Docket

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

November 1, 1988

Docket No. 50-498

Mr. J. H. Goldberg
Group Vice-President, Nuclear
Houston Lighting & Power Company
P. O. Box 1700
Houston, Texas 77001

Dear Mr. Goldberg:

SUBJECT: ISSUANCE OF AMENDMENT NO. 2 TO FACILITY OPERATING LICENSE NPF-76 - SOUTH TEXAS PROJECT, UNIT 1 (TAC NO. 65086)

The Commission has issued the enclosed Amendment No. 2 to Facility Operating License No. NPF-76 for the South Texas Project, Unit 1. The amendment consists of changes to the Technical Specifications in response to your application dated March 8, 1988 as supplemented on March 26, 1988.

The amendment changes the Appendix A Technical Specifications to reflect approval of the expansion of the spent fuel pool capacity from the current 196 fuel assemblies to 1969 fuel assemblies. The approval is granted based upon the storage of non-consolidated fuel and the installation of all new racks prior to the storage of any spent fuel in the spent fuel pool. If you change your plans and decide to store spent fuel in the pool before completing installation of the new racks, you should submit documentation to the staff for prior review addressing all significant changes from the request that the staff is now approving.

It was noted during the staff review that while the proposed surveillance program for monitoring the Boraflex in the spent fuel pool was acceptable, no corrective action was proposed in the event that Boraflex degradation was observed. It is recommended that a plan of corrective actions be developed and implemented.

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CP1 cc

A copy of the Safety Evaluation supporting the amendment is also enclosed. Notice of Issuance will be included in the Commission's next Bi-weekly Federal Register notice.

Sincerely,

/s/

George F. Dick, Jr., Project Manager
Project Directorate - IV
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 2 to NPF-76
2. Safety Evaluation

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| PD4/LA | PD4/PM | OGC-Rockville | PD4/D |
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A copy of the Safety Evaluation supporting the amendment is also enclosed. Notice of Issuance will be included in the Commission's next Bi-weekly Federal Register notice.

Sincerely,

George F. Dick Jr.

George F. Dick, Jr., Project Manager
Project Directorate - IV
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
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2. Safety Evaluation

cc w/enclosures:
See next page

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

HOUSTON LIGHTING & POWER COMPANY

CITY PUBLIC SERVICE BOARD OF SAN ANTONIO

CENTRAL POWER AND LIGHT COMPANY

CITY OF AUSTIN, TEXAS

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 2
License No. NPF-76

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

   A. The application for amendment by Houston Lighting & Power Company (HL&P) dated March 8, 1988 as supplemented on March 26, 1988, 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, as amended, 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 license 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. NPF-76 is hereby amended to read as follows:

   2. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 2 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DKWiggin, act

Jose A. Calvo, Director
Project Directorate - IV
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 1, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 2

FACILITY OPERATING LICENSE NO. NPF-76

DOCKET NO. 50-498

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

| Remove | Insert |
|---|---|
| 5-6 | 5-6 |
| 5-7 | 5-7 |
| - | 5-9 |

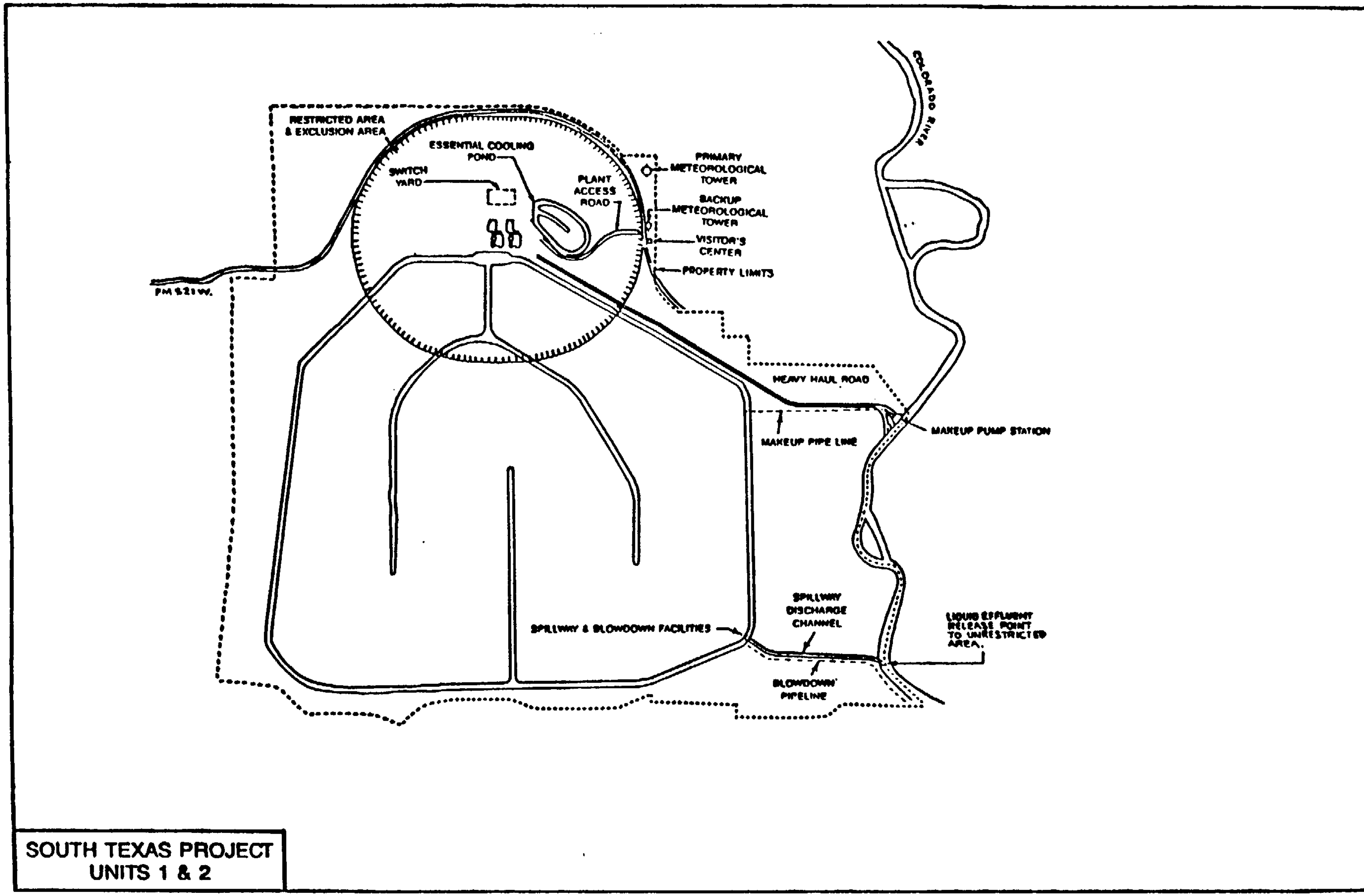


FIGURE 5.1-4

RESTRICTED AREA AND SITE BOUNDARY FOR RADIOACTIVE LIQUID EFFLUENTS

## 5.3 REACTOR CORE

### FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 168 inches. The initial core loading shall have a maximum enrichment of 3.5 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.5 weight percent U-235.

### CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 57 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 158.9 inches of absorber material. The absorber material shall be hafnium. All control rods shall be clad with stainless steel tubing.

## 5.4 REACTOR COOLANT SYSTEM

### DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

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

b. For a pressure of 2485 psig, and

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

### VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 13,814 $\pm$ 100 cubic feet at a nominal $T_{avg}$ of 561°F.

## 5.5 METEOROLOGICAL TOWER LOCATION

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

## 5.6 FUEL STORAGE

### CRITICALITY

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

a. A $k_{eff}$ equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of

0.0185 Δk for Region 1 uncertainties and tolerances and 0.0259 Δk for Region 2 uncertainties and tolerances.

b. A nominal 10.95 inches center to center distance between fuel assemblies in Region 1 of the storage racks and a nominal 9.15 inches center to center distance between fuel assemblies in Region 2 of the storage racks.

c. Neutron absorber (Boraflex) installed between spent fuel assemblies in the storage racks in Region 1 and Region 2.

d. Region 1 of the spent fuel storage racks can be used to store fuel which has U-235 enrichment less than or equal to a nominal 4.5 weight percent. Region 2 can be used to store fuel which has achieved sufficient burnup such that storage in Region 1 is not required. The initial enrichment vs. burnup requirements of Figure 5.6-1 shall be met prior to storage of fuel assemblies in Region 2.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 62 feet-6 inches.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1969 fuel assemblies.

