehensive structural evaluation must deal with these without sacrificing conservatism.

The three-dimensional single rack dynamic model introduced by Holtec International in the Enrico Fermi Unit 2 rack project (ca. 1980) and used in some 30 rerack projects since that time (Table 6.4.1) addresses the above mentioned array of parameters. The details of this methodology are published in the permanent literature.[6] Briefly, handling of the array of variables for the single rack 3-D model is discussed below.

<u>Interface Coefficient of Friction</u>

Parametric runs are made with upper bound and lower bound values of the coefficient of friction. The limiting values are based on experimental data which have been found to be bounded by the values 0.2 and 0.8.

<u>Impact Phenomena</u>

Compression-only gap elements are used to provide for opening and closing of interfaces such as the pedestal-to-bearing pad interface and for the fuel assembly-to-cell wall interface.

<u>Fuel Loading Scenarios</u>

The fuel assemblies are conservatively assumed to rattle in unison, which obviously exaggerates the contribution of impact against the cell wall. The different patterns of possible fuel assembly loadings in the rack are simulated by orienting the center of gravity column of the assemblage of fuel assemblies with respect to the module geometric center of gravity in an appropriate manner.

<u>Fluid Coupling</u>

The contribution of fluid coupling forces is ascertained by prescribing the motion of the racks (adjacent to the one being analyzed). The most commonly used assumption when dealing with a single rack is that the adjacent racks vibrate out-of-phase with respect to the rack being analyzed.

Despite the above simplifying assumptions, targeted for accuracy and conservatism, a large menu of cases is run to foster confidence in the calculated safety margins. Most safety analyses reported in previous dockets (Table 6.4.1) over the past decade have relied on the single rack 3-D model. From a conceptual standpoint, aspects of the 3-D single rack model are satisfactory except for the fluid coupling effect. One intuitively expects relative motion of free-standing racks in the pool to be poorly correlated, given the random harmonics in the impressed slab motion. Single rack analyses cannot model this interactive behavior between racks. However, as described later, analytical and experimental research in this field has permitted rack analyses to be extended to the entire array of racks in the pool simultaneously. Holtec International extended Fritz's classical two-body fluid coupling model[15] to multiple bodies and utilized it to perform the first two-dimensional multi-rack analysis (Diablo Canyon, ca. 1987). Subsequently, laboratory experiments were conducted to validate the multi-rack fluid coupling theory. This technology was incorporated in the computer code DYNARACK which now could handle simultaneous simulation of racks in the pool. This development was first utilized in Chin Shan, Oyster Creek and Shearon Harris plants[7,8] and, subsequently, in numerous other rerack projects. The Whole Pool Multi-Rack (WPMR) 3-D analyses have corroborated the accuracy of the single rack 3-D solutions in predicting the maximum structural stresses and serve to improve predictions of rack kinematics.

The WPMR analysis methodology is the vehicle available to establish the presence or absence of specific rack-to-rack impacts during the seismic event.

Recognizing that the analysis work effort must deal with both stress and displacement criteria, the sequence of model development and analysis steps that are undertaken are summarized in the following.

a. Prepare 3-D dynamic models suitable for a time-history analysis of the new maximum density racks.

b. Perform 3-D dynamic analyses on limiting module geometry types (from those present in the spent fuel pool) and include various physical conditions (such as coefficient of friction and extent of cells containing fuel assemblies).

c. Perform stress analysis of high stress areas for the limiting case of the single rack dynamic analysis runs made in the foregoing steps. Demonstrate compliance with ASME Code Section III, Subsection NF limits on stress and displacement.

d. Prepare a WPMR dynamic model of rack modules in the pool including fluid coupling interactions among them, as well as fluid coupling interactions between racks and pool walls. This 3-D simulation is referred to as a WPMR model.

e. Perform 3-D WPMR analyses to demonstrate that kinematic criteria for the spent fuel rack modules are satisfied and that resultant structure loads confirm the validity of the single rack structural qualification. The principal kinematic criteria are 1) no rack-to-pool wall impact, and 2) no rack-to-rack impact in the cellular region of the racks containing active fuel.

As shown in Figure 6.4.1, a total of 24 free-standing rack modules in two sizes (7x8 and 7x9) are arrayed in the WBN pool at relatively close spacings (the intermodule gaps vary from 1.0 to 3.7 inches). The peripheral modules are additionally equipped with "baby racks" which are free standing for support but are fastened to the main racks near the top and at the base for stability. In this manner, the reracked WBN pool is modeled as an assemblage of 34 free-standing racks. Figure 6.4.2 is a representative sketch of a full size high density fuel rack assemblage submerged in water.

6.4.2 Input Loadings

The primary loading causing a limiting stress state in the spent fuel racks is the seismic loading. The seismic loading induces other loadings in the racks, as discussed in Section 6.3.

In order to prepare an acceptable set of acceleration time-histories, Holtec International's proprietary code GENEQ[9] is utilized.

6.4.3 Acceptance Criteria for Spent Fuel Rack Design

6.4.3.1 Kinematic and Stress Criteria

There are two sets of criteria to be satisfied by the rack modules:

a. Kinematic Criteria

In order to be qualified as a physically stable structure it is necessary to demonstrate that an isolated rack in water would not overturn when an event of magnitude 1.1 times the governing faulted seismic loading condition is applied.[11]

b. Stress Limit Criteria

Stress limits must not be exceeded under the postulated load combinations provided in Section 6.4.4 herein.

The stress limits presented below are derived from the ASME Code, Section III, Subsection NF.[3] Parameters and terminology are in accordance with the ASME Code. Material properties are obtained from the ASME Code Appendices[10] and are listed in Table 6.4.2.

(i) Normal and Upset Conditions (Level A or Level B)

a. Allowable stress in tension on a net section is:

$F_t = 0.6\ S_y$ (S_y = yield stress at temperature)

(F_t is equivalent to primary membrane stress)

b. Allowable stress in shear on a net section is:

$F_v = 0.4\ S_y$

c. Allowable stress in compression on a net section

$$F_a = \frac{\left[1 - \frac{(k\ell)^2}{r^2}/2C_c^2\right]S_y}{\{(\frac{5}{3}) + [3(\frac{k\ell}{r})/8C_c] - [(\frac{k\ell}{r})^3/8C_c^3]\}}$$

where:

$$C_c = [\frac{(2\pi^2 E)}{S_y}]^{1/2}$$

ℓ = unsupported length of component

k = length coefficient which gives influence of boundary conditions; e.g.

k = 1 (simple support both ends)
= 2 (cantilever beam)
= 1/2 (clamped at both ends)

E = Young's Modulus

r = radius of gyration of component

$k\ell/r$ for the main rack body is based on the full height and cross section of the honeycomb region.

d. Maximum allowable bending stress at the outermost fiber of a net section, due to flexure about one plane of symmetry is:

$F_b = 0.60\ S_y$ (equivalent to primary bending)

e. Combined bending and compression on a net section satisfies:

$$\frac{f_a}{F_a} + \frac{C_{mx}\ f_{bx}}{D_x\ F_{bx}} + \frac{C_{my}\ f_{by}}{D_y\ F_{by}} < 1.0$$

where:

f_a = Direct compressive stress in the section

f_{bx} = Maximum bending stress along x-axis

f_{by} = Maximum bending stress along y-axis

C_{mx} = C_{my} = 0.85

$$D_x = 1 - \frac{f_a}{F'_{ex}}$$

$$D_y = 1 - \frac{f_a}{F'_{ey}}$$

$$F'_{ex,ey} = \frac{12\ \pi^2\ E}{23\left(\frac{k\ell}{r}\right)^2_{x,y}}$$

and subscripts x,y reflect the particular bending plane.

f. Combined flexure and axial compression (or tension) on a net section:

$$\frac{f_a}{0.6\,S_y} + \frac{f_{bx}}{F_{bx}} + \frac{f_{by}}{F_{by}} < 1.0$$

The above requirements are to be met for both direct tension or compression.

(ii) Level D Service Limits

Section F-1334 of ASME Section III, Appendix F,[10] states that limits for the Level D condition are the minimum of 1.2 (S_y/F_t) or $(0.7S_u/F_t)$ times the corresponding limits for the Level A condition. S_u is ultimate tensile stress at the specified rack design temperature.

6.4.3.2 Dimensionless Stress Factors

For convenience, the stress results are presented in dimensionless form. Dimensionless stress factors are defined as the ratio of the actual developed stress to the specified limiting value. The limiting value of each stress factor is 1.0 for Levels A, B, and D (where $1.2S_y < .7S_u$). Stress factors reported are:

R_1	=	Ratio of direct tensile or compressive stress on a net section to its allowable value (note pedestals only resist compression)
R_2	=	Ratio of gross shear on a net section in the x-direction to its allowable value
R_3	=	Ratio of maximum bending stress due to bending about the x-axis to its allowable value for the section
R_4	=	Ratio of maximum bending stress due to bending about the y-axis to its allowable value for the section
R_5	=	Combined flexure and compressive factor (as defined in Section 6.4.3.1(i)e)
R_6	=	Combined flexure and tension (or compression) factor (as defined in Section 6.4.3.1(i)f)
R_7	=	Ratio of gross shear on a net section in the y-direction to its allowable value

6.4.4 Loads and Loading Combinations for Spent Fuel Racks

The applicable loads and their combinations which must be considered in the seismic analysis of rack modules are excerpted from References 11 and 12 and are presented in the following:

Loading Combination	Service Level
$D + L$ $D + L + T_o$ $D + L + T_o + E$	Level A
$D + L + T_a + E$ $D + L + T_o + P_f$	Level B
$D + L + T_a + E'$ $D + L + T_o + F_d$	Level D The functional capability of the fuel racks should be demonstrated.

Abbreviations are those used in Section 3.8.4 of the Standard Review Plan[1] and the OT Position Paper on "Review and Acceptance of Spent Fuel Storage and Handling Applications."[2]

D	=	Dead weight-induced loads (including fuel assembly weight)
L	=	Live Load (not applicable for the fuel rack, since there are no moving objects in the rack load path)
F_d	=	Force caused by the accidental drop of the heaviest load from the maximum possible height specified in the FSAR.
P_f	=	Upward force on the racks caused by postulated stuck fuel assembly
E	=	Operating Basis Earthquake
E'	=	Safe Shutdown Earthquake
T_o	=	Differential temperature induced loads (normal operating or shutdown condition based on the most critical transient or steady state condition)
T_a	=	Differential temperature induced loads (the highest temperature associated with the postulated abnormal design conditions)

T_a and T_o produce local thermal stresses. The worst thermal stress field in a fuel rack is obtained when an isolated storage location has a fuel assembly generating heat at maximum postulated rate and surrounding storage locations contain no fuel. Heated water makes unobstructed contact with the inside of the storage walls, thereby producing maximum possible temperature difference between adjacent cells. Secondary stresses produced are limited to the body of the rack; that is, support pedestals do not experience secondary (thermal) stresses.

6.5 SEISMIC EVALUATION OF RACKS

6.5.1 Synthetic Time-Histories

The synthetic time-histories in three orthogonal directions (N-S, E-W, and vertical) are generated in accordance with the provisions of SRP 3.7.1.[1] A preferred criterion for the synthetic time-histories in Reference 1 calls for both the response spectrum and the power spectral density corresponding to the generated acceleration time-history to envelope their target (design basis) counterparts with only finite enveloping infractions. The time-histories for the WBN spent fuel pool have been generated to satisfy this preferred (and more rigorous) criterion. Figures 6.4.3-6.4.11 (a total of 9 figures) contain the graphical plots of the three time-histories for the SSE event, proof of response spectrum enveloping (3 figures), and proof of enveloping of the target spectral density (3 figures). The synthetic time-histories also satisfy the requirements of statistical independence mandated by Reference 1. These artificial time-histories are used in the non-linear dynamic simulations of the racks.

The OBE seismic accelerations for WBN are set at 50% of their SSE counterparts.

6.5.2 Modeling for Dynamic Simulation

The dynamic modeling of the rack structure is prepared with special consideration of the nonlinearities and parametric variations.

A rack may be completely loaded with fuel assemblies (which corresponds to greatest total mass), or it may be completely empty. The coefficient of friction, μ, between pedestal supports and pool floor is indeterminate. According to Rabinowicz,[13] results of 199 tests performed on austenitic stainless steel plates submerged in water show a mean value of μ to be 0.503 with standard deviation of 0.125. Upper and lower bounds (based on twice standard deviation) are 0.753 and 0.253, respectively. Analyses are therefore performed for coefficient of friction values of 0.2 (lower limit), 0.8 (upper limit) and random friction values clustered about a mean of 0.5. The bounding values of μ = 0.2 and 0.8 have been found to bracket the upper limit of module response in previous rerack projects.

Lift-off of support pedestals and subsequent liner impacts are modeled using impact (gap) elements, and Coulomb friction between rack and pool liner is simulated by piecewise linear (friction) elements. Rack elasticity, relative to the rack base, is included in the model with linear springs representing beam-like action, twisting, and extensions. These special attributes of rack dynamics require strong emphasis on modeling of linear and nonlinear springs, dampers, and compression-only gap elements. The term "nonlinear spring" is a generic term to denote the mathematical element representing the case where restoring force is not linearly proportional to displacement. In the fuel rack simulations, the Coulomb friction interface between rack support pedestal and liner is typical of a nonlinear spring.

Three-dimensional dynamic analyses of single rack modules require a key modeling assumption. This relates to location and relative motion of neighboring racks. The gap between a peripheral rack and adjacent pool wall is known, with motion of the pool wall prescribed. However, another rack, adjacent to the rack being analyzed, is also free-standing and subject to motion during a seismic event. To conduct the seismic analysis of a given rack, its physical interface with neighboring modules must be specified. There are two ways to consider the spacings between racks in single rack analysis. The first is to specify that neighboring racks move 180 out-of-phase in relation to the subject rack. Thus, the available gap before inter-rack impact occurs is 50% of the physical gap. This "opposed-phase motion" assumption increases the likelihood of inter-rack impacts and is thus conservative. However, it also increases the relative contribution of fluid coupling, which depends on fluid gaps and relative movements of bodies, making overall conservatism a less certain assertion. The alternative approach is to assume that all racks move in-phase. The entire array of racks move together as one body. Therefore, the critical dimensions are the boundary gaps between the fuel racks and the adjacent pool walls. This method of analysis predicts larger rack displacements and higher stress ratios, but the likelihood of inter-rack impacts is decreased. Three-dimensional WPMR analyses carried out on several previous plants demonstrate that single rack simulations underpredict rack displacement during seismic responses.[8] Nevertheless, 3-D analyses of single rack modules permit detailed evaluation of stress fields and serve as a kinematic benchmark check for the much more involved WPMR analysis.

Particulars of modeling details and assumptions for the 3-D Single Rack analysis for the new fuel racks and for the WPMR analysis for the entire array of racks are given in the following.

6.5.3 The 3-D 22-DOF Model for Single Rack Module Analysis of Maximum Density Racks

6.5.3.1 Assumptions

a. The fuel rack structure motion is captured by modeling the rack as a 12 degree-of-freedom structure. Movement of the rack cross-section at any height is described by six degrees of freedom of the rack base and six degrees of freedom at the rack top. In this manner, the beam-like response of the module, relative to the baseplate, is captured in the dynamic analyses once suitable springs are introduced to couple the rack degrees of freedom. Rattling fuel assemblies within the rack are modeled by five lumped masses located at H, .75H, .5H, .25H, and at the rack base (H is the rack height measured above the baseplate). Each lumped fuel mass has two horizontal displacement degrees of freedom. Vertical motion of the fuel assembly mass is assumed equal to rack vertical motion at the baseplate level. The centroid of each fuel assembly mass can be located off-center, relative to the rack structure centroid at that level, to simulate a partially loaded rack.

b. Seismic motion of a fuel rack is characterized by random rattling of fuel assemblies in their individual storage locations. The fuel assemblies are assumed to move in-phase within a rack. This exaggerates computed dynamic loading on the rack structure and, therefore, yields conservative results.

c. Fluid coupling between rack and fuel assemblies, and between rack and wall, is simulated by appropriate inertial coupling in the system kinetic energy. Inclusion of these effects uses the methods of References 14 and 15 for rack/assembly coupling and for rack-to-rack coupling. Fluid coupling terms for rack-to-rack coupling are based on either in-phase or opposed-phase motion of adjacent modules.

d. Fluid damping and form drag are conservatively neglected.

e. Sloshing is found to be negligible at the top of the rack and is therefore neglected in the analysis of the rack.

f. Potential impacts between the cell walls of the new racks and the contained fuel assemblies are accounted for by appropriate compression-only gap elements between masses involved. The possible incidence of rack-to-wall or rack-to-rack impact is simulated by gap elements at the top and bottom of the rack in two horizontal directions. Bottom gap elements are located at the baseplate elevation. The initial gaps reflect the presence of baseplate extensions.

g. Pedestals are modeled by gap elements in the vertical direction and as "rigid links" for transferring horizontal stress. Each pedestal support is linked to the pool liner (or bearing pad) by two friction springs. The spring rate for the friction springs includes any lateral elasticity of the stub pedestals. Local pedestal vertical spring stiffness accounts for floor elasticity and for local rack elasticity just above the pedestal.

h. Rattling of fuel assemblies inside the storage locations causes the gap between fuel assemblies and cell wall to change from a maximum of twice the nominal gap to a theoretical zero gap. Fluid coupling coefficients are based on the nominal gap.

6.5.3.2 Model Details for Spent Fuel Racks

Figure 6.5.1 shows a schematic of the dynamic model where p_i represents translational degrees of freedom, and q_i represents rotational degrees of freedom. H is the height of the rack above the baseplate.

Table 6.5.1 lists the degrees of freedom for the single rack model. Translational and rotational degrees of freedom, 1, 2, 3, 7, 8, 9 and 4, 5, 6,10, 11, 12, respectively, describe the rack motion; rattling fuel masses (nodes 1^*, 2^*, 3^*, 4^*, 5^* in Figure 6.5.1) are described by translational degrees of freedom 13-22. $U_i(t)$ represents pool floor slab displacement seismic time-history.

Figures 6.5.2 and 6.5.3, respectively, show inter-rack impact springs (to track potential for impact between racks or between rack and wall) and fuel assembly/storage cell impact springs at one location of rattling fuel assembly mass.

Figure 6.5.4 shows the modeling technique and degrees of freedom associated with rack elasticity. In each bending plane a shear and bending spring simulate elastic effects.[16] Linear elastic springs coupling rack vertical and torsional degrees of freedom are also included in the model.

Additional details concerning fluid coupling and determination of stiffness elements are provided below.

6.5.4 Fluid Coupling Effect

In its simplest form, the so-called "fluid coupling effect"[14,15] can be explained by considering the proximate motion of two bodies under water. If one body (mass m_1) vibrates adjacent to a second body (mass m_2), and both bodies are submerged in frictionless fluid, then Newton's equations of motion for the two bodies are:

$$(m_1 + M_{11})\ddot{X}_1 + M_{12}\ddot{X}_2 = \text{applied forces on mass } m_1 + O(X_1^2)$$

$$M_{21}\ddot{X}_1(m_2 + M_{22})\ddot{X}_2 = \text{applied forces on mass } m_2 + O(X_2^2)$$

$\ddot{X}_1$, $\ddot{X}_2$ denotes absolute accelerations of masses m_1 and m_2, respectively, and the notation $O(X^2)$ denotes nonlinear terms.

M_{11}, M_{12}, M_{21}, and M_{22} are fluid coupling coefficients which depend on body shape, relative disposition, etc. Fritz[15] gives data for M_{ij} for various body shapes and arrangements. The fluid adds mass to the body (M_{11} to mass m_1), and an inertial force proportional to acceleration of the adjacent body (mass m_2). Thus, acceleration of one body affects the force field on another. This force field is a function of inter-body gap, reaching large values for small gaps. Lateral motion of a fuel assembly inside a storage location encounters this effect. For example, fluid coupling is between nodes 2 and 2* in Figure 6.5.1. The rack analysis also contains inertial fluid coupling terms which model the effect of fluid in the gaps between adjacent racks. Terms modeling effects of fluid flowing between adjacent racks in a single rack analysis are computed assuming that all racks adjacent to the rack being analyzed are vibrating in-phase. Thus, the modeled rack is enclosed by a hydrodynamic mass computed based on the peripheral rack-to-wall gaps. Rack-to-rack gap elements have initial gaps set to 100% of the physical gap between the outermost racks and the adjacent pool walls.

6.5.5 Stiffness Element Details

The cartesian coordinate system associated with the rack has the following nomenclature:

x = Horizontal coordinate along the short direction of rack rectangular planform

y = Horizontal coordinate along the long direction of the rack rectangular planform

z = Vertical coordinate upward from the rack base

Table 6.5.2 lists the spring elements used in the 3-D 22-DOF single rack model.

If the simulation model is restricted to two dimensions (one horizontal motion plus vertical motion, for example), for the purpose of model clarification only, then Figure 6.5.5 describes the configuration. This simpler model is used to elaborate on the various stiffness modeling elements.

Gap elements modeling impacts between fuel assemblies and rack have local stiffness K_i in Figure 6.5.5. In Table 6.5.2, for example, gap elements 5 through 8 act on the rattling fuel mass at the rack top. Support pedestal spring rates K_S are modeled by elements 1 through 4 in Table 6.5.2. Local compliance of the concrete floor is included in K_S. Friction elements in the two orthogonal directions at each pedestal are shown in Figure 6.5.1. Friction at support/liner interface is modeled by the piecewise linear friction springs with suitably large stiffness K_f up to the limiting lateral load, μN, where N is the current compression load at the interface between support and liner. At every time-step during transient analysis, the current value of N (either zero if the pedestal has lifted off the liner, or a compressive finite value) is computed.

The gap element K_S, modeling the effective compression stiffness of the structure in the vicinity of the support, includes stiffness of the pedestal, local stiffness of the underlying pool slab, and local stiffness of the rack cellular structure above the pedestal.

The previous discussion is limited to a 2-D model solely for simplicity. Actual analyses incorporate 3-D motions and include the stiffness elements listed in Table 6.5.2.

6.5.6 Whole Pool Multi-Rack (WPMR) Model

6.5.6.1 General Remarks

The single rack 3-D (22-DOF) models for the new racks outlined in the preceding subsection are used to evaluate structural integrity and physical stability of the rack modules. Prescribing the motion of the racks adjacent to the module being analyzed is an assumption in the single rack simulations which cannot be defended on the grounds of conservatism. For closely spaced racks, demonstration of kinematic compliance is further verified by including the modules in one comprehensive simulation using a WPMR model. In WPMR analysis, the rack modules are modeled simultaneously and the coupling effect due to this multi-body motion is included in the analysis.

6.5.6.2 Multi-Body Fluid Coupling Phenomena

During the seismic event, the racks in the pool are subject to the input excitation simultaneously. The motion of each free-standing module would be autonomous and independent of others as long as they did not impact each other and no water were present in the pool. While the scenario of inter-rack impact is not a common occurrence, the effect of water the so-called fluid coupling effect is a universal factor. As noted in References 14 and 15, the fluid forces can reach rather large values in closely spaced rack geometries. It is, therefore, essential that the contribution of the fluid forces be included in a comprehensive manner. This is possible only if the racks in the pool are allowed to execute 3-D motion in the mathematical model. For this reason, single rack or even multi-rack models involving only a portion of the racks in the pool, are inherently inaccurate. The WPMR model removes this intrinsic limitation of the rack dynamic models by simulating the 3-D motion of all modules simultaneously. The fluid coupling effect, therefore, encompasses interaction between every set of racks in the pool, i.e., the motion of one rack produces fluid forces on other racks and on the pool walls. Stated more formally, both near-field and far-field fluid coupling effects are included in the analysis.

The derivation of the fluid coupling matrix[17] relies on the classical inviscid fluid mechanics principles, namely the principle of continuity and Kelvin's recirculation theorem. While the derivation of the fluid coupling matrix is based on no artificial construct, it has been, nevertheless, verified by an extensive set of shaketable experiments.[17]

For the cases where a slender, burnup credit "baby" rack is fastened to a larger freestanding PaR rack, there is no fluid coupling interaction between the baby and the mother rack. In such cases, including the "baby" rack in the simulation adds to the inertia load burden without providing a concomitant hydrodynamic benefit. Therefore, 3-D WPMR analyses which include baby racks, would bound the results of the scenario where the "baby" racks are absent.

