UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 28, 1993

Docket Nos. 50-327
and 50-328

Tennessee Valley Authority
ATTN: Dr. Mark O. Medford, Vice President
Nuclear Assurance, Licensing & Fuels
3B Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Dr. Medford:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M83068 AND M83069) (TS 92-01)

The Commission has issued the enclosed Amendment No. 167 to Facility Operating License No. DPR-77 and Amendment No. 157 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant, Units 1 and 2, respectively. These amendments are in response to your application dated March 27, 1992, which was supplemented by letters dated May 11, May 28, September 8, and October 8, 1992; February 18 and April 1, 1993.

The amendments incorporate the technical specification changes that are necessary for expansion of the spent fuel pool storage capacity to 2091 fuel assemblies and addition of a fuel rack storage module to be located in the cask loading area of the cask pit to accommodate no more than 225 additional fuel assemblies. The new racks increase the total spent fuel storage capacity to 2316 fuel assemblies and extend the projected storage capacity into the year 2005 or 2006.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

David E. LaBarge, Senior Project Manager
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 167 to License No. DPR-77
2. Amendment No. 157 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures:
See next page

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and 50-328

April 28, 1993

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Nuclear Assurance, Licensing & Fuels
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A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

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Original signed by

David E. LaBarge, Senior Project Manager
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 167 to License No. DPR-77
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cc w/enclosures:
See next page

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Tennessee Valley Authority
ATTN: Dr. Mark O. Medford

Sequoyah Nuclear Plant

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AMENDMENT NO.167 FOR SEQUOYAH UNIT NO. 1 - DOCKET NO. 50-327 and
AMENDMENT NO.157 FOR SEQUOYAH UNIT NO. 2 - DOCKET NO. 50-328
DATED: April 28, 1993

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 167
License No. DPR-77

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated March 27, 1992, which was supplemented by letters dated May 11, May 28, September 8, and October 8, 1992; February 18 and April 1, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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P PDR

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-77 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 167, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance. It will be implemented when the proper plant conditions can be established that will accommodate the corresponding modifications. The staff requests that the licensee inform the Commission by letter when implementation has been completed.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 28, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 167

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	**INSERT**
3/4 9-1a	3/4 9-1a
3/4 9-7	3/4 9-7
3/4 9-7a	3/4 9-7a
B3/4 9-2	B3/4 9-2
5-5	5-5
-	5-5a
-	5-5b
-	5-5c
-	5-5d
-	5-5e

SURVEILLANCE REQUIREMENTS (Continued)

4.9.1.3 One of the following valve combinations shall be verified closed under administrative control at least once per 72 hours:

Combination A	Combination B	Combination C	Combination D
a. 1-81-536	a. 1-81-536	a. 1-81-536	a. 1-81-536
b. 1-62-922	b. 1-62-922	b. 1-62-907	b. 1-62-907
c. 1-62-916	c. 1-62-916	c. 1-62-914	c. 1-62-914
d. 1-62-933	d. 1-62-940	d. 1-62-921	d. 1-62-921
	e. 1-62-696	e. 1-62-933	e. 1-62-940
	f. 1-62-929		f. 1-62-929
	g. 1-62-932		q. 1-62-932
	h. 1-FCV-62-128		h. 1-62-696
			i. 1-FCV-62-128

4.9.1.4 The boron concentration in the spent fuel pool shall be determined by chemical analysis to be greater than or equal to 2,000 parts per million (ppm) at least once per 72 hours during fuel movement and until the configuration of the assemblies in the storage racks is verified to comply with the criticality loading criteria specified in Design Feature 5.6.1.1.c and 5.6.1.1.d.

4.9.1.5 The boron concentration in the cask loading area of the cask pit shall be determined by chemical analysis to be greater than or equal to 2000 parts per million (ppm) at least once per 72 hours during fuel movement in that area and until the assemblies in that storage rack are verified to comply with the criticality loading criteria specified in Design Feature 5.6.1.1.e.

3/4.9.7 CRANE TRAVEL - SPENT FUEL PIT AREA

LIMITING CONDITION FOR OPERATION

3.9.7 Loads traveling over fuel assemblies in the spent fuel pit area shall be restricted as follows:

a. Spent fúel storage pool:

Loads in excess of 2100 pounds* shall be prohibited from travel over fuel assemblies in the spent fuel storage pool.

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 it.

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

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7.1 Crane interlocks and physical stops which prevent crane hook travel over the storage pool shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

4.9.7.2 When fuel is stored in the cask pit area, verify administrative requirements concerning the impact shield are met prior to moving loads in excess of 2100 pounds across the cask pit area.

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*The spent fuel pool transfer canal gate and the spent fuel pool divider gate may travel over fuel assemblies in the spent fuel pool.

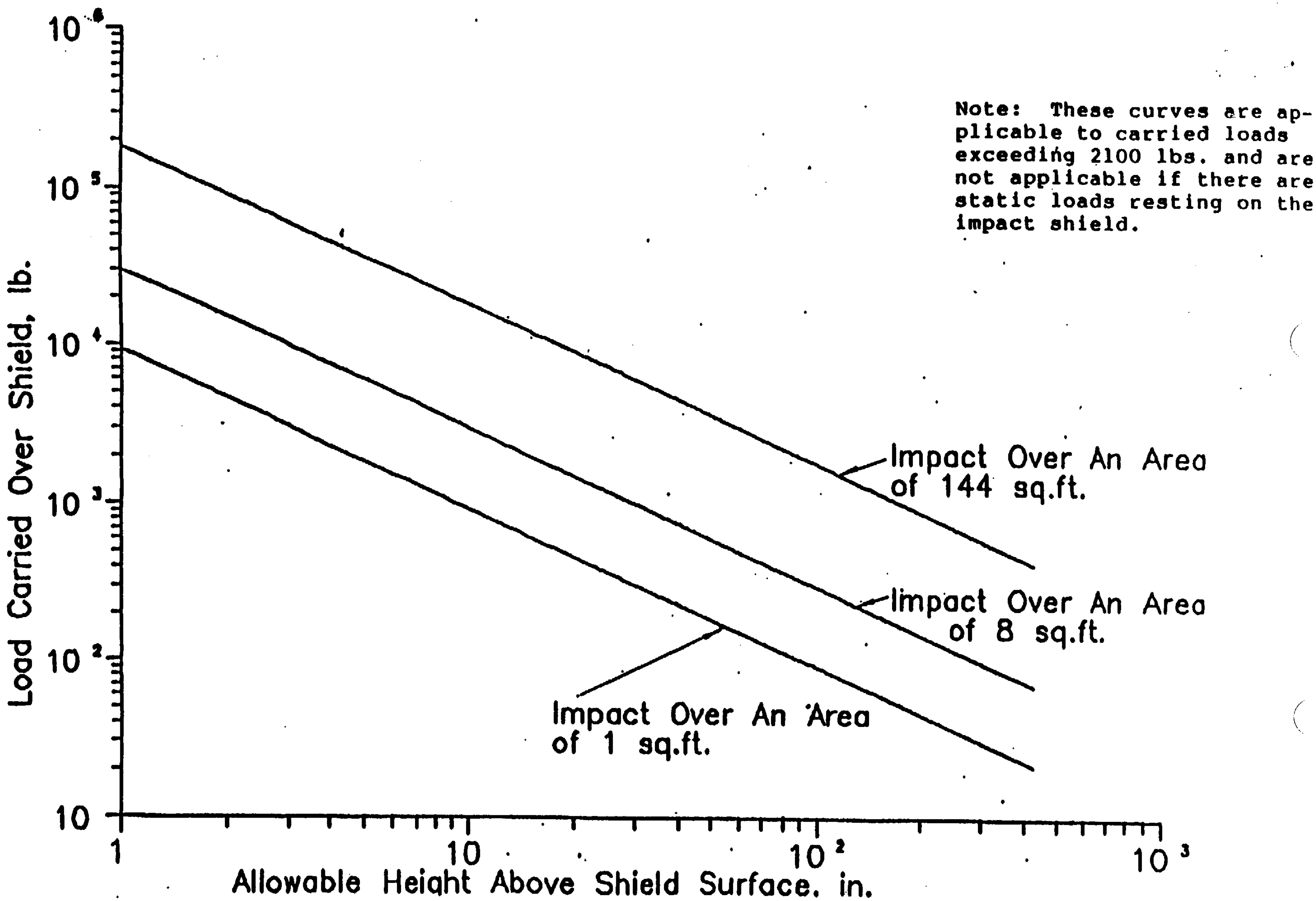

FIGURE 3.9-1 RELATIONSHIP BETWEEN LOAD, ALLOWABLE HEIGHT AND IMPACT AREA FOR OBJECTS TO BE CARRIED OVER THE IMPACT SHIELD

3/4.9.6 MANIPULATOR CRANE

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

3/4.9.7 CRANE TRAVEL - SPENT FUEL PIT AREA

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped: 1) the activity release will be limited to that contained in a single fuel assembly, and 2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analyses. Assurance against load drops over fuel stored in the cask loading area of the cask pit is achieved by observance of the calculated load criteria which will prevent penetration of the impact shield in the event of a load drop.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that; 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and 2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification. The minimum required flow rate of 2000 gpm ensures decay heat removal, minimizes the probability of losing an RHR pump by air-entrainment from pump vortexing, and minimizes the potential for valve damage due to cavitation or chatter. Losing an RHR pump is a particular concern during reduced RCS inventory operation. The 2000 gpm value is limited by the potential for cavitation in the control valve and chattering in the 10-inch check valve.

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

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

5.6.1.1 The spent fuel storage racks are designed for fuel enriched to 5 weight percent U-235 and shall be maintained with:

a. A K_{eff} equivalent to less than 0.95 when flooded with unborated water.*.

b. A nominal 8.972 inch center-to-center distance between fuel assemblies placed in the storage racks.

c. A three region arrangement in the spent fuel storage pool with the following definitions:

   1. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.95 ± 0.05 wt% U-235, or spent fuel regardless of the fuel burnup.

   2. Region 2 is designed to accommodate fuel of 4.95% initial enrichment burned to at least 50 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 in MWD/KgU is given by Y; when $Y = -23.761 + 22.075E - 2.0165E^2 + 0.1152E^3$, where E is the initial enrichment in the axial zone of highest enrichment.

   3. Region 3 is designed to accommodate fuel of 4.95% 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 $Y = -25.7425 + 18.76E - 1.3933E^2 + 0.0666E^3$, where E is the initial enrichment in the axial zone of highest enrichment.

   An empty cell is less reactive than any cell containing fuel and therefore may be used as a Region 1, Region 2, or Region 3 cell in any arrangement.

d. The following arrangement of regions apply in the spent fuel storage pool:

   1. Region 1 fuel assemblies located along the periphery of the storage modules adjacent to the pool walls must be isolated from

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*For some accident conditions, the presence of dissolved boron in the pool water may be taken into account by applying the double contingency principle which requires two unlikely, independent, concurrent events to produce a criticality accident.