## 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.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

| COMPONENT | CYCLIC OR TRANSIENT LIMIT | DESIGN CYCLE OR TRANSIENT |
|---|---|---|
| Reactor Coolant System | 200 heatup cycles at $\leq$ 100°F/h and 200 cooldown cycles at $\leq$ 100°F/h. | Heatup cycle - $T_{avg}$ from $\leq$ 200°F to > 550°F. Cooldown cycle - $T_{avg}$ from $\geq$ 550°F to $\leq$ 200°F. |
| | 200 pressurizer cooldown cycles at $\leq$ 200°F/h. | Pressurizer cooldown cycle temperatures from $\geq$ 650°F to $\leq$ 200°F. |
| | 80 loss of load cycles, without immediate Turbine or Reactor trip. | > 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER. |
| | 40 cycles of loss-of-offsite A.C. electrical power. | Loss-of-offsite A.C. electrical ESF Electrical System. |
| | 80 cycles of loss of flow in one reactor coolant loop. | Loss of only one reactor coolant pump. |
| | 400 Reactor trip cycles. | 100% to 0% of RATED THERMAL POWER. |
| | 10 auxiliary spray actuation cycles. | Spray water temperature differential > 621°F. |
| | 200 leak tests. | Pressurized to $\geq$ 2485 psig. |
| | 10 hydrostatic pressure tests. | Pressurized to $\geq$ 3110 psig. |
| Secondary Coolant System | 1 steam line break. | Break in a > 6-inch steam line. |
| | 10 hydrostatic pressure tests. | Pressurized to $\geq$ 1600 psig. |

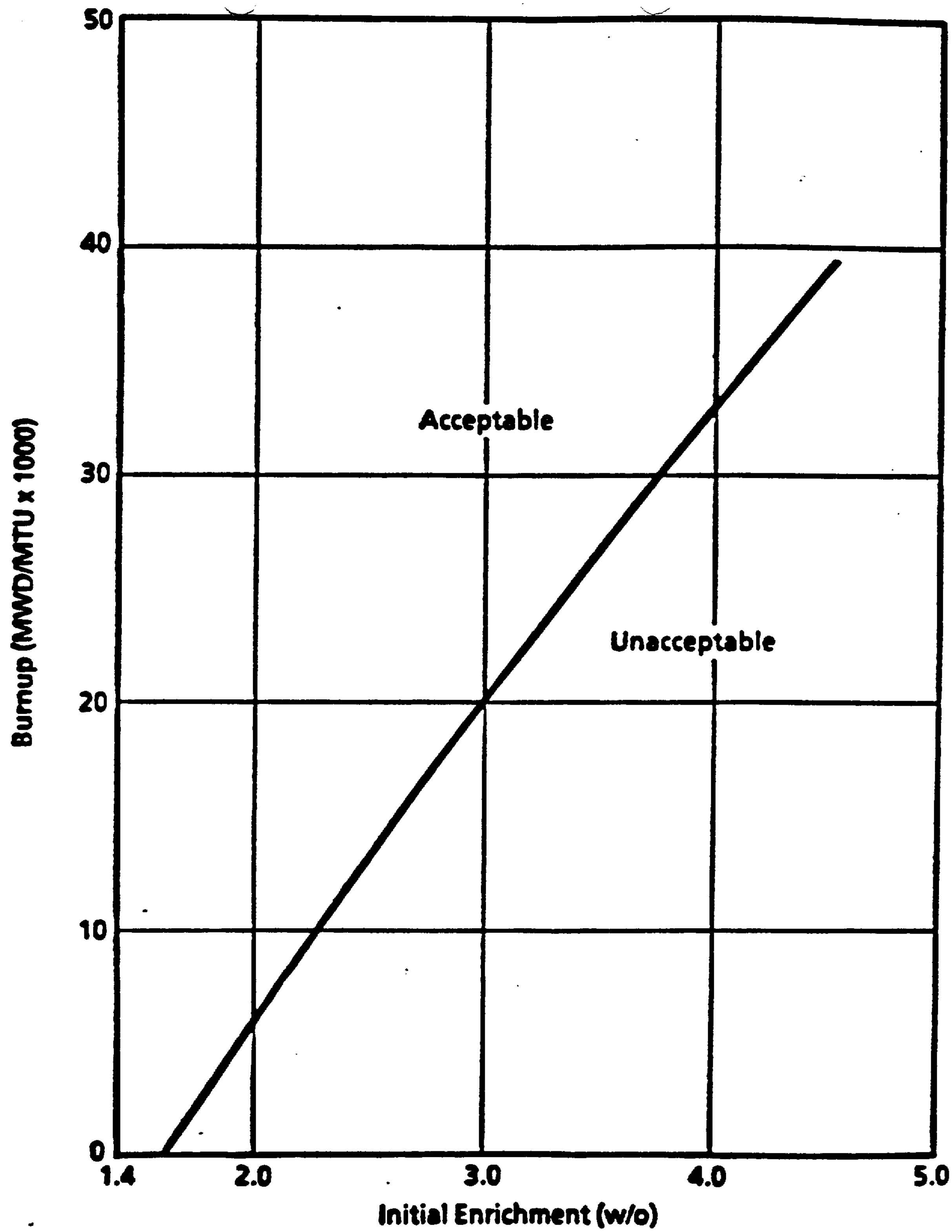


**Figure 5.6-1**

**SOUTH TEXAS PROJECT SPENT FUEL RACKS**
**REGION 2 REQUIRED BURNUP AS A FUNCTION OF INITIAL ENRICHMENT**

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 2 TO

FACILITY OPERATING LICENSE NO. NPF-76

HOUSTON LIGHTING & POWER COMPANY

CITY PUBLIC SERVICE BOARD OF SAN ANTONIO

CENTRAL POWER AND LIGHT COMPANY

CITY OF AUSTIN, TEXAS

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 2 TO

FACILITY OPERATING LICENSE NO. NPF-76

HOUSTON LIGHTING & POWER COMPANY

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DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

## 1. INTRODUCTION

### 1.1 Licensee Submittal and Staff Review

This report presents the NRC staff safety evaluation for the reracking of the spent fuel pool at the South Texas Project, Unit No. 1. By letter dated March 8, 1988 as supplemented March 26, 1988, Houston Lighting & Power Company (HL&P, the licensee) submitted an application to increase the storage capacity of the spent fuel pool, including the appropriate and necessary changes to the Technical Specifications. The licensee requested the increase in storage capacity because the spent fuel pool contained only 196 total storage cells. That storage capacity was adequate only for Unit 1 initial fueling and testing, and the initial part of fuel cycle 1, which is currently in progress.

The March 8, 1988 request for the amendment was noticed in the Federal Register on June 23, 1988 (53 FR 23707) as a Consideration of Issuance of Amendment to Facility Operating License and Opportunity for Hearing. The notice was supplemented on September 14, 1988 (53 FR 35570).

The application is based on the licensee's "High Density Spent Fuel Racks Safety Analysis Report" which was submitted as an enclosure to the March 8, 1988 application. During its review, the staff requested additional information from the licensee. The additional information was provided by letters dated August 9, 10, 19, 30, and September 21, 22, and 29, 1988.

This report was prepared by the staff of the Office of Nuclear Reactor Regulation. The principal contributors to this report are:

| | |
|---|---|
| H. Ashar | Structural and Geosciences Branch |
| L. Kopp | Reactor Systems Branch |
| J. Martin | Radiation Protection Branch |

| | |
|---|---|
| W. LeFave | Plant Systems Branch |
| J. Wing | Chemical Enegineering Branch |
| G. Dick | Project Directorate IV |

### 1.2 Summary Description of Reracking

The amendment would authorize the licensee to increase the spent fuel pool storage capacity from 196 to 1969 fuel assemblies. The proposed expansion is to be achieved by reracking the spent fuel pool into two discrete regions. New, high-density storage racks (free-standing) will be used. The existing storage racks (free-standing) will be removed, packaged and stored on-site.

The spent fuel pool is a stainless-steel lined reinforced concrete pool and is an integral part of the Fuel Handling Building (FHB). The pool walls are 5 feet, 3 inches to 7 feet, 9 inches thick and the basemat is 6 feet, 0 inches thick. The walls and floor are lined with a $\frac{1}{4}$-inch thick stainless steel liner to ensure the leaktight integrity of the pool. The liner plate welds are backed with fabricated members to collect water leakage from the pool. Any leakage entering the formed channels is directed to the Liquid Waste Processing System via the FHB pump.

Region I of the spent fuel pool includes 6 modules (racks) having a total of 288 storage cells. The nominal center-to-center spacing is 10.95 inches. All cells can be utilized for storage and each cell can accept new fuel assemblies with enrichments up to 4.5 weight percent U-235 or spent fuel assemblies that have not achieved burnup adequate for storage in Region 2. Region 2 includes 14 modules (racks) having a total of 1681 storage cells. The nominal center-to-center spacing is 9.15 inches. All cells can be utilized for storage and each cell can accept spent fuel assemblies with various initial enrichments that have achieved minimum burnups. Each cell in each region is designed to accomodate a single PWR fuel assembly, or equivalent.

The high-density spent fuel storage rack cells are fabricated from ASTM A240 Type 304L stainless steel plates. The Region 1 racks are a welded honeycomb array of square boxes separated by narrow rectangular water boxes. Strips of Boraflex neutron absorber are affixed on the outside face of the long sides. Stainless steel sheets are welded over the Boraflex sheets to hold them in a fixed position on the box. In Region 2, the Boraflex strips are located between adjacent walls. To provide space for the Boraflex strips between the cells, a double row of matching flat round raised areas are stamped into the side walls of each cell. The cells are welded together at the raised areas to hold the Boraflex in place. The cells are welded to individual assembly bases and to one another. The final rack arrangement is shown in Figure 1. Figures 2 and 3 show the cell design for the Region 1 and Region 2 racks, respectively.

The fuel rack module assembly consists of the storage cells (and integrally welded base plates) welded together and mounted on the support pedestals. The pedestals (four per rack module) are provided with holes and passages for flow to holes in the storage cell bottom plates. Figure 4 illustrates support arrangements. The tops of the support plates are welded to the fuel cell base plates. The leveling screws transmit the loads to the pool floor embedments, provide a sliding contact and provide for the leveling adjustment of the rack.

The new racks are not doubled-tiered and all racks will sit on the spent fuel pool floor.