6.5.6.3 Coefficients of Friction

To eliminate the last significant element of uncertainty in rack dynamic analyses, the friction coefficient ascribed to the support pedestal/pool bearing pad interface are made consistent with Rabinowicz's data.[13] Friction coefficients, developed by a random number generator with Gaussian normal distribution characteristics, are imposed on each pedestal of each rack in the pool. The assigned values are then held constant during the entire simulation in order to obtain reproducible results.[(1)] Thus, in this manner, the WPMR analysis results are brought closer to the realistic structural conditions.

6.5.6.4 Modeling Details

Figures 6.5.6 and 6.5.7 show the rack and pedestal numbering scheme used for WPMR analysis.

In WPMR analysis, a 16 degrees of freedom discretization set is used to model each rack plus contained fuel. The rack structure is modeled by twelve degrees of freedom. A dynamically consistent portion of contained fuel assemblies is assumed to rattle within the rack, while the remainder of the contained fuel is assumed as a distributed mass attached to the rack.

The rattling portion of the contained fuel is modeled by four horizontal degrees of freedom. Thus, the WPMR model involves the racks in the spent fuel pool with each individual rack and its fuel modeled as an 16-DOF structure.

The WPMR model includes gap elements representing compression-only pedestals, representing impact potential at fuel assembly-fuel rack interfaces, and at rack-to-rack or rack-to-wall locations at top and bottom corners of each rack module. Each pedestal has two friction elements associated with force in the vertical compression element. Values used for spring constants for the various stiffness elements are equal to the values used in the 22-DOF model.

6.5.7 Governing Equations of Motion

Using the structural model discussed in the foregoing, equations of motion corresponding to each degree of freedom are obtained using Lagrange's Formulation.[18] The system kinetic energy includes contributions from solid structures and from trapped and surrounding fluid. The final system of equations obtained have the matrix form:

$$[M]\,\{\ddot{q}\} = \{Q\} + \{G\}$$

(1) It is noted that DYNARACK has the capability to change the coefficient of friction at any pedestal at each instant of contact based on a random reading of the PC-clock cycle. However, exercising this option would yield results that could not be reproduced. Therefore, the random choice of coefficients is made only once per run.

where:

[M]	-	total mass matrix (including structural and fluid mass contributions). The size of this matrix will be 22x22 for a single rack analysis or 16n x 16n for a WPMR analysis (n = number of racks in the spent fuel pool).
{q}	-	the nodal displacement vector relative to the pool slab displacement (double dot stands for second derivatives with respect to time)
{G}	-	a vector dependent on the given ground acceleration
{Q}	-	a vector dependent on the spring forces (linear and nonlinear) and the coupling between degrees of freedom

The above column vectors have length 22 or 16n.

The matrix [Q] of the governing equation of motion includes a damping term. Structural damping follows established practice and is incorporated into the elastic portion of the model by introducing a structural damping matrix formed by associating linear structural damping coefficients with every linear spring in the model. Therefore, the Q matrix contains damping terms linearly proportional to velocity in addition to spring terms.

The equations can be rewritten as follows:

$$\{\ddot{q}\} = [M]^{-1}\{Q\} + [M]^{-1}\{G\}$$

This equation set is mass uncoupled, displacement coupled at each instant in time; numerical solution uses a central difference scheme built into the proprietary computer program DYNARACK.[19] The proprietary program DYNARACK has been validated against exact solutions, experimental data, and solutions obtained using alternate numerical schemes.

6.5.8 Structural Evaluation of Racks

The seismic/structural analysis of the modules was carried out using the methodology described in the preceding subsection. Based on considerations of location, aspect ratio and weight, the 7x9 rack modules were selected for single rack 3-D simulations. Single rack analyses were performed for a number of bounding scenarios.[11] Table 6.5.3 contains a listing of the single rack analyses performed.

In addition to the single rack analyses, a number of WPMR analyses were also carried out.[12] Table 6.5.4 provides a list of the WPMR simulations.

The results of the analyses can be summarized as follows:

(i) When fully loaded with intact fuel, the maximum stress develops, as expected, in the support-pedestal-to baseplate-junction. The largest compressive pedestal load in any pedestal in the pool based on multi-rack analysis is 120,000 lbs for SSE. Similarly, for the OBE case, as determined from multi-rack analysis, the largest compressive load is 86,300 lbs. for intact fuel.

(ii) The maximum stress factors for the SSE event are less than the allowable limit of 1.0. The largest stress factor R6 for the cellular region is 0.552 for intact fuel loading conditions. The largest stress factor R5 for a pedestal just below the baseplate is 0.684 for intact fuel loading conditions.

(iii) No rack-to-wall impact is indicated.

(iv) The rack modules exhibit large margins against overturning as evidenced by the maximum rack movements for the case with a 1.1 seismic amplifier.

(v) No rack-to-rack impact is indicated for intact fuel.

6.5.9 Fatigue Analysis of Racks

Because the spent fuel racks are deeply submerged in the spent fuel pool, in-service inspection of the critically loaded components of the racks cannot be performed reliably. Therefore, it is important to ensure that the cyclic stresses produced during the seismic events do not produce low cycle fatigue failure in the racks. To make this evaluation, the amplitude of the maximum stress intensity variation and the number of cycles during the 30 second seismic event are determined. The number of cycles is conveniently obtained from the time-history plots of the pedestal loads. Computation of the peak stress intensity, on the other hand, requires a detailed finite element model[20] of the most heavily loaded region, namely the pedestal-to-rack interface region. Figure 6.5.8 shows a schematic of the finite element model. The stress intensity at the critical rack location is plotted versus time for the SSE and the OBE seismic events in Figure 6.5.9 and 6.5.10. respectively. The cumulative damage factor is 0.732, well below the ASME Code limit of 1.0.

6.5.10 Analysis of Attachments for Burnup Credit Racks

Each Holtec "baby" rack is designed with at least four pedestals. Therefore, these racks are by themselves stable structures under static conditions. To further prevent any rack instability during a seismic event, the "baby" racks are attached to the adjacent PaR racks. The "baby" racks are connected to their parent racks at the pedestal level and at the top of the rack. Figures 3.3.1 and 3.3.2 show sketches of the "baby" rack attachments. The upper connection is achieved by placing two parallel bars through the cell region of the PaR rack which together with plate and channel provide a hook receptacle. The hook or joining attachment is welded to the periphery of the "baby" rack. WPMR analysis shows that the peak force that develops in any connecting bar is 28,450 lbs. This value results in a maximum tensile stress of 17,046 psi. The allowable stress limit according to Subsection NF of the ASME Code for Level D conditions is 1.2 times the yield stress of the material. For stainless steel, the allowable stress is 25,560 psi which is much greater than the calculated value.

6.6 REFERENCES

1. USNRC Standard Review Plan, NUREG-0800 (SRP 3.7.1, Revision 2, 1989).

2. "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, and January 18, 1979 amendment thereto.

3. ASME Boiler & Pressure Vessel Code Section III, Subsection NF, (1995).

4. American Concrete Institute, "Building Code Requirements for Structural Concrete," ACI 318-77.

5. USNRC Regulatory Guide 1.29, "Seismic Design Classification," Revision 3, 1978.

6. Soler, A.I. and Singh, K.P., "Seismic Responses of Free Standing Fuel Rack Constructions to 3-D Motions," Nuclear Engineering and Design, Vol. 80, pp. 315-329 (1984).

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10. ASME Boiler & Pressure Vessel Code, Section III, Appendices (1986).

11. Holtec Report HI-961485, "Single Rack Analysis of High Density Spent Fuel Racks."

12. Holtec Report HI-961505, "Seismic/Structural Analysis of Watts Bar Spent Fuel Racks," (WPMR)

13. Rabinowicz, E., "Friction Coefficients of Water Lubricated Stainless Steels for a Spent Fuel Rack Facility," MIT, a report for Boston Edison Company, 1976.

14. Singh, K.P. and Soler, A.I., "Dynamic Coupling in a Closely Spaced Two-Body System Vibrating in Liquid Medium: The Case of Fuel Racks," 3rd International Conference on Nuclear Power Safety, Keswick, England, May 1982.

15. Fritz, R.J., "The Effects of Liquids on the Dynamic Motions of Immersed Solids," Journal of Engineering for Industry, Trans. of the ASME, February 1972, pp 167-172.

16. Levy, S. and Wilkinson, J.P.D., "The Component Element Method in Dynamics with Application to Earthquake and Vehicle Engineering," McGraw Hill, 1976.

17. Paul, B., "Fluid Coupling in Fuel Racks: Correlation of Theory and Experiment," (Proprietary), NUSCO/Holtec Report HI-88243.

18. "Dynamics of Structures," R.W. Clough and J. Penzien, McGraw Hill (1975).

19. Soler, A.I., DYNARACK Validation Manual, Holtec Proprietary Report HI-91700, Revision 0, October, 1991.

20. ANSYS 5.2, Swanson Analysis Systems, 1995.

TABLE 6.4.1

PARTIAL LISTING OF FUEL RACK APPLICATIONS USING DYNARACK

PLANT	DOCKET NUMBER(s)	YEAR
Enrico Fermi Unit 2	USNRC 50-341	1980
Quad Cities 1 & 2	USNRC 50-254, 50-265	1981
Rancho Seco	USNRC 50-312	1982
Grand Gulf Unit 1	USNRC 50-416	1984
Oyster Creek	USNRC 50-219	1984
Pilgrim	USNRC 50-293	1985
V.C. Summer	USNRC 50-395	1984
Diablo Canyon Units 1 & 2	USNRC 50-275, 50-323	1986
Byron Units 1 & 2	USNRC 50-454, 50-455	1987
Braidwood Units 1 & 2	USNRC 50-456, 50-457	1987
Vogtle Unit 2	USNRC 50-425	1988
St. Lucie Unit 1	USNRC 50-335	1987
Millstone Point Unit 1	USNRC 50-245	1989
D.C. Cook Units 1 & 2	USNRC 50-315, 50-316	1992
Indian Point Unit 2	USNRC 50-247	1990
Three Mile Island Unit 1	USNRC 50-289	1991
James A. FitzPatrick	USNRC 50-333	1990
Shearon Harris Unit 2	USNRC 50-401	1991
Hope Creek	USNRC 50-354	1990
Kuosheng Units 1 & 2	Taiwan Power Company	1990
Ulchin Unit 2	Korea Electric Power Co.	1990

TABLE 6.4.1 (continued)

PARTIAL LISTING OF FUEL RACK APPLICATIONS USING DYNARACK

PLANT	DOCKET NUMBER(s)	YEAR
Laguna Verde Units 1 & 2	Comision Federal de Electricidad	1991
Zion Station Units 1 & 2	USNRC 50-295, 50-304	1992
Sequoyah	USNRC 50-327, 50-328	1992
LaSalle Unit 1	USNRC 50-373	1992
Duane Arnold Energy Center	USNRC 50-331	1992
Fort Calhoun	USNRC 50-285	1992
Nine Mile Point Unit 1	USNRC 50-220	1993
Beaver Valley Unit 1	USNRC 50-334	1992
Salem Units 1 & 2	USNRC 50-272, 50-311	1993
Limerick	USNRC 50-352, 50-353	1994
Ulchin Unit 1	KINS	1995
Yonggwang Units 1 & 2	KINS	1996
Kori-4	KINS	1996
Connecticut Yankee	USNRC 50-213	1996
Angra Unit 1	Brazil	1996
Sizewell B	United Kingdom	1996

TABLE 6.4.2 RACK MATERIAL DATA (200 F)			
Holtec Racks			
Component: Material	Young's Modulus, E (psi)	Yield Strength, S_y (psi)	Ultimate Strength, S_u (psi)
Cell Structure: SA240-304L.	27.7×10^6	21,300	66,200
Female Pedestal: SA240-304L	27.7×10^6	21,300	66,200
Male Pedestal: SA564-630 (age hardened at 1100°F)	27.7×10^6	106,300	140,000
Flux Trap Racks			
Component: Material	Young's Modulus, E (psi)	Yield Strength, S_y (psi)	Ultimate Stremgth, S_u (psi)
Poison Can: 304 S.S.	27.7×10^6	41,700	66,200
Grid Castings: CF-3M	27.7×10^6	27,800	66,200
Pedestal: 17-4 PH (age hardened at 1100°F)	27.7×10^6	115,000	140,000

TABLE 6.5.1

DEGREES OF FREEDOM

LOCATION (Node)	DISPLACEMENT			ROTATION		
	U_x	U_y	U_z	θ_x	θ_y	θ_z
1	p_1	p_2	p_3	q_4	q_5	q_6
2	p_7	p_8	p_9	q_{10}	q_{11}	q_{12}
Point 2 is assumed attached to rigid rack at the top most point.						
2*	p_{13}	p_{14}				
3*	p_{15}	p_{16}				
4*	p_{17}	p_{18}				
5*	p_{19}	p_{20}				
1*	p_{21}	p_{22}				

where the relative displacement variables q_i are defined as:

$p_i = q_i(t) + U_1(t)$ $\quad$ i = 1,7,9,11,13,15,17
$p_i = q_i(t) + U_2(t)$ $\quad$ i = 2,8,10,12,14,16,18
$p_i = q_i(t) + U_3(t)$ $\quad$ i = 3,19

$U_i(t)$ are the three known earthquake displacements.

TABLE 6.5.2 NUMBERING SYSTEM FOR GAP ELEMENTS AND FRICTION ELEMENTS		
I. Nonlinear Springs (Gap Elements) (40 Total)		
Number	Node Location	Description
1	Support S1	Z compression-only element
2	Support S2	Z compression-only element
3	Support S3	Z compression-only element
4	Support S4	Z compression-only element
5	2,2*	X rack/fuel assembly impact element
6	2,2*	X rack/fuel assembly impact element
7	2,2*	Y rack/fuel assembly impact element
8	2,2*	Y rack/fuel assembly impact element
9-24	Other rattling masses for nodes 1*, 3*, 4* and 5*	
25-32	Bottom edge of rack cross section	Inter-rack impact elements
33-40	Top edge of rack cross section	Inter-rack impact elements
II. Non-Linear Springs (Friction Elements)		
1	Support S1	X direction friction
2	Support S1	Y direction friction
3	Support S2	X direction friction
4	Support S2	Y direction friction
5	Support S3	X direction friction
6	Support S4	Y direction friction
7	Support S4	X direction friction
8	Support S4	Y direction friction

TABLE 6.5.3 LIST OF SINGLE RACK SIMULATIONS (All simulations include baby rack and consolidated fuel except as noted)				
Run No.	Rack Size	Seismic Event	Fuel Loading	Coefficient of Friction
1	7 x 9	1.0 x SSE	Fully Loaded	0.2
2	7 x 9	1.0 x SSE	Half Loaded Along X Axis	0.2
3	7 x 9	1.0 x SSE	Half Loaded Along Y Axis	0.2
4	7 x 9	1.0 x SSE	Fully Loaded	0.8
5	7 x 9	1.0 x SSE	Half Loaded Along X Axis	0.8
6	7 x 9	1.0 x SSE	Half Loaded along Y Axis	0.8
7	7 x 9	1.1 x SSE	Fully Loaded	0.8
8	7 x 9	1.2 x OBE (0.6 x SSE)	Limiting Case from Run Nos. 1 through 6	
9	7 x 9*	1.0 x SSE	Limiting Case from Run Nos. 1 through 6	
10	7 x 9**	1.0 x SSE	Limiting Case from Run Nos. 1 through 6	
11	7 x 9***	1.0 x SSE	Limiting Case from Run Nos. 1 through 6	
12	15 x 15†	1.0 x SSE	Limiting Case from Run Nos. 1 through 6	

* 7 x 9 rack is loaded with consolidated fuel. No baby rack attached. Effective fuel assembly weight equals 3000 lbf.

** 7 x 9 rack and attached baby rack are loaded with intact fuel. Effective fuel assembly weight equals 2424 lb.

*** Effective fuel assembly weight equals 2873 lbf. Three loading scenarios represented by this weight are listed below. Other combinations which produce an effective weight equal to 2873 lbf are also possible.

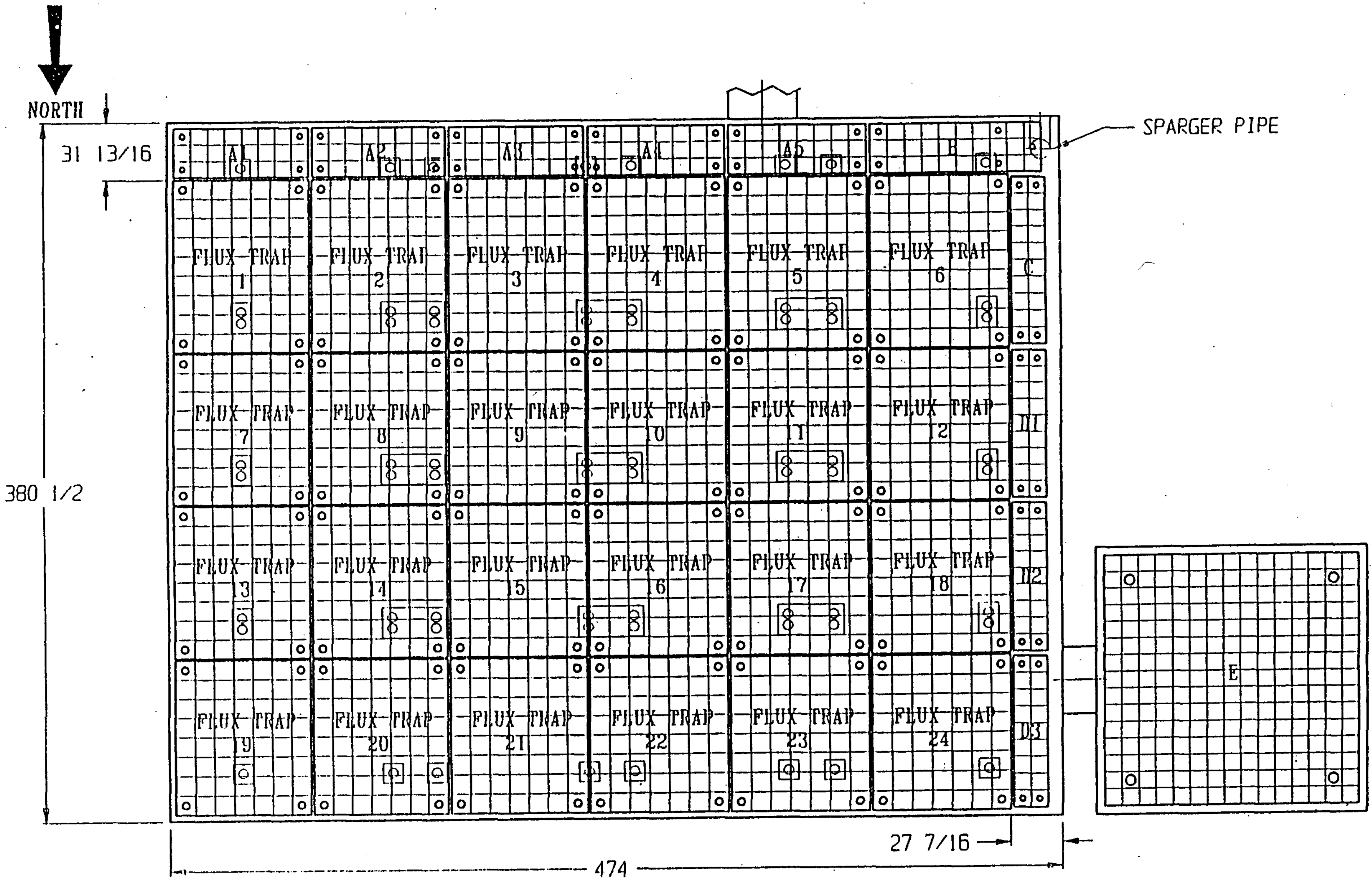
1) A 7 x 9 rack loaded with intact fuel (10%) and consolidated fuel (90%). No baby rack attached.
2) A 7 x 9 rack and attached baby rack loaded with intact fuel (75%) and consolidated fuel (25%).
3) A 7 x 9 rack loaded with intact fuel (65%) and consolidated fuel (35%). Attached baby rack loaded with intact fuel.

† 15 x 15 rack in the cask pit is loaded with consolidated fuel. No baby rack attached. Effective fuel assembly weight equals 3000 lbf.

TABLE 6.5.4

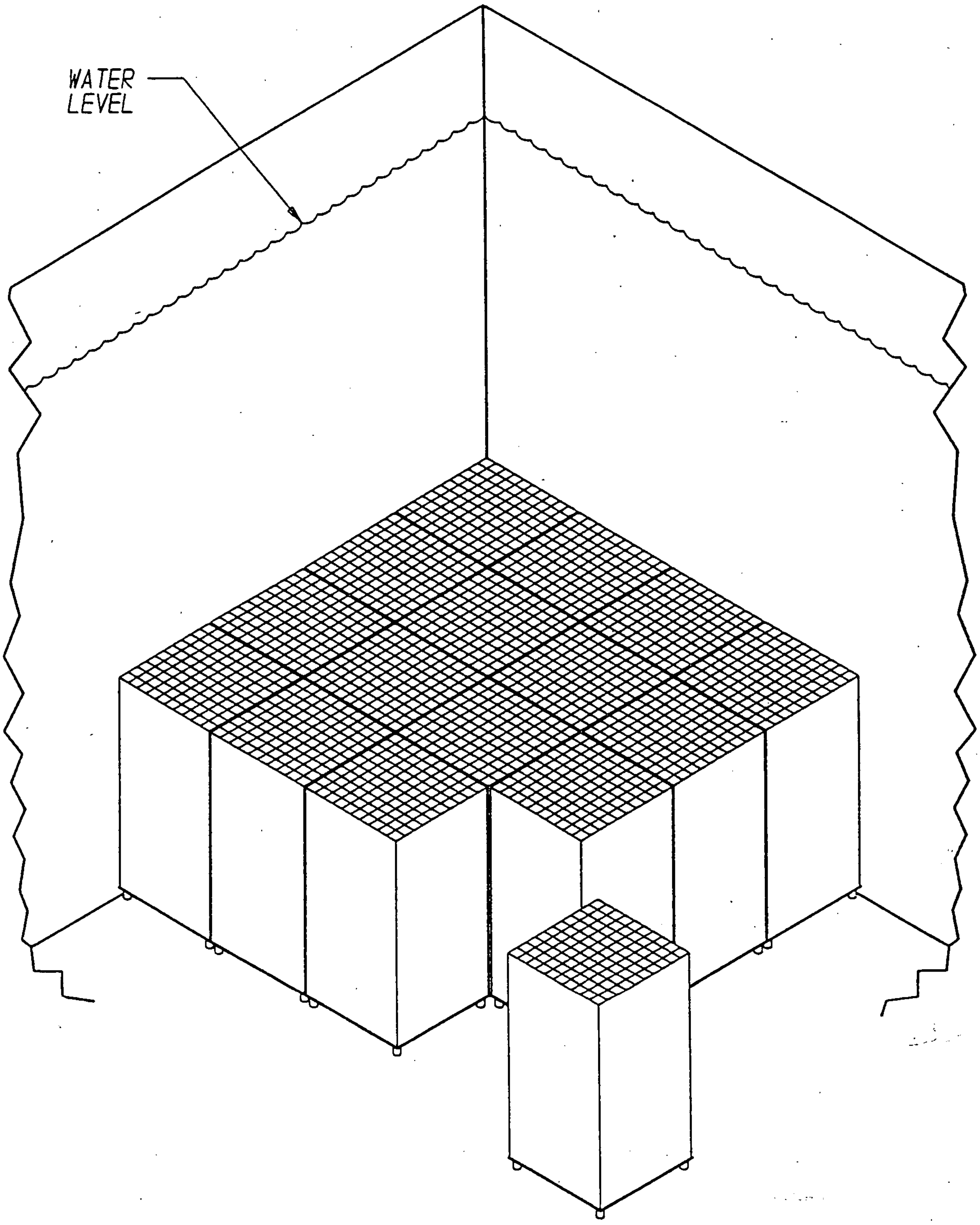
LIST OF WHOLE POOL MULTI-RACK (WPMR) SIMULATIONS

Run No.	Seismic Event	Fuel Loading	Coefficient of Friction
1	1.0 x SSE	Fully loaded with intact fuel, 1700 lbf per assembly	Guassian distribution with a mean of 0.5 (upper and lower limits of 0.8 and 0.2)
2	1.0 x SSE	Fully loaded with consolidated fuel, 3000 lbf per assembly	Guassian distribution with a mean of 0.5 (upper and lower limita of 0.8 and 0.2)
3	1.0 x SSE	Fully loaded with fuel (75% consolidated, 25% intact), 2675 lbf per assembly	Guassian distribution with a mean of 0.5 (upper and lower limits of 0.8 and 0.2)
4	1.0 x OBE (0.5 x SSE)	Fully loaded with intact fuel, 1700 lbf per assembly	Guassian distribution with a mean of 0.5 (upper and lower limits of 0.8 and 0.2)