5.6 FUEL STORAGE

each other and from the inner Region 3 cells by at least one (1) Region 2 spent fuel assembly (i.e., fuel of 50 MWD/KgU burnup or equivalent).

2. Region 1 fuel assemblies located along the wide water-gaps** between storage modules must be isolated from each other and from the inner Region 3 cells by at least one (1) Region 2 spent fuel assembly (i.e., fuel of 50 MWD/KgU assembly average burnup or equivalent).

3. Region 1 fuel assemblies located along the narrow water-gaps** between storage modules must be isolated from each other by at least two (2) Region 2 spent fuel assemblies and from the inner Region 3 cells by at least one (1) Region 2 spent fuel assembly (i.e., fuel of 50 MWD/KgU assembly average burnup or equivalent).

4. A checkerboard pattern of fresh fuel and empty cells may be used throughout any storage module, or internal to any storage module in lieu of Region 3 fuel as shown in Figure 5.6-2.

Figure 5.6-1 shows a typical arrangement of regions. Figure 5.6-2 illustrates internal module checkerboarding of fresh fuel with empty cells in a portion of the fuel pool. Figure 5.6-3 illustrates the two burnup-enrichment equations (5.6.1.1.c.2 and 5.6.1.1.c.3) in graphical form.

e. Only spent fuel meeting the Region 3 burnup requirements shall be stored in any module in the cask loading area of the cask pit.

CRITICALITY - NEW FUEL

5.6.1.2 The new fuel pit storage racks are designed and shall be maintained with a nominal 21.0 inch center-to-center distance between new fuel assemblies such that k_{eff} will not exceed 0.98 when fuel having an enrichment of 4.5 weight percent U-235 is in place and optimum achievable moderation is assumed.

DRAINAGE

5.6.2 The spent fuel pit is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 722 ft.

**The nominal gap (2-1/8 inches) running in the E-W direction between the adjacent modules is referred to as the "wide gap." The N-S direction gap (1.5 inch) is referred to as the "narrow gap."

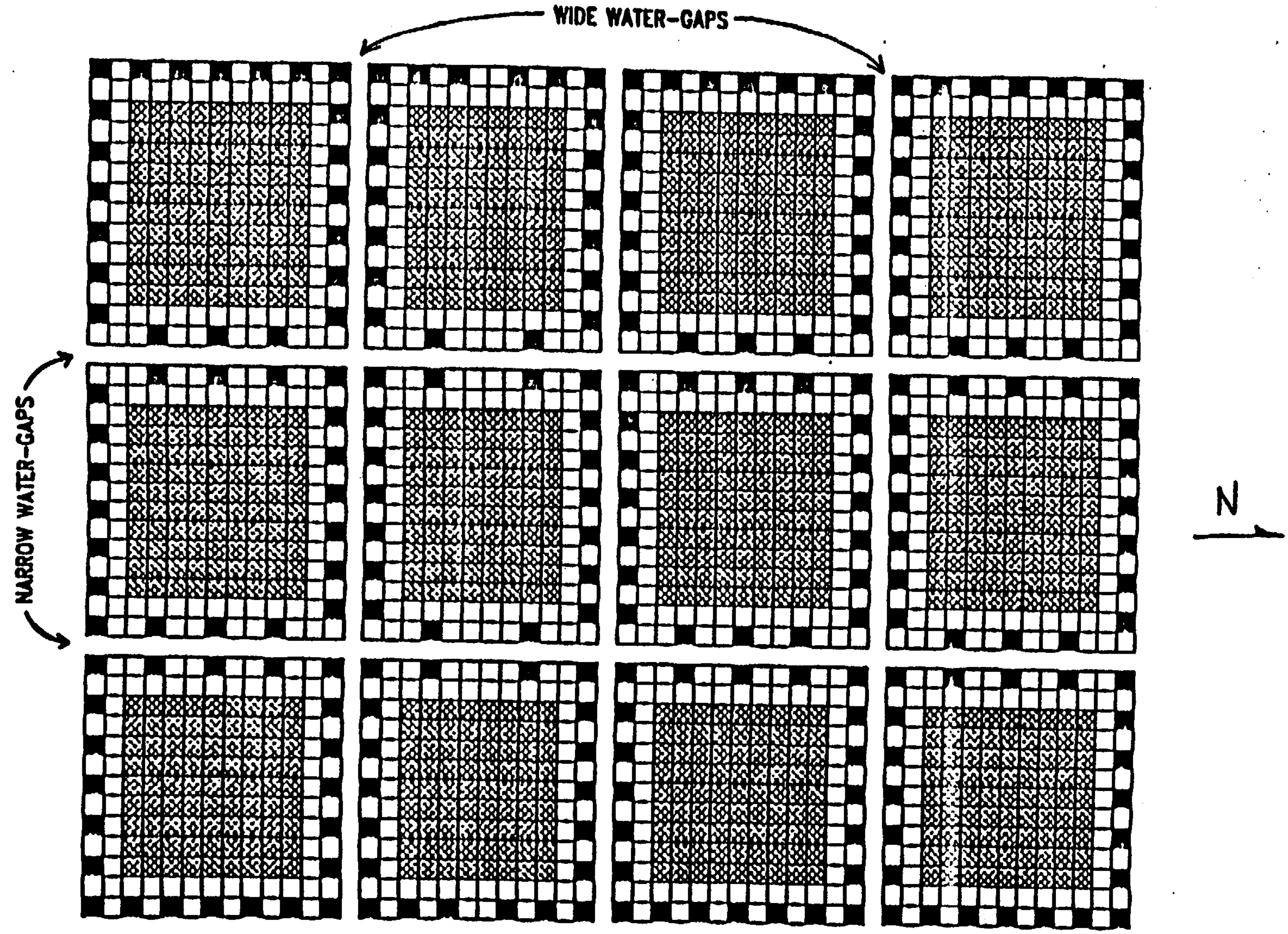
5.6 FUEL STORAGE

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2091 fuel assemblies. In addition, no more than 225 fuel assemblies will be stored in a rack module in the cask loading area of the cask pit.

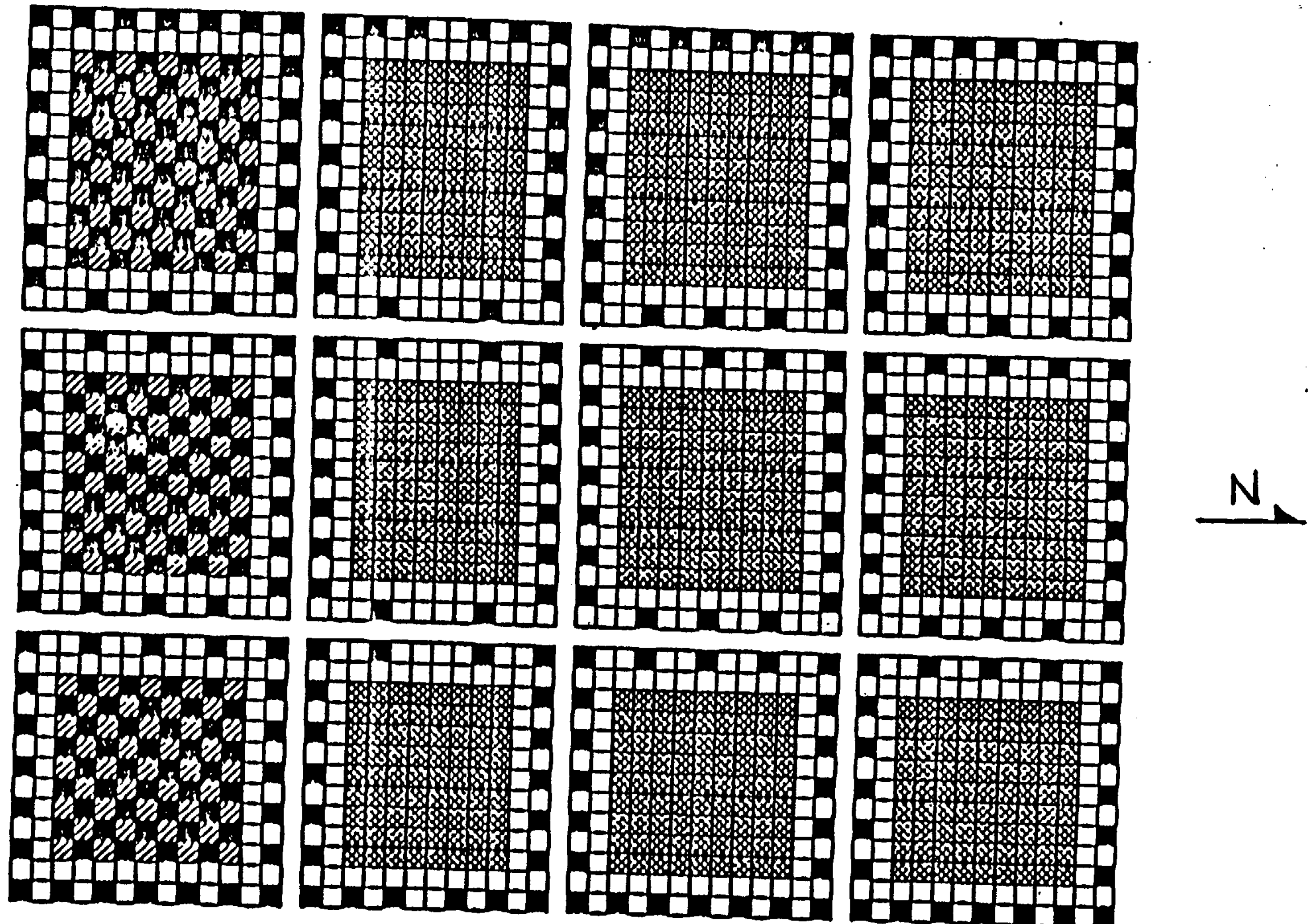
5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