The proposed expansion of the spent fuel pool storage capacity to 1969 fuel assemblies will provide adequate storage until the year 2020, while maintaining full core offload capability. In addition, it is expected that the expansion will be adequate until a federal repository is available for spent fuel.

The proposed request is for the storage of a single fuel assembly in each storage location of the high density racks. However, most of the analyses have been performed with the consolidated fuel weight in the storage locations. For the sake of analysis, the conservative assumptions have been made to simulate gaps and spring constants. The staff finds the approach acceptable for evaluating the proposed reracking.

However, this safety evaluation approves HL&P's request, that is the storage of non-consolidated fuel.

## 2. CRITICALITY CONSIDERATIONS

### 2.1 Criticality Analysis

#### 2.1.1 Calculation Methods

The calculation of the effective multiplication factor, $K_{eff}$, makes use of the PDQ-7 two-dimensional four-group diffusion theory computer code with neutron cross sections generated by the LEOPARD code. These codes were benchmarked against a series of critical experiments with characteristics similar to the South Texas spent fuel pool racks. These comparisons resulted in a model bias of + 0.0067 and a 95/95 probability/confidence uncertainty of ± 0.0027 for the Region 1 racks and a model bias of + 0.0057 and a 95/95 uncertainty of ± 0.0086 for the Region 2 racks.

In order to calculate the criterion for acceptable burnup for storage in Region 2, calculations were made for fuel of several different initial enrichments and, at each enrichment, a limiting reactivity value, which included an additional factor for uncertainty in the burnup analysis, was established. Burnup values which yielded the limiting reactivity values were then determined for each enrichment from which the acceptable burnup domain for storage in Region 2, as shown in proposed technical specification Figure 5.6-1 (Figure 5 of this SE), was obtained. The staff finds this procedure acceptable.

#### 2.1.2 Treatment of Uncertainties

A correction for the reactivity effect of pool temperature is included as well as a geometric modeling effect bias to account for mesh spacing and smeared stainless steel-water composition effects.

For the Region 1 analysis, the total uncertainty is the statistical combination of the calculational uncertainty and manufacturing and mechanical uncertainties due to variations in Boraflex thickness, inner stainless steel storage box dimension, stainless steel thickness, and fuel enrichment and density.

In the Region 2 analysis, the same uncertainties are considered. In addition, an uncertainty due to the burnup analysis is estimated and combined statistically with the other uncertainties.

The staff concludes that the appropriate uncertainties have been considered and have been calculated in an acceptable manner. In addition, these uncertainties were determined at least at a 95% probability 95% confidence level, thereby meeting the NRC requirements, and are acceptable.

2.1.3 Results of Analysis

For Region 1, the rack multiplication factor is calculated to be 0.9250, including uncertainties at the 95/95 probability/confidence level, when fuel having an enrichment of 4.5 weight percent U-235 is stored therein. Although the pool is normally flooded with water borated to 2500 ppm, unborated water was assumed in the analysis.

For Region 2, the rack multiplication factor is calculated to be 0.9478 for the most reactive irradiated fuel permitted to be stored in the racks, i.e., fuel with the minimum burnup permitted for each initial enrichment as shown in Figure 5. The design will accept fuel of 4.5 weight percent U-235 initial enrichment burned to 40.0 MWD/kgU. The analysis of the Region 2 racks also assumed full flooding by unborated water.

Therefore, the results of the criticality analyses meet the staff's acceptance criterion of $K_{eff}$ no greater than 0.95 including all uncertainties at the 95/95 probability/confidence level.

Most abnormal storage conditions will not result in an increase in the $K_{eff}$ of the racks. For example, loss of a cooling system will result in a decrease in the $K_{eff}$ value since reactivity decreases with decreasing water density.

It is possible to postulate events, such as an inadvertent misplacement of a fresh fuel assembly either into a Region 2 storage cell or outside and adjacent to a rack module, which could lead to an increase in pool reactivity. However, for such events credit may be taken for the Technical Specification requirement of at least 2500 ppm of boron in the refueling canal during refueling operations. The reduction in the $K_{eff}$ value caused by the boron more than offsets the reactivity addition caused by credible accidents.

The staff considered the possibility of irradiation induced axial shrinkage of the Boraflex panels as documented in NRC Information Notice No. 87-43. Based on this, the licensee has performed analyses to determine the reactivity effects of potential Boraflex shrinkage on the South Texas spent fuel pool.

Several scenarios were evaluated ranging from shrinkage of the top and bottom of each Boraflex panel with a corresponding exposure of active fuel at each end, to a single tear in each panel at the active fuel mid-plane. The results indicate that sufficient margin is available in both the Region 1 and Region 2 rack design to accommodate at least 8 inches of shrinkage at each end. In addition, for Region 1, a mid-plane gap of up to 4.5 inches in every panel would not prevent the maintenance of a $k_{eff}$ less than 0.95. For Region 2, mid-plane gaps in every panel of up to 3.2 inches could be accommodated. If the mid-plane gaps are assumed to occur in only two of the four panels in each Region 2 cell, gaps as large as 10 inches could be accommodated without preventing the maintenance of a $k_{eff}$ less than 0.95. Therefore, although it is not likely that significant gap formation will occur in the Boraflex panels, the staff believes that there would be sufficient time to detect such anomalies and provide appropriate actions before any significant adverse reactivity effects occur.

## 2.2 Technical Specification Changes

The following Technical Specification (TS) changes have been made as a result of the proposed spent fuel pool storage modifications. The staff finds these changes acceptable.

1. TS 5.6.1.1 and 5.6.1.2 are combined into one Specification (5.6.1). The new TS 5.6.1 correctly accounts for the uncertainties and tolerances assumed in the criticality analyses as well as the nominal center-to-center fuel assembly spacing, the maximum allowable U-235 enrichment, and the installation of Boraflex between spent fuel assemblies.

2. Figure 5.6-1 has been added to specify the initial enrichment vs. burnup requirements to be met prior to storage of fuel assemblies into Region 2.

3. The spent fuel pool storage capacity has been increased from 196 to 1969 fuel assemblies in TS 5.6.3.

The TS changes are effective as of the date of this Safety Evaluation.

## 2.3 Conclusion

Based on the review described above, the staff finds that the criticality aspects of the design of the South Texas Unit 1 spent fuel racks are acceptable and meet the requirements of General Design Criterion (GDC) 62 for the prevention of criticality in fuel storage and handling. The staff concludes that fuel from Unit 1 may be safely stored in Region 1 provided that the enrichment does not exceed 4.5 weight percent U-235. Any of these fuel assemblies may also be stored in Region 2 provided they meet the burnup and enrichment limits specified in Figure 5.6-1 of the South Texas Technical Specifications.

## 3.0 MATERIAL COMPATABILITY AND CHEMICAL STABILITY

### 3.1 Discussion

The staff has reviewed the compatability and chemical stability of the materials (except the fuel assemblies) wetted by the pool water, in accordance with Section 9.1.2 of the Standard Review Plan (NUREG-0800, July 1981). The STP-1 pool contains oxygen-saturated demineralized water which has 2500 parts per million of boron as boric acid. The pool is lined with stainless steel and has two adjacent regions of storage. The principal construction materials for the proposed new racks in the spent fuel storage pools are ASTM A-240 Type 304L austenitic stainless steel for structure and Boraflex for neutron absorption. The racks are welded honeycomb arrays of square stainless steel boxes forming individual cells for fuel storage. Each of the four sides of a given storage cell has a Boraflex assembly, except those sides that are nearest to the storage pool walls.

In Region 1, the Boraflex assembly consists of a thin rectangular stainless steel water box with a Boraflex sheet affixed on one side of the box and another Boraflex sheet on the other. A thin stainless steel plate is welded over each of the two Boraflex sheets on the water box. The entire assembly is removable from the storage cell. In Region 2, a Boraflex sheet is positioned between two adjacent walls of the square storage cells. The Boraflex sheets in Region 2 are not removable. In both regions, single sheets of Boraflex are used, and the Boraflex sheets are not mechanically fastened to any surface or structure.

The licensee proposed an inservice surveillance program for the Boraflex material, using sample coupons that are made of the same material composition, fabricated by the same method, certified to the same criteria, cut to the same physical dimensions, and encased in the same material as the removable Boraflex assemblies. A minimum of one such coupon and a string of foot-long samples of the same material will be provided for the storage pool in Unit 1. Evaluation of the coupon performance will include visual inspection and measurements of the neutron attenuation, hardness, and physical dimensions. Initial surveillance of the specimens will be performed after five years of exposure to the storage pool environment. Based on the results of the initial surveillance, the licensee will determine the schedule and extent of additional surveillance. The licensee, however, has provided no corrective actions to take if degradation of the Boraflex assemblies is found, but will evaluate available plant data on Boraflex performance from the nuclear industry to modify the surveillance program when warranted and justified.

### 3.2 Evaluation

The stainless steel in the storage pool liners and rack assemblies is compatible with the oxygen-saturated borated water and radiation environment of the spent fuel pool. In this environment, corrosion of Type 304L stainless steel is not expected to exceed a rate of $6 \times 10^{-7}$ inch per year. This corrosion rate is negligible for even the thinnest stainless steel walls of 0.03 inch in the rack assemblies. Contact corrosion or galvanic attack between the stainless steel in the pool liners or rack assemblies and the Inconel/Zircaloy in the fuel assemblies to be stored will not be significant, because all these materials are protected by passivating oxide films. Boraflex is composed of non-metallic materials and, therefore, will not develop a galvanic potential with the metal components.