POOL LAYOUT FOR WATTS BAR

Figure 6.4.1



TYPICAL ARRAY OF HIGH DENSITY SPENT FUEL RACKS

Figure 6.4.2

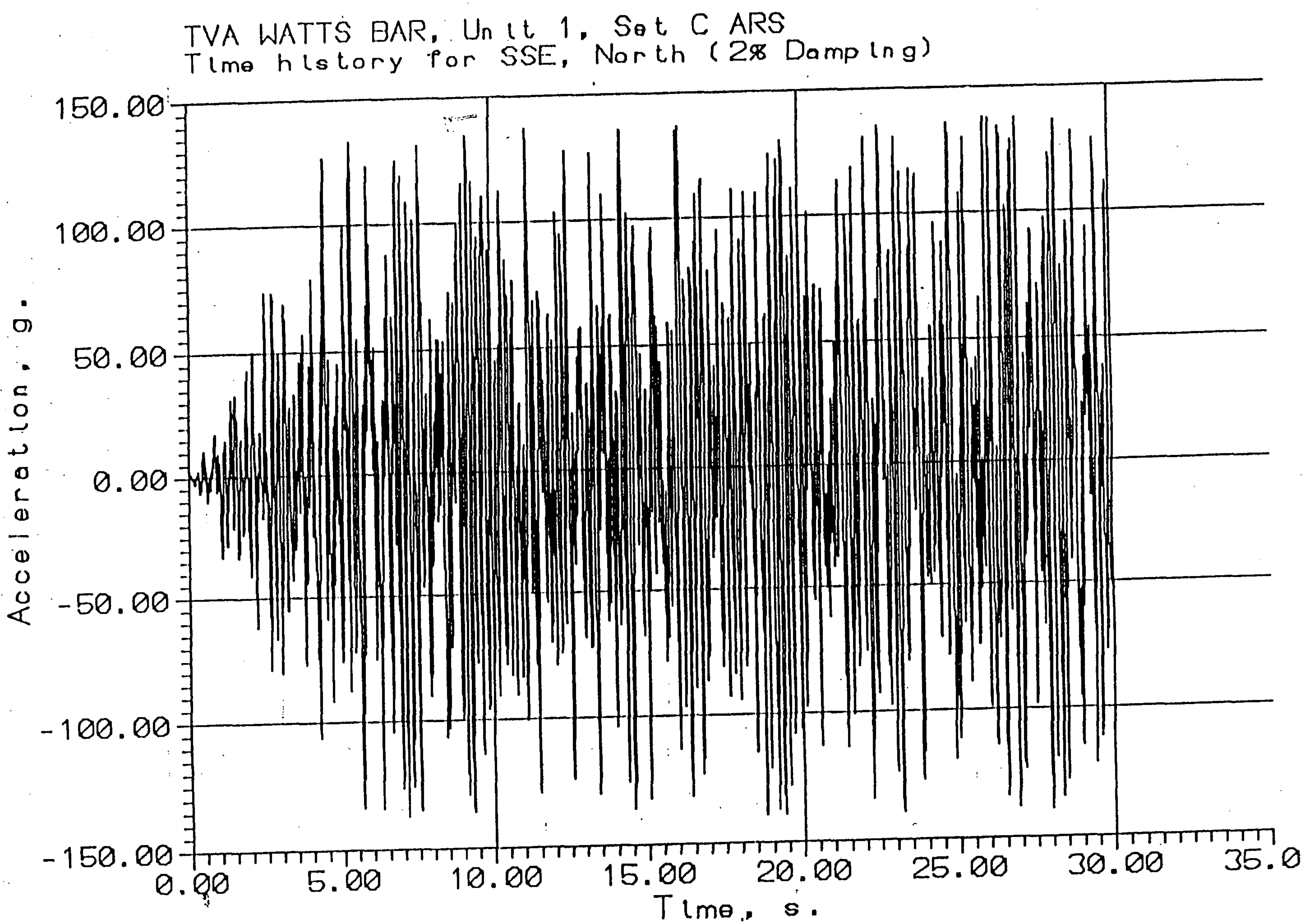

TVA WATTS BAR, Unit 1, Set C ARS
Time history for SSE, North (2% Damping)
Acceleration, g.
150.00
100.00
50.00
0.00
-50.00
-100.00
-150.00
0.00
5.00
10.00
15.00
20.00
25.00
30.00
35.0
Time, s.

Figure 6.4.3

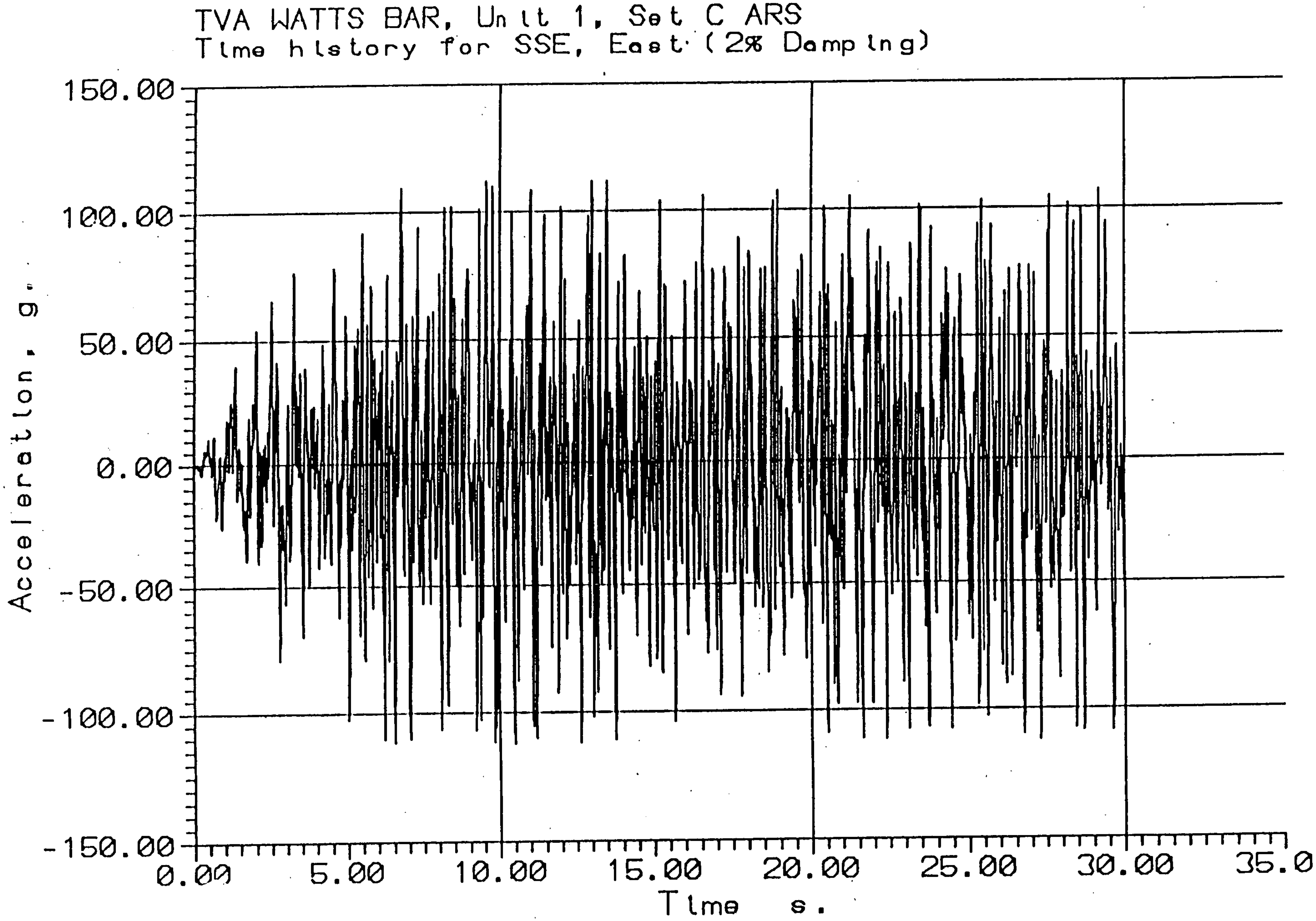
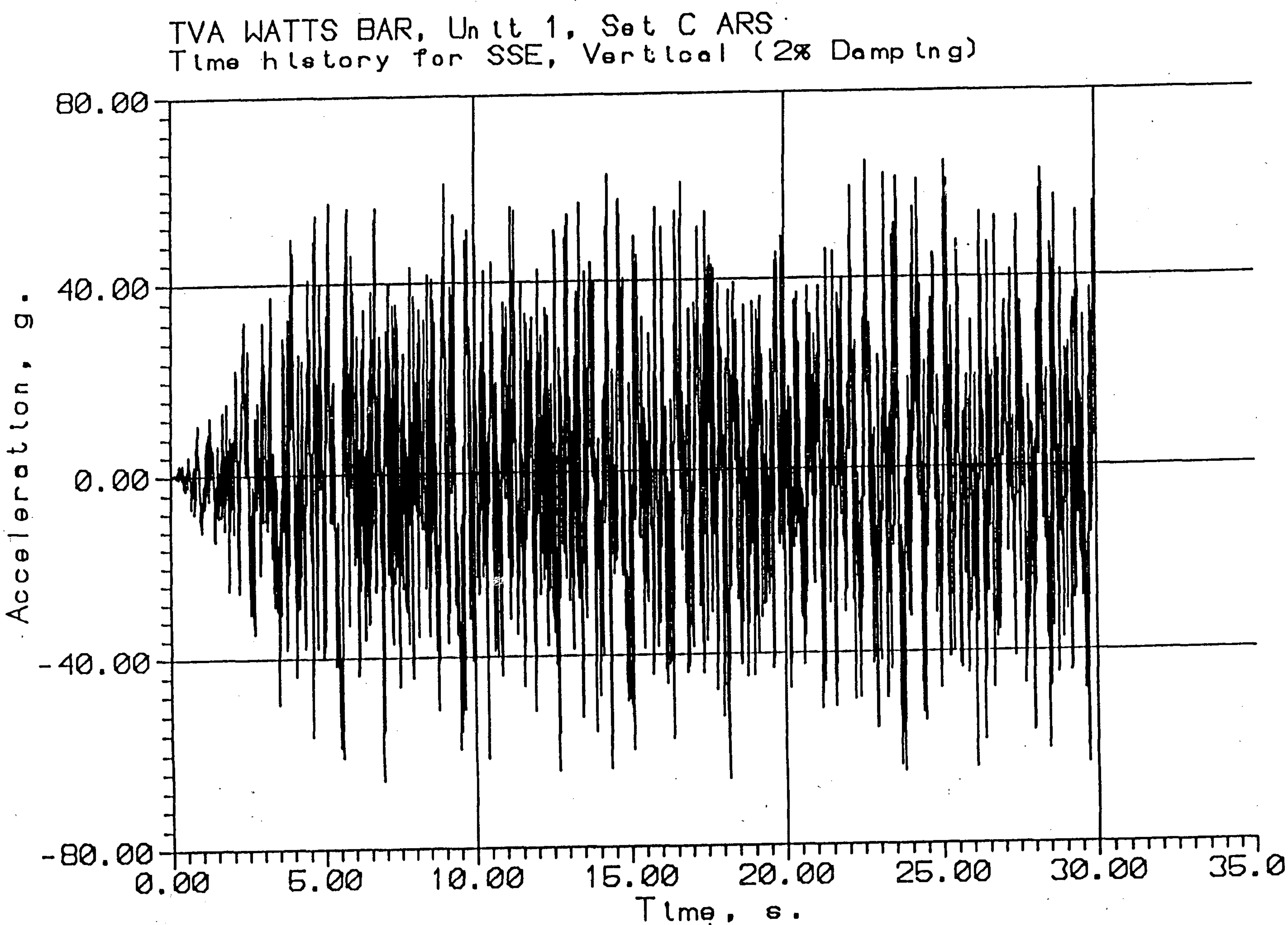

TVA WATTS BAR, Unit 1, Set C ARS
Time history for SSE, East (2% Damping)
150.00
100.00
50.00
0.00
-50.00
-100.00
-150.00
Acceleration, g.
0.00
5.00
10.00
15.00
20.00
25.00
30.00
35.0
Time s.

Figure 6.4.4



TVA WATTS BAR, Unit 1, Set C ARS
Time history for SSE, Vertical (2% Damping)
80.00
40.00
0.00
-40.00
-80.00
Acceleration, g.
0.00
5.00
10.00
15.00
20.00
25.00
30.00
35.0
Time, s.

Figure 6.4.5

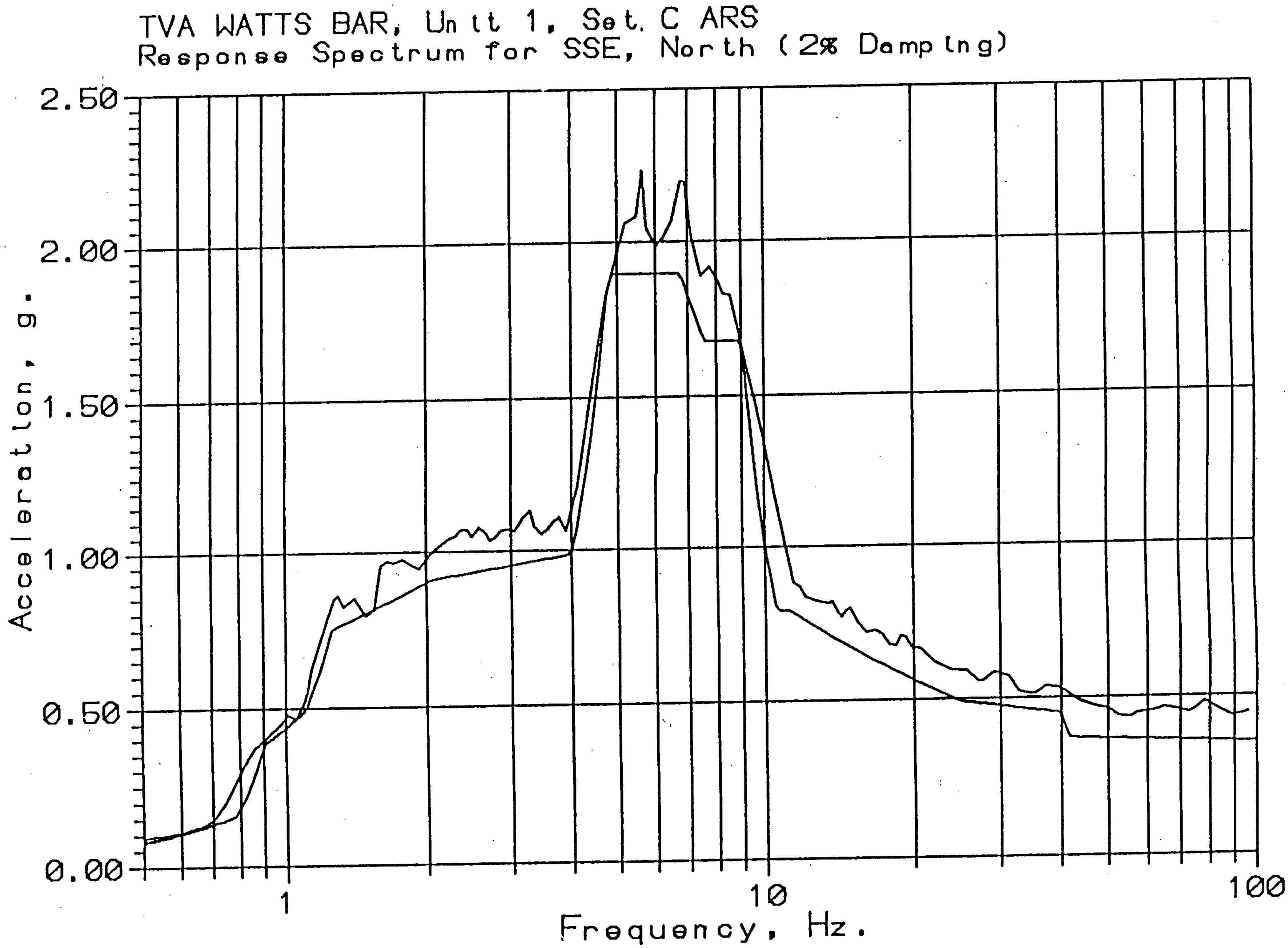

TVA WATTS BAR, Unit 1, Set C ARS
Response Spectrum for SSE, North (2% Damping)
2.50
2.00
1.50
1.00
0.50
0.00
Acceleration, g.
1
10
100
Frequency, Hz.

Figure 6.4.6

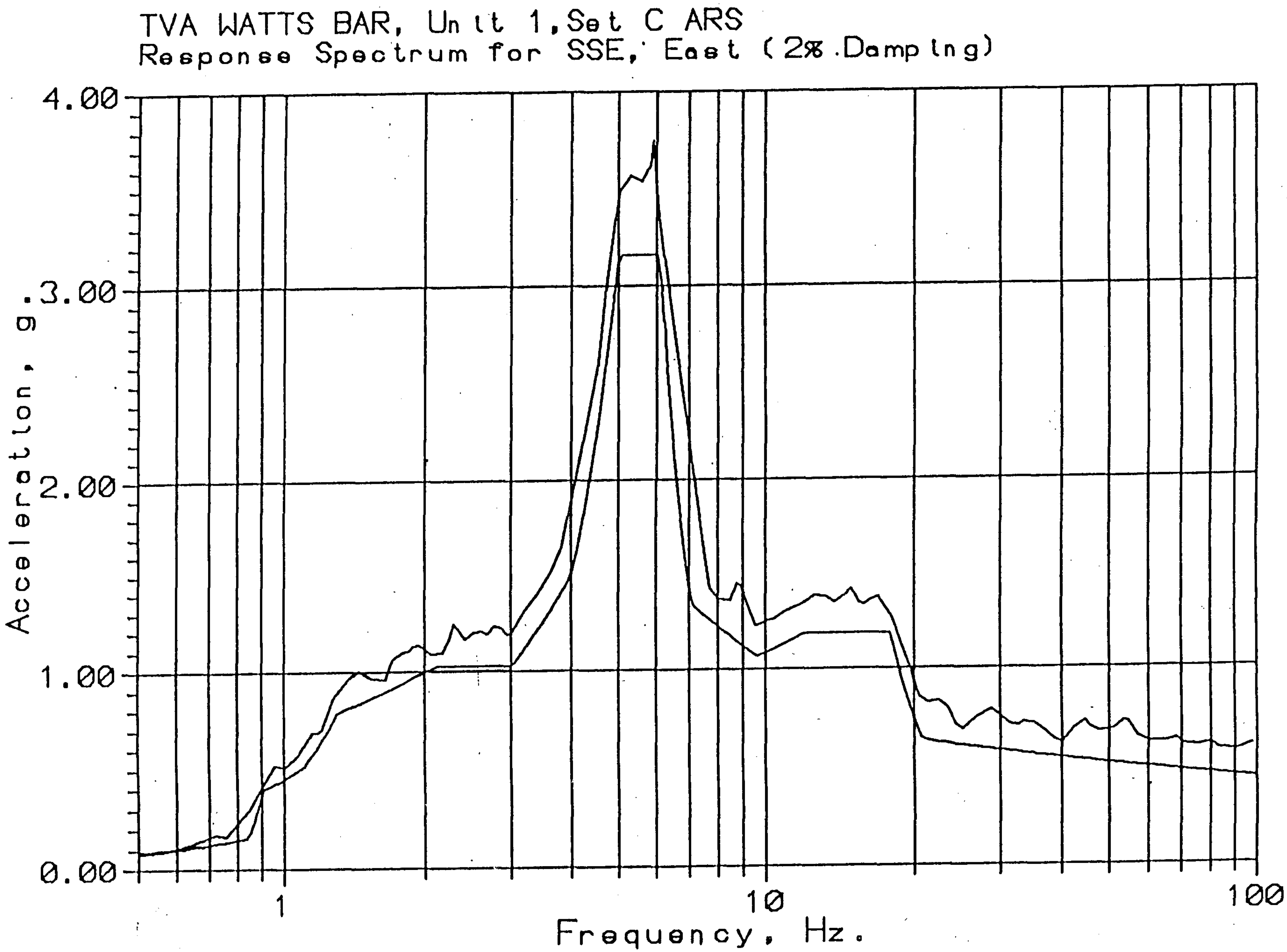
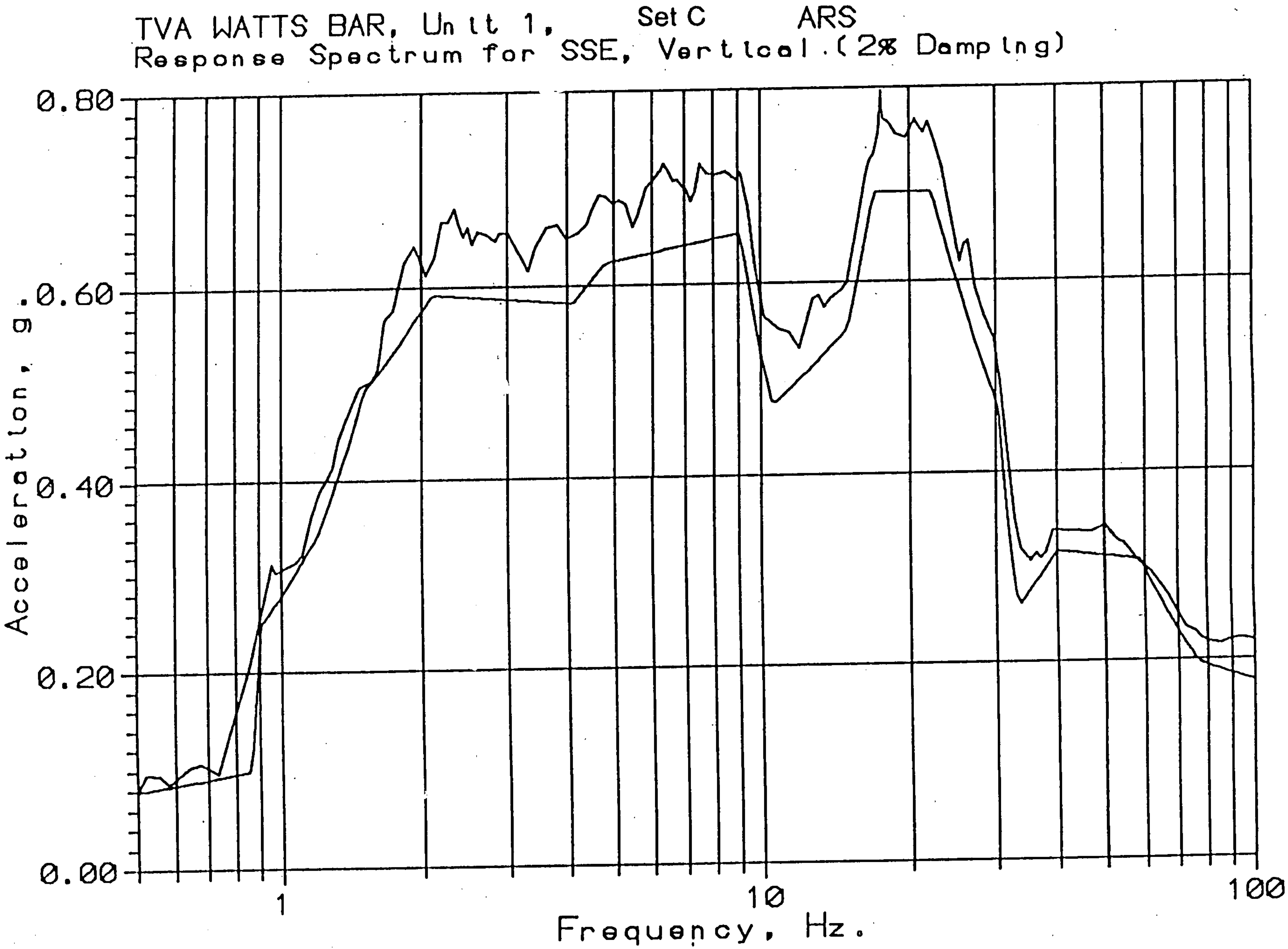

TVA WATTS BAR, Unit 1, Set C ARS
Response Spectrum for SSE, East (2% Damping)
4.00
3.00
2.00
1.00
0.00
Acceleration, g.
1
10
100
Frequency, Hz.