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



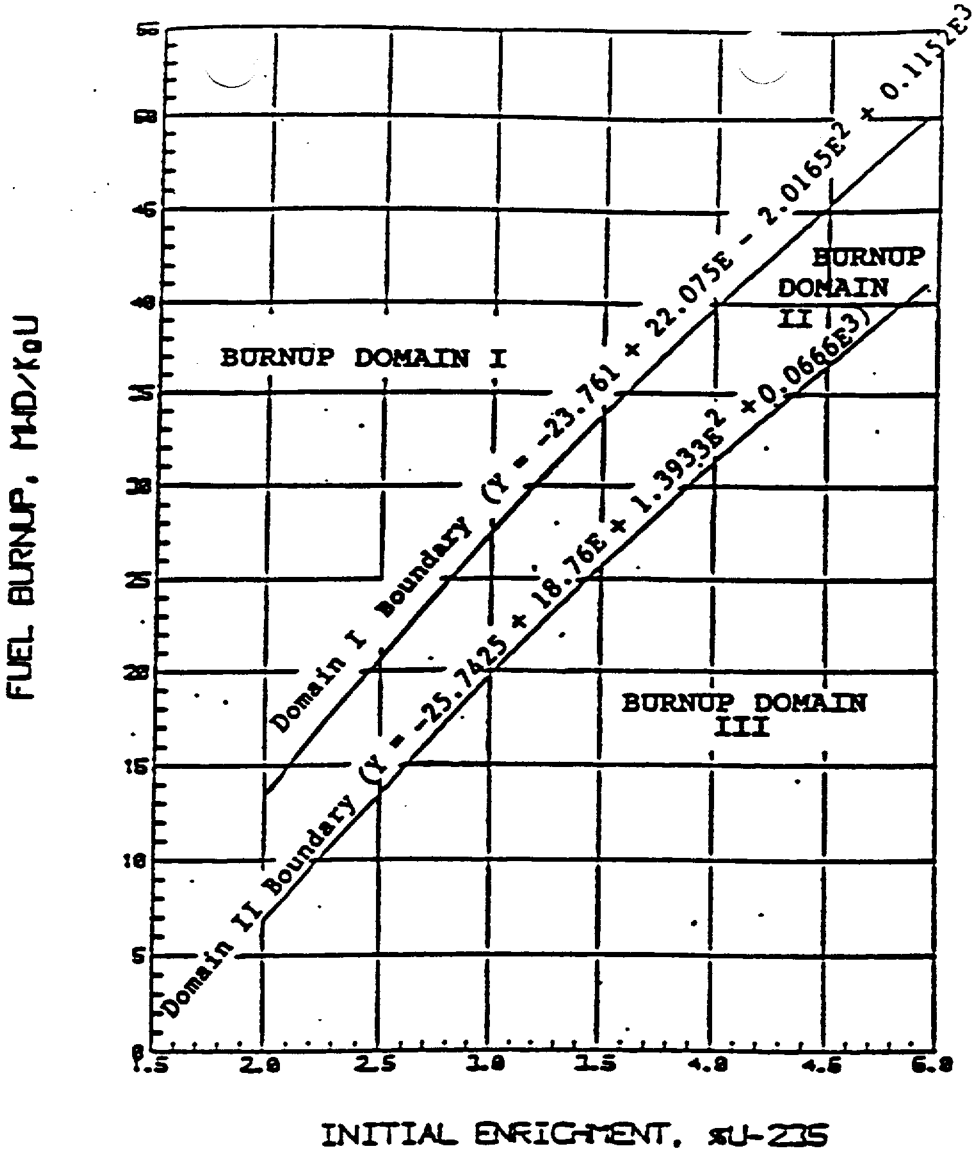
■ 250 Region 1 cells □ 826 Region 2 cells ▩ 1015 Region 3 cells

FIGURE 5.6-1 REFERENCE SPENT FUEL POOL LAYOUT



250 + 130 Region 1 cells
131 Empty Cells
826 Region 2 cells
754 Region 3 cells

FIGURE 5.6-2 EXAMPLE OF CHECKERBOARD SPENT FUEL STORAGE ARRAY



Domain	I:	Fuel assemblies with initial enrichment-burnup combinations in Domain I may be placed in either Region 1, 2, or 3 storage cells.
Domain	II:	Fuel assemblies with initial enrichment-burnup combinations in Domain II shall be placed only in Region 1 or 3 storage cells.
Domain	III:	Fuel assemblies with initial enrichment-burnup combinations in Domain III shall be placed only in Region 1 storage cells

FIGURE 5.6-3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 157
License No. DPR-79

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated March 27, 1992, which was supplemented by letters dated May 11, May 28, September 8 and October 8, 1992; February 18 and April 1, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-79 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 157, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance. It will be implemented when the proper plant conditions can be established that will accommodate the corresponding modifications. The staff requests that the licensee inform the Commission by letter when implementation has been completed.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 28, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 157

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
3/4 9-2	3/4 9-2
3/4 9-8	3/4 9-8
3/4 9-8a	3/4 9-8a
B3/4 9-2	B3/4 9-2
5-5	5-5
-	5-5a
-	5-5b
-	5-5c
-	5-5d
-	5-5e

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

4.9.1.3 One of the following valve combinations shall be verified closed under administrative control at least once per 72 hours:

Combination A	Combination B	Combination C	Combination D
a. 2-81-536	a. 2-81-536	a. 2-81-536	a. 2-81-536
b. 2-62-922	b. 2-62-922	b. 2-62-907	b. 2-62-907
c. 2-62-916	c. 2-62-916	c. 2-62-914	c. 2-62-914
d. 2-62-933	d. 2-62-940	d. 2-62-921	d. 2-62-921
	e. 2-62-696	e. 2-62-933	e. 2-62-940
	f. 2-62-929		f. 2-62-929
	g. 2-62-932		g. 2-62-932
	h. 2-FCV-62-128		h. 2-62-696
			i. 2-FCV-62-128

4.9.1.4 The boron concentration in the spent fuel pool shall be determined by chemical analysis to be greater than or equal to 2000 ppm at least once per 72 hours during fuel movement and until the configuration of the assemblies in the storage racks is verified to comply with the criticality loading criteria specified in Design Feature 5.6.1.1.c and 5.6.1.1.d.

4.9.1.5 The boron concentration in the cask loading area of the cask pit shall be determined by chemical analysis to be greater than or equal to 2000 parts per million (ppm) at least once per 72 hours during fuel movement in that area and until the assemblies in that storage rack are verified to comply with the criticality loading criteria specified in Design Feature 5.6.1.1.e.

3/4.9.7 CRANE TRAVEL - SPENT FUEL PIT AREA

LIMITING CONDITION FOR OPERATION

3.9.7 Loads traveling over fuel assemblies in the spent fuel pit area shall be restricted as follows:

a. Spent fuel storage pool:

   Loads in excess of 2100 pounds* shall be prohibited from travel over fuel assemblies in the spent fuel storage pool.

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 it.

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

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7.1 Crane interlocks and physical stops which prevent crane hook travel over the storage pool shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

4.9.7.2 When fuel is stored in the cask pit area, verify administrative requirements concerning the impact shield are met prior to moving loads in excess of 2100 pounds across the cask pit area.

*The spent fuel pool transfer canal gate and the spent fuel pool divider gate may travel over fuel assemblies in the spent fuel pool.

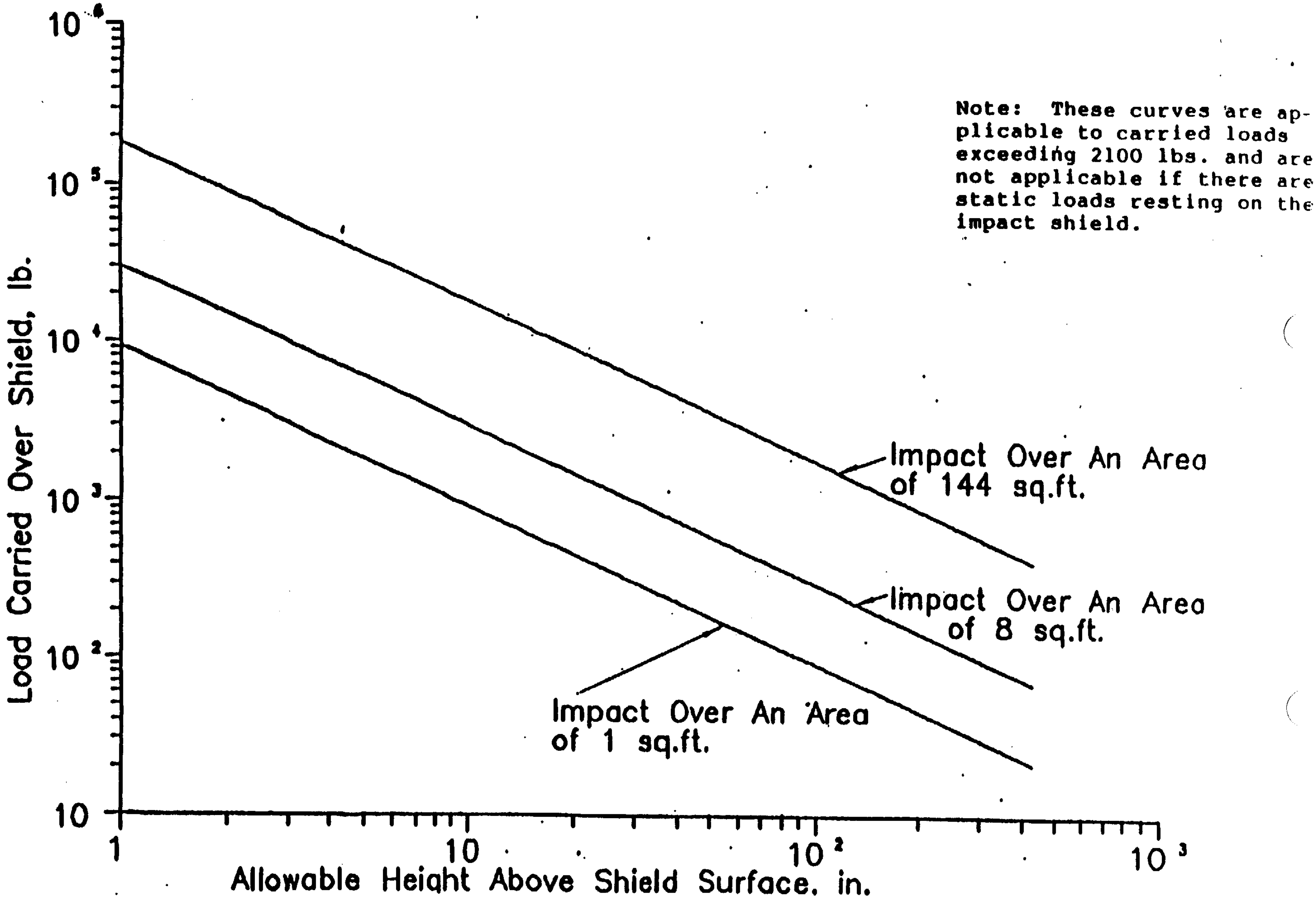


FIGURE 3.9-1 RELATIONSHIP BETWEEN LOAD, ALLOWABLE HEIGHT AND IMPACT AREA FOR OBJECTS TO BE CARRIED OVER THE IMPACT SHIELD

3/4.9.6 MANIPULATOR CRANE

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

3/4.9.7 CRANE TRAVEL - SPENT FUEL PIT AREA

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped 1) the activity release will be limited to that contained in a single fuel assembly, and 2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analyses. Assurance against load drops over fuel stored in the cask loading area of the cask pit is achieved by observance of the calculated load criteria which will prevent penetration of the impact shield in the event of a load drop.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that; 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and 2) sufficient coolant circulation is maintained thru the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification. The minimum required flow rate of 2000 gpm ensures decay heat removal, minimizes the probability of losing an RHR pump by air-entrainment from pump vortexing, and minimizes the potential for valve damage due to cavitation or chatter. Losing an RHR pump is a particular concern during reduced RCS inventory operation. The 2000 gpm value is limited by the potential for cavitation in the control valve and chattering in the 10-inch check valve.