Space is available to allow escape of any gas which may be generated from the polymer binders in the Boraflex during heating and irradiation, thus preventing possible bulging or swelling of the Boraflex assemblies. Boraflex, an elastomer of methylated polysiloxane filled with boron carbide powder, is used as a neutron absorber (poison) in the spent fuel storage facilities of many nuclear power plants. It has undergone extensive testing to determine the effects of gamma irradiation in various environments and to verify its structural integrity and suitability as a neutron absorbing material. The evaluation tests have shown that Boraflex is unaffected by the pool water environment and will not be degraded by corrosion. Tests were performed at the University of Michigan, exposing Boraflex to $1.03 \times 10^{11}$ rads of gamma radiation with substantial concurrent neutron flux in borated water. These tests indicated that Boraflex maintains its neutron attenuation capabilities after being subjected to an environment of borated water and gamma irradiation. Irradiation will cause some loss of flexibility and shrinkage of the Boraflex.

Long-term borated water soak tests at high temperatures were also conducted. The tests show that Boraflex withstood a borated water immersion at 240°F for 251 days without visible distortion or softening. The Boraflex showed no evidence of swelling or loss of ability to maintain a uniform distribution of boron carbide. The spent fuel pool water temperature under normal operating conditions will be approximately 105°F which is well below the 240°F test temperature. In general, the rate of a chemical reaction decreases exponentially with decreasing temperature. Therefore, the staff does not anticipate any significant deterioration of the Boraflex at the normal pool operating conditions over the design life of the spent fuel racks.

The tests have shown that neither irradiation, environment, nor Boraflex composition have a discernible effect on the neutron transmission of the Boraflex material. The tests also have shown that Boraflex does not possess leachable halogens that might be released into the pool environment in the presence of radiation. Similar conclusions are reached regarding the leaching of elemental boron from the Boraflex. Boron carbide of the grade normally present in the Boraflex will typically contain 0.1 weight percent of soluble boron. The test results have confirmed the encapsulation capability of the silicone polymer matrix to prevent the leaching of soluble species from the boron carbide.

Recently, anomalies ranging from minor physical changes in color, size, hardness, and brittleness to gap formation of up to four inches wide were observed in Boraflex panels that have been used in three nuclear power plants. The exact mechanisms that caused the observed physical degradations of Boraflex have not been confirmed. But the staff can postulate that gamma radiation from the spent fuel initially induced crosslinking of the polymer in Boraflex, producing shrinkage of the Boraflex material. When crosslinking became saturated, scissioning (a process in which bonds between atoms are broken) of the polymer predominated as the accumulated radiation dose increased. Scissioning produced porosity which allowed the spent fuel pool water to permeate the Boraflex material. Scissioning and water permeation could embrittle the Boraflex material. In short, gamma radiation from spent fuel is the most probable cause of the observed physical degradations, such as changes in color, size, hardness, and brittleness. The staff does not have sufficient information to determine conclusively what caused the gap formation in some Boraflex panels. However, it is conceivable that if the two ends of a full-length Boraflex panel are physically restrained, then shrinkage caused by gamma radiation can break up the panel and lead to gap formation.

The staff determined that reasonable assurance exists that physical restraints are absent in the Boraflex panels of the South Texas Project, because the Boraflex sheets are not mechanically fastened to any structure. It is not likely that gaps will form to any significant extent in the Boraflex panels during the projected life of the Boraflex assemblies. However, minor physical degradations can take place in the Boraflex from irradiation.

In the unlikely event of gap formation in the Boraflex panels that would lead to loss of neutron absorbing capability, the monitoring program will detect such degraded Boraflex panels, and the licensee would have sufficient time to replace them in Region 1 of the storage pools. In Region 2 where the Boraflex sheets are not removable, the licensee can either place new Boraflex sheets in the affected empty storage cells or restrict the use of the affected cells for fuel storage if degraded Boraflex is found.

### 3.3 Conclusions

Based on the above discussion, the staff concludes that the corrosion of the spent fuel pool components due to the pool environment will be of little significance during the life of the facility. Components in the spent fuel storage pools are constructed of alloys which have a low differential galvanic potential between them and have a high resistance to general corrosion, localized corrosion, and galvanic corrosion.

The staff further concludes that the environmental compatibility of the materials used in the spent fuel storage pools is adequate based on the test data cited in Section 3.2 and actual service experience at operating reactor facilities.

The staff has reviewed the proposed surveillance program for monitoring the Boraflex in the spent fuel storage pool and concludes that the program can reveal deterioration that might lead to loss of neutron absorbing capability during the life of the spent fuel racks. However, if a significant loss of neutron absorbing capability is found in any Boraflex panel, the licensee should take corrective actions such as replacement of the degraded Boraflex panel, insertion of a new Boraflex sheet in the affected storage cell, or restriction of use of the affected cell for fuel storage.

The staff finds that the proposed monitoring program and the selection of appropriate materials of construction by the licensee meet the requirements of 10 CFR Part 50, Appendix A, GDC 61 regarding the capability to permit appropriate periodic inspection and testing of components, and GDC 62 regarding prevention of criticality by the use of boron poison and by maintaining structural integrity of components, and are, therefore, acceptable.

## 4. STRUCTURAL DESIGN

For this portion of the review, the primary focus was assuring the structural integrity of the fuel, 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. The major areas of concern and their resolution are discussed in the following paragraphs.

### 4.1 Fuel Handling Building and Spent Fuel Pool

The Fuel Handling Building analysis and design was reviewed and accepted during the Operating License stage. HL&P, however, performed the seismic analysis of the fuel handling building incorporating the revised mass of the proposed reracking (consolidated fuel). The soil structure interaction analysis and the input motion were considered in the same way as in the original analysis. A comparative review of the output tables indicated less than 5% differences in modal frequencies. Also, a comparison of acceleration response spectra indicated negligible differences in spectral accelerations at the spent fuel pool floor level. The analysis thus confirmed the validity of the basic input criteria for the seismic analysis of the high density racks. HL&P also recalculated the differences in soil bearing pressure and the factor of safety against liquification due to the added mass. The soil bearing pressure increased from 20.2 ksf to 22 ksf against the allowable of 32 ksf under dead load and Safe Shutdown Earthquake (SSE) combination. The minimum safety factor of 1.4 against liquification remained unaffected.

HL&P also performed a detailed seismic analysis of spent fuel pool areas affected by the proposed reracking. HL&P demonstrated that the minimum safety factors at various critical sections of the pool walls and floor slab were higher than 1.0 for all conditions of loading considering the consolidated fuel weight. However, the design margin for transverse shear in the spent fuel pool floor slab was marginally above 1.0 for standard fuel. For consolidated fuel, HL&P demonstrated a similar margin when the confirmatory-basis response spectra described in the FSAR Section 3.7.2.4 was used. This evaluation pertains to the use of standard (single fuel assembly per storage location) fuel, for which the staff considers the design to be adequate.

The staff had expressed a concern regarding the integrity of pool floor liner plate before the rack pedestals start sliding under the postulated earthquake loading. The licensee performed a rigorous analysis of the liner-concrete interface and its anchorage at the embedded plates and leak chases. With conservatively estimated shear load, and considering a very low coefficient of friction of 0.15 between the liner and the concrete, the licensee demonstrated that margins of safety in excess of 1.0 exist for stresses in liner plate, embedded plates, anchor studs, anchor bolts and connecting welds. The staff finds the licensee's conclusion regarding the integrity of liner plate under postulated loadings to be acceptable.

### 4.2 Rack Analysis and Design

Tables 1, 2 and 3 provide the rack module data, dimensions and pertinent modeling parameters. HL&P's analysis is based on one set of synthetic time-histories. The staff expressed concern regarding the adequacy of energy content at the frequencies of interest, when used for non-linear rack analysis. HL&P generated the Power Spectral Density (PSD) functions for the floor input motion used in the rack analysis and compared them with the target PSD obtained by the method in NUREG/CR-3509, "Power Spectral Density Functions Compatible with Regulatory Guide 1.60, Response Spectra." A typical comparison is shown in Figure 6. In general, and particularly in the low frequency range of interest, the computed PSD exceeded the target PSD by a good margin. A dip at 6.3 Hz was indicative of the characteristics of the design response spectra. At frequencies higher than 12 Hz, the dips are expected when PSDs for in-structure motion are compared to the target PSD for ground motion. On an overall basis, the comparison indicated adequate energy content for time-histories being used for the rack analysis.

Requirements for seismic and impact loads are discussed in Section 3 of Appendix D of SRP Section 3.8.4. There it is stated that seismic excitation along three orthogonal directions should be imposed simultaneously for the design of the new rack system. HL&P's original rack analyses were based on the square root of the sum of the squares (SRSS) combination for the rack responses due to the three components of the earthquake to be considered.

The rack responses (displacement, forces) were separately calculated for two horizontal directions using a proprietary computer code "RACKOE". The responses due to the vertical component were calculated using an equivalent static method with a dynamic load factor of 1.5. This procedure appeared to provide bounding calculations for forces at the pedestal, but the staff expressed concern regarding the ability of the procedure to provide a realistic assessment of the displacements under the three components of an earthquake. "RACKOE" as used by HL&P is a two-dimensional code, capable of performing two-dimensional dynamic analysis with simultaneous seismic input in two directions. HL&P performed multi-rack analyses of 3-racks in a row (E-W direction) with varying loading conditions for each rack with simultaneous application of E-W and vertical time histories. The multi-rack model is shown in Figure 6. The gap used between the racks in these analyses was 1.0 inch. The results indicated that the maximum relative displacement between the two adjacent racks was 0.73 inches and 0.41 inches for the coefficients of friction of 0.2 and 0.8 respectively. None of the cases showed rack-to-rack or rack-to-wall interactions. The cross coupling effects were ignored in these calculations resulting in calculated displacements that are larger than the actual displacements.