Figure 6.4.7



TVA WATTS BAR, Unit 1, Set C ARS
Response Spectrum for SSE, Vertical (2% Damping)
0.80
0.60
0.40
0.20
0.00
Acceleration, g
1
10
100
Frequency, Hz.

Figure 6.4.8

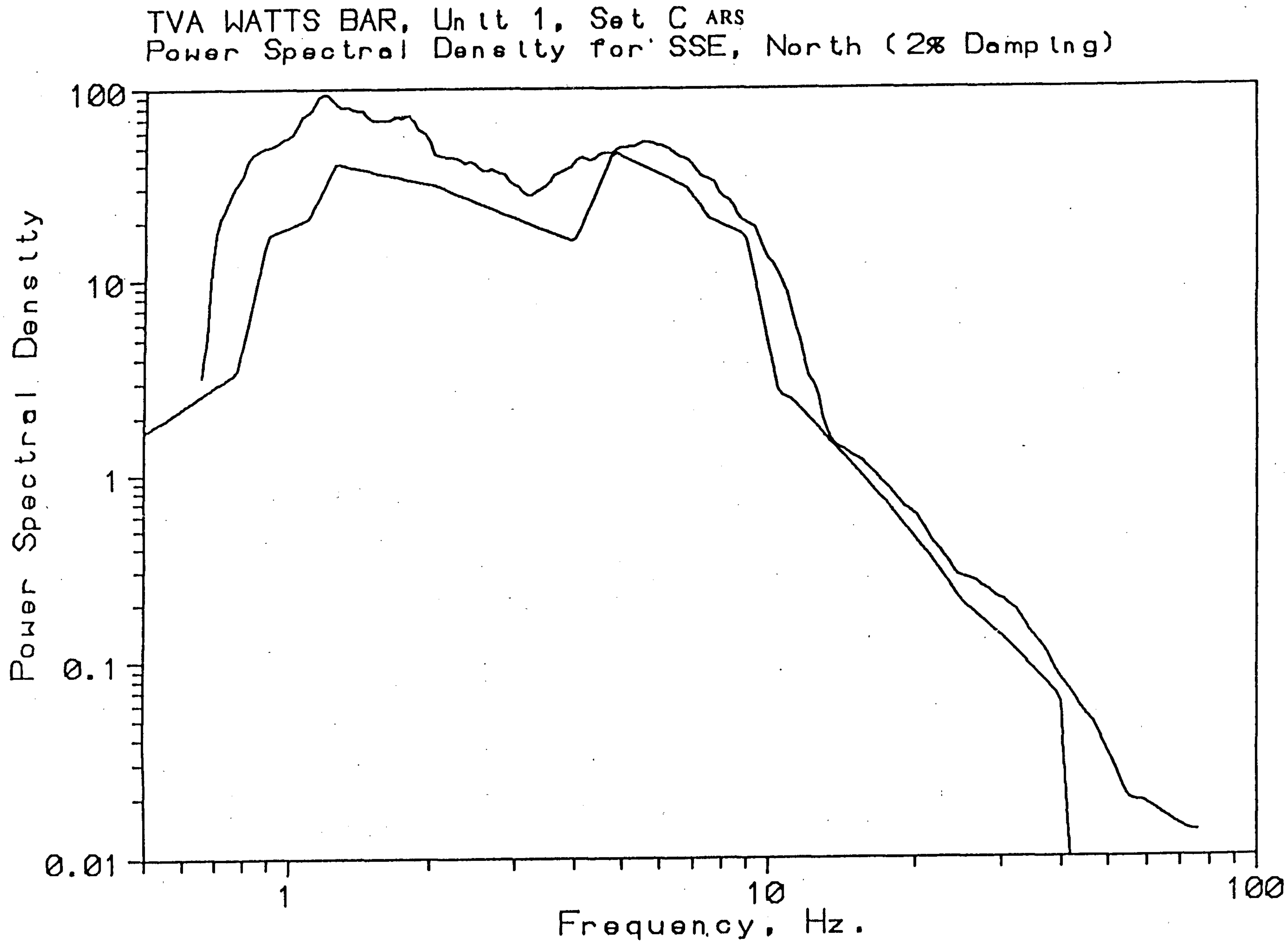

TVA WATTS BAR, Unit 1, Set C ARS
Power Spectral Density for SSE, North (2% Damping)
Power Spectral Density
100
10
1
0.1
0.01
1
10
100
Frequency, Hz.

Figure 6.4.9

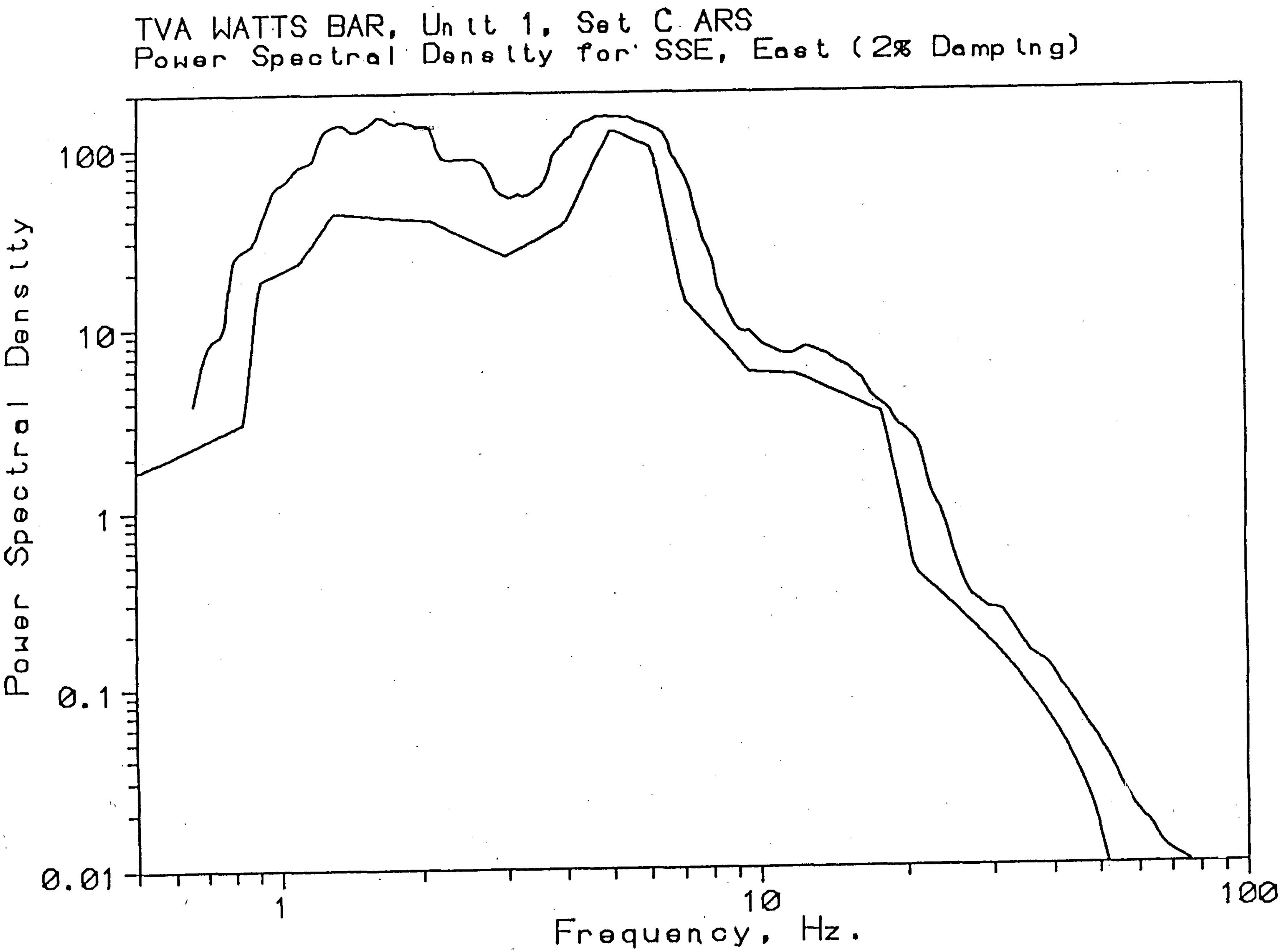
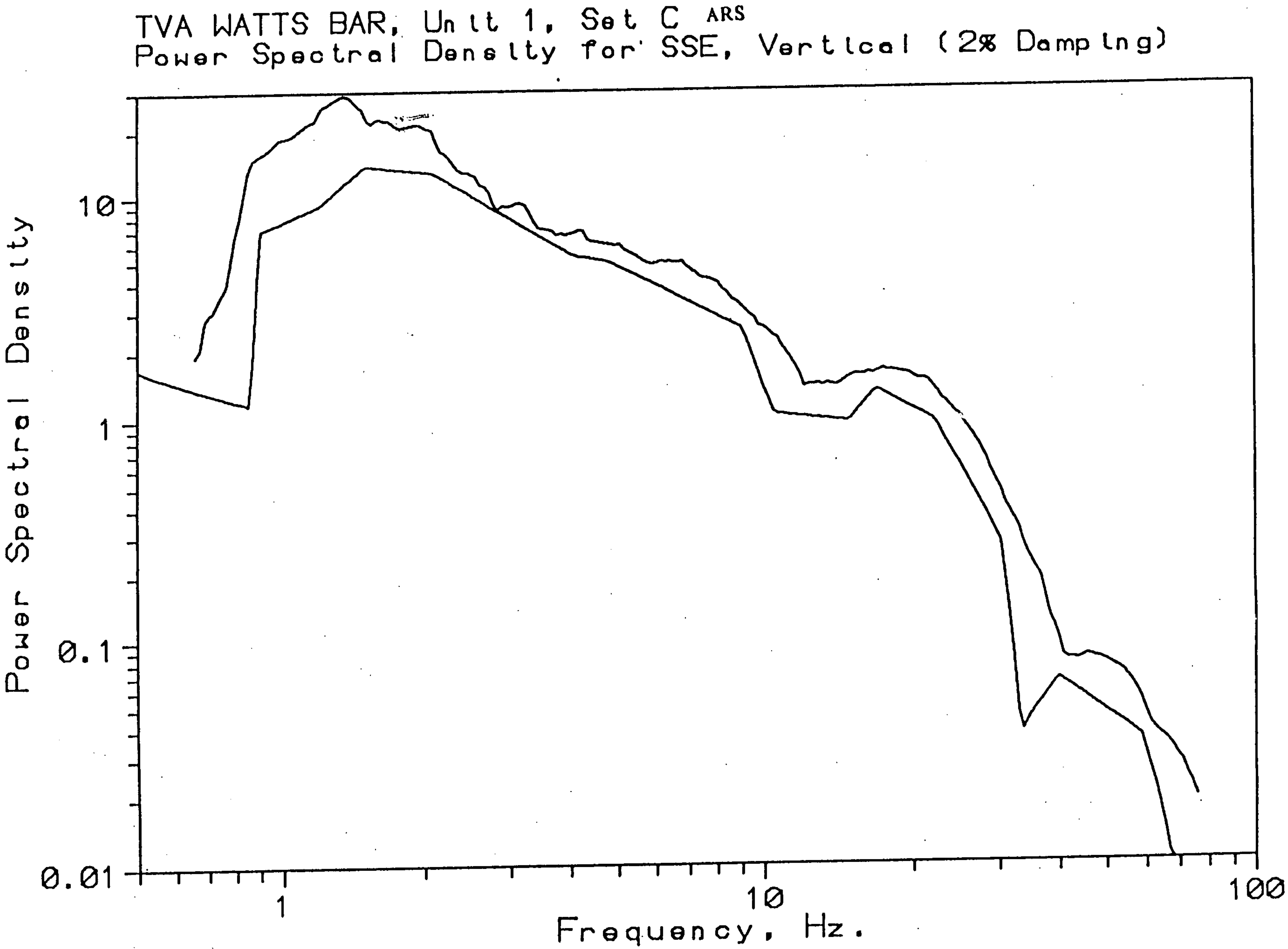
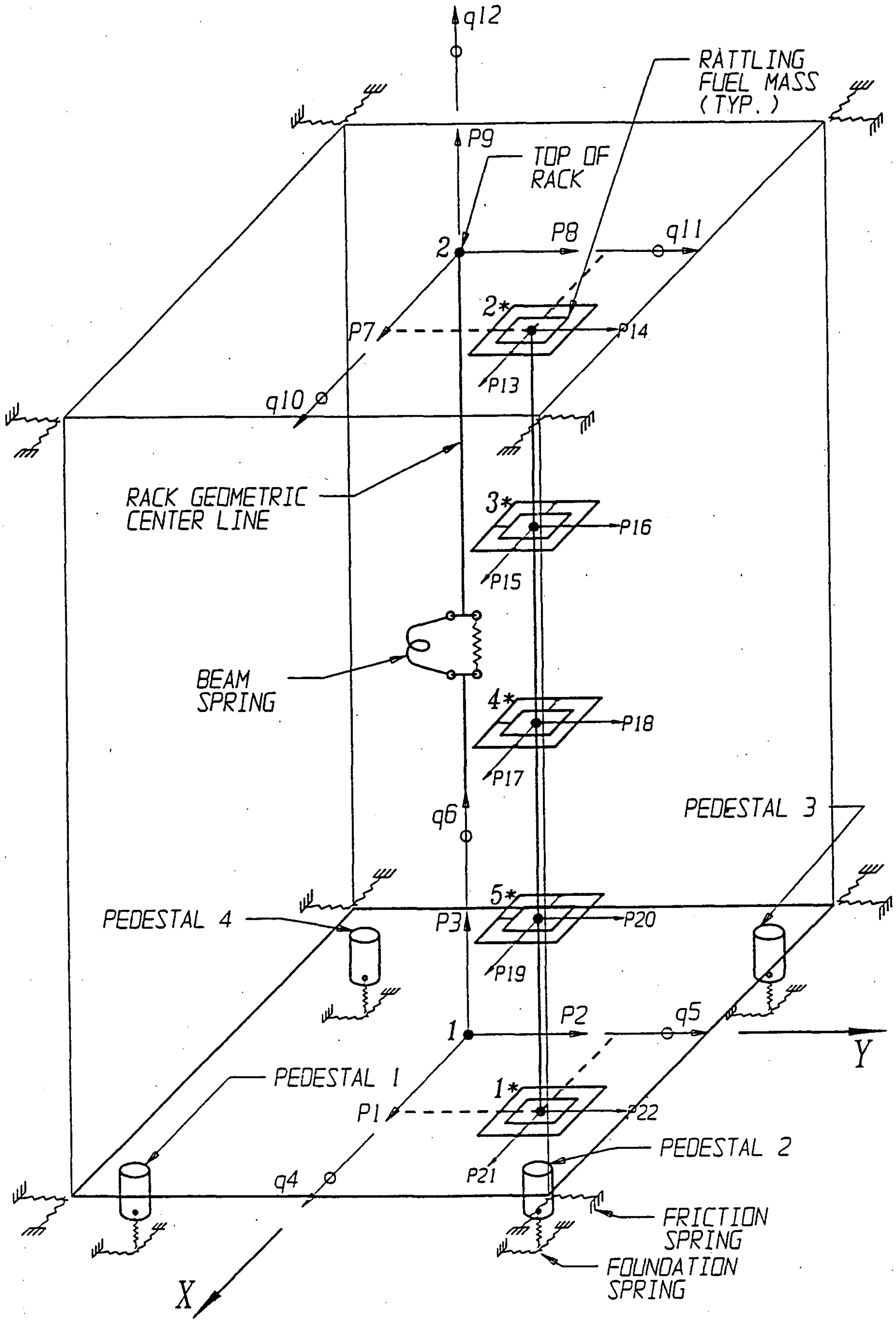

TVA WATTS BAR, Unit 1, Set C ARS
Power Spectral Density for SSE, East (2% Damping)
Power Spectral Density
100
10
1
0.1
0.01
1
10
100
Frequency, Hz.

Figure 6.4.10



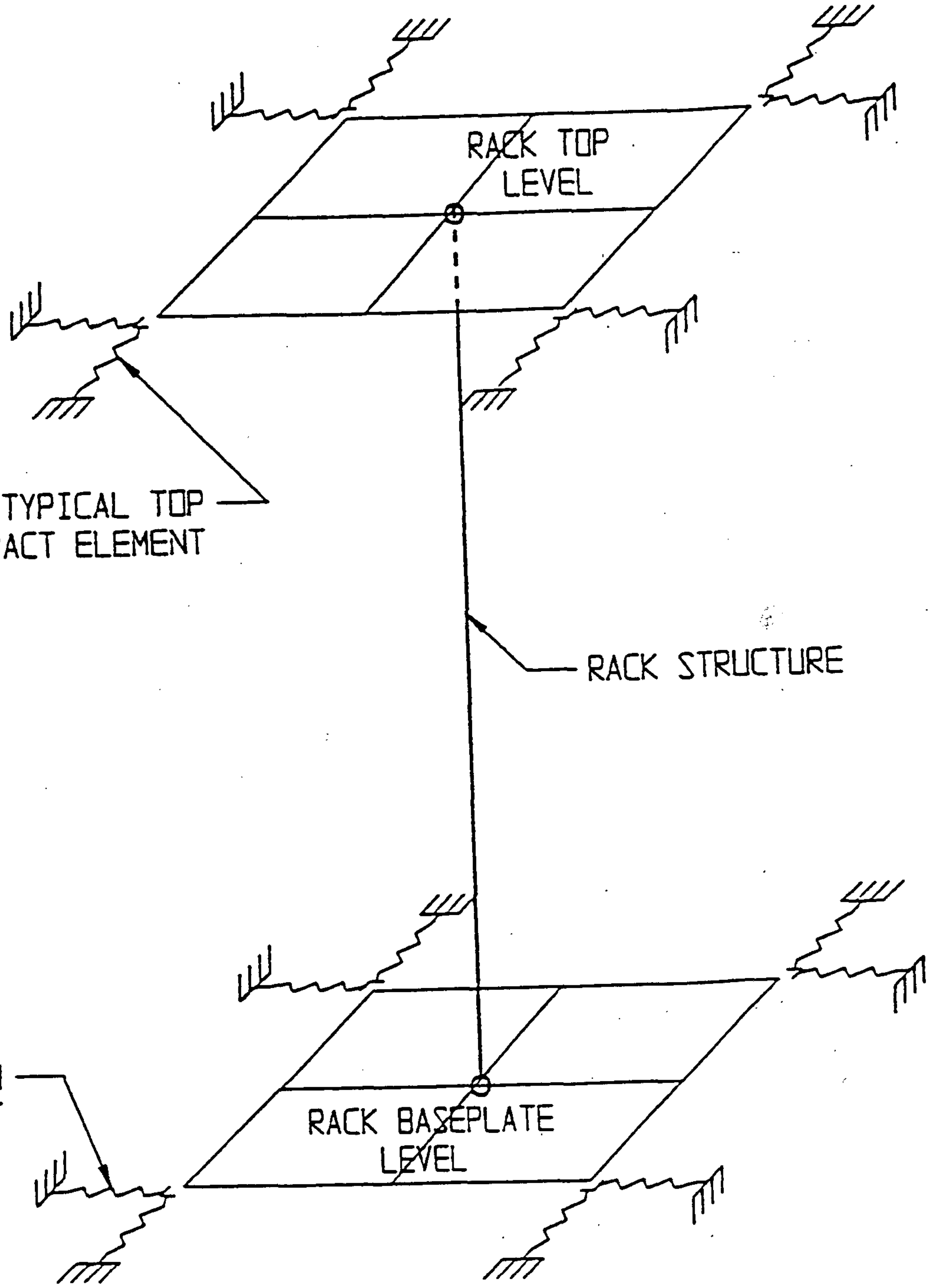
TVA WATTS BAR, Unit 1, Set C ARS
Power Spectral Density for SSE, Vertical (2% Damping)
Power Spectral Density
10
1
0.1
0.01
1
10
100
Frequency, Hz.

Figure 6.4.11



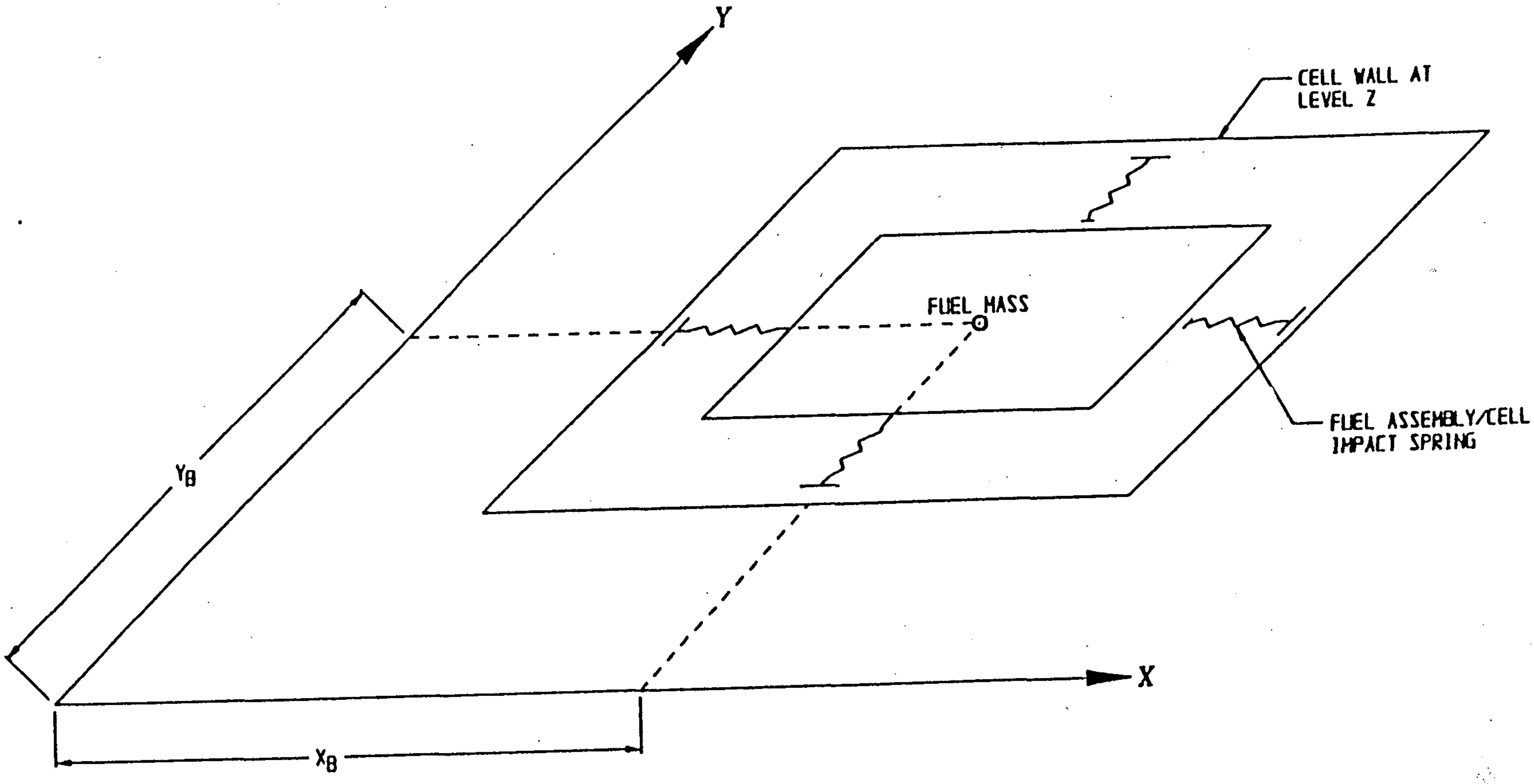
SCHEMATIC OF THE DYNAMIC MODEL FOR DYNARACK