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

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

5.6.1.1 The spent fuel storage racks are designed for fuel enriched to 5 weight percent U-235 and shall be maintained with:

a. A K_{eff} equivalent to less than 0.95 when flooded with unborated water.*

b. A nominal 8.972 inch center-to-center distance between fuel assemblies placed in the storage racks.

c. A three region arrangement in the spent fuel storage pool with the following definitions:

   1. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.95 ± 0.05 wt% U-235, or spent fuel regardless of the fuel burnup.

   2. Region 2 is designed to accommodate fuel of 4.95% initial enrichment burned to at least 50 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 in MWD/KgU is given by Y; when $Y = -23.761 + 22.075E - 2.0165E^2 + 0.1152E^3$, where E is the initial enrichment in the axial zone of highest enrichment.

   3. Region 3 is designed to accommodate fuel of 4.95% 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 $Y = -25.7425 + 18.76E - 1.3933E^2 + 0.0666E^3$, where E is the initial enrichment in the axial zone of highest enrichment.

   An empty cell is less reactive than any cell containing fuel and therefore may be used as a Region 1, Region 2, or Region 3 cell in any arrangement.

d. The following arrangement of regions apply in the spent fuel storage pool:

   1. Region 1 fuel assemblies located along the periphery of the storage modules adjacent to the pool walls must be isolated from

*For some accident conditions, the presence of dissolved boron in the pool water may be taken into account by applying the double contingency principle which requires two unlikely, independent, concurrent events to produce a criticality accident.

5.6 FUEL STORAGE

each other and from the inner Region 3 cells by at least one (1) Region 2 spent fuel assembly (i.e., fuel of 50 MWD/KgU burnup or equivalent).

2. Region 1 fuel assemblies located along the wide water-gaps** between storage modules must be isolated from each other and from the inner Region 3 cells by at least one (1) Region 2 spent fuel assembly (i.e., fuel of 50 MWD/KgU assembly average burnup or equivalent).

3. Region 1 fuel assemblies located along the narrow water-gaps** between storage modules must be isolated from each other by at least two (2) Region 2 spent fuel assemblies and from the inner Region 3 cells by at least one (1) Region 2 spent fuel assembly (i.e., fuel of 50 MWD/KgU assembly average burnup or equivalent).

4. A checkerboard pattern of fresh fuel and empty cells may be used throughout any storage module, or internal to any storage module in lieu of Region 3 fuel as shown in Figure 5.6-2.

Figure 5.6-1 shows a typical arrangement of regions. Figure 5.6-2 illustrates internal module checkerboarding of fresh fuel with empty cells in a portion of the fuel pool. Figure 5.6-3 illustrates the two burnup-enrichment equations (5.6.1.1.c.2 and 5.6.1.1.c.3) in graphical form.

e. Only spent fuel meeting the Region 3 burnup requirements shall be stored in any module in the cask loading area of the cask pit.

CRITICALITY - NEW FUEL

5.6.1.2 The new fuel pit storage racks are designed and shall be maintained with a nominal 21.0 inch center-to-center distance between new fuel assemblies such that k_{eff} will not exceed 0.98 when fuel having an enrichment of 4.5 weight percent U-235 is in place and optimum achievable moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 722 ft.

**The nominal gap (2-1/8 inches) running in the E-W direction between the adjacent modules is referred to as the "wide gap." The N-S direction gap (1.5 inch) is referred to as the "narrow gap."

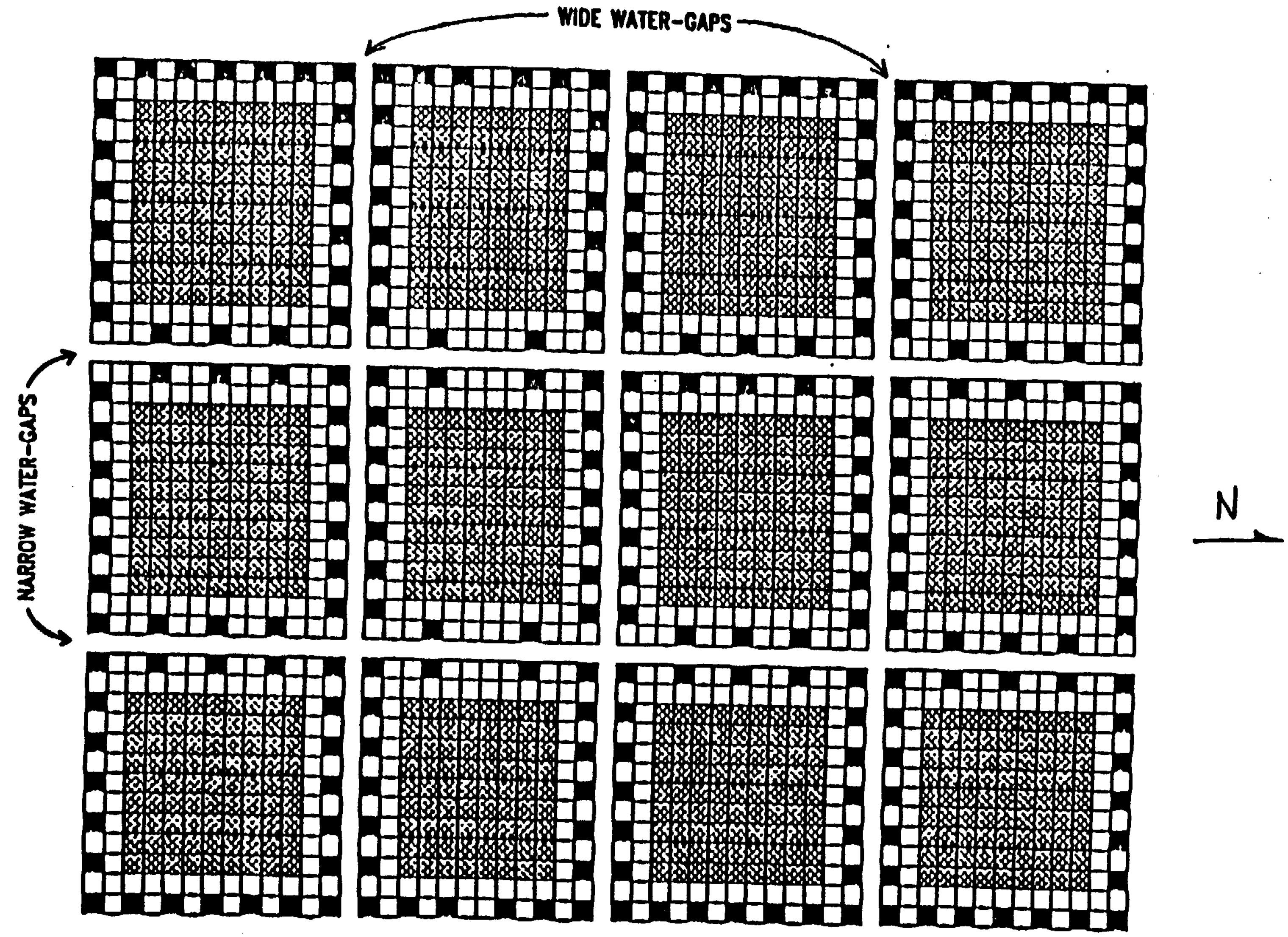
5.6 FUEL STORAGE

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2091 fuel assemblies. In addition, no more than 225 fuel assemblies will be stored in a rack module in the cask loading area of the cask pit.

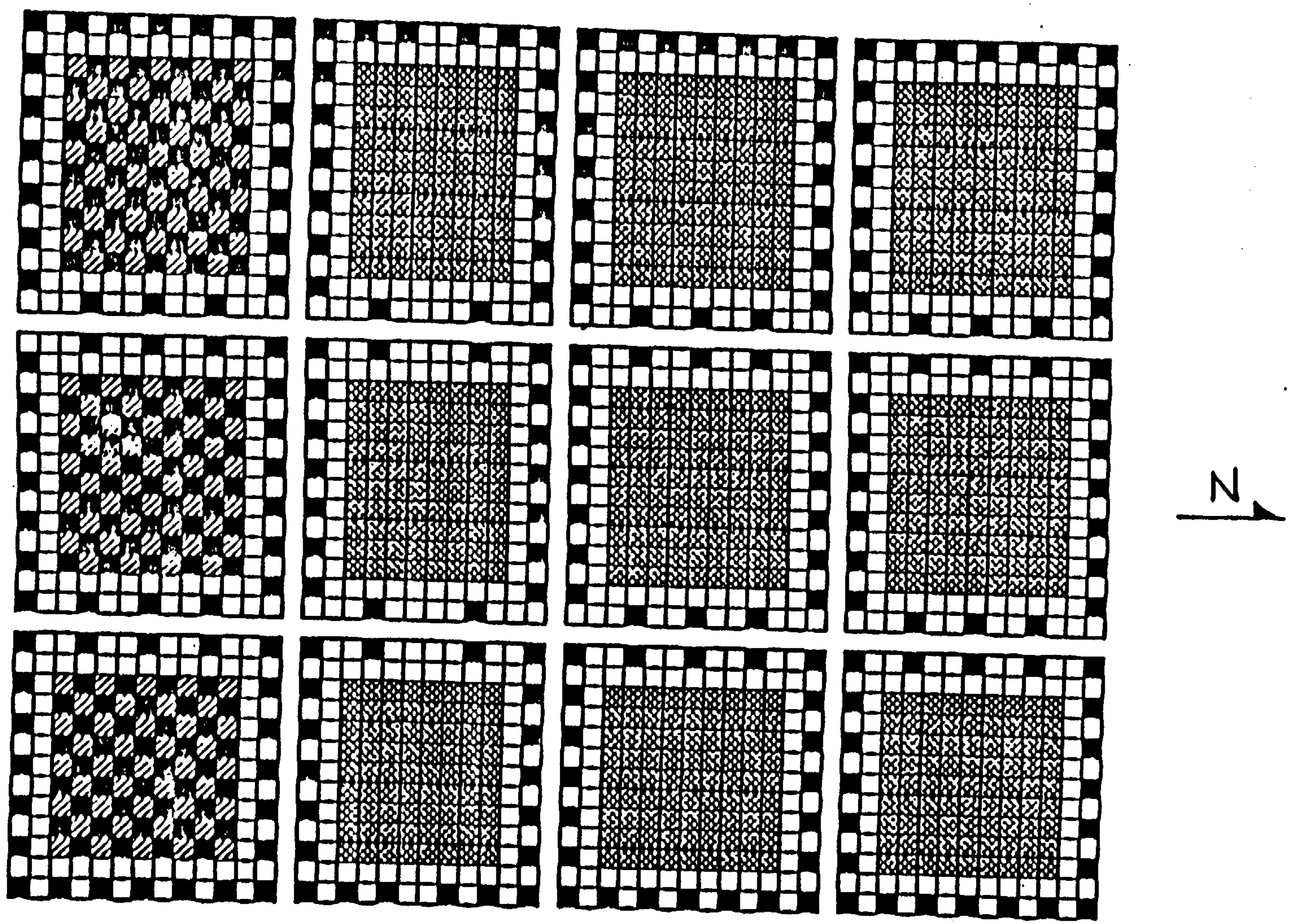
5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