Based on the conservatively computed maximum loadings, stresses in various critical components of the rack modules were computed for load combinations recommended in Table 1 of Appendix D of SRP 3.8.4. The stresses were compared against the requirements of Subsection NF of ASME Code and minimum safety factors as ratios of the allowable divided by the actual stresses were computed. Table 4 is a summary of the safety factors at critical rack locations.

The staff was concerned that after a seismic event the racks could move, creating gaps between the racks different from the 1 inch initial spacing such that the configuration of the racks would no longer be bounded by the seismic analysis. In response, the licensee committed to modify the plant procedures to include a requirement to perform a walkdown of the spent fuel pool to check the rack configuration after a seismic event (i.e., Operating Basis Earthquake or greater).

Based on the results of the HL&P's seismic analysis, the staff concludes that during an SSE, the fuel racks will maintain their structural integrity, fuel assemblies will not sustain damage, and rack displacements will not be large enough to result in rack-to-rack or rack-to-wall impact.

### 4.3 Fuel Handling Accident Consideration

HL&P performed structural analyses and evaluations of four postulated fuel handling accidents:

A. Dropped Fuel Accident 1

A consolidated fuel canister was assumed to have dropped from 14 inches above the top of a rack module and directly impacted the bottom plate of a fuel cell. The final velocity and total energy were considered assuming no energy dissipation in the canister. Based on this consideration the bottom plate weld could fail, thus allowing the bottom plate and the fuel canister to impact the pool liner. However, HL&P demonstrated that the impact energy would not perforate the liner. HL&P used Ballistic Research Laboratory (BRL) formula for predicting the liner penetration/perforation. Considering the conservative approach used by HL&P and that the staff's evaluation concerns single fuel assemblies, the analysis results are acceptable to the staff.

B. Dropped Fuel Accidents 2 and 3

In Accident 2, HL&P considered a drop of a consolidated fuel canister from 14 inches above the top of a rack. The top portion of the rack could sustain some plastic (permanent) deformation. However, HL&P's calculations confirmed that the safe, subcritical configuration of the stored fuel would not be compromised due to such an accident. The staff accepts the HL&P's findings for the purpose of this evaluation.

Accident 3, which postulates an inclined drop of a fuel canister, would not be as severe as Accident 2, as the impact energy would be distributed over a large area of a rack module.

C. Jammed Fuel Assembly

HL&P considered the rack stresses when 4000 lbs. of force was applied to unjam a fuel assembly in a storage location. This force was considered at any height of the fuel storage cell. HL&P's calculations indicated the stresses resulting from application of such a force to be within the acceptable criteria. The staff finds the postulation of the force to be reasonable and the final conclusions to be acceptable.

In any of the postulated accidents, damage to the dropped or jammed fuel assembly is possible. According to HL&P, the consequences of such damage are bounded by the design basis fuel-handling accidents described in the licensee's FSAR Section 15.7.4.

### 4.4 Conclusions

The staff concludes that the structural design, rack design and fuel handling accident considerations are acceptable.

## 5. SPENT FUEL POOL COOLING AND LOAD HANDLING

The licensee's submittal was reviewed in accordance with the requirements of GDC 2, 44, and 61, and the guidelines of NUREG-0800, "Standard Review Plan" (SRP) and NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

### 5.1 Decay Heat Generation Rate

The licensee stated in the March 8, 1988 submittal that the calculation of the decay heat generation rate was in accordance with the guidelines of NUREG-0800, SRP Section 9.1.3 and Branch Technical Position ASB 9-2. For the normal maximum heat load case the licensee assumed the pool was filled with one-third core refuelings every 12 months (maintaining a full core discharge capability) with the final one-third core being placed in the pool at 140 hours (Case A) and at 80 hours (Case B) after shutdown. The two cases of 140 hours and 80 hours were calculated because the South Texas plant has a fast refueling option which has the capability to offload one-third of a core in 80 hours. The specific recommendation in SRP Section 9.1.3 is 150 hours (Case C). The licensee calculated heat loads and fuel pool temperatures (one pool cooling train and two pool cooling train operation) for both the 140 hour and 80 hour cases and for the SRP Section 9.1.3 assumptions of one-third core after 150 hours, one-third core at one year, plus one-third core after 400 days. The maximum calculated pool temperatures with one and two trains operating are:

| | 1 Cooling Train | 2 Cooling Trains |
|---|---|---|
| Case A | 145.7°F | 126.0°F |
| Case B | 150.7°F | 129.2°F |
| Case C | 131.2°F | 118.7°F |

For the abnormal maximum heat load case (Case D), the licensee assumed the same conditions as in Cases A and B except that the last one-third core offload had been in the pool for 36 days plus a full core offload 120 hours after shutdown. The recommendations of SRP Section 9.1.3 are one-third core in the pool for 400 days, one-third for 36 days and one full core at 150 hours after shutdown. The calculated pool water temperature for Case D is 155.4°F with two pool cooling trains operating.

To verify the licensee's calculated spent fuel heat loads, the staff performed an independent calculation for the maximum abnormal storage condition of Case D using BTP ASB 9-2 guidelines. The staff calculated a heat load of 58.03 MBtu/hr as compared with 63.15 MBtu/hr. Because the licensee's calculated value was based on conservative assumptions as compared with the staff's (the licensee assumed last refueling was greater than 1/3 core leaving no empty storage spaces) and not appreciably different based on the high rate of decay heat energy, the staff finds that the licensee has properly calculated the heat generation rate in accordance with the SRP.

## 5.2 Spent Fuel Pool Cooling System

The spent fuel pool cooling system (SFPCS) consists of two seismic Category I, Quality Group C cooling water trains each with one pump and one heat exchanger. After the spent fuel pool water is cooled in the heat exchangers, it is purified by the non-seismic Category I cleanup system. In the event of a loss of the SFPCS, there are several sources of pool makeup water available including a seismic Category I source from the low-head safety injection pumps.

In its April 1986 Safety Evaluation Report (SER), NUREG-0781, for South Texas Units 1 and 2, the staff concluded that the SFPCS met the acceptance criteria of SRP Section 9.1.3 including GDC 2 and was acceptable. The bases for this conclusion have not changed as a result of the proposed reracking, except with regard to the requirements of GDC 44, "Cooling Water". The change in the basis for GDC 44 is due to the new decay heat loads which are higher for the increased storage capacity.

As indicated in Section 5.1, the design of the SFPCS still meets the 140°F fuel pool water temperature recommendation of SRP Section 9.1.3 when calculating the maximum normal heat load using the assumptions identified in the SRP. Under the higher heat load conditions identified using the licensee's more conservative assumptions for South Texas, the recommended pool temperature of 140°F for single train operation is acceptable because:

a. The assumptions used in the calculations are more conservative than staff guidelines;
b. The SFPCS is a safety-related system;
c. For the worst case (Case A) the 140°F could be exceeded for only 11.5 days;
d. With two trains operating, the pool temperatures for Cases A and B are well below 140°F;
e. The 140°F is a recommended limit and the likelihood of exceeding that recommendation is low given the conditions and conservatisms assumed in the calculation; and
f. The effect of pool water temperature slightly above 140°F on spent fuel storage safety is negligible.

For the abnormal maximum heat load (Case D), the SFPCS will maintain pool water temperature at or below 155.4°F with two trains of cooling which is well below the recommended no boiling limit of SRP Section 9.1.3 under these conditions.

As a result of its review, the staff finds that the SFPCS still meets the requirements of GDC 44 with respect to providing adequate pool cooling under maximum normal heat load conditions following a single failure.

5.3 Loss of Cooling

In the event that all SFP cooling is lost, the spent fuel pool temperature will increase until boiling occurs. The licensee has estimated the time from the loss of pool cooling until the pool boils for the four cases identified above. The times for the various cases including the boil-off rates are:

a. Case A - 8.29 hours at 54 gpm
b. Case B - 6.67 hours at 62 gpm
c. Case C - 15.49 hours at 35 gpm
d. Case D - 2.86 hours at 135 gpm

The assured (seismic Category I) pool makeup water source from the low pressure injection system has a capability in excess of the above boil-off rates. This seismic Category I makeup source is adequate to provide water for the higher boil-off rate of the expanded storage capacity and, therefore, still meets the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena."

5.4 Building Ventilation

The seismic Category I fuel handling building (FHB) ventilation exhaust system is designed to limit offsite doses in the event of a fuel handling accident. The staff's evaluation and conclusions regarding the consequences of a fuel handling accident identified in Section 15 of NUREG-0781 have not changed as a result of the proposed increased storage capacity because the accident analysis is based on the activity released from the last one third of a core placed in the pool. Thus, the FHB ventilation exhaust system continues to be acceptable.

5.5 Heavy Load Handling

The new and old spent fuel storage racks are considered to be heavy loads and will be moved by the FHB overhead crane. The FHB overhead crane is a single-failure-proof crane which meets the guidelines of NUREG-0554, "Single Failure-Proof Cranes for Nuclear Power Plants" and NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants," as indicated by the staff's acceptance in NUREG-0781.

Because the reracking will take place prior to the Unit 1 first refueling, heavy loads will not be carried over spent fuel during the reracking operation. The methods and equipment used for the reracking will be in accordance with Section 5.1.1 of NUREG-0612, which includes the identification of safe load paths for heavy loads, procedures for load handling, and operator training.