Figure 6.5.1



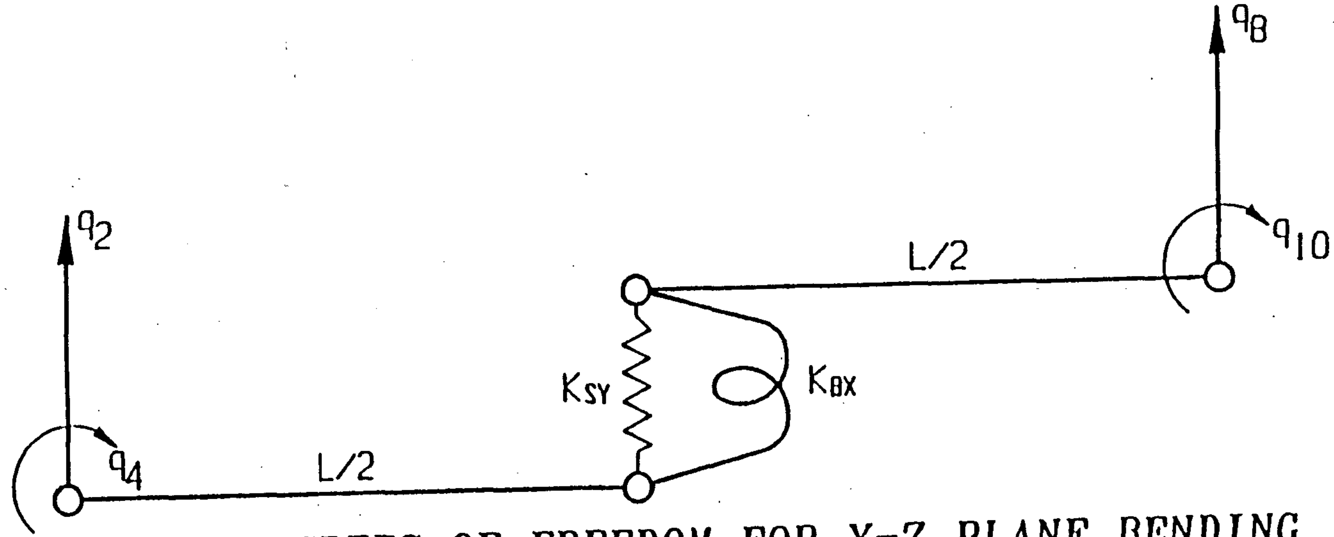
RACK-TO-RACK IMPACT SPRINGS

Figure 6.5.2

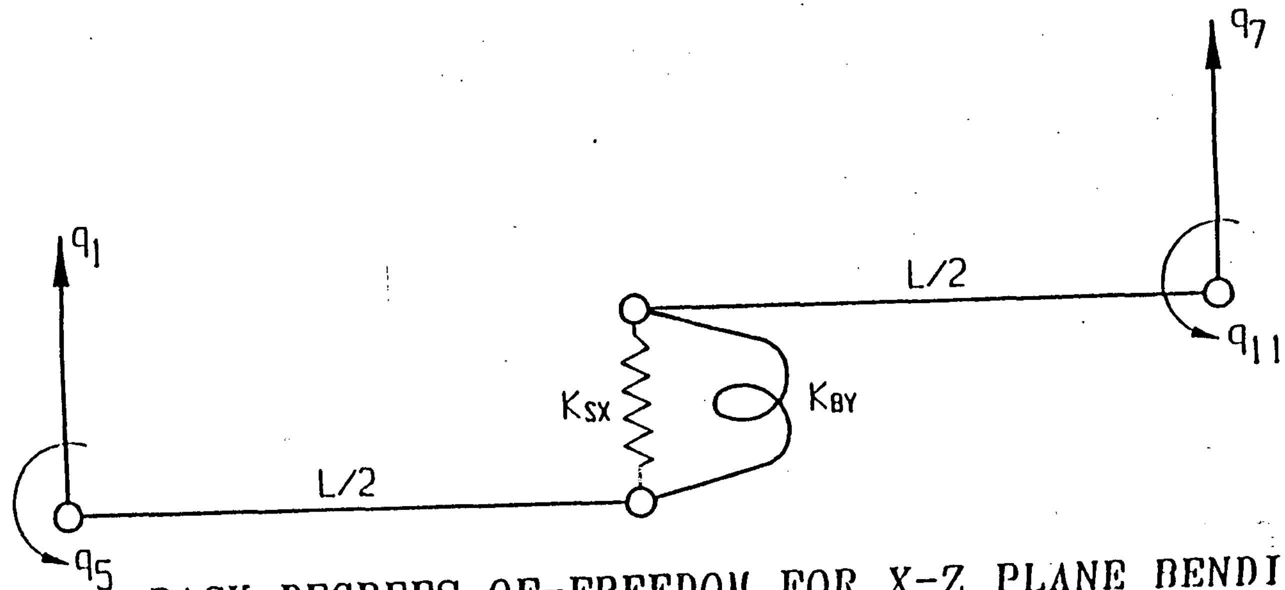


FUEL-TO-RACK IMPACT SPRINGS AT LEVEL OF RATTLING MASS

Figure 6.5.3

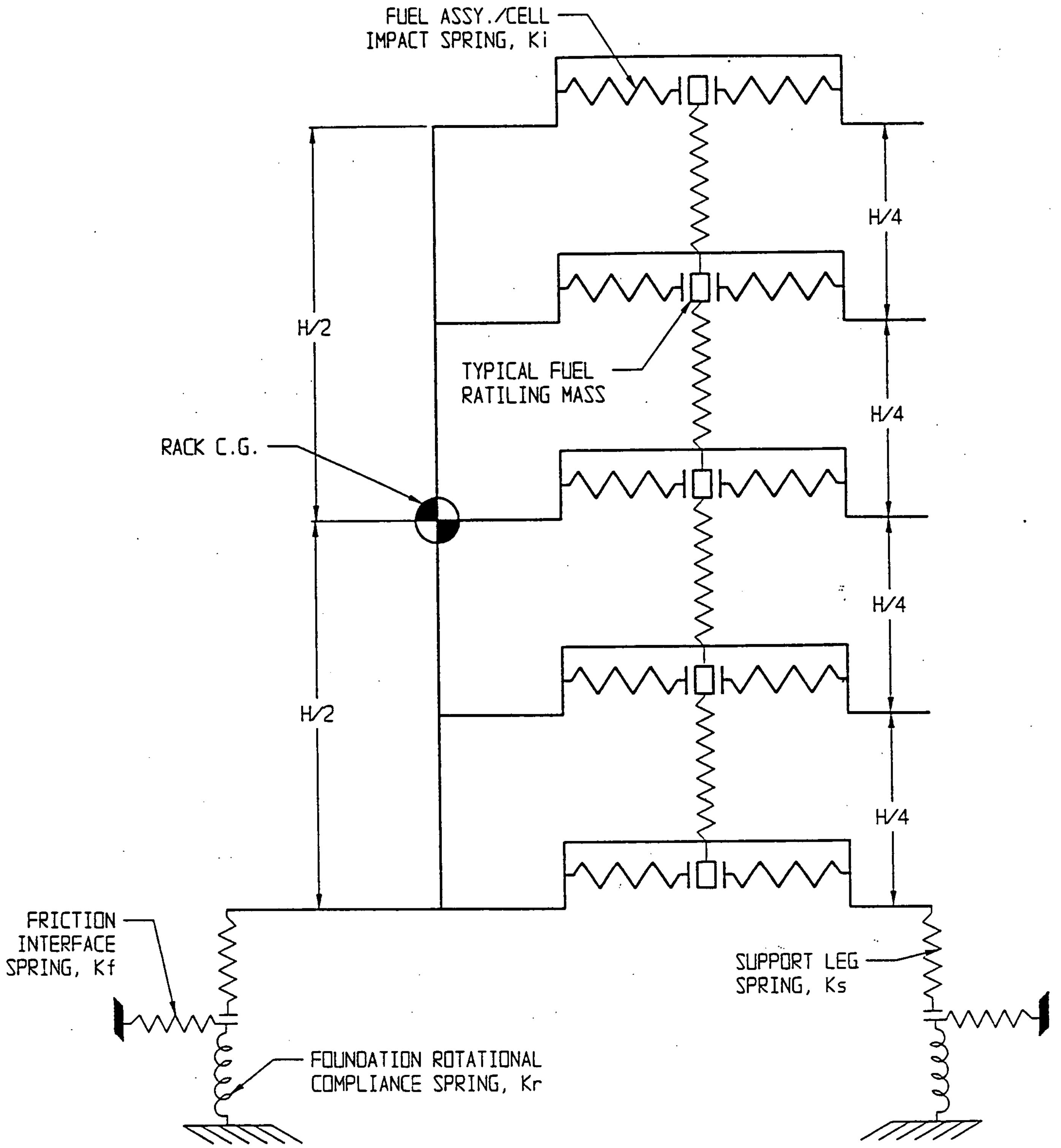


RACK DEGREES-OF-FREEDOM FOR Y-Z PLANE BENDING
WITH SHEAR AND BENDING SPRING



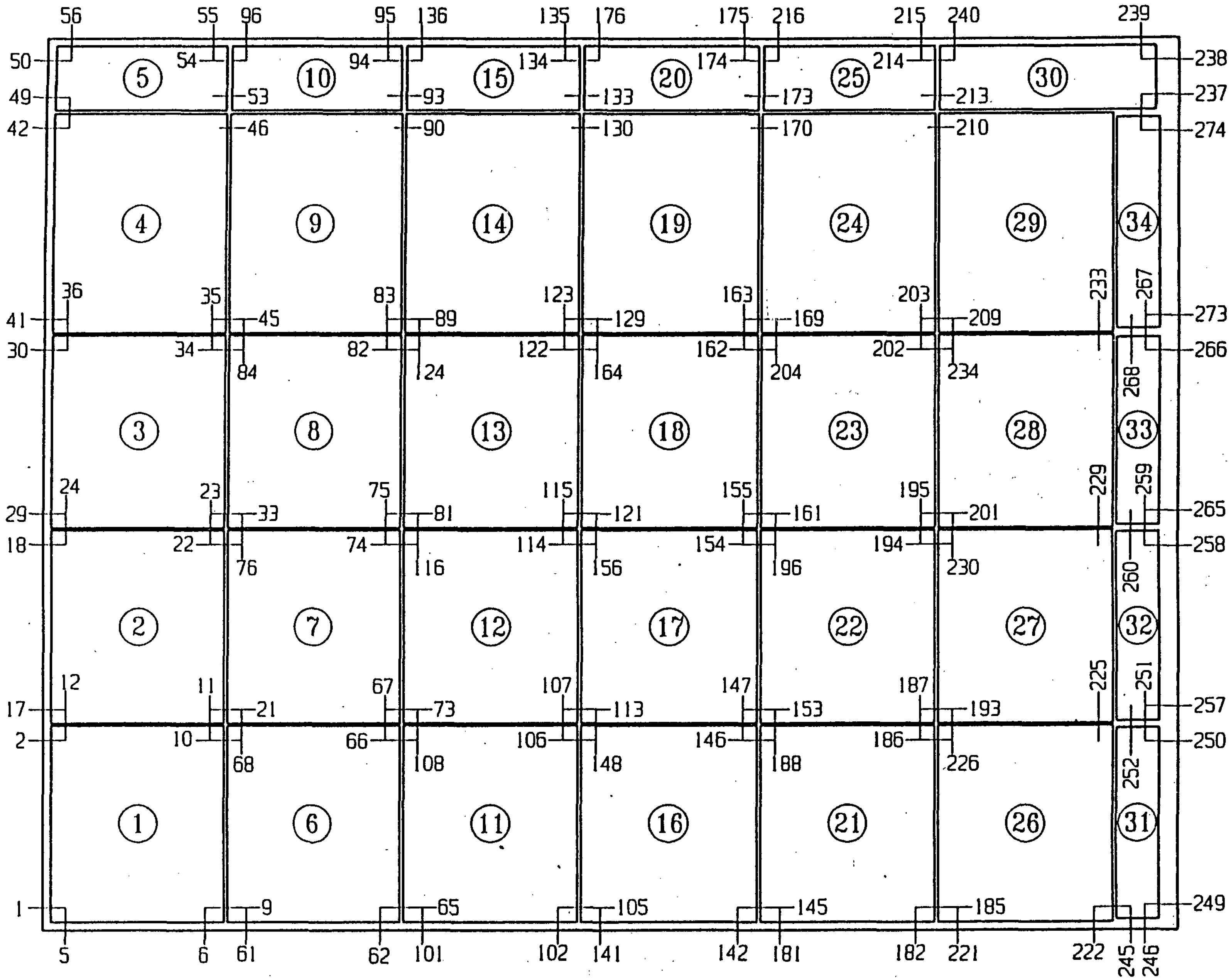
RACK DEGREES-OF-FREEDOM FOR X-Z PLANE BENDING
WITH SHEAR AND BENDING SPRING

Figure 6.5.4



2-D VIEW OF THE SPRING-MASS SIMULATION

Figure 6.5.5



WPMR NUMBERING SCHEME (BOTTOM)

Figure 6.5.6

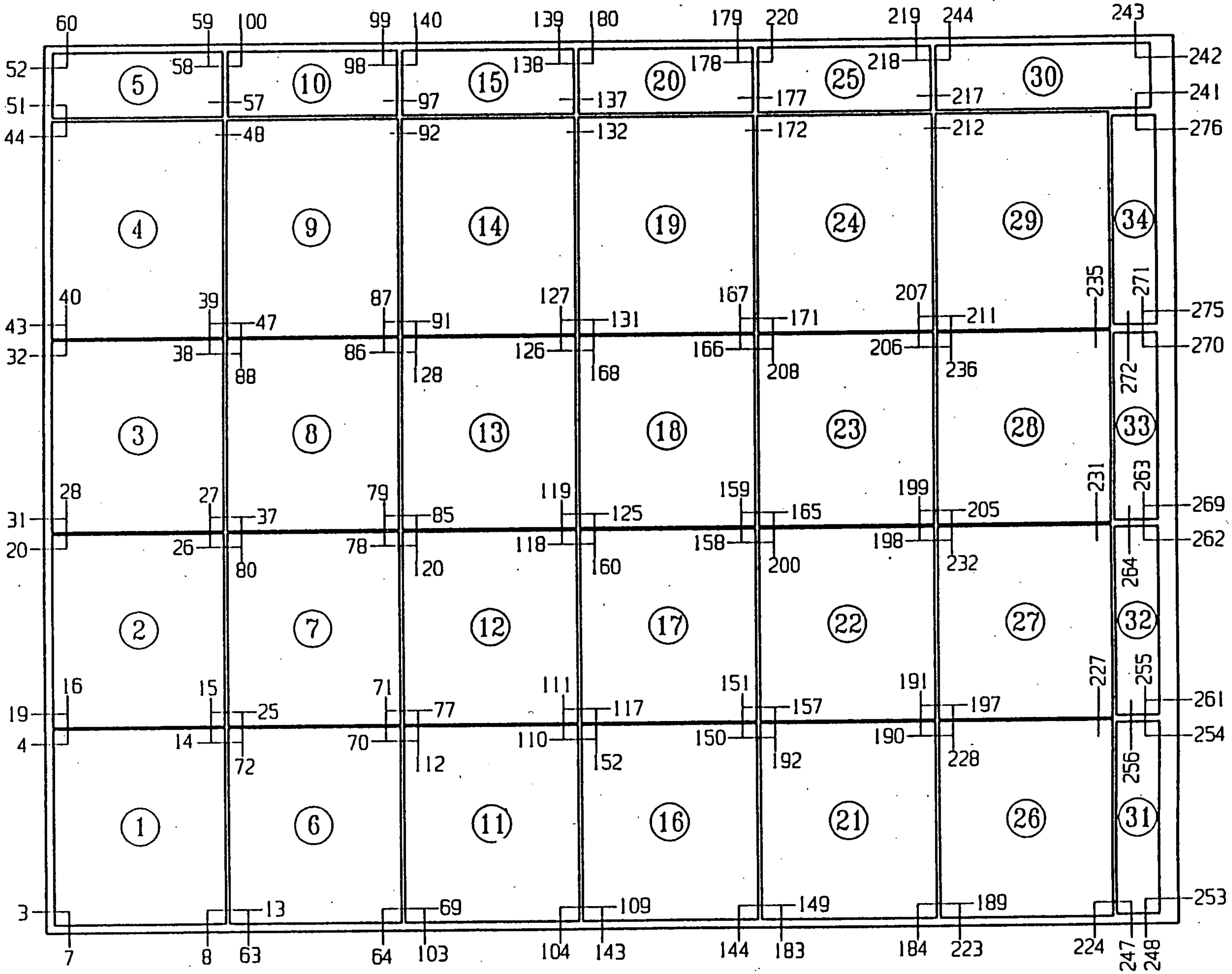


Figure 6.5.7

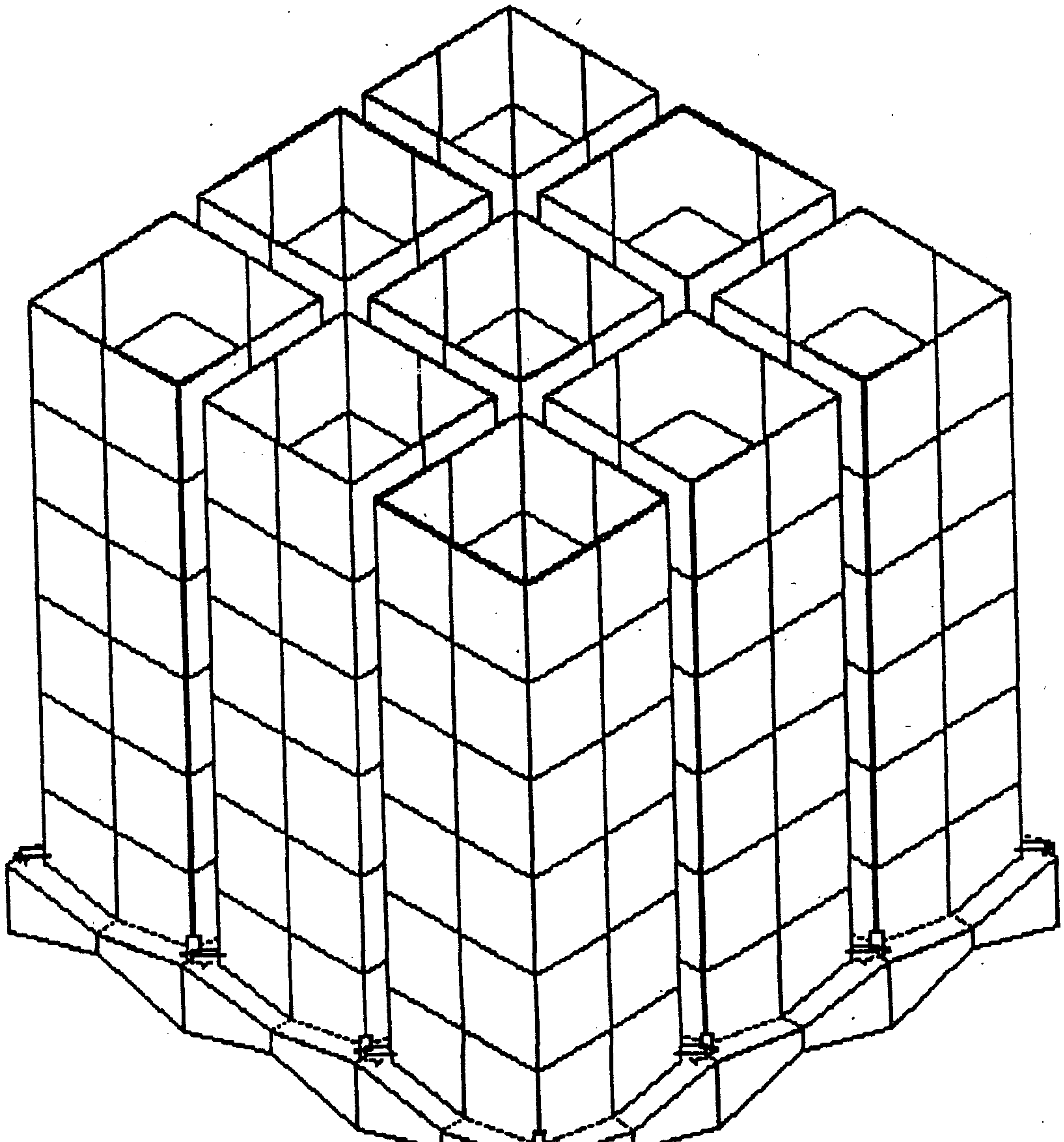


Figure 6.5.8 Rack Fatigue Model

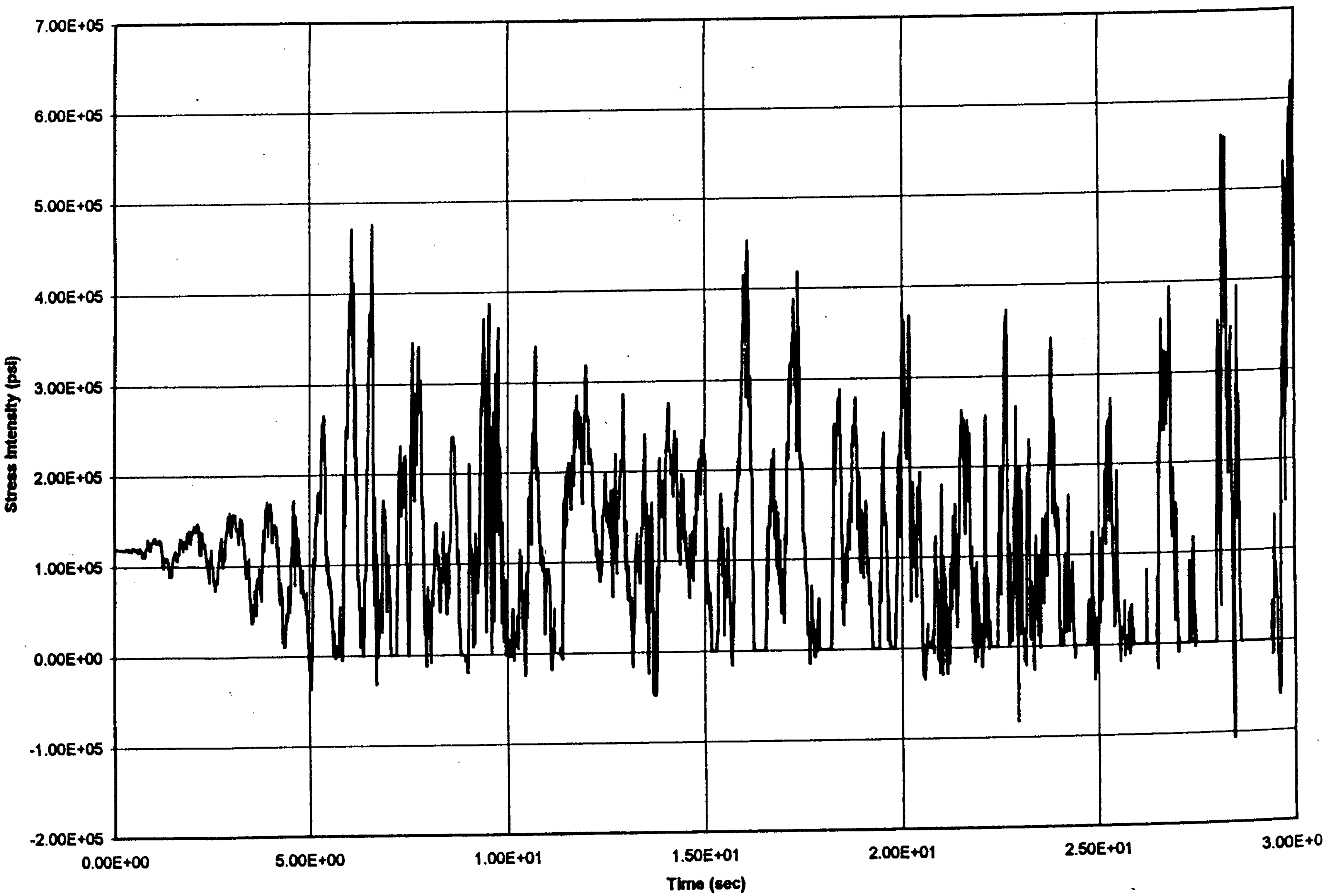


Figure 6.5.9 SSE Stress Intensity Time-History

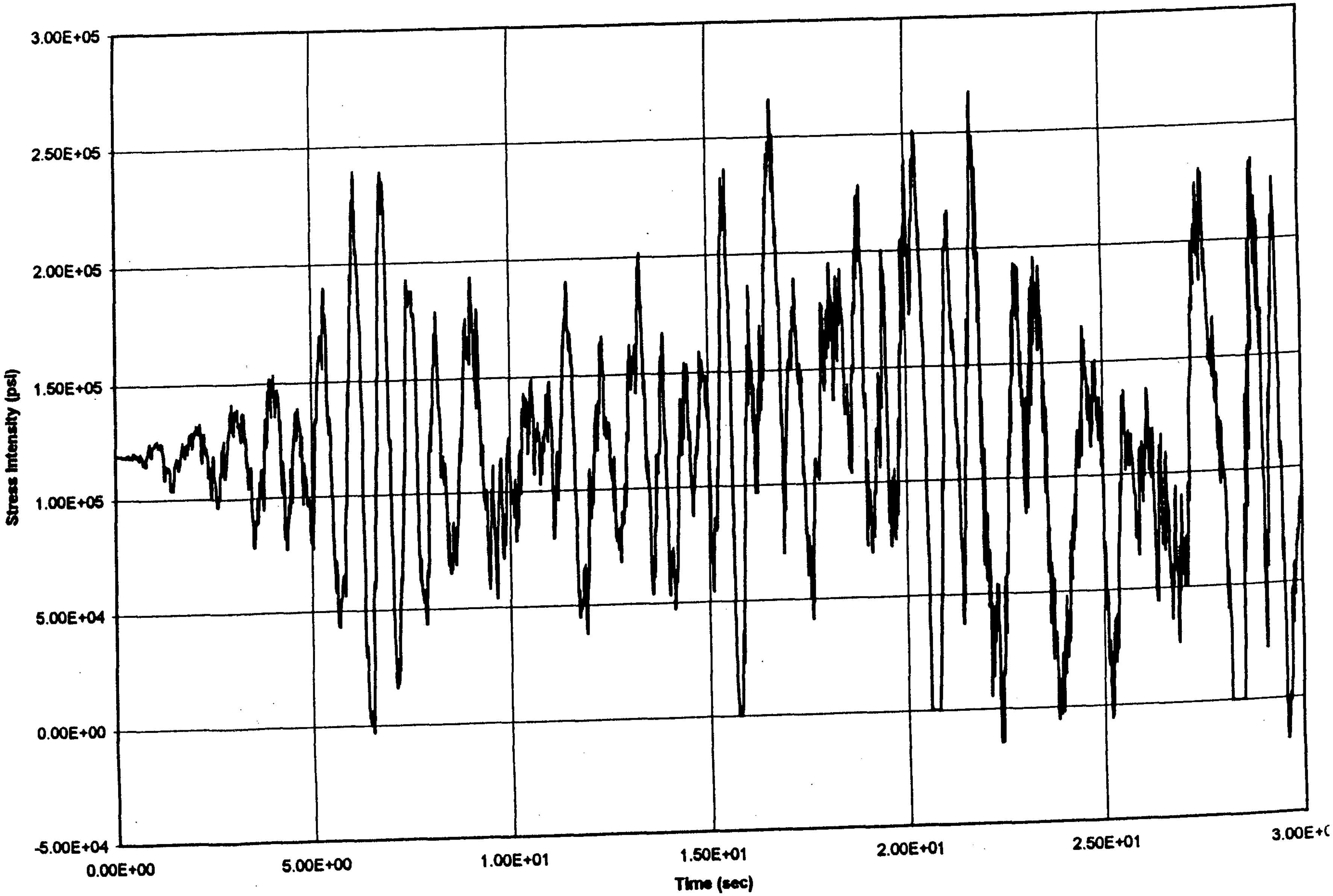


Figure 6.5.10 OBE Stress Intensity Time-History

CHAPTER 7

ACCIDENT ANALYSES AND MISCELLANEOUS STRUCTURAL EVALUATIONS

7.0 ACCIDENT ANALYSES AND MISCELLANEOUS STRUCTURAL EVALUATIONS

7.1 INTRODUCTION

This section provides results of accident analyses and miscellaneous evaluations performed to demonstrate regulatory compliance of the fuel racks. Accident events considered are taken from Reference 1.