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



■ 250 Region 1 cells □ 826 Region 2 cells ▩ 1015 Region 3 cells

FIGURE 5.6-1 REFERENCE SPENT FUEL POOL LAYOUT



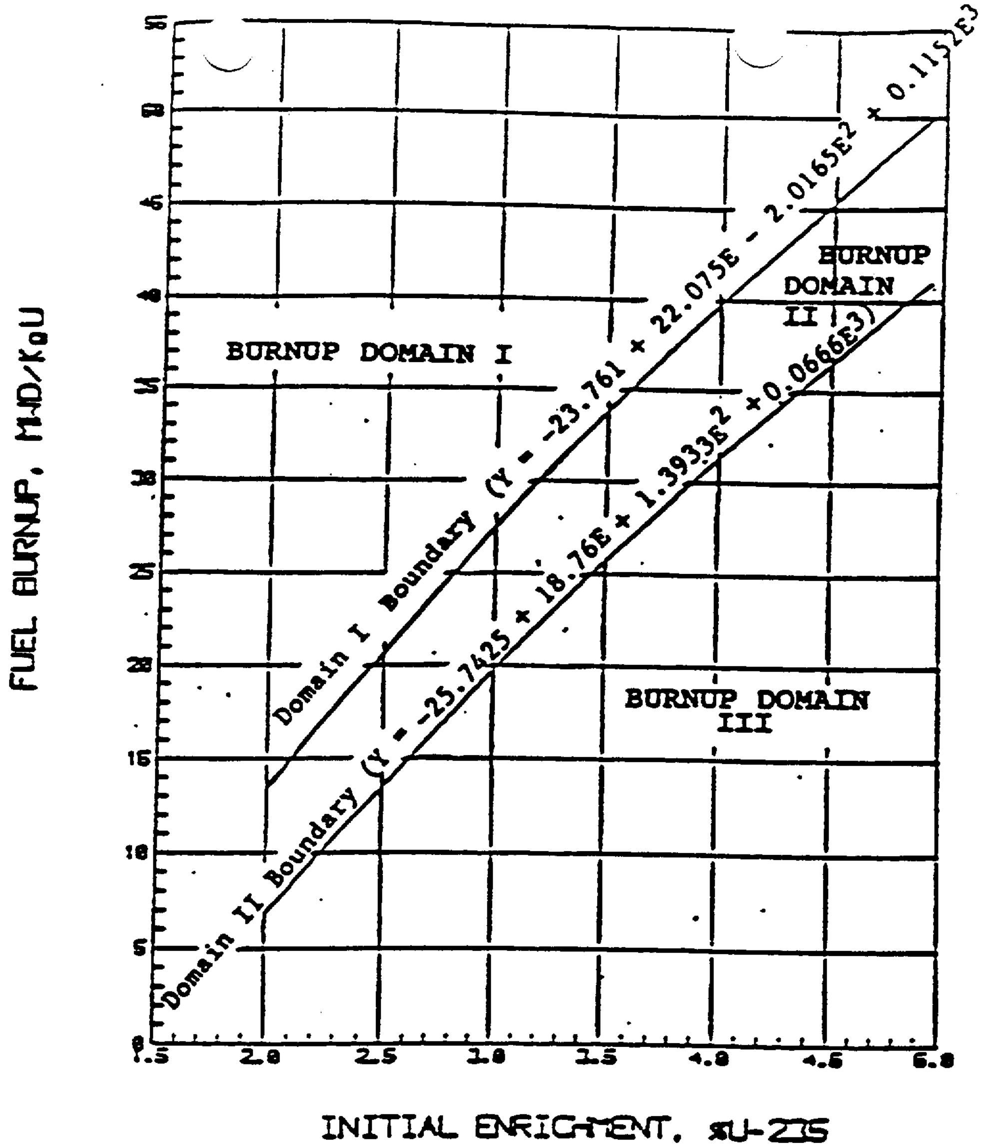
■ 250 + 130 Region 1 cells

▨ 131 Empty Cells

□ 826 Region 2 cells

▩ 754 Region 3 cells

FIGURE 5.6-2 EXAMPLE OF CHECKERBOARD SPENT FUEL STORAGE ARRAY



Domain I: Fuel assemblies with initial enrichment-burnup combinations in Domain I may be placed in either Region 1, 2, or 3 storage cells.

Domain II: Fuel assemblies with initial enrichment-burnup combinations in Domain II shall be placed only in Region 1 or 3 storage cells.

Domain III: Fuel assemblies with initial enrichment-burnup combinations in Domain III shall be placed only in Region 1 storage cells

FIGURE 5.6-3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ENCLOSURE 3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 167 TO FACILITY OPERATING LICENSE NO. DPR-77

AND AMENDMENT NO. 157 TO FACILITY OPERATING LICENSE NO. DPR-79

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

1.0 INTRODUCTION

By letter dated March 27, 1992, and supplemented by letters dated May 11, May 28, September 8 and October 8, 1992; February 18 and April 1, 1993, the Tennessee Valley Authority (TVA or the licensee) requested an amendment to change the Technical Specifications (TS) for the Sequoyah Nuclear Plant, Units 1 and 2. These changes would reflect expansion of the spent fuel pool (SFP) storage capacity by installation of new storage racks. The new racks would increase the total spent fuel storage capacity to 2316 fuel bundles and extend the projected storage capacity for spent fuel into the year 2005 or 2006. In its submittal, TVA provided a licensing report for the proposed SFP rerack for staff review that was prepared by Holtec International. The May 11, May 28, September 8, October 8, 1992, February 18, 1993, and April 1, 1993, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The present Sequoyah SFP storage racks have a total storage capacity of 1386 cells. Since the full core has 193 fuel assemblies for both Units, maintaining full core off load capability from one reactor implies that 1193 storage cells (1386 minus 193) be available for normal off load storage. Consideration of previous and future fuel assembly discharges indicates that Sequoyah will lose full core discharge capability (for one reactor) in 1996. Therefore, to preclude this situation, and to ensure that sufficient spent fuel storage capacity continues to exist, TVA plans to install new high density spent fuel storage racks whose design incorporates Boral as a neutron absorber in the cell walls, thereby allowing for more dense storage of spent fuel.

Each storage cell is composed of single Boral absorber panels positioned between two 8.75-inch I.D., 0.060-inch thick stainless steel boxes. The fuel assemblies are located in the center of each storage cell on a nominal lattice spacing of 8.97 inches. The Boral absorber has a nominal thickness of 0.102 inches and a nominal boron-10 areal density of 0.0324 gm/sq-cm. The new racks would provide a storage capacity of 2091 fuel assemblies.

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The SFP high density packing arrangement consists of 12 racks: 6 racks of a 13-by-14 array of cells, 2 racks of a 12-by-14 array of cells, 3 racks of a 13-by-13 array of cells, and one rack of a 12-by-13 array of cells. The racks are constructed from American Society for Testing and Materials (ASTM) A240-Type 304L stainless steel with adjustable support spindles made from A564-Type 630 precipitation-hardened stainless steel. Holes in the bottom and the base plates of the cells will allow for natural convection of SFP water to cool the fuel bundles. The neutron-absorbing material is Boral with a B_{10} loading of 0.030 gm per sq cm. Boral consists of finely divided particles of boron carbide uniformly distributed in Type 1100 aluminum alloy powder, clad in 1100 aluminum alloy, which is pressed, sintered, and hot rolled into 0.102-inch thick sheets. The Boral sheets are sheared into strips that are 7.5 inches wide and 144 inches long. The Boral strips are sandwiched between the cells, which are welded together around the perimeter of the strip. The Boral strips surrounding the outside perimeter of the racks are contained within closed pockets of stainless steel, which are spot welded to the outer cells. The Boral strips are exposed to pool water and vented. The openings allow gases, produced from radiolysis and water-aluminum chemical reactions, to escape.

In the submittal, TVA also proposed installation of a fuel rack storage module of similar design as the SFP modules to be located in the cask loading area of the cask pit, consisting of a 15-by-15 square array of cells that will hold 225 fuel bundles. This cask pit is located adjacent to the SFP, and will be covered with a removable impact shield during the reracking of the SFP to protect the fuel bundles stored in it from falling objects. Addition of this rack will increase the total spent fuel storage capacity to 2316 fuel bundles and extend the projected storage capacity into the year 2005 or 2006. TVA will schedule this activity after reracking the SFP.

The licensee proposed a surveillance program to monitor the performance of the Boral in the SFP and cask pit. To conduct this program, the licensee will mount 12 test coupons to simulate the inservice geometry, physical mounting, materials, and water flow conditions, that the Boral will be exposed to in the storage racks. The coupons will be exposed to slightly higher radiation doses than will the Boral in the racks.

2.0 EVALUATION

2.1 Reactivity Analysis

Three separate storage regions are provided in the SFP with independent criteria defining the highest potential reactivity in each of the three regions. Region 1 is designed to accommodate new (fresh) fuel with a maximum enrichment of 4.95 (nominal) + 0.05 weight percent (w/o) U-235 of spent fuel regardless of its discharge burnup. This configuration is intended primarily to facilitate a full core offload from one Unit, if needed, plus a normal discharge.

Region 2 is designed to accommodate fuel of maximum nominal initial enrichment up to 4.95 w/o U-235 which has accumulated minimum irradiation levels to at least 50 MWD/KgU (assembly average), or fuel of other enrichments with a burnup yielding an equivalent reactivity.

Region 3 is designed to accommodate fuel of 4.95 w/o nominal initial enrichment burned to at least 41 MWD/KgU (assembly average), or fuel of other enrichments with a burnup yielding an equivalent reactivity. These acceptable burnup domains are depicted in a graph supplied by the licensee in their submittal as Figure 4.2.3.

In the configurations analyzed, the fresh fuel cells were located alternately along the periphery of the storage rack (where neutron leakage reduces reactivity) or along the boundary between two storage modules (where the water gap provides a flux-trap which reduces reactivity). Those located along the outer periphery were isolated from each other and from the inner Region 3 cells by at least one Region 2 spent fuel assembly (i.e., fuel of 50 MWD/KgU burnup or equivalent). The fresh fuel assemblies located along the wide water-gap between storage modules were isolated from each other and from the inner Region 3 cells by at least one Region 2 spent fuel assembly (i.e., fuel of 50 MWD/KgU burnup or equivalent). Fresh fuel assemblies located along the narrow water-gap between storage modules were isolated from each other by at least two Region 2 spent fuel assemblies and from the inner Region 3 cells by at least one Region 2 spent fuel assembly (i.e., fuel of 50 MWD/KgU burnup or equivalent). In addition, a checkerboard loading pattern of fresh fuel intermixed with empty cells in any storage module was analyzed.