The spent fuel shipping cask cannot be carried over the spent fuel pool due to crane travel limitations, so a cask drop accident will not affect spent fuel or the spent fuel pool cooling system, as previously determined by the staff. Therefore, storage of spent fuel in the new proposed high density storage racks will not affect the staff's previous acceptance of the spent fuel cask drop analysis as contained in NUREG-0781. As a result of its review, the staff finds that heavy loads handling will be performed in accordance with the guidelines of NUREG-0612 and therefore the requirements of GDC 61, Fuel Storage and Handling and Radioactivity Control," are met as they relate to proper load handling to ensure against an unacceptable release of radioactivity or a criticality accident as a result of a postulated heavy load drop.

5.6 Conclusion

Based on the above, the staff concludes that the proposed expansion of the South Texas, Unit 1 spent fuel pool storage capacity complies with the requirements of General Design Criteria 2, 44 and 61, the guidelines of NUREG-0612 and applicable acceptance criteria of the Standard Review Plan with respect to the capability to provide adequate spent fuel pool cooling and the safe handling of heavy loads. The staff, therefore, concludes that the proposed spent fuel pool expansion is acceptable with respect to spent fuel pool cooling and load handling.

6. SPENT FUEL POOL CLEANUP SYSTEM

The spent fuel pool cleanup system at STP-1 is an integral part of the spent fuel pool cooling system. The system is designed to maintain water quality and clarity and to remove decay heat generated by the spent fuel assemblies in the spent fuel pool and in the temporary in-containment storage area. The cleanup system is also designed to purify water in the refueling cavity and the refueling water storage tank. The system includes all components and piping from inlet to exit from the spent fuel pool, in-containment storage area, refueling cavity, and piping used for fuel pool makeup, from the refueling water storage tank and the cleanup filters/demineralizers to the point of discharge to the radwaste system. The spent fuel pool cooling and cleanup system consists of two maximum normal heat load full-capacity fuel pool cooling trains (each with a pump and heat exchanger), two demineralizer purification trains, a spent fuel

pool surface skimmer loop, and a reactor cavity filtration system. The spent fuel pool cooling pumps can be powered from the Class 1E emergency sources.

Radioactivity and impurity levels in the water of a spent fuel pool increase primarily during the refueling operations as a result of fission product leakage from defective fuel elements being discharged into the pool and to a lesser degree during other spent fuel handling operations. The reracking of the spent fuel pool at the South Texas Project, Unit 1 will not increase the refueling frequency and fraction of the core replaced after each fuel cycle. Therefore, the frequency of operating the spent fuel pool cleanup system is not expected to increase.

Similarly, the chemical and radionuclide composition of the spent fuel pool water will not change as a result of the proposed reracking. Following the discharge of spent fuel from the reactor into the pool, the fission product inventory in the spent fuel and in the pool water will decrease by radioactive decay. Furthermore, experience also shows that there is no significant leakage of fission products from spent fuel stored in pools after the fuel has cooled for several months. Thus, the increased quantity of spent fuel to be stored in the South Texas Project, Unit 1 fuel pool will not increase significantly the total fission product activity in the spent fuel pool water during the operation of the pool.

7. RADIATION PROTECTION AND ALARA CONSIDERATIONS

In as much as the new spent fuel racks will be installed in the SFP before the pool is used for storage of spent fuel, there will be no additional occupational radiation exposure associated with the reracking of the spent fuel pool.

HL&P has considered any increases in exposure from spent fuel storage, airborne radiation, solid radioactive waste (resins, filters, and corrosion product crude) and concluded that no significant increases are expected. The staff has reviewed HL&P's analysis and finds it acceptable.

The radiological protection of workers during fuel handling operations will not change because the spent fuel will remain covered by 23 feet of water, as before, and spent fuel will be covered by at least 10 feet of water during spent fuel handling operations.

Based on the review of the HL&P's submittal, the staff concludes that the projected activities and estimated person-rem doses for this project are reasonable. HL&P intends to take ALARA considerations into account, and to implement reasonable dose-reducing activities. The staff concludes that HL&P will be able to maintain individual occupational radiation exposures within the applicable limits of 10 CFR Part 20, and maintain ALARA doses, consistent with the guidelines of Regulatory Guide 8.8. Therefore, the proposed radiation protection aspect of the SFP rerack is acceptable.

## 8. ACCIDENT ANALYSES

The staff has reviewed the accident analysis that could occur at STP-1 in conjunction with the proposed reracking. The applicable accidents were cask drop, loads over the spent fuel, and spent fuel pool boiling.

The proposed changes do not affect the previously approved cask drop analysis. Crane design and building arrangement prevent movement of the cask over the fuel pool and prevent interference of the cask crane bridge, trolley, and hoist with fuel racks or building structures. The rail for the cask handling crane stops at the edge of the cask loading pool, which is more than 25 ft. from the spent fuel pool boundary. Building arrangement, crane control, and lifting rig design restrict vertical lift of the cask to an elevation such that the cask will not be higher than 30 feet above the floor in the Fuel Handling Building. In accordance with 10 CFR Part 71, the spent fuel shipping cask is designed to sustain a free-fall of 30 feet onto an unyielding surface followed by a specified puncture, fire, and immersion in water with the release of no more than a specified small quantity of radioactivity.

The spent fuel cask crane is not capable of traveling over the spent fuel pool.

In the spent fuel pool boiling accident, it was assumed that a loss of spent fuel pool cooling occurred after a refueling where 1/3 of the core had been removed and placed in the spent fuel pool. As a result of boiling it was assumed that the greatest contribution to iodine leakage was from the off-loaded 1/3 core. The dose consequences at the exclusion zone boundary (0-2 hours) and low population zone boundary (0-30 days) were 0.0002 and 0.54 thyroid-rem respectively.

The potential doses resulting from the accidents considered were well below the allowable 10 CFR Part 100 guidelines. Therefore, the accident analysis aspect of the spent fuel pool rerack is acceptable.

## 9. RADIOACTIVE WASTE TREATMENT

The plant contains a radioactive waste management system designed to provide for the controlled handling and treatment of liquid, gaseous, and solid wastes.

The radioactive waste management system was evaluated in staff Safety Evaluation Report (SER) dated April 1986 (NUREG-0781). There will be no changes in the system described in the SER because of the proposed SFP rerack.

## 10. SUMMARY OF STAFF EVALUATION

The staff has reviewed and evaluated HL&P's request for the expansion of the spent fuel pool capacity. Based on the considerations discussed in this safety evaluation, the staff concludes that the analyses of the spent fuel rack modules and the spent fuel pool are in compliance with the acceptance criteria set forth in the FSAR and consistent with the current licensing practice, and therefore are acceptable.

The approval is based on the storage of non-consolidated fuel and the installation of all racks prior to storage of any spent fuel in the spent fuel pool. If HL&P should change its plans and decide to store spent fuel in the pool before completing the installation of the new racks, it should submit documentation to the staff for prior review addressing all significant changes from the request the staff is now approving.

It was noted during the staff review that while the proposed surveillance program for monitoring Boraflex in the spent fuel pool was acceptable, no corrective action was proposed in the event that Boraflex degradation was observed. It is recommended that a plan of corrective actions be developed and implemented.

## 11. ENVIRONMENTAL CONSIDERATIONS

The March 8, 1988 request for amendment was noticed in the Federal Register on June 23, 1988 (53 FR 23707) as a Consideration of Issuance of Amendment to Facility Operating License and Opportunity for Hearing. It was supplemented on September 14, 1988 (53 FR 35570). No hearing requests were received.

A separate Environmental Assessment has been prepared pursuant to 10 CFR Part 51. The Notice of Issuance of Environmental Assessment and Finding of No Significant Impact was published in the Federal Register on October 28, 1988 (53 FR 43788).

## 12. CONCLUSIONS

The staff has reviewed and evaluated the licensee's request for amendment for the South Texas Project, Unit 1 regarding the expansion of the spent fuel pool. Based on the considerations discussed in this safety evaluation, the staff concludes that:

(1) this amendment will not, (a) significantly increase the probability or consequences of accidents previously evaluated, (b) create the possibility of a new or different accident from any accident previously evaluated, (c) significantly reduce a margin of safety; and therefore, the amendment does not involve significant hazards considerations;