The following accident and miscellaneous structural evaluations are considered:

- Refueling accident - drop of fuel assembly with its handling tool, and drop of gate on rack.
- Local cell wall buckling.
- Fuel rack subjected to external forces.
- Structural adequacy of the impact shield for the cask loading area of the cask pit.

7.2 REFUELING ACCIDENTS

The FSAR[2] states that the fuel handling system devices and equipment have provisions to avoid dropping or jamming fuel assemblies while conducting refueling operations. The combined weight of a fuel assembly plus handling tool is approximately 2100 lbs. Controls on crane movement are such that the top of an active fuel assembly can only be raised to within approximately 10 feet of the top of normal water level. Despite the handling system provisions and the controls imposed on the crane, a conservative accident evaluation of the fuel racks should include the effect of a fuel assembly falling. Drop accidents focusing on the integrity of the rack structure due to such drops are considered for the bounding rack cases. The consequences of dropping a fuel assembly as it is being moved over stored fuel is discussed below. Based on the highest lift of a fuel assembly,[2] the maximum distance from the bottom of a fuel assembly, traveling over fuel racks, to the top of the rack is 36 inches.

7.2.1 Dropped Fuel Assembly - Accident I

A fuel assembly plus its handling tool (2100 lbs.) is dropped from 36 inches above the top of an empty storage location away from a rack support pedestal and impacts the base of the rack module. Local failure of the baseplate or bottom casting is acceptable; however, the rack design should ensure that gross structural failure of the rack does not occur and that the subcriticality of the adjacent stored fuel assemblies is not violated. Calculated results show that there will be no change in spacing between cells for either the PaR flux trap modules or the Holtec burnup credit modules.[3] For the burnup credit racks, local deformation of the baseplate in the area of the impact will occur, but the dropped assembly will be contained and will not impact the pool liner. It is shown that the baseplate deformation of the Holtec racks is less than 3.67 inches and this value is less than the distance between the baseplate and the liner. The load transmitted to the pool liner through the support pedestal by such an accident is well below the loads caused by the seismic event results provided in Chapter 6.

Local failure of the PaR rack bottom casting occurs during a "straight deep drop" accident away from the pedestal locations. The rack design allows local failure in that the amount of casting material present at the base of each cell is insufficient to support the postulated impact load. A finite element analysis using DYNA 2-D[3] shows that the local failure of the bottom casting grid structure absorbs only 12 percent of the total impact energy. The pool liner is impacted following failure of the bottom casting. Local damage of the liner and its supporting concrete structure in the leak chase area was investigated using the LS-DYNA3D computer code[4] to address the nonlinear elasto-plastic problem. The results show that there is no rupture of the liner.[5]

7.2.2 Dropped Fuel Assembly - Accident II

Bounding pedestal parameters were used from the PaR and Holtec rack modules to address the case of the "straight deep drop" accident over a pedestal. [3] The resulting impact transmits a load of 191,000 lbs. to the slab through the pedestal. The magnitude of this impact is less than the peak pedestal load, 300,000 lbs., obtained from the seismic analysis[7] for the PaR racks. Furthermore, the impact load is less than the calculated peak pedestal load from the single rack analyses under OBE conditions (198,000 lbs.). In that analysis, the pedestals were shown to satisfy the allowable stress limits for Level A conditions. This accident, therefore, is not limiting for either of the spent fuel rack types. The bearing pressure on the pool slab, 2,432 psi, is below the allowable concrete pressure, 2,890 psi.

7.2.3 Dropped Fuel Assembly - Accident III

For the "straight shallow drop" of a fuel assembly and its handling tool on the top of the rack modules, a very conservative energy balance calculation was used together with the more conservative physical parameter values from the two types of racks.[3] For example, the storage cell wall thickness of the PaR racks and the Holtec racks are 0.09 inches and 0.06 inches, respectively. Calculations were based on the smaller dimension, 0.06 inches. Permanent deformation of the rack is acceptable, but such deformation is required to be limited to the top region such that the rack cross-sectional geometry at the level of the top of the active fuel region (and below) is not altered. Analysis results demonstrate that permanent damage to any fuel storage cell is limited to a maximum depth of 3.06 inches below the top of the rack. This is less than the distance from the top of the rack to the beginning of the active fuel region (approximately 20 inches). Therefore, there will be no effect on the subcriticality of fuel stored in adjacent cells as a result of this accident.

7.3 DROPPED GATE

The drop of the 3820 lb. gate from eight feet above the top of the racks was also evaluated.[3] It was determined that permanent damage to a fuel storage cell is limited to a maximum depth of 5.325 inches below the top of the rack. Again, there will be no effect on the subcriticality of fuel stored in adjacent cells as a result of this accident.

7.4 LOCAL BUCKLING OF FUEL CELL WALLS

The allowable local buckling stresses in the fuel cell walls are obtained by using classical plate buckling analysis. The following formula for the critical stress is used based on a width of cell "b": [6]

$$\sigma_{cr} = \frac{\beta \pi^2 E t^2}{12 b^2 \left(1 - \mu^2\right)}, \text{ where}$$

σ_{cr} is the limiting vertical compressive stress in the tube, E = 27.6 x 10^6 psi, μ = 0.3, (Poison's ratio), t = .060 inches (limiting value from PaR racks and Holtec racks), and b = 8.75 inches. The factor β is suggested to be 4.0 for a long panel.[6] Near a pedestal, additional cell wall strength is provided by added strip material which increases effective thickness of the region prone to buckling to .1045 inches in the highly loaded region.

For the given data,

$$\sigma_{cr} = 14{,}232 \text{ psi}$$

It should be noted that this stability calculation is based on the applied stress being uniform along the entire length of the cell wall. In the actual fuel rack, the compressive stress comes from consideration of overall bending of the rack structures during a seismic event and as such is negligible at the rack top and maximum at the rack bottom. It is conservative to apply the above equation to the rack cell wall if we compare σ_{cr} with the maximum compressive stress anywhere in the cell wall. The maximum compressive stress in the outermost cell is obtained by multiplying the limiting value of the stress factor R_6 (for the cell cross-section just above the baseplate) by the allowable stress. Thus, from the whole pool multi-rack analyses, $\sigma = R_6$ x allowable stress = 0.552 x 21,300 psi = 11,758 psi under faulted conditions.

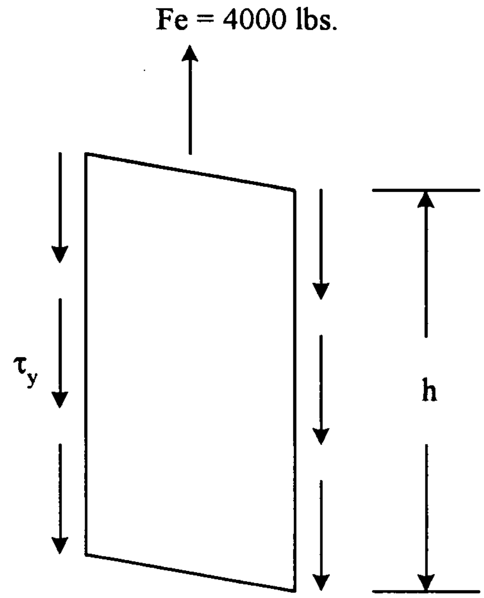
7.5 EXTERNAL FORCES

The capability of the racks to withstand a vertical or inclined (at 45 degrees) force of 4000 pounds (bridge crane uplift limit) at any location, without affecting the subcriticality of the stored fuel, was evaluated. The critical location for load application is to have this load applied near the top of the rack along or against a single cell wall. Again, the object of the investigation is to show that damage is confined to a region above the active fuel. If the vertical load is resisted only by shear stress, and the yield stress in shear is forty percent of the yield stress in simple tension, then (using static values only):

$$\sigma_y = 21{,}300 \text{ psi}$$

$$\tau_y = 0.4\sigma_y \qquad \tau_y = 8.52\ 10^3 \text{ psi}$$

The depth h of the cell that can support the applied load is obtained from



t = 0.060 in.

$$h = \frac{F_e}{2\tau_{yt}}$$

h = 3.912 in.

Since the damage is above the active fuel area (minimum of 12 inches above the active area), the application of Fe vertically is not a concern.

If the load is applied vertically anywhere else along a cell wall, the stress developing in the wall is

w = 8.75 in.

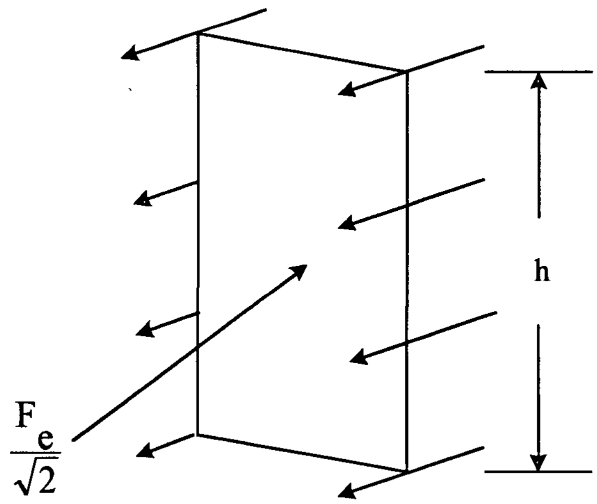
$$\sigma = \frac{F_e}{w\,t} \qquad \sigma = 7.619 \times 10^3 \text{ psi}$$

The stress is below yield and will cause no permanent damage to the cell.

If the load is applied inclined at 45 degrees, then there is a horizontal load component that must be supported. Realistically, this load can only be applied at the top of the rack. Therefore, it is again necessary to show that any damage is confined to a region above the active fuel area. If h is the depth of the "damaged" region, tear out of a cell wall was considered to show that the damaged region is less than the distance from the rack top to the edge of the neutron absorber.

$$h = \frac{F_e}{2\sqrt{2}\,\tau_y t}$$

$h = 2.766$ in.



It is therefore concluded that there is no concern since any damage will not violate the active fuel envelope.

7.6 ANALYSIS OF THE IMPACT SHIELD FOR CASK PIT

To maximize the storage capacity of the spent fuel pool, a spent fuel storage rack containing 225 cells (15x15 cells) is proposed to be installed in the 12 feet x12 feet cask loading area of the cask pit of the WBN spent fuel storage pool. After installation of the rack in the cask pit, the pit will be equipped with a removable impact shield (SA-36 material) to prevent accidental dropping of any object on the fuel rack. The proposed impact shield is shown in Figure 2.4.1. It consists of panel cover plates attached to a frame made of wide flange beams. This shield is designed to withstand a total load of 288,000 lbs. uniformly applied on the whole shield, or a total of 70,000 lbs. uniformly applied on one of the panel plates. The panel plate thickness was determined by a limit load analysis, and the dimensions of the wide flange beams are chosen so that the maximum stresses in the frame for the postulated load cases are within the corresponding allowables.[8] The ANSYS finite element program is used to perform the frame stress analysis. The results are summarized below:

(1) Panel plate can resist a uniform load of 70,000 lbs. on one panel or a concentrated load of 7,952 lbs. applied at any point without sustaining a plastic collapse.

(2) Maximum direct plus bending stress in the frame beams is 51,961 psi, which is below 90% of the ultimate material strength. Maximum average shear stress is 2,850 psi, which is less than the postulated allowable (36,000 psi).

(3) Maximum average compression stress on concrete wall at the bearing locations is 329 psi, which is considerably lower than the allowable (2,975 psi).

Figure 7.6.1 shows the allowable drop height as a function of the heavy load weight with the cross-sectional area of the load as parameter. From this figure, the allowable drop height for any heavy load can be ascertained by interpolation. Tabular information is also available.

7.7 REFERENCES

1. TVA Specification No. 3344 - WBNP 92, Revision 2, page 38, 39.

2. TVA Watts Bar Nuclear Plant Final Safety Analysis Report," Section 9.1.

3. Holtec International, "Mechanical Accident Analysis Report for Watts Bar Nuclear Plant - Tennessee Valley Authority," Project No. 10371, Report No. HI-961457, TVA Calculation WCG-1-1811.

4. LS-DYNA3D, Version 932, Livermore Software Technology Corporation, May 1, 1995.

5. Holtec International, "Evaluation of the Spent Fuel Pool Liner for the Fuel Assembly Drop Accident, Project No. 10371, Report No. HI-961514, TVA Calculation WCG-1-1818.

6. "Strength of Materials," S. P. Timoshenko, 3rd Edition, Part II, pp 194-197 (1956).

7. "Spent Fuel Storage Racks Design Report for Sequoyah Nuclear Plant Units 1 and 2, Grid Seismic Static Analysis Design Report," Revision 2, PaR System Corp., DR-9001-8, April, 1980.

8. Design and Analysis of Impact Shield, TVA-SQN Calculation No., SCG1S469, Revision 1, Holtec Report No. HI-91712, Revision 4, SQN DCN No. M08736A, page 34.

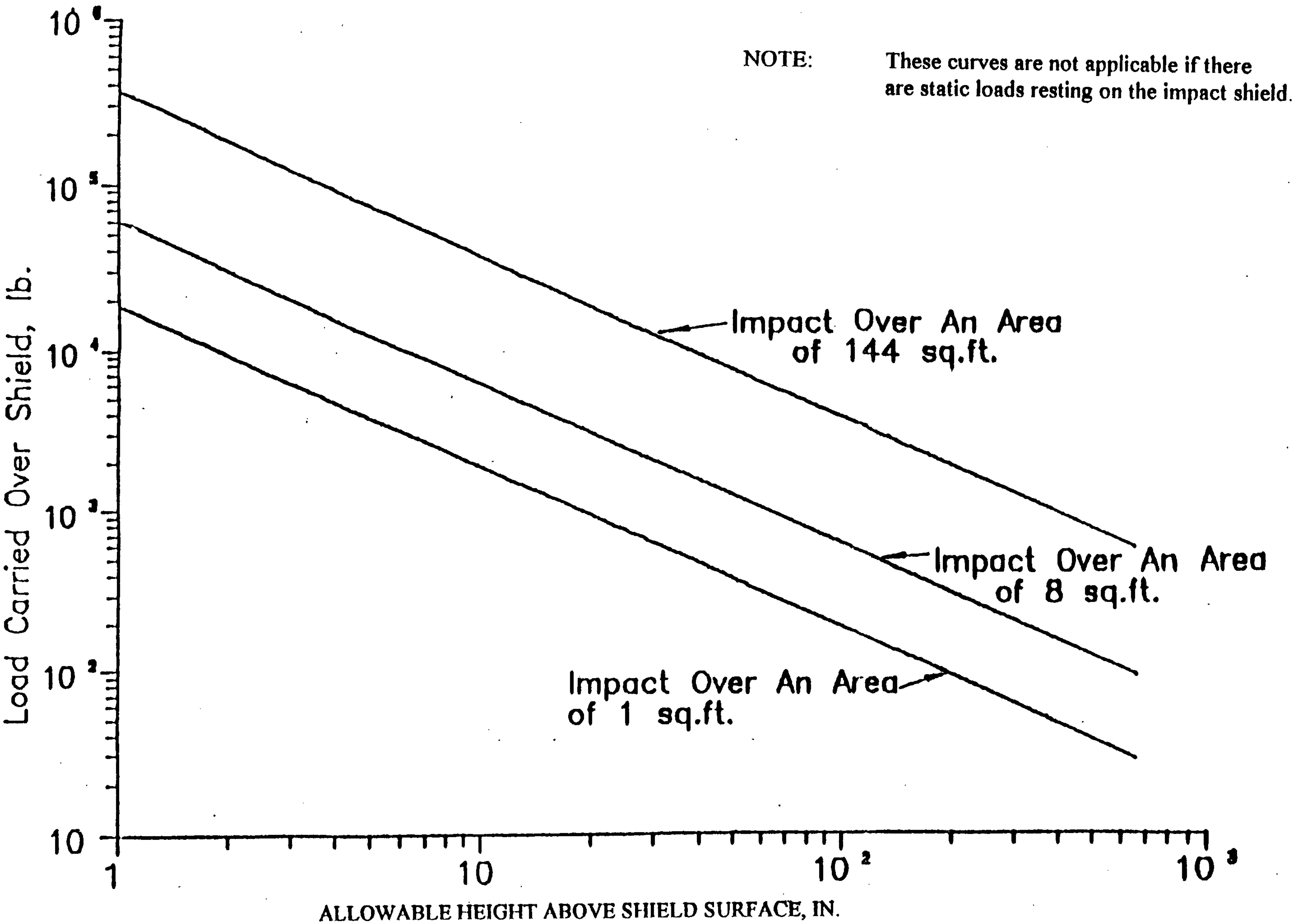


FIGURE 7.6.1 Relationship Between Load, Allowable Height and Impact Area for Dropped Object

CHAPTER 8

FUEL POOL STRUCTURE INTEGRITY CONSIDERATIONS

8.0 FUEL POOL STRUCTURE INTEGRITY CONSIDERATIONS

8.1 INTRODUCTION

The WBN spent fuel pool is a safety related, seismic category I, reinforced concrete structure. In this section, the analysis to demonstrate the structural adequacy of the pool structure, as required by Section IV of the USNRC OT Position Paper,[1] is discussed. Since the spent fuel pool storage area at WBN is identical to that at SQN, the detailed analysis performed for SQN[9] will be invoked for acceptance of the WBN pool. A comparison of WBN and SQN conditions was performed in Reference 10 and will be discussed later.

The spent fuel pool slab is over 25 feet thick reinforced concrete supported on rock, and is therefore not a candidate region of potential deficiency in structural strength. In the interest of revalidation of original design work, a reanalyzes of the pool structure for the new loadings due to reracking, is performed.[9] The design bases are the Safe Shutdown Earthquake evaluated for the N-S and vertical and for the E-W and vertical seismic loadings separately. The original design basis ACI Code[2] and the Working Stress Design (WSD) Method are also invoked to maintain consistency with the previous qualification effort. However, the method of analysis is upgraded from the classical static analyses to the Response Spectrum Method. The latter methodology enables a realistic characterization of hydrodynamic sloshing loads during seismic events which could only be incorporated with a large margin of uncertainty in the static analysis procedures.

Pool structural loading involves the following discrete components:

a) Static Loading

1) Dead weight of pool structure plus pool water (including hydrostatic pressure on the pool walls). Combining the hydrostatic and structure dead weight is in conformance with Reference 2.

2) Dead weight of rack modules and fuel assemblies stored in the modules.

b) Dynamic Loading

1) Vertical loads transmitted by the rack support pedestals to the slab during an SSE or OBE seismic event.

2) Inertia loads due to the slab, pool walls and contained water mass and sloshing loads which arise during a seismic event.

3) Hydrodynamic loads caused by rack motion in the pool during a seismic event.

c) Thermal Loading

Mean temperature rise and temperature gradient across the pool slab and the pool walls due to temperature differential between the pool water and the atmosphere external to the slab and walls. Both normal and pool boiling (loss of cooling accident) conditions are considered.

The spent fuel pool region is pictorially illustrated in Figure 8.1.1. The pool structure is analyzed using the finite element method. The results for the above load components are combined using load combinations mandated by ACI[2] and consequently, the structural components are investigated using the "working stress" design method. It is demonstrated that for the critical load combinations, design criteria requirements are maintained when the fuel pool is assumed to be fully loaded with high density fuel racks with the storage locations occupied by fuel assemblies. The general purpose finite element code ANSYS[3] is utilized to perform the analysis.

The critical regions examined for the fuel pool are the slab and the critical wall sections adjoining the pool slab. Both moment and shear capacities of the critical wall regions are checked for structural integrity. Local punching and bearing integrity of the slab in the vicinity of a rack module support pedestal pad is evaluated. Structural capacity evaluations are carried out in accordance with the requirements of the American Concrete Institute (ACI).[2]

8.2 GENERAL FEATURES OF THE MODEL

The fuel pool model is constructed using information from plant auxiliary building structural drawings. A description of the pool structure modeled for analysis is given in the following:

For WBN, the fuel pool slab is a 25.71 foot thick reinforced concrete slab supported on rock with inside dimensions of 53 feet 1/4 inches long and 31 feet 8-1/2 inches wide. The top of the slab is located at Elevation 708.71 and its long direction is considered aligned along the plant east-west direction. The north edge of the slab has a 7 feet thick vertical reinforced wall which rests on the slab and which extends above the slab to Elevation 757. The south edge of the slab has a 6-foot thick wall from the slab to Elevation 757. The south wall separates the fuel pool from the fuel transfer canal. The canal is not modeled; however, the discontinuity in the wall structure in the center of the south wall (with respect to the transfer canal) is included. The west wall is a 6-foot 11-1/2-inch thick wall extending from the slab to Elevation 757. The east edge of the slab has a 7-foot thick wall extending up to Elevation 757. The cask area is located in a 12-foot x 31 feet 8-1/2 inch space at the west end of the pool. The cask area is separated from the spent fuel storage area by a 1-foot 6-inch thick wall aligned along the north-south direction. This dividing wall extends to Elevation 757, and has an opening to transfer fuel assemblies extending down to Elevation 725.12. The wall modeling is done to Elevation 757; free edges are assumed at this level. Figure 8.1.1 shows a schematic of the above geometry which is modeled and analyzed using the finite element method. Dimensions described above are the same for SQN except the SQN slab is 20.71 feet thick.