The analysis of the reactivity effects of fuel storage in Regions 1, 2 and 3 was performed with the KENO-5a Monte Carlo computer code using the 27-group SCALE neutron cross-section library. Since the KENO-5a code package does not have burnup capability, depletion analyses and the determination of small reactivity increments due to manufacturing tolerances were made with the two-dimensional transport theory code, CASMO-3. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. The staff finds the analysis methods used acceptable.

The criticality analyses were performed with several assumptions which tend to maximize the rack reactivity. These include:

(1) Moderator is assumed to be unborated water of the maximum density (1.0 gm/cc) at a temperature of 4°C.

(2) The effective multiplication factor (keff) of an infinite radial array of fuel assemblies was used except for the boundary storage cells where leakage is inherent.

(3) Neutron absorption in minor structural members is neglected.

The staff finds that appropriate conservative assumptions were made.

For the nominal storage cell design, uncertainties due to boron loading tolerances, boral width tolerances, tolerances in cell lattice spacing, stainless steel thickness tolerances, fuel enrichment and density tolerances, and eccentric fuel positioning and minimum water-gap tolerance were accounted for. These uncertainties were appropriately determined at least at the

95 percent probability, 95 percent confidence (95/95 probability/confidence) level. In addition, a calculational bias and uncertainty were determined from benchmark calculations, as well as an allowance for uncertainty in depletion calculations and the effect of the axial distribution in burnup. The final maximum calculated reactivity resulted in a keff of 0.943 when combined with all known uncertainties. This meets the NRC criterion of keff no greater than 0.95 including all uncertainties at the 95/95 probability/confidence level and is, therefore, acceptable. For fresh fuel checkerboarded with empty cells, the resulting 95/95 keff was 0.938, which also meets the NRC criterion.

The calculated maximum reactivities include a burnup-dependent allowance for uncertainty in depletion calculations and, as mentioned above, provide an additional margin below the limiting keff criterion of no greater than 0.95. Although not included in the criticality analyses, subsequent decay of Pu-241 with long-term storage results in a significant decrease in reactivity. This will provide an increasing subcriticality margin and further compensate for any uncertainty in the depletion calculations. In their submittal, the licensee supplied a graph (Figure 4.2.3) that defines the acceptable burnup domains for spent fuel and illustrates the limiting burnup for fuel of various initial enrichments for both Region 2 and Region 3, both of which assume that the fresh fuel (Region 1) is enriched to a nominal 4.95 w/o U-235. This reactivity equivalencing method is the standard one used for storage rack reactivity evaluations and is acceptable.

Most abnormal storage conditions will not result in an increase in the keff of the racks. However, it is possible to postulate events, such as the inadvertent misloading of an assembly with a burnup and enrichment combination outside of the acceptable area in Figure 4.2.3, which could lead to an increase in reactivity. However, for such events, credit may be taken for the presence of at least 2000 ppm (parts per million) of boron in the pool water required by TS, since the NRC does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (Double Contingency Principle). The reduction in keff caused by the boron more than offsets the reactivity addition caused by credible accidents. In fact, the licensee has confirmed that a minimum boron concentration of only 690 ppm boron would be adequate to assure that the limiting keff of 0.95 is not exceeded.

It is not physically possible to install a fuel assembly outside and adjacent to a storage module in the SFP. If it were possible, this event could also lead to a reactivity increase. However, for the storage module to be installed in the cask loading area of the cask pit, there would be sufficient room for such an extraneous assembly. However, the proposed TS 5.6.1.1.e allows this module to only contain spent fuel meeting the Region 3 burnup requirements. For a postulated drop of an assembly adjacent to this module, calculations have shown that the maximum keff remains well below the 0.95 limit, even in the absence of soluble boron. In addition, the proposed TS Surveillance Requirement 4.9.1.5 requires the boron concentration in the cask loading area of the cask pit be verified to be at least 2000 ppm once every 72 hours during fuel movement in that area.

2.2 Materials/Chemical Engineering Analysis

The austenitic stainless steel used in fabricating the modules is compatible with the borated water and the radiation environment of the SFP. The borated water is controlled to stay within an acidity range of 4.5 to 6.5 ph. Oxygen dissolved in the water will help to passivate the stainless steel and aluminum (surface of the Boral). In this environment, austenitic stainless steel will exhibit extremely low rates of corrosion for even the thinnest stainless steel elements of the pool liner or racks. Galvanic attack between the stainless steel in the pool liner or racks and Zircaloy in the fuel assemblies or Boral will not be significant since these materials are protected by passive oxide films. Concentration of chloride is maintained below the limit at which significant initiation of stress corrosion cracking can occur.

Boral has been tested extensively to study the effects of gamma irradiation in various environments and to verify its structural integrity and suitability as a neutron-absorbing material. It has been qualified for 1.0 E 11 rads of gamma radiation while maintaining its neutron attenuation capability. Tests have shown that Boral does not possess leachable halogens or elemental boron that could be released into the pool environment in the presence of radiation. Corrosion of Boral in water with acidity in the range of 4.5 to 7.0 ph is insignificant.

Surveillance coupons containing Boral will reveal behavior of Boral in the SFP over time. The staff reviewed the description of the proposed surveillance program for monitoring the Boral in the SFP and concludes that the program can reveal deterioration that might lead to a loss of the neutron absorbing capability during the life of the spent fuel racks. Any such deterioration would occur gradually. In the unlikely event of Boral deteriorating in the pool, the monitoring program will enable the licensee to detect such deterioration and allow the licensee time to take suitable corrective actions.

2.3 Radiation Protection

2.3.1 Occupational Exposure Controls

In the licensee's March 27, 1992 license amendment request, TVA estimated that the total collective dose for planned reracking activities would be between 6 and 12 person-rem. The licensee stated that all work related to the SFP rerack will be performed in accordance with TVA approved written procedures.

As part of the ALARA planning, the licensee plans to make use of remote handling tools as much as possible. The tooling and handling equipment designed for the reracking work has been used in previous rerack projects, and has proven its effectiveness in helping the licensee to meet as low as reasonably achievable radiation exposure goals. To minimize possible contamination (e.g., from hot particles) to personnel and plant facilities from the existing SFP racks after removal, high pressure water decontamination of these racks will be conducted underwater in accordance with approved procedures. The entire operation will be covered by the existing radiation protection program under the direction of the Radiological Engineer. All pool-side and in-pool work activities will be surveyed by Radiological

Controls (RC) personnel. RC personnel will also have the authority to stop work in the event of any unsafe or questionable operations. The staff finds these procedures acceptable.

Past operational experience involving rerack operations at other facilities has shown that there is a negligible increase in airborne radioactivity in the SFP area. This, coupled with the licensee's experience involving fuel movements during refueling outages, indicates that neither the current health physics program, nor area monitoring systems, need significant modification.

All plant personnel involved in this work will be covered by applicable Radiation Work Permits. Also, the appropriate protective clothing, respiratory protective and air sampling equipment, will be available as needed; as will be personnel radiation monitoring equipment such as thermoluminescence dosimeters, pocket dosimeters, and extremity film badges.

The major work effort to be expended for the rerack work will be the removal of the old racks. This is expected to require 800 person-hours, including diving operations. Divers would be used when necessary for the efficient removal of certain underwater appurtenances. Detailed procedures and radiological controls will be implemented to ensure minimum cumulative radiation dose to the diver.

The total collective dose projected for the entire SFP rerack modification is estimated to be between 6 to 12 person-rem. This estimate is consistent with the historical range of collective doses for SFP reracking operations. The staff has found that such estimates are generally conservative.

Based on our review of the Sequoyah proposal, we conclude that the projected activities and estimated collective doses for this project appear achievable and that the licensee will be able to maintain individual occupational radiation exposures within the limits of 10 CFR Part 20 and maintain doses as low as reasonably achievable. Therefore, the proposed radiation protection aspects of the SFP rerack are acceptable.

2.3.2 Design Basis Accident (DBA) Analysis

The licensee's analysis also was performed for a fuel enrichment to approximately 5 w/o U-235, allowing fuel burnup up to 60,000 megawatt days per metric ton (MWD/MT). The licensee has evaluated the effect of the change on the calculated consequences of a spectrum of postulated design basis accidents and concludes that the effect of the proposed TS change is small, and that the calculated consequences are within regulatory requirements and staff guideline dose values.

The staff has reviewed this request, and has audited the licensee's dose estimates. The staff has also reviewed its contractor's report "Assessment of the Use of Extended Burnup Fuel in Light Water Reactors", NUREG/CR-5009, Pacific Northwest Laboratories, February 1988. The staff agrees with the conclusion that the effect of increasing fuel burnup to as much as 60,000 MWD/MT would be to increase thyroid doses for a postulated fuel handling design basis accident by about twenty percent. The licensee's reanalysis of the fuel handling accident reflects this small increase. The licensee also referenced NUREG/CR-5009 in their submittal of March 27, 1992, and concluded that the DBAs previously analyzed in their Final Safety Analysis (FSAR) bounded any potential radiological consequences of a DBA associated with the extended fuel burnup. Table 1 presents the fuel handling accident thyroid and whole body doses, presented in the operating license SER, dated March 1979.

TABLE 1
Radiological Consequences of Fuel
Handling Design Basis Accident (rem)

	Exclusion Area		Low Population Zone	
	Thyroid	Whole Body	Thyroid	Whole Body
Original Estimates (SER-1979 Table 15.1)	20	1	<1	<1
Estimates for Higher Fuel Burnup*	24	1	<1	<1
Regulatory Requirement (10 CFR Part 100)	300	25	300	25

* Factor of 1.2 greater than original estimate for iodine.

2.4 Plant Systems Analysis

2.4.1 Control of Heavy Loads

The Sequoyah SFP currently contains 24 medium-density rack modules with a total of 1386 storage cells. During the period from September 1993 to October 1994, which encompasses the time period proposed for the reracking operation, approximately 897 out of the 1386 cell locations will be occupied with spent fuel. The licensee determined that, by temporarily storing a number of fuel assemblies in the cask loading pit, a sufficient number of unoccupied cells will be present in the pool to permit relocation of all fuel, such that the existing rack modules can be emptied and removed from the pool, and new rack modules installed in a programmed manner. The new rack modules will not be anchored to the pool floor.

In the licensing report prepared for the licensee by Holtec International, the licensee specifically committed to employ in the reracking process, a remotely engagable lifting device designed to meet the criteria of Section 5.1.1 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," July 1980. The licensee also stated that operator training, crane inspection, safe load path development, and procedure development for the reracking operation, will comply with the criteria of Section 5.1.1 of NUREG-0612.