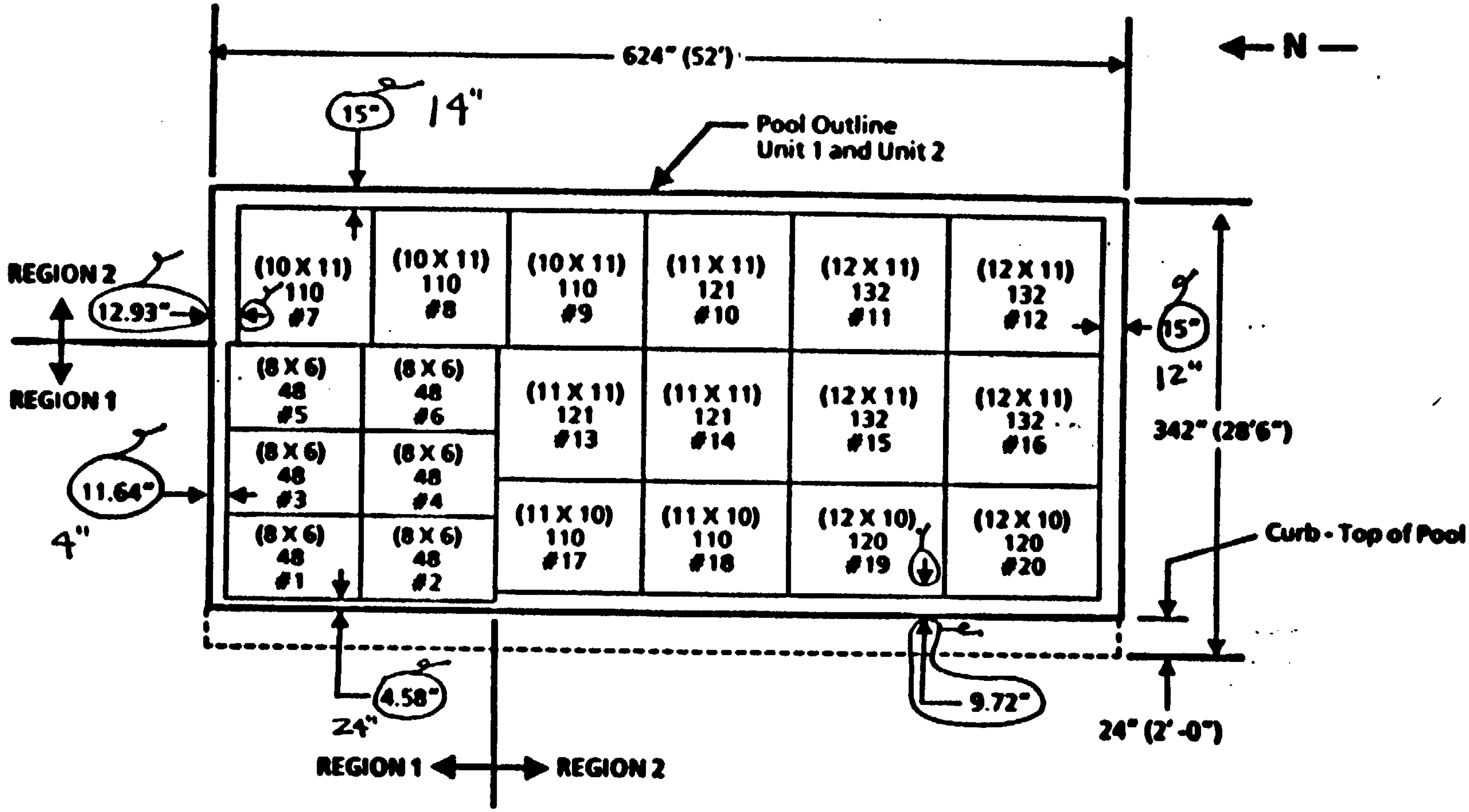
(2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and

(3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

The amendment is in effect as of the date of the staff's Safety Evaluation.

Dated: November 1, 1988

High Density Spent Fuel Storage Racks
South Texas Electric Generating Station
Houston Lighting & Power Company




| | | |
|---|---|---|
| **Region 1** | **F.A. Storage =** | **288** |
| **Region 2** | **F.A. Storage =** | **1681** |
| **Total** | **F.A. Storage =** | **1969** |