For SQN, the pool is assumed to be loaded with high density fuel racks having a total of 2091 cells. The WBN pool has a maximum capacity of 1610 cells. For SQN analysis purposes, each cell is assumed to contain a 1550 lb. weight fuel assembly. The hydrodynamic load arising from the kinematic action of the cask pit rack is not included in the analysis. The relatively large gap (nominally 4.5 inches) between the cask pit rack and the pool wall makes the fluid coupling effect between them negligible in comparison to other loadings (such as hydrostatic pressure). The intermediate wall (between the cask pit and the fuel pool) is rendered into a non-load bearing wall by not installing the gate which separates the two regions. The cask pit will, therefore, also be filled with water making the net hydrostatic pressure on the intermediate wall zero. The finite element model is only of the five vertical walls; the slab is not modeled but its effect on the vertical walls is considered by assuming complete mechanical fixity at wall-slab interfaces. The depth of the slab (20.71 feet at SQN; 25.71 feet at WBN) is such that it may be considered as a rigid body for the purposes of structural requalification of the vertical walls; the slab is only structurally examined to demonstrate satisfaction of local punching and bearing requirements. Growth of the slab is considered in the analysis of the fuel pool walls under thermal loading.

Fluid sloshing effects are included by using a fluid model based on masses and springs in accordance with Reference 4. The sloshing fluid mass is connected to the slab walls by weak springs tuned to reproduce the sloshing frequency. The remainder of the fluid mass is coupled to the structure using stiff springs.

The finite element model is constructed using the ANSYS classical shell element STIF63 in the ANSYS finite element code. The shell element thickness in the various regions of the structure is the actual thickness of the structure at the location. The finite element model is prepared for the analysis of both mechanical load and thermal load. The effects of structural reinforcement and the properties of the concrete (cracked or uncracked) are accounted for in the finite element model by establishing an appropriate effective modulus for each shell element. Effective moduli are defined for each local in-plane axis for the shell elements. The different moduli reflect the fact that different reinforcement geometries may be used in perpendicular directions of the plate-like sections when the different concrete section assumptions (cracked or uncracked) are applied to the structure. Only major reinforcement which affects the plate and shell behavior of the structure is incorporated into the definition of the effective moduli; additional local reinforcement in various areas of the pool structure are neglected in the defining of the effective moduli. However, such local reinforcement may be accounted for in the stress evaluation after results are obtained for local bending moments. The variation of the extent of reinforcement is taken into account in the finite element model by defining different material types as necessary to reflect the varying values of effective moduli in different regions. Uncracked section properties are assumed for the initial mechanical load analyses. For the thermal analyses, it is shown that the thermal gradients will always yield a cracked section if the uncracked stiffness is used; that is, an iterative solution is used to show that cracked section properties should be used for all finite element analyses for thermal loading.

The effective properties for the elements used in the finite element model are calculated using standard procedures for reinforced concrete sections to define equivalent effective homogeneous materials having the appropriate stiffness and strength.

8.3 LOADING CONDITIONS

To evaluate response due to the different load mechanisms outlined in Section 8.1, the following finite element analyses are carried out. Loading cases are defined below which enable us to obtain the moments and shears for required combined loadings[1] by linear combination.

Case 1: Dead loading from concrete, reinforcement and 41.29 feet of hydrostatic head. The loading is applied as a 1.0g vertical gravitational load for the structure and a surface pressure on the walls to simulate the hydrostatic head. The cask pit is assumed to be full of water. The fuel transfer canal is assumed empty for this analysis to maximize south wall lateral loading.

Case 2: Seismic horizontal loading due to pool structure mass and contained water mass. A response spectrum analysis is performed with the contained fluid modeled as impulsive mass and rigid mass.[4] The input response spectra loading are the two horizontal design basis seismic response spectra at 5% damping. Subsequent load combinations reflect the postulated 2-D analysis carried out for each horizontal direction separately.

Case 3: Seismic horizontal load due to hydrodynamic effects from fluid coupling caused by rack motion relative to the walls. The level of horizontal pressure loading is obtained from the results of spent fuel rack analyses outlined in Chapter 6. This analysis is carried out for the SSE pressure distribution. Due to the small contribution from this effect, the loading is also used for the OBE condition in the load combinations. For conservative analysis of the intermediate wall, as noted previously, no rack is assumed in the cask pit.

Case 4: A mean temperature rise plus a thermal gradient applied across the walls to simulate the heating effect of the water in the pool. This gradient is calculated based on appropriate surface heat transfer coefficients and on maximum wall temperature deduced from the pool bulk temperature calculations for the licensing basis as discussed in Chapter 5. The cask area is assumed to be full and the fuel transfer canal is assumed to be empty for thermal stress analysis.

Case 5: A mean temperature rise plus a thermal gradient applied across the walls to simulate the case of pool boiling. For this abnormal case thermal analysis, the cask area is assumed to be full and the fuel transfer canal is assumed to be empty.

For subsequent discussion of structural integrity checks using various mandated load combinations, the above individual finite element load cases are referred to as load cases 1-5, respectively. For the pool structural analysis, the following load combinations are evaluated:

Load Combinations	Allowable WSD Stresses
Case I = D	$f_c = .45f'_c$ (concrete)
Case Ia = D + T_N	$f_s = .4f_y$ (steel)
Case IIa = D + E	$f_c = 0.45\ f'_c$
Case IIb = D + E + T_N	$f_s = 0.50\ f_y$
Case IIIa = D + E'	$f_c = 0.75\ f'_c$
Case IIIb = D+ E' + T_N	$f_s = 0.90\ f_y$

Case IV = D + T_A $\qquad$ $f_c = 0.75\ f'_c$
$f_s = 0.90\ f_y$

In the above loading combinations, the load conditions are defined as follows:

D = Dead weight and hydrostatic load
E' = Safe Shutdown Earthquake (SSE)
E = Operating Basis Earthquake (OBE)
T_N = Normal Thermal Load
T_A = Accident Thermal Load (pool boiling)

The appropriate load conditions are formed from the results of finite element cases defined at the beginning of this subsection as follows:

D = case 1

E' = 1.0 x case 2 + 1.0 x case 3

E = OBE amplifier (1/2) x case 2 + 1.0 x Case 3

T_N = case 4

T_A = case 5

Load combinations are formed using absolute values where necessary so as to maximize critical combined stress resultants.

8.4 RESULTS OF ANALYSES

The ANSYS postprocessing capability was used in the evaluation of the SQN spent fuel pool to form the appropriate load combinations identified above and to establish the critical bending moments in various sections of the pool structure.[9] The following limit strengths for concrete and for reinforcement were used in the computation of allowable stresses.[5]

concrete f'_c = 4000 psi (compression)
reinforcement = f_y = 60000 psi (tension/compression)

Due to the structural similarity of the SQN and WBN spent fuel pools (identical geometry and individual load set) and the elastic nature of the material considered in the 3-D finite-element analysis, the numerical results obtained for each individual load case from the SQN analysis are used in the assessment of the WBN spent fuel pool.[10] Each set of numerical results corresponding to a specific individual load case is amended by a specific global coefficient reflecting the ratio of magnitudes of the SQN pool and WBN pool for that specific load case. The final combinations are calculated to numerically simulate the most critical bending moment fields and then compared to the reinforced concrete sectional capacities. The safety margin for bending is then calculated as the minimum allowable bending moment divided by the calculated bending moment. Table 8.4.1 summarizes the results obtained from the analyses and shows minimum safety margins on each wall of the spent fuel storage pool. The analyses conclude that the WBN spent fuel pool is structurally adequate for the loads associated with the rerack effort.

8.5 POOL LINER INTEGRITY ANALYSIS

The pool liner integrity is verified by performing a comparison between the SQN and WBN spent fuel pool components and attributes (i.e., liner, racks, bearing pads, etc). The SQN calculation[11] is evaluated for the WBN single rack and WPMR results to verify applicability to WBN. Computations show that the maximum stress in the liner is well below the ultimate stress of the liner material.[6] The computations provide a safety factor of 3.0 which is greater than the required 1.5. Therefore, it is demonstrated that the pool liner will not tear or rupture under any design loading conditions.

An evaluation of the potential for liner failure due to cyclic fatigue was also made. Analyses show that the cumulative usage or damage factor is less than the ASME Code limit of 1.0. Therefore, it is demonstrated that there is no fatigue failure for the pool liner when experiencing one SSE and five OBE seismic events.

8.6 BEARING PAD ANALYSIS

To protect the pool slab and liner from high localized dynamic loadings, stainless steel bearing pads are placed between the pedestal base and the slab. Fuel rack pedestals impact on these bearing pads during a seismic event, and pedestal loading is transferred to the liner and slab. Bearing pad dimensions are set to ensure that the average pressure on the slab surface due to a static load plus a dynamic load does not exceed the American Concrete Institute[7] limit on bearing pressures. The bearing pad thickness is set to ensure the stress limit is met and to ensure proper pad thickness to safely act as a bridge over a leak chase or weld seam. Reference 8 substantiates that the results of the detailed finite element analysis performed for the SQN bearing pad design is applicable to WBN. The analysis performed by this calculation demonstrates that the bearing pad design is acceptable based on the worst case scenario of: (1) the maximum pedestal vertical load, (2) the relative location of this load on the pad, and (3) the location of the leak chase channel under the pad. The calculated resultant stresses are shown to be well within the allowable stresses.

8.7 CONCLUSIONS

Critical regions affected by loading the fuel pool completely with high density racks are examined for structural integrity under bending and shearing action. It is determined that adequate safety factors exist assuming that all racks are fully loaded with fuel and that the factored load combinations are checked against the appropriate structural design strengths. The peripheral space between the cask pit rack and the walls is sufficiently large to attenuate the fluid coupling forces to negligible levels, making their evaluation essentially superfluous. Likewise, the intermediate wall, which is surrounded by water on both sides, is not subject to any appreciable structural loads and is not considered in this evaluation.

It is also shown that local frictional loading on the liner in both the main pool and in the cask pit results in stresses that are low enough so that liner fatigue is not a concern.

8.8 REFERENCES

1. OT Position for Review and Acceptance of Spent Fuel Handling Applications, by B.K. Grimes, USNRC, Washington, D.C., April 14, 1978, and January 18, 1979 amendment.

2. ACI 318-63, 318-71, Building Code Requirements for Reinforced Concrete, American Concrete Institute, Detroit, Michigan.

3. ANSYS User's Manual, Swanson Analysis Rev. 4.4A, 1990.

4. "Nuclear Reactors and Earthquakes," U.S. Department of Commerce, National Bureau of Standards, National Technical Information Service, Springfield, Virginia (TID 7024).

5. TVA Watts Bar Nuclear Plant Final Safety Analysis Report, Table 3.8.4-1 (sheet 3).

6. TVA Calculation WCG-1-1810, "Analysis of WBN Unit 1 Spent Fuel Pool Liner in Support of the Installation of High-Density SFP Storage Racks."

7. American Concrete Institute, "Building Code Req uirements for Structural Concrete," ACI-318-77.

8. TVA Calculation WCG-1-1825, "Pedestal Bearing Pad Design for the Watts Bar Spent Fuel Storage Racks."

9. TVA Calculation SCG1S431, "Calculation Package for SQN Spent Fuel Pool Concrete Structural Analysis."

10. TVA Calculation WCG-1-1819, "WBN Spent Fuel Pool Structural Qualification."

11. TVA Calculation SCG1S429, "Analysis of SQN Spent Fuel Liner in Support of Installation of High Density SFP Storage Racks."

TABLE 8.4.1

SAFETY FACTORS FOR BENDING OF POOL STRUCTURE REGIONS

REGION	SAFETY MARGIN*
North Wall	>1.22
East/West Wall	>1.23
South Wall	>1.05

* Above the limits prescribed by the working stress design method.

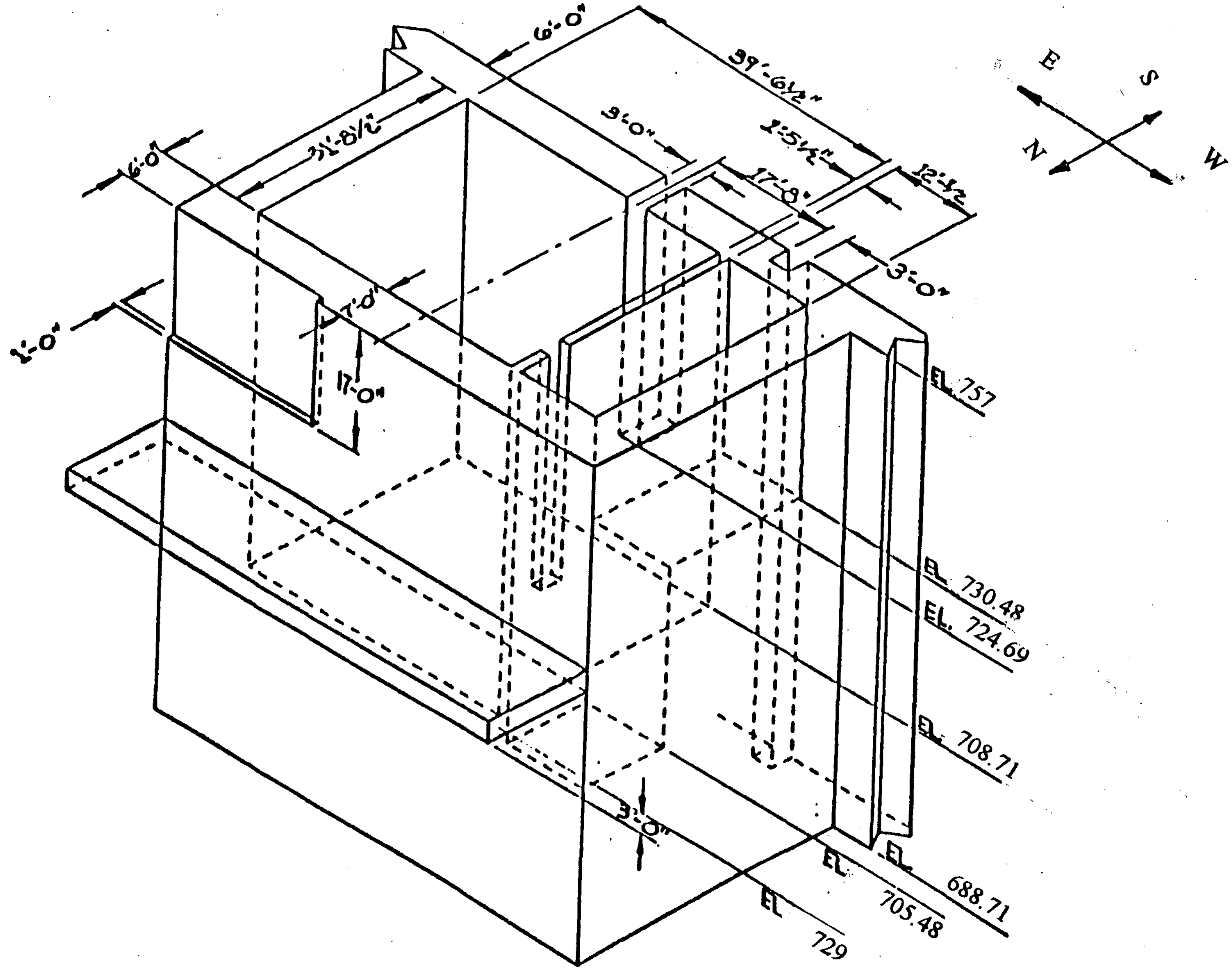


Figure 8.1.1 ISOMETRIC VIEW OF WATTS BAR FUEL POOL STRUCTURE

CHAPTER 9

RADIOLOGICAL EVALUATION

9.0 RADIOLOGICAL EVALUATION

9.1 FUEL HANDLING ACCIDENT

The potential radiological consequences of a fuel handling accident in the WBN Auxiliary Building have been determined.

9.1.1 Assumptions and Source Term Calculations

Evaluation of the accident was based on fuel of 5.0 wt% initial enrichment with 1,000 effective full power days of operation. In addition, an initial assessment, was made regarding the radiological consequences of 1500 effective full power days of operation. The reactor was assumed to have been operating at 3565 Mw thermal power prior to shutdown; this yields a (conservative) specific power of 40.00 kw/kgU. Except for fuel enrichment and discharge burnup, the assumptions used in the evaluations are the same as those previously reviewed and accepted by NRC in the WBN Safety Evaluation Report (SER).

As in the FSAR evaluation, the fuel handling accident was conservatively assumed to result in the release of the gaseous fission products contained in the fuel/cladding gaps of the rods in the peak-power fuel assembly at the time of the accident. Gap inventories of fission products available for release were estimated using the release fractions identified in Regulatory Guide 1.25.[1] Dose calculations were performed for a fuel decay time of 100 hours.

The gaseous fission products that have significant impacts on the offsite doses following short fuel decay periods, are the short-lived nuclides of iodine and xenon, which reach saturation inventories during in-core operation. These inventories depend primarily on the fuel specific power over the few months immediately preceding reactor shutdown. In the highest power assembly, the specific power and hence the inventory of iodine and xenon will be proportional to the peaking factor (assumed to be 1.65 per Regulatory Guide 1.25).

At the conservative (short) delay time of 100 hours used in the WBN calculations, most of the thyroid dose comes from Iodine-131, while most of the whole-body dose comes from Xenon-133. At longer cooling times, Iodine-131 remains the dominant isotope for thyroid dose, while the major contributor to whole-body dose becomes Krypton-85 (the shorter-lived Xenon-133 having decayed to very low levels). The doses after long decay periods are low compared to the doses calculated here for the very short decay time of 100 hours. Though the single iodine and xenon isotopes are the major contributors to offsite doses, the contributions from other isotopes are calculated and included in the overall dose values.

The present evaluation uses values for atmospheric diffusion factor (x/Q) and for filter efficiencies that are used in the FSAR Chapter 15 accident analysis. Initial core specific inventories (Curies per assembly) of fission products were estimated with the ORIGEN code[2]. Using ORIGEN input, core fission product inventory after 100 hours of decay time was determined by TVA code, "Source Transport Program" (STP). The results of the STP calculations for isotopes of interest are given in Table 9.1, while the percentages of the core inventories released from the fuel to the fuel rod gaps under the assumptions of Regulatory Guide 1.25 are listed in Table 9.2

The core isotopic inventory as determined by STP was input into TVA code FENCDOSE to calculate site boundary doses. FENCDOSE uses the following equations to calculate the thyroid, gamma and beta dose:

The whole body beta and gamma doses are calculated as follows:

$$D_{\beta\infty} = \sum_i \sum_j 0.23\,(Q_{ij})\,(X/Q)_j \bar{E}_{i\beta} \qquad (1)$$

$$D_{\gamma\infty} = \sum_i \sum_j 0.25\,(Q_{ij})\,(X/Q)_j \bar{E}_{i\gamma} \qquad (2)$$

where:

$D_{\beta\infty}$ = beta dose from a semi-infinite cloud (rem)

$D_{\gamma\infty}$ = "whole body" gamma dose from a semi-infinite cloud (rem)

Q_{ij} = integrated concentration of isotope i over time period j (Curie)

$(X/Q)_j$ = meteorological diffusion factor for time period j (sec/m^3)

$\bar{E}_{i\beta}$ = average beta energy for isotope i (MeV/disintegration)

$\bar{E}_{i\gamma}$ = average gamma energy for isotope i (MeV/disintegration)

The thyroid (inhalation) dose is given by

$$D_I = \sum_i \sum_j B_j\,(Q_{ij})\,(X/Q)_j\,C_i \qquad (3)$$

where

D_I = inhalation dose (rem)

B_j = breathing rate (m^3/sec)

Q_{ij} = integrated concentration of isotope i over time period j (Curie)

$(X/Q)_j$ = meteorological diffusion factor for time period j (sec/m^3)

C_i = dose conversion factor for isotope i (rem/Ci) (Ref. 2)

The following assumptions are used in performing the calculations:

1. The fuel handling accident (FHA) occurs at 100 hours after shutdown, consistent with the FSAR and the Technical Requirements Manual.

2. All of the rods in one fuel assembly are assumed to be damaged.[1]

3. All activity is assumed to be released to the environment over a two hour time period.[1]

4. All of the gap activity in the damaged rods is released which consists of 10% of the total noble gases other than Kr-85, 30% of Kr-85, and 10% of the total radioactive iodine in the rods at the time of the accident.[1]

5. The values assumed for individual fission product inventories are calculated assuming 3565 Mw power operation at the end of core life immediately preceding shutdown with a radial peaking factor of 1.65.[1]

6. From Regulatory Guide 1.25, the iodine gap inventory is composed of inorganic species (99.75%) and organic species (0.25%).

7. The pool decontamination factors for the inorganic iodine is assumed to be 133, and organic iodine is assumed to be 1.[1]

8. The retention of noble gases in the pool is negligible.[1]

9. It is assumed that the radioisotopes do not mix with the surrounding building atmosphere (Regulatory Guide 1.25). It is assumed that all radioisotopes enter the exhaust ductwork immediately. This is modeled in the STP computer run through step release.

10. The filter efficiency is 99% for all iodines.

9.1.2 Results

The WBN 2-hour site boundary doses from the specified fuel handling accident are tabulated below.

Thyroid dose, rem	=	1.8141
Beta dose, D_g (rem)	=	1.9614
Gamma dose, D_τ	=	0.675

These potential doses are well within the exposure guideline values of 10 CFR Part 100, paragraph 11. As defined in Standard Review Plan 15.7.4, Radiological Consequences of Fuel Handling Accidents, "well within" means 25 percent or less of the 10 CFR 100 guidelines, or values of 75 rad for thyroid doses and 6.25 rem for whole-body doses. The potential doses at WBN from the conservative scenarios presented here easily meet the criteria for "well within." This conclusion is also valid for 1500 effective full power days of operation.

9.2 SOLID RADWASTE

The necessity for spent fuel pool polisher resin replacement is determined primarily by the requirement for water clarity, and the resin is normally changed about once per refueling. No significant increase in the volume of solid radioactive wastes is expected with the expanded storage capacity. During reracking operations involving the peripheral racks, a small amount of additional reasons may be generated by the pool cleaning system on a one-time basis.

9.3 GASEOUS RELEASES

Gaseous releases from the fuel storage area are combined with other plant exhausts. Normally, the contribution from the fuel storage area is negligible compared to the other releases and no significant increases are expected as a result of the expanded storage capacity.

9.4 PERSONNEL EXPOSURES

During normal operations, personnel working in the fuel storage area are exposed to radiation from the spent fuel pool. PWR operating experience has shown that the area radiation dose rates, which originate primarily from radionuclides in the pool water, are generally 1 to 3 mrem/hr, with an occasional reading of 5 mrem/hr. Dose rates on the pool bridge crane platform are 4 to 5 mrem/hr. These doses may temporarily increase slightly during refueling operations. However, the reracking is not expected to significantly affect these doses.

Normal operational radiation levels in zones surrounding the pool are not expected to be significantly affected. Existing shielding around the pool (water depth and concrete walls) provide more than adequate protection, despite the slightly closer approach of the new racks to the walls of the pool.

Representative concentrations of radionuclides expected in the pool water are shown in Table 9.4. When the peripheral storage racks are added, the concentrations might be expected to increase due to crud deposits spalling from spent fuel assemblies which are shuffled. However, industry experience to date has not indicated a major increase as a consequence of fuel shuffling during reracking.

PWR operating experience has also shown that there have been negligible concentrations of airborne radioactivity and no increases are expected as a result of the expanded storage capacity. Area monitors for airborne activities are available in the immediate vicinity of the spent fuel pool.

No increase in radiation exposure to operating personnel is expected; therefore, neither the current health physics program nor the area monitoring system needs to be modified.

9.5 ANTICIPATED EXPOSURE DURING RERACKING

There will be no spent fuel or other contaminated material in the WBN spent fuel storage pool when the PaR System racks are installed. These storage racks were previously in service at TVA's Sequoyah Nuclear Plant and are contaminated. Future installation of the new, uncontaminated peripheral "baby" racks will be done remotely while spent fuel is stored in the pool. The operations involved in reracking will utilize detailed procedures prepared with full consideration of ALARA principles. Similar operations have been performed in a number of facilities in the past, and there is every reason to believe that reracking can be safely and efficiently accomplished at the WBN plant, with minimum radiation exposure to personnel.

Total occupational exposure for the reracking operation is estimated to be between 1.5 and 2.5 person-rem. This is believed to be a reasonable estimate for planning purposes. The existing radiation protection program at WBN is adequate for the reracking operations. Where there is a potential for airborne activity, continuous air samplers will be in operation. Personnel will wear protective clothing and, if necessary, respiratory protective equipment. Activities will be governed by a Radiation Work Permit, and personnel monitoring equipment will be assigned to each individual. As a minimum, this will include thermoluminescent dosimeters and electronic dosimeters. Additional personnel monitoring equipment (i.e., extremity badges) may be utilized as required. Work, personnel traffic, and the movement of equipment will be monitored and controlled to minimize contamination and to assure that exposures are maintained ALARA.