The licensee has developed a preliminary fuel movement strategy so that spent fuel elements are concentrated in specific locations in the pool, and a minimum of 3 feet of lateral clearance is maintained between any heavy load and racks containing spent fuel. In a supplementary submittal, the licensee concluded that the spent fuel will have decayed sufficiently at the time of heavy load movement during the reracking so that an accidental heavy load drop would result in off site doses less than one-quarter of 10 CFR Part 100 limits. The licensee also determined that a postulated accidental heavy load drop will not result in a fuel configuration with an effective multiplication factor (Keff) greater than 0.95, or affect the ability to adequately cool and shield the spent fuel in the pool due to leakage from the SFP in excess of makeup capacity. These determinations indicate that, although heavy loads will be moved above the SFP floor and adjacent to racks containing spent fuel, the reracking process will comply with the recommended guidelines of Section 5.1 of NUREG-0612 with regard to control of heavy loads in PWR SFP areas.

The licensee has committed to administratively impose restrictions on the handling of heavy loads. Racks undergoing transfer in or out of the pool will be empty. Certain peripheral rack modules are planned to be partially loaded with fuel assemblies approximately 20 inches from their final location. These rack modules will then be lifted approximately 4 inches above the pool floor and moved to their final locations. Movement of the impact shield will be controlled such that its position relative to the cask pit walls will prevent the shield from falling into the cask pit.

2.4.2 SFP Thermal-Hydraulics

The licensing report states that the decay heat load calculation for the SFP was performed in accordance with the provisions of Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling," Rev. 2, July 1981. In order to evaluate the total decay heat load, an inventory of 1773 fuel assemblies (accumulated through scheduled discharges from September 1982 to April 2002) was assumed to be present in the SFP. In addition to the heat load from this inventory, heat loads resulting from the following spent fuel off-load scenarios were added:

1A. The entire core (193 fuel assemblies) from one unit was assumed to be transferred to the SFP after 288 hours of decay in the reactor vessel. The fuel transfer time was assumed to be 36 hours. Thirty days later, 113 of these fuel assemblies were reloaded into the reactor vessel over a period of 21.1 hours. The total duration of the outage was assumed to be 60 days. Two SFP cooling trains are assumed to be operating. This scenario is representative of normal refueling operations at Sequoyah.

1B. Identical to scenario 1A, except only one SFP cooling train is assumed to be operating.

2. Eighteen days following the outage described for scenario 1A, the other unit performs an identical outage.

3. Sixty days following the outage described in scenario 2, the first unit was assumed to have an unplanned shutdown. The entire core was assumed to be transferred to the SFP in a 36-hour period following 288 hours of decay in the reactor vessel.

A period of approximately 3.5 years of full power operation was assumed for all fuel assemblies. Except where noted otherwise, two trains of SFP cooling were assumed to be in operation for the above scenarios. The cumulative SFP inventory was assumed to be 2126 fuel assemblies following scenario 3.

A transient analysis was performed to evaluate bulk pool temperature for each of these scenarios. Convective heat transfer and evaporative cooling from the pool surface, and heat removal through operating SFP heat exchangers, were credited in the analysis. The heat removal rate through operating SFP heat exchangers was calculated based on a temperature effectiveness factor obtained by rating the heat exchanger on a proprietary thermal-hydraulic computer code. In obtaining the temperature effectiveness value, the heat exchanger was assumed to be fouled to the design maximum extent, and 5 percent of the tubes were assumed to be plugged.

The most limiting scenario with regard to bulk pool temperature was found to be scenario 1B. The calculated maximum bulk pool temperature for this scenario was determined to be 174.9°F, 336 hours following reactor shutdown. Scenario 1A, identical to scenario 1B except that both SFP cooling trains are in operation, was found to result in a maximum bulk pool temperature of 138.0°F. In response to staff questions with regard to decay time, the licensee reevaluated these scenarios 1A and 1B, assuming fuel movement is begun immediately following the TS required minimum of 100 hours decay in the reactor vessel. This reevaluation produced estimated maximum bulk SFP temperatures of 177.2°F and 139.5°F for scenarios 1B and 1A, respectively. Calculated maximum bulk pool temperatures for scenarios 2 and 3 were found to be 142.6°F and 146.7°F, respectively.

The licensee report also evaluated the transient response of the SFP following a loss of all forced cooling. The loss of cooling was assumed to occur coincident with the maximum bulk temperature reached for each scenario evaluated. The response was evaluated assuming no makeup water addition. Of the scenarios evaluated, scenario 1 (which assumed a single SFP cooling train was in operation) was found to be most limiting due to the higher temperature at the time cooling is lost. For this scenario, bulk boiling conditions were determined to exist in the SFP 3.4 hours following the loss of forced cooling.

In order to verify that no void formation occurs and cladding integrity is not threatened, a model was developed to calculate the maximum local water and cladding temperature. The model was used to determine the location of minimum flow in an idealized, axially symmetric arrangement of fuel assemblies. The calculation assumed that the fuel assembly located in the minimum flow region is the most thermally limited. As an additional conservatism, the fuel assembly cladding was assumed to have a crud deposit which covered the entire surface. For both unblocked and 50 percent blocked flow conditions, the calculation indicated no incidence of nucleate boiling and no potential for fuel cladding damage.

The staff evaluated the heat load imposed on the SFP cooling system by the following two fuel offload scenarios: (1) a single active failure assumed coincident with a SFP inventory consisting of one standard refueling offload of 80 fuel assemblies after 150 hours decay, one standard refueling offload after one year decay, and the postulated existing inventory of 2017 fuel assemblies at time of the 27th discharge; and (2) a full-core offload of 193 fuel assemblies after 150 hours decay, one standard refueling offload after 36 days decay, and the postulated existing inventory of 2017 fuel assemblies at the time of the 27th discharge. It was further assumed that there were no equipment failures. These scenarios correspond to the maximum projected fuel loading at the time a full core offload reserve capacity is lost.

The staff calculated heat loads for the fuel inventories described above utilizing the methodology of Branch Technical Position ASB 9-2. The staff assumed a period of approximately 3.5 years of full power operation for all stored fuel. Since the staff performed a steady-state calculation, the total heat load was assumed to be constant.

Using these heat load values, the heat exchanger temperature effectiveness factor calculated by the licensee, and design data from the Sequoyah FSAR, the staff then calculated SFP steady-state temperatures assuming heat removal only through the SFP heat exchanger. Calculated maximum steady-state temperatures for the normal discharge with single failure and the full-core offload scenarios were determined to be 141.0°F and 139.9°F respectively.

The staff's acceptance criterion for SFP temperature under maximum normal heat load conditions coincident with a single failure of the SFP cooling system is based on preventing thermal damage to the SFP and SFP support system components. The FSAR specifies a minimum design temperature of 200°F for SFP support system components. The licensing report included a structural analysis of the SFP. This analysis concluded that adequate safety margins exist to protect the SFP structure from damage due to the thermal stresses induced by the peak calculated bulk SFP temperature or bulk SFP boiling. The calculated maximum temperatures for the staff's normal discharge scenario, and licensee scenarios 1A, 1B and 2 are such that no damage to the SFP or its support system components would be expected.

The staff's acceptance criterion for SFP temperature under maximum abnormal heat loads was also found to be satisfied. The acceptance criterion is based on preventing bulk boiling in the SFP. This acceptance criterion applies to the full-core offload scenario evaluated by the staff. The calculated maximum SFP temperatures for the staff's full-core offload scenario and licensee scenario 3 indicate that bulk boiling would not be expected to occur in the SFP under the resultant abnormal heat loads.

For the scenarios evaluated by the licensee, a minimum time of 3.4 hours to reach bulk boiling conditions in the SFP following a loss of all forced cooling was calculated. Based on the availability of a number of alternate sources of makeup water, the staff concludes that adequate time is available to provide makeup water to the SFP prior to the onset of bulk boiling and a subsequent loss of coolant inventory.

2.5 Structural Integrity Analysis

This evaluation addresses the adequacy of the structural aspects on the use of high density spent fuel racks. The primary areas of review associated with the proposed application are focused toward assuring the structural integrity of the fuel cells, rack modules, and the spent fuel pool floor and walls under the postulated loads (Appendix D of SRP 3.8.4) and fuel handling accidents.

2.5.1 Spent Fuel Storage Pool

The spent fuel pool is a reinforced concrete structure and is designed as a Seismic Category I structure. A single pool serves both units. Pool slab is approximately 21 inches in thickness with inside dimension of 53 feet by 32 feet. The pool is supported on rock. Wetted inside surfaces of the pool are lined with stainless steel to ensure water tight integrity.

The loading on the pool structure consists of static, dynamic and thermal loads. Static loads include weight of the pool structure, water in the pool, weight of the fuel assemblies and rack modules. For the dynamic loads, both safe shutdown earthquakes (SSEs) and operating bases earthquakes (OBEs) are considered. In addition, stresses generated from a thermal gradient across the thick concrete walls and slab due to temperature differential between the pool water and the atmosphere external to the slab and walls are investigated. The effects of the rack motion on the pool steel liner due to an earthquake is also incorporated in the analysis.

The spent fuel pool was analyzed using the finite element method with working stress approach. The results of the load components were combined in accordance with the FSAR commitment.

Structural concrete capacities in terms of moment and shear were evaluated in accordance with the Building Code Requirements of the American Concrete Institute, ACI 318, referred to in the FSAR. Concrete strength capacities were then compared with anticipated loads on the pool structure discussed above. The staff reviewed a table that summarized the results obtained from the finite element analyses and the safety margins against allowable on each section of the structure.

The increase of the number of fuel assemblies from the existing fuel did not alter the safety margin appreciably because they only contributed to a small part of the total dead weight (approximately 6 percent) since the massive concrete structure and water in the pool are the major dead weight contributors.

The staff, therefore, concludes that the spent fuel pool would continue to support the loads under normal, severe environmental conditions, as well as accident conditions, and maintain its integrity.

2.5.2 Refueling Accidents

The licensee has investigated the consequences of dropping a new or spent fuel assembly as it is being moved over the stored fuel. The case considered is when a fuel assembly is dropped from an elevation of 36 inches above the top of the rack and impacts the base of the module. Next, a fuel assembly is dropped from an elevation of 36 inches above the rack and hitting the top of the rack. The height of the fuel assembly drop is limited to 36 inches by the constraint of the fuel transfer equipment. A drop of a fuel assembly directly on the pool liner is unlikely since prior to placing the fuel assemblies into the rack, all the high density racks are to be placed in the pool filling the pool floor.