Figure 1

**SPENT FUEL POOL RACK LAYOUT**
**(TYPICAL UNIT 1 OR UNIT 2)**

**(Reference FSAR Figure 9.1.2-2)**

* Dimensions from the racks to the edge of the pool are minimums

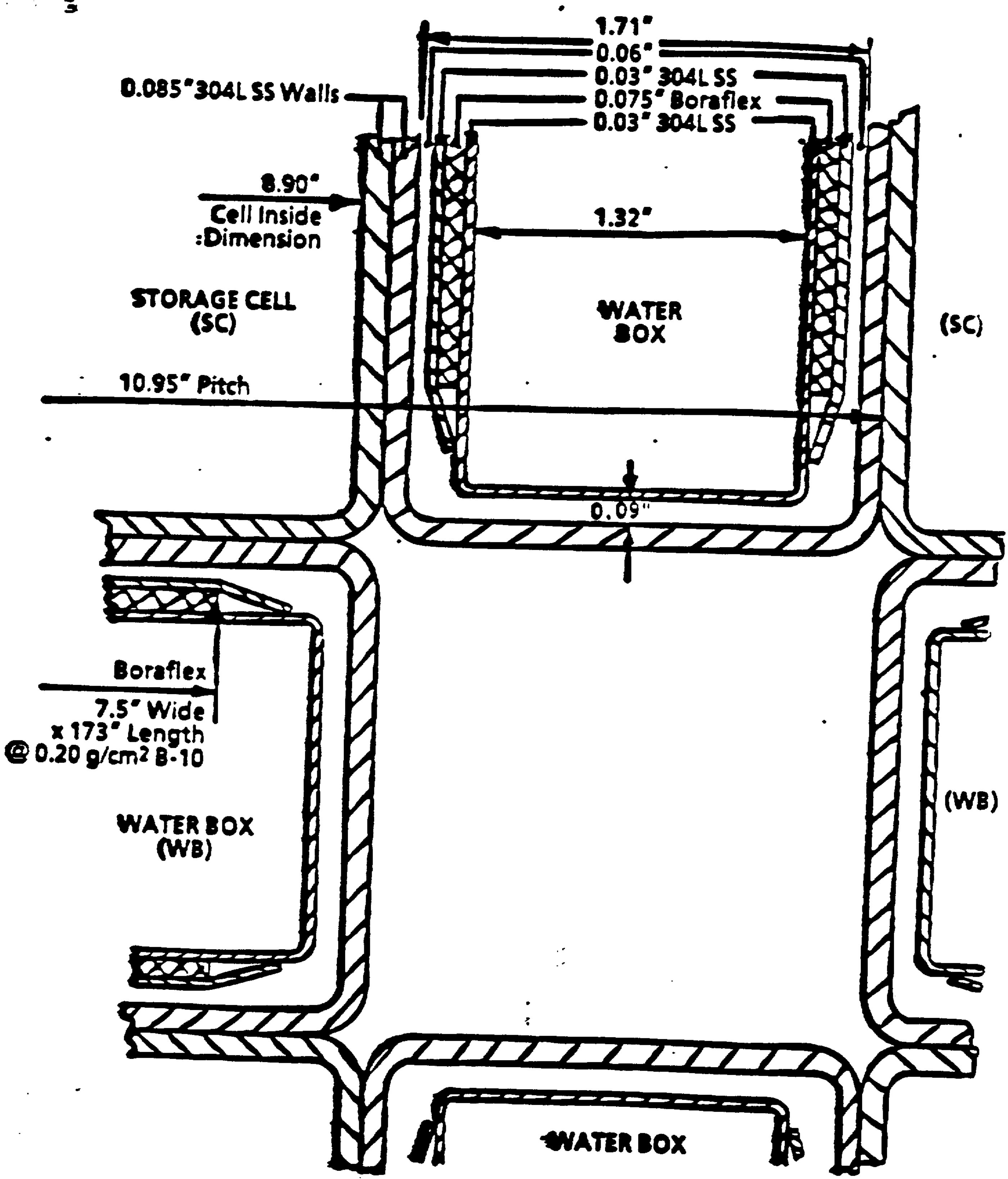



Figure 2

REGION 1 RACK CELL DIMENSIONS

Not to Scale

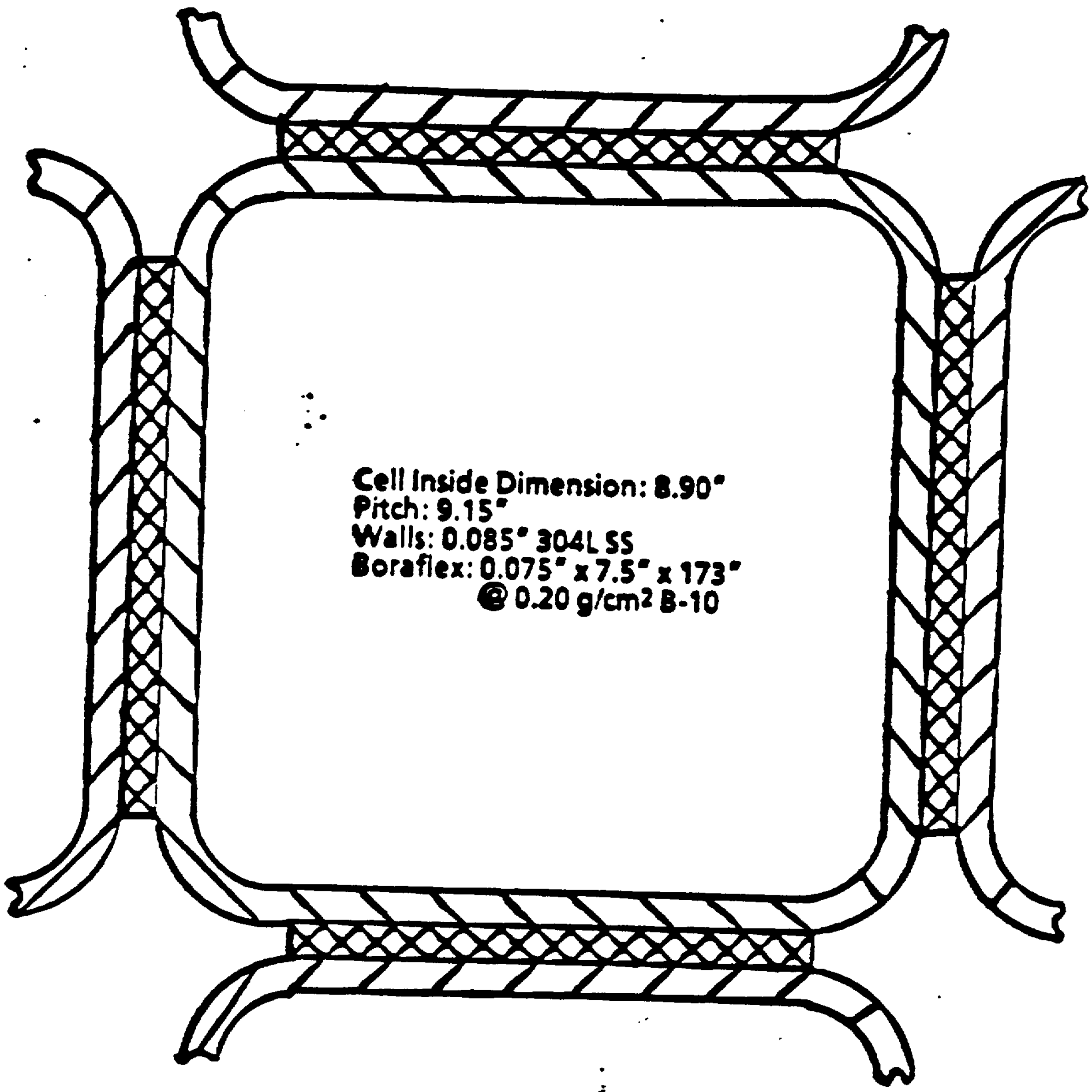



Figure 3

**REGION 2 RACK CELL DIMENSIONS**

Not to Scale

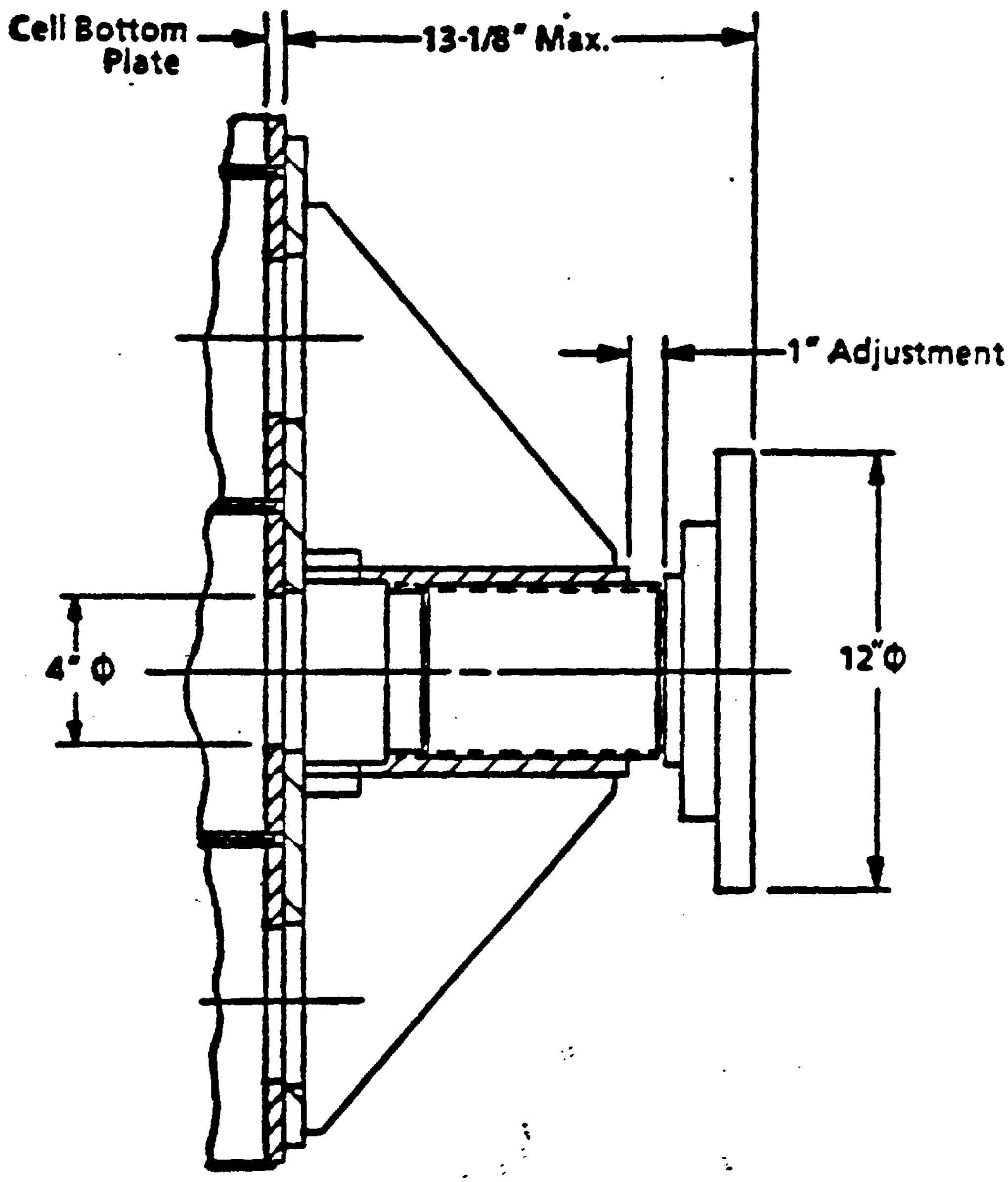



Figure 4

ADJUSTABLE RACK PEDESTAL

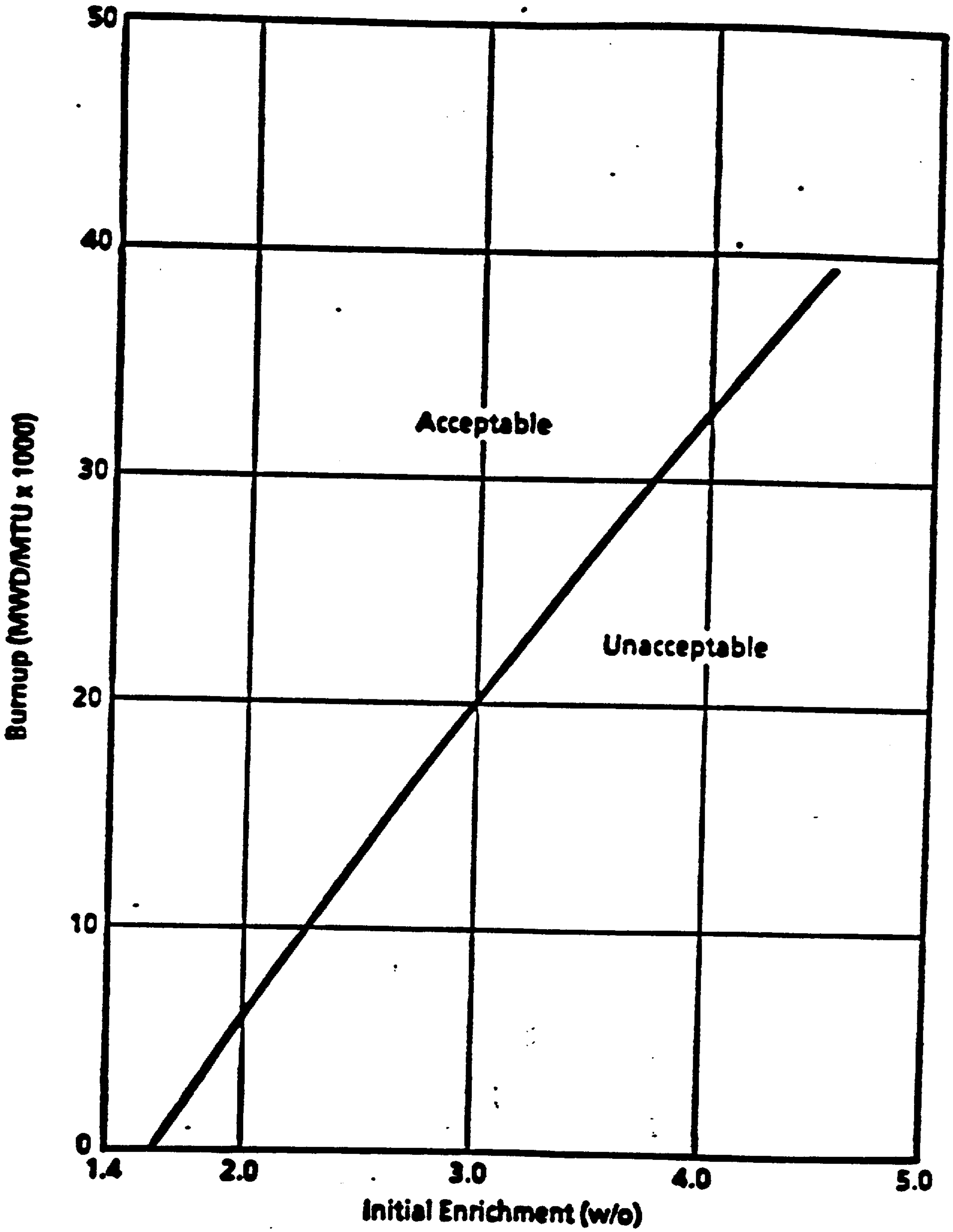



Figure 5

SOUTH TEXAS PROJECT SPENT FUEL RACKS
REGION 2 REQUIRED BURNUP AS A FUNCTION OF INITIAL ENRICHMENT