9.6 REFERENCES

1. Regulatory Guide 1.25 (AEC Safety Guide 25), "Assumptions Used For Evaluating The Potential Radiological Consequences of a Fuel Handling Accident In the Fuel Handing and Storage Facility for Boiling and Pressurized Water Reactors."

2. SAS2H Input for Computing Core Activities of 4.5, 5.0, and 5.5 Weight % uranium 235 Fuel for Sequoyah Nuclear Plant, ORNL/M-3739, Oak Ridge National Laboratory, August 1994.

TABLE 9.1

RESULTS OF STP CALCULATIONS FOR RADIONUCLIDES OF IODINE, KRYPTON, AND XENON AT 100-HOURS DECAY TIME

Radionuclide		Curies
KRM	83	6.42E-13
KRM	85	3.904E-03
KR	85	2.647E+03
KR	87	8.736E-20
KR	88	1.0100E-06
KR	89	0.000E+00
XEM	131	7.0290E+02
XEM	133	1.4580E+03
XE	133	9.6000E+04
XEM	135	7.6350E-05
XE	135	2.8800E+01
XE	138	0.0000E+00
I	131	4.2400E+00
I	132	5.565E-13
I	133	4.4740E-01
I	134	0.0000E+00
I	135	3.3350E-04
I*	131	1.4180E+00
I*	132	1.8610E-13
I*	133	1.4910E-01
I*	134	0.0000E+00
I*	135	1.1100E-04

*Organic Species

TABLE 9.2

DATA AND ASSUMPTIONS FOR THE EVALUATION
OF THE FUEL HANDLING ACCIDENT

Core power level, Mw(t)	3565
Fuel enrichment, wt% U	5.0
Effective Full Power Days Operation	1,000
Specific power, kw/kgU	40.00
Power peaking factor	1.65
Number of failed fuel rods	All rods in 1 of 193 assemblies
Core inventory released to gap, %	
Iodine-131	10*
Other iodines	10*
Krypton-85	30*
Xenon-133	10*
Other xenons	10*
Iodine composition, %	
Elemental	99.75
Organic	0.25
Pool decontamination factors	
Elemental iodine	133
Organic iodine	1
Noble gases	1
Filter Efficiencies	
Elemental iodine	99%
Organic iodine	99%
Noble gases	0%
Atmospheric diffusion factor (X/Q), sec/m^3	6.07×10^{-4}
Breathing rate, m^3/sec	3.47×10^{-4}

*From Regulatory Guide 1.25

TABLE 9.3

RADIONUCLIDE PROPERTIES USED IN THE
FUEL HANDLING ACCIDENT ANALYSIS

Nuclide	Dose Conversion, Rads/Curie
Iodine-131	1.48×10^6
Iodine-132	5.35×10^4
Iodine-133	4.0×10^5
Iodine-134	2.5×10^4
Iodine-135	1.24×10^5

ISOTOPE	GAMMA ENERGY (MEV/DIS)	BETA ENERGY (MEV/DIS)
KRM-83	0.0025	0.0371
KRM-85	0.1586	0.2529
KR-85	0.0022	0.2506
KR-87	0.7928	1.3237
KR-88	1.9629	0.3750
KR-89	2.0837	1.2310
XEM-131	0.0201	0.1428
XEM-133	0.0416	0.1898
XE-133	0.0454	0.1354
XEM-135	0.4318	0.0950
XE-135	0.2470	0.3168
XE-138	1.1830	0.6058
I-131	0.3810	0.1943
I-132	2.3332	0.5143
I-133	0.6100	0.4080
I-134	2.5928	0.6102
I-135	1.5802	0.3680

TABLE 9.4

REPRESENTATIVE CONCENTRATIONS OF RADIONUCLIDES
IN THE SPENT FUEL POOL WATER

Nuclide	Concentration, $\mu C_i/ml$
Co-58	3×10^{-5}
Co-60	4×10^{-4}
Cs-134	1×10^{-5}
Cs-137	3×10^{-5}

CHAPTER 10

ENVIRONMENTAL COST/BENEFIT ASSESSMENT

10.0 ENVIRONMENTAL COST/BENEFIT ASSESSMENT

10.1 INTRODUCTION

Article V of the USNRC OT position paper[1] requires the submittal of a cost/benefit analysis for the chosen fuel storage capacity enhancement method. This section discusses factors considered by TVA before selecting reracking as the most viable alternative.

10.2 IMPERATIVE FOR INCREASING SPENT FUEL STORAGE

The specific need to increase the limited existing spent fuel storage capacity at WBN is based on the projected continual increase in inventory in the spent fuel pool and the advisability of maintaining full-core off-load capability. The WBN fuel pool will lose the capacity to accept a discharge of one full core (193 fuel assemblies) late in the year 2001 or early 2002 after four cycles of operation. The capacity to accept a normal discharge batch would be lost two cycles later. The need for flexibility to load new fuel into the pool for component shuffling during a refueling outage advances the date for increased spent fuel pool storage capacity to one cycle (18 months) earlier.

The projected loss of storage capacity in the WBN pool would affect TVA's ability to operate the WBN Unit 1 reactor. There are no commercial independent spent fuel storage facilities operating in the United States. Since the cost of spent fuel reprocessing is not offset by the salvage value of the residual uranium, reprocessing represents an added cost for the nuclear fuel cycle which already includes the Nuclear Waste Policy Act (NWPA) Nuclear Waste Fund fees. In any event, there are no domestic reprocessing facilities. TVA does not have an existing or planned contractual arrangement for third-party fuel storage or fuel reprocessing. There are no acceptable alternatives to developing additional onsite spent fuel storage capacity for WBN. Replacement power costs an average of approximately $380,000 per day. Shutting down WBN is many times more expensive than increasing onsite spent fuel storage capacity.

10.3 APPRAISAL OF ALTERNATIVE OPTIONS

TVA has determined that reracking by transferring the PaR racks which were previously in service at the SQN is the most viable option for WBN in comparison to other spent fuel storage alternatives.

The key considerations in evaluating the alternative options were:

- Minimize the effects on plant systems and operations by reducing the amount of fuel handling as well as the attendant potential impacts on safety and as low as reasonably achievable (ALARA) radiation exposures.

- Maturity of the technology and the extent of industry experience.

- Maximize flexibility to:

1. Implement subsequent actions for further increasing onsite spent fuel storage capacity.

2. Interface with Department of Energy technology choices for shipment, storage, and ultimate disposal of the spent fuel

- Minimize overall capital and operational and maintenance (O&M) costs.

Reracking with the PaR racks from SQN was found by TVA to be the most attractive option at this time with respect to the foregoing criteria when compared to the following alternative technologies.

- Wet Storage

1. Reracking with ultra high density new racks.
2. Rod consolidation.
3. Transshipment (pool-to-pool).

- Dry Storage

1. Metal casks.
2. Concrete casks.
3. Concrete vaults.
4. Multi-purpose canisters/overpacks

10.4 PROJECT COST ESTIMATE

The total cost for the WBN rerack project is estimated to be approximately $2.7 million and includes engineering design, handling and transport, installation, and an allowance for contingencies. Comparative estimates of the costs per incremental fuel assembly storage space for the alternative technologies in 1995 dollars are:

Reracking with Sequoyah's PaR racks	$3,000
Reracking with ultra high density racks	$6,000 - $7,000
Rod Consolidation	$14,000-27,000
Transshipment	$14,000-18,000
Metal Casks	$36,000-52,000
Concrete Casks	$18,000-38,000
Concrete Vaults	$18,000-30,000
Multi-purpose Canisters/Overpacks	$30,000-35,000

10.5 RESOURCE COMMITMENT

The expansion of the spent fuel pool capacity will not require the commitment of significant additional primary resources because the existing SQN surplus racks are being utilized. Selection of the reracking alternative normally requires the following resources:

Stainless steel	300 tons
Boral neutron absorber	25 tons, of which 13 tons is boron carbide powder and 12 tons are aluminum.

For WBN, the stainless bearing pads will be the principal resource commitment, approximately 4 tons of stainless steel.

10.6 ENVIRONMENTAL CONSIDERATIONS

Due to the additional heat-load arising from increased spent fuel pool inventory, the anticipated maximum bulk pool temperature will rise by less than 10°F due to the proposed increase in the spent fuel inventory in the spent fuel pool. The total heat-load for the unplanned emergency core off-load (worst case) is less than 35 million BTU/HR, which is less than one percent of the total plant heat loss to the environment.

The increased bulk pool temperature will result in an increase in the pool water evaporation rate. This increase is within the capacity of the existing WBN heating, ventilating, and air conditioning (HVAC) system. The net result of the increased heat loss and water vapor emission to the environment is negligible.

In Chapter 9 of this report, an assessment of the impact of the expanded storage capacity on pool radwaste volume is considered. During actual installation of the replacement and new racks, no additional resins are expected to be generated by the pool cleanup system. It is concluded that the effect of the proposed capacity increase is insignificant.

Spent fuel has never been stored in the existing WBN racks. Since the replacement of these racks is planned to be complete before the end of the first operating fuel cycle, there will be no significant addition to the plant's low specific activity (LSA) waste output.

10.7 REFERENCES

1. "OT Position Paper for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, and January 18, 1979 amendment thereto.

ENCLOSURE 3

Proposed Technical Specification Change

Watts Bar Nuclear Plant Unit 1

Docket No. 50-390

(WBN-TS-96-010)

Determination of No Significant Hazards Considerations

ENCLOSURE 3

WATTS BAR NUCLEAR PLANT UNIT 1
NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Description of Proposed License Amendment

The proposed amendment would revise the WBN Unit 1 Technical Specifications to increase the enrichment and storage capacity of the spent fuel pool racks. The proposed modification increases the WBN spent fuel pool storage capacity from 484 fuel assemblies to 1835 fuel assemblies. The initial enrichment of the fuel to be stored in the spent fuel storage racks will be increased from 3.5 weight percent to 5 weight percent. This modification would also change the spacing of stored fuel assembly center-to-center spacing from a nominal 10.72 inches to 10.375 inches in 24 flux trap rack modules and 8.972 inches in ten smaller burnup credit rack modules to be installed peripherally along the south and west pool walls and in a single 15 x 15 burnup credit rack to be installed in the cask pit.

In addition to the above proposed revisions, two limiting conditions for operation (LCOs) will be added to require that the combination of initial enrichment and burnup of each spent fuel assembly be stored is in the acceptable region and to require boron concentration of the cask pit to be greater than or equal to 2000 parts per million (ppm) during fuel movement in the flooded cask pit.

The WBN Unit 1 Technical Specification Bases and the Technical Requirements Manual (TRM) would be revised to support these changes.

Basis for No Significant Hazards Consideration Determination

The Nuclear Regulatory Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2).create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed below for the proposed amendment.

(1) **Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.**

The following potential scenarios were considered:

1. A spent fuel assembly drop.

2. Drop of the transfer canal gate or the cask pit divider gate.

3. A seismic event.

4. Loss of cooling flow in the spent fuel pool.

5. Installation activities.

The effect of additional spent-fuel pool storage cells fully loaded with fuel on the first four potential accident scenarios listed above has been reviewed. It was concluded that after installation activities have been completed, the presence of additional fuel in the pool does not increase the probability of occurrence of these four events.

With regard to installation activities, the existing WBN TRM prohibit loads in excess of 2059 pounds from travel over fuel assemblies in the storage pool and require the associated crane interlocks and physical stops be periodically demonstrated operable. During installation, racks and associated handling tools will be moved over the spent fuel pool, however there will be no fuel in the pool when the 24 flux trap rack modules are installed. A three foot lateral free zone clearance from stored spent fuel will be maintained during installation of the ten smaller burnup credit rack modules. Installation work in the spent fuel pit area will be controlled and performed in strict accordance with specific written procedures.

NRC guidance provides that, in lieu of providing a single failure-proof crane system, the control-of-heavy-loads guidelines can be satisfied by establishing that the potential for a heavy load drop is extremely small. Storage rack movements to be accomplished with the WBN Auxiliary Building crane will conform with NUREG-0612 guidelines in that the probability of a drop of a storage rack is extremely small. The crane has a tested capacity of 125 tons. The maximum weight of any existing, replacement, or new storage rack and its associated handling tool is less that 20 tons. Therefore, there is ample safety factor margin for movements of the storage racks by the Auxiliary Building crane. Special lifting devices, which have redundancy or a rated capacity sufficient to maintain adequate safety factors, will also be utilized in the movements of the storage racks. In accordance with NUREG-0612, Appendix B, the safety margin ensures that the probability of a load drop is extremely low.

Future load travel over fuel stored in a rack specifically designed for the cask loading area of the cask pit will be prohibited unless an impact shield, which has been specifically designed for this purpose, is covering the area. Loads that are permitted when the shield is in place must meet analytically determined weight, travel height, and cross-sectional area criteria that preclude penetration of the shield. A TR has been proposed that incorporates the previously mentioned load criteria.

Also a rack changeout sequence will be developed that addresses removal of the existing racks, movement of the new racks into the Auxiliary Building, initial staging on the refueling floor, and final installation in the pool. The changeout sequence objectives will include establishing lift heights, travel distances, and number of lifts to be as low as reasonably achievable. Accordingly, it is concluded that the proposed installation activities will not significantly increase the probability of a load-handling accident. The consequences of a load-handling accident are unaffected by the proposed installation activities.

The consequences of a spent fuel assembly drop were evaluated, and it was determined that the racks will not be distorted such that the racks would not perform their safety function. The criticality acceptance criterion, K_{eff} 0.95, is not violated, and the calculated doses are well within 10 CFR Part 100 guidelines. Thus, the consequences of this type of accident are not changed from previously evaluated spent fuel assembly drops that have been found acceptable by NRC.

A TRM change has been proposed that would permit the transfer-canal gate and the divider gate for the cask pit to travel over fuel assemblies in the spent fuel pool. Rack damage is restricted to an area above the active fuel region.

The consequences of a seismic event have been evaluated. The replacement racks are designed and fabricated and the new racks will be fabricated to meet the requirements of applicable portions of the NRC regulatory guides and published standards. Design margins have been provided for rack tilting, deflection, and movement such that the racks do not impact each other or the spent fuel pool walls in the active fuel region during the postulated seismic events. The free-standing racks will maintain their integrity during and after a seismic event. The fuel assemblies also remain intact and therefore no criticality concerns exist.

The spent fuel pool system is a passive system with the exception of the fuel pool cooling train and heating, ventilating, and air-conditioning (HVAC) equipment. Redundancies in the cooling train and HVAC hardware are not reduced by the planned fuel storage modification. The potential increased heat load resulting from any additional storage of spent fuel is well within the existing system cooling capacity. Therefore, the probability of occurrence or malfunction of safety equipment leading to the loss of cooling flow in the spent fuel pool is not significantly affected. Furthermore, the consequences of this type incident are not significantly increased from previously evaluated cooling system loss of flow malfunctions. Thermal-hydraulic scenarios assume the reracked pool is approximately 90 percent full with spent fuel assemblies. From this starting point, the remaining storage capacity is utilized by analyzing both normal and unplanned full core offloads using conservative assumptions and previously established methods. Calculated values include maximum pool water bulk temperature, coincident maximum pool water local temperature, the maximum fuel cladding temperature, time-to-boil after loss of cooling paths, and the effect of flow blockage in a storage cell.

Although the proposed modification increases the pool heat load, results from the above analyses yield a maximum bulk temperature of approximately 160 degrees Fahrenheit which is below the bulk boiling temperature. Also the maximum local water temperature is below nucleate boiling condition values. Associated results from corresponding loss of cooling evaluations give minimums of 5.3 hours before boiling begins and 45 hours before the pool water level drops to the minimum required for shielding spent fuel. This is sufficient time to begin utilization of available alternate sources of makeup cooling water. Also, the effect of the increased thermal loading on the pool structure was evaluated and determined to be acceptable.

(2) **Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously analyzed.**

The proposed modification has been evaluated in accordance with the guidance of the NRC position paper entitled, "OT Position for Review and Acceptance of Spent-Fuel Storage and Handling applications;" appropriate NRC regulatory guidelines; appropriate NRC standard review plans; and appropriate industry codes and standards. Proven analytical technology was used in designing the planned fuel storage expansion and will be utilized in the installation process. Basic reracking technology has been developed and demonstrated in applications for fuel pool capacity increases that have already received NRC staff approval.

Proposed TSs for the spent fuel storage racks use burnup credit and fuel assembly administrative placement restrictions for criticality control. These restrictions are described in the proposed change to the design features section of the TSs by reference to the Spent Fuel Pool Modifications report. Additional evaluations were required to ensure that the criticality criterion, $k_{eff} \leq 0.95$, is maintained. These include evaluation for the abnormal placement of unirradiated (fresh) fuel assemblies of 5.0 weight percent (wt%) enrichment into a storage cell location designed for lower enrichment or irradiated fuel. Soluble boron, for which credit is permitted under these abnormal conditions, ensures that reactivity is maintained substantially less than the design requirement. For example, if the flux trap Programmed and Remote System Corporation (PaR) racks are inadvertently all loaded with fresh assemblies of the maximum 5.0 wt% fuel instead of observing the 3.8 wt% and 6.75 MWD/KgU controls, the worth of the 2000 ppm borated water is sufficient to lower the k_{eff} of the storage racks to 0.83. The existing and proposed TSs require boron concentration in the pool and cask pit to be ≥ 2000 ppm during fuel movement. An analytical determination of the reactivity worth of 2000 ppm borated water in the spent fuel storage pool predicted the change in k_{eff} to be approximately 17 percent Δk_{eff}. Although no credit for soluble boron was proposed in the TSs, it was also determined by an independent calculation that a minimum concentration of 520 ppm soluble boron allows the unrestricted storage of 5.0 wt% enriched fuel in the PaR flux trap racks.

The Holtec-designed peripheral "baby" racks and the 15 x 15 racks in the cask loading area can safely and conservatively store fuel of 5 wt% initial enrichment burned to 41 MWD/kgU or lower enriched fuel with lower burnup, i.e., fuel of equivalent reactivity. Evaluations have confirmed that, for the abnormal placement of a fresh fuel assembly of

5.0 wt% in these racks, the criticality criterion is maintained with the existing and proposed TS requirements of 2000 ppm soluble boron.

Although these changes required addressing additional aspects of a previously analyzed accident, the possibility of a previously unanalyzed accident is not created. It is therefore concluded that the proposed reracking does not create the possibility of a new or different kind of accident from any previously analyzed.

(3) **Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.**

The design and technical review process applied to the reracking modification included addressing the following areas:

1. Nuclear criticality considerations

2. Thermal-hydraulic considerations.

3. Mechanical, material, and structural considerations.

The established acceptance criterion for criticality is that the neutron multiplication factor shall be less than or equal to 0.95, including all uncertainties. The results of the criticality analyses for the rack designs demonstrate that this criterion is satisfied. The methods used in the criticality analysis conform to the applicable portions of NRC guidance and industry codes, standards, and specifications. In meeting the acceptance criteria for criticality in the spent fuel pool and the cask loading area, such that k_{eff} is always less than 0.95 at a 95/95 percent probability tolerance level, the proposed amendment does not involve a significant reduction in the margin of safety for nuclear criticality.

Conservative methods and assumptions were used to calculate the maximum fuel temperature and the increase in temperature of the water in the spent fuel pit area. The thermal-hydraulic evaluation used methods previously employed. The proposed storage modification will increase the heat load in the spent fuel pool, but the evaluation shows that the existing spent fuel cooling system will maintain the bulk pool water temperature at or below 160 degrees Fahrenheit. Thus it is demonstrated that the worst-case peak value of the pool bulk temperature is considerably lower than the bulk boiling temperature. Evaluation also shows that maximum local water temperatures along the hottest fuel assembly are below the nucleate boiling condition value. Thus there is no significant reduction in the margin of safety for thermal hydraulic or spent fuel cooling considerations.

The mechanical, material, and structural design of the spent fuel racks is in accordance with applicable portions of NRC's position in "OT Position for Review and Acceptance of

Spent-Fuel Storage and Handling applications," dated April 14, 1978 (as modified January 18, 1979), as well as other applicable NRC guidance and industry codes The primary safety function of the spent fuel racks is to maintain the fuel assemblies in a safe configuration through normal and abnormal loading conditions. Abnormal loadings that have been evaluated with acceptable results and discussed previously include the effect of an earthquake and the impact because of the drop of a fuel assembly. The rack materials used are compatible with the fuel assemblies and the environment in the spent fuel pool. The structural design for the new racks provides tilting, deflection, and movement margins such that the racks do not impact each other or the spent fuel pit walls in the active fuel region during the postulated seismic events. Also the spent fuel assemblies themselves remain intact and no criticality concerns exist. In addition, finite element analysis methods were used to evaluate the continued structural acceptability of the spent fuel pit. The analysis was performed in accordance with "Building Code Requirements for Reinforced Concrete," (ACI 318-63,77). Therefore, with respect to mechanical, material, and structural considerations, there is no significant reduction in a margin of safety.

Summary

Based on the above analysis, TVA has determined that operation of WBN in accordance with the proposed amendment would not (1) involve a significant increase in the probability of consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Therefore, operations of WBN in accordance with the proposed amendments as described do not involve significant hazard considerations as defined in 10 CFR 50.92 and that the criteria of 10 CFR 50.91 have accordingly been met.

TVA has also reviewed the NRC examples of licensing amendments considered not likely to involve significant hazards considerations as provided in the final adoption of 10 CFR 50.92 published on page 7751 of the Federal Register, Volume 51, No. 44, March 6, 1986. Example (X) provides four criteria that, if satisfied by a reracking request, indicate that it is likely no significant hazards considerations are involved. The criteria and how TVA's amendment request for WBN complies are indicated below.

Criterion (1):

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

ENCLOSURE 3

WATTS BAR NUCLEAR PLANT UNIT 1
NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Proposed Amendment:

The WBN reracking involves replacing the existing racks with a design that allows slightly closer spacing between stored fuel assemblies and also provides additional rack storage on the pool floor where space permits.

Criterion (2):

The storage expansion method does not involve rod consolidation or double tiering.

Proposed Amendment:

The WBN racks are not double tiered, and the racks will sit on the floor of the spent fuel pool. Additionally, the amendment application does not involve consolidation of spent fuel.

Criterion (3):

The k_{eff} of the pool is maintained less than or equal to 0.95.

Proposed Amendment:

The design of the spent fuel racks contains a neutron absorber, Boral, to allow close storage of spent fuel assemblies while ensuring that the k_{eff} remains less than 0.95 under normal operating conditions with unborated water in the pool and less than 0.95 under abnormal conditions with soluble boron in the pool.

Criterion (4):

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

Proposed Amendment:

The construction processes and analytical techniques used in the fabrication and design are substantially the same as those of numerous other rack installations. Thus, no new or unproven technology is utilized in the construction or analysis of the high-density, spent fuel racks at WBN. TVA's Contractor, Holtec International, has previously supplied licensable racks of very similar design for about 10 other reracking projects.

ENCLOSURE 4

Proposed Environmental Impact Evaluation

Watts Bar Nuclear Plant Unit 1

Docket No. 50-390

ENCLOSURE 4

WATTS BAR NUCLEAR PLANT UNIT 1
PROPOSED ENVIRONMENTAL IMPACT EVALUATION

The proposed changes do not involve an unreviewed environmental question because operation of WBN Unit 1 in accordance with this change would not:

1. involve a significant hazards consideration,

2. a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or

3. a significant increase in individual or cumulative occupational exposure.

Therefore, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed changes is not required.

NOTE: The bases for the above statements are discussed in Chapter 9 (Radiological Evaluations) and Chapter 10 (Environmental Cost Benefit Assessment of Enclosure 2, Spent Fuel Pool Modification for Increased Storage Capacity)

ENCLOSURE 5

COMMITMENT LIST

Watts Bar Nuclear Plant Unit 1

Docket No. 50-390

(WBN-TS-96-010)

ENCLOSURE 5

WATTS BAR NUCLEAR PLANT UNIT 1
COMMITMENT LIST

1. Procedures and modifications to install the Sequoyah PaR racks will be in accordance with the "Spent Fuel Pool Modification for Increased Storage Capacity Report" submitted with Technical Specification Request WBN-TS-96-010.

2. Procedures and modifications to install the Holtec burnup credit racks in the WBN spent fuel pool and the cask pit area will be in accordance with the "Spent Fuel Pool Modification for Increased Storage Capacity Report" submitted with Technical Specification Request WBN-TS-96-010.