In the case of fuel assembly drop on a rack module base plate, the licensee determined that there will be no change in the spacing between cells. Local deformation of the baseplate in the neighborhood of the impact will occur, but the dropped assembly will not impact the pool liner. When a fuel assembly is dropped on the top of the rack, the licensee's analysis indicated that although local deformation could occur, it would be confined to a region above the active fuel zone; thus, it does not alter the subcriticality status of the fuel assemblies. The licensee also determined that the case of dropping a mass of 3820 pounds pool gate from a height of 24 inches above the top of the rack is less severe than the two previous cases already considered. The staff considers the licensee's evaluation of the postulated refueling accidents acceptable.

2.5.3 High Density Racks

The spent fuel storage racks are Seismic Category I equipment and, therefore, are required to remain functional during and after an earthquake. The function of the rack is to maintain structural integrity and provide the minimum spacing between the adjacent fuel storage cells. The racks are free-standing and they are neither anchored to the pool floor, the pool wall, nor structurally interconnected. Each rack module is provided with leveling pads which support the rack. The fuel rack structure is a folded metal plate assemblage of thin gauge material approximately 180 inches long vertically and width and depth dimensions are typically 117 inches and 128 inches; it is welded to a baseplate and supported on four legs. In this application, other than the size of the rack, the structural design is the same regardless of whether it is freshly discharged fuel, older spent fuel or new fuel.

The rack was modeled by a system of springs and lumped masses. There are many aspects of the modeling that make rack dynamic response analyses highly nonlinear. There are gap spring elements that simulate gaps between a fuel assembly and a rack cell. Sliding elements are provided to model sliding action of the rack module with respect to the pool floor. Water in the pool is modeled by equivalent hydrodynamic masses. These elements are integrated into a computer model suitable for analyses by a code named DYNARACK which executes time history integration of the governing nonlinear differential equations of motion of a 24-degree of freedom system consisting of rack and fuel assemblies. The licensee also provided multi-rack analyses as well as evaluations for different combinations of loading patterns, such as, half empty and fully loaded cases.

Based on the above described analyses, the licensee concluded that the racks would neither impact each other nor the vertical pool walls. The stresses in the rack components (fuel cells, baseplate, pedestals and connecting welds) were found to be less than half the allowables except for the weld connecting baseplate to the pedestal, where the stress was found to be 75 percent of the allowable.

After reviewing additional information provided to assist the evaluation, the staff concluded that the verification process employed for the analysis, including the DYNARACK code, was rather limited. Specifically, there were no realistic physical tests to verify the analytical results obtained by the code. In addition, the licensee has not demonstrated that the errors from the analytical method associated with numerical integration of the governing nonlinear differential equation of motion are within an acceptable range, and that the stability of the analytical-numerical solution of the response was controlled or accounted for. The staff, for these reasons, performed additional independent assessments to augment the licensee's analyses. The results of the staff assessments are discussed below.

The staff investigated the actual safety margin against overturning of the rack. It was found that the estimated safety factor against overturning for the rack was much greater than the allowable factor of 1.1 provided in the SRP Section 3.8.5, and, therefore, is acceptable. The investigation was based on the conservation of energy principle, whereby the kinetic energy of the rack, induced by an earthquake, is equated to the potential energy that is needed to raise the rack to a position where the center of gravity of the rack moves beyond the line connecting the two supporting legs of the rack. The procedure provides an overall assessment of the stability of the free-standing racks under vibratory ground motion induced by an earthquake.

Another aspect of the rack response that the staff considered is the structural integrity of the rack itself with regard to lateral impact. The maximum impact that can be realized is when one considers an extreme bounding condition where there is no friction between the rack support and pool floor. The licensee's simulated test indicated that there was no impact between the boxes and between the box and the container wall. In the test, the boxes simulate the racks and the container simulates spent fuel pool. The container is filled with water and the boxes are placed in the container. For the test, the boxes are suspended from above, thus providing no contact between the bottom of the box and the floor of the container. Dynamic motion is applied only to the container. The test demonstrated that water between the objects provided a significant cushion, thus, preventing impact between closely spaced objects. Even if there is a certain degree of isolated impacts among the racks or between racks and pool wall in an actual case, the impact forces are not expected to be significant because of damping effect of the water. Therefore, the basic function of the rack to keep the fuel bundles upright would not be compromised. The test report was made available to the staff at the time of Indiana Michigan Power application for the D.C. Cook plant. However, the test result is generic in nature and applicable to this reracking application.

In the submittal, the licensee indicated that the spent fuel pool floor is on rock. Review of the Final Safety Analysis Report indicates that the ground maximum velocity level is below 20 inches per sec. This is far below the maximum allowable estimated to be 100 inches per sec. before tip-over of a rack.

Based on the review of the licensee submittal, a detailed examination of the licensee's dynamic response analyses and the staff's independent assessment, it is concluded that the rack modules will perform their function and maintain their structural integrity during and after an earthquake in combination with other applicable loads. However, since the racks could shift their relative locations during a seismic event, and the gap size is an important parameter in the seismic analysis, the staff requests that the licensee evaluate the spacings between the racks and the walls after an earthquake event that exceeds the postulated Operating Basis Earthquake (OBE).

3.0 CONCLUSIONS

Based on the review described above, the staff finds the criticality aspects of the proposed modifications to the Sequoyah SFP storage racks are acceptable and meet the requirements of General Design Criterion (GDC) 62 for the prevention of criticality in fuel storage and handling.

Corrosion of the proposed fuel storage racks, because of the SFP environment, should be of little significance during the life of the facility. The surveillance program proposed by the licensee would reveal any deterioration in the neutron absorbing capability of Boral. If significant degradation is found, the licensee would have sufficient time to take the appropriate corrective measures.

The staff finds that the licensee selected the appropriate materials of construction and planned the proposed Boral surveillance program to meet the requirements of 10 CFR 50, Appendix A, General Design Criterion 61 for ensuring the capability to permit appropriate periodic inspecting and testing of components and General Criterion 62 for preventing criticality by using neutron absorbers and by maintaining the structural integrity of components.

The staff concludes that the only potential increased doses resulting from a DBA with extended burnup to 60,000 MWD/T is the thyroid dose resulting from fuel handling accidents. However, these doses remain well within the 300 rem thyroid exposure guideline values set forth in 10 CFR Part 100. This small calculated increase is not significant.

The licensee has committed to use in the reracking process, cranes and associated lifting devices which conform to the criteria of Section 5.1.1 of NUREG-0612. This commitment satisfies the requirements of General Design Criterion 61 of Appendix A to 10 CFR Part 50 with regard to the design of heavy load handling systems. Therefore, the staff finds the licensee has committed to employ an acceptable heavy load handling system in the reracking process.

The licensee has also committed to develop operator training programs, crane inspection plans, safe load paths, and procedures for the reracking operation which comply with the criteria of Section 5.1.1 of NUREG-0612. In addition to imposing administrative restrictions on the handling of heavy loads near spent fuel, the licensee has also demonstrated compliance with the general guidelines of Section 5.1 of NUREG-0612. These actions are consistent with the defense-in-depth approach of NUREG-0612 and the requirements of GDCs 4, 61 and 62.

The staff evaluation of the heat load imposed on the SFP cooling system by the modification, including the thermal affects on the SFP structural material, was adequately analyzed by the licensee. This analysis concluded that adequate safety margins exist to protect the SFP structure from damage due to the thermal stresses induced by the peak calculated bulk SFP temperature or bulk SFP boiling. The staff's acceptance criterion for SFP temperature under maximum abnormal heat loads was also found to be satisfied. The acceptance criterion is based on preventing bulk boiling in the SFP. Based on this evaluation, the staff concludes that the requirements of GDC 61 are met with regard to providing adequate cooling for the postulated spent fuel inventory and assuring an adequate coolant inventory is maintained under accident conditions. Therefore, the increased fuel storage capacity of the SFP is acceptable with regard to SFP thermal-hydraulic concerns.

From a structural analysis consideration, the increase of the number of fuel assemblies from the existing number of fuel assemblies, the staff has concluded that the licensee's design of the spent fuel rerack modules and the spent fuel pool are adequate to withstand the effects of the required design basis environmental, abnormal and accident loads and able to maintain integrity of the fuel assemblies and fuel rods.

The licensee has committed to incorporate a requirement to determine the gap between the peripheral racks and the spent fuel pool walls following an OBE and perform an evaluation of any change in this gap. This satisfies the concern regarding any potential for rack movement following an OBE.

4.0 TECHNICAL SPECIFICATION CHANGES

The following TS changes have been proposed to support the proposed SFP reracking:

(1) TS 4.9.1.4 - Reference would be made to the arrangement of regions which apply in the SFP as specified in TS 5.6.1.1.d.

(2) TS 4.9.1.5 - A surveillance requirement would be added to require chemical analysis of the boron concentration in the cask loading area of the cask pit to be greater than or equal to 2000 ppm at least once per 72 hours during fuel movement in that area and until the assemblies in that storage rack are verified to comply with the criticality loading criteria specified in TS 5.6.1.1.e.

(3) TS 5.6.1.1 - The specification would be replaced with a new one which allows a change in the placement of fuel assemblies from a nominal 10.375-inch, center-to-center distance to a nominal 8.972-inch, center-to-center distance between fuel assemblies placed in the proposed new storage racks. This change also creates the previously described three-region storage arrangement in the

SFP with accompanying definitions and an explanation of where fuel assemblies would be stored according to initial fuel enrichment and burnup parameters. Figure 6.5-1 would be added to illustrate a typical arrangement of fuel regions. Figure 5.6-2 would be added to illustrate internal module checkerboarding of fresh fuel with empty cells. Figure 5.6-3 would be added to illustrate graphically the two fuel burnup versus enrichment equations.

(4) TS 5.6.3 - The specification would change the current spent fuel storage capacity from 1386 fuel assemblies to 2091 fuel assemblies and provide an additional storage capacity of no more than 225 fuel assemblies in a proposed fuel rack storage module to be located in the cask loading area of the cask pit, for a total of 2316 fuel assemblies.

Based on the results of the analysis described above that determined that the spent fuel pool expansion was acceptable, staff review of these changes to the technical specifications has determined that they are appropriate to support the modifications and are consistent with the stated criteria. Therefore, the proposed technical specification changes regarding the spent fuel storage capacity increase are acceptable.

The licensee also proposed changes to TS 3/4.9.7 requiring placement of an impact shield over fuel stored in the cask loading area, restricting the load, height and impact area of loads greater than 2100 pounds carried over the impact shield covering the cask pit, clarifying the surveillance requirement to demonstrate crane interlocks and physical stops are operable, and adding a surveillance requirement to verify that the administrative restrictions associated with the impact shield are met prior to carrying heavy loads over the cask pit area. These changes are consistent with the guidance of Regulatory Guide 1.13 and the requirements of GDCs 4, 61 and 62 with regard to protecting the stored fuel from the effects of accidental heavy load drops. Therefore, the proposed changes to TS 3/4.9.7 are acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on January 28, 1993 (58FR6423). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of the amendments will not have a significant effect on the quality of the human environment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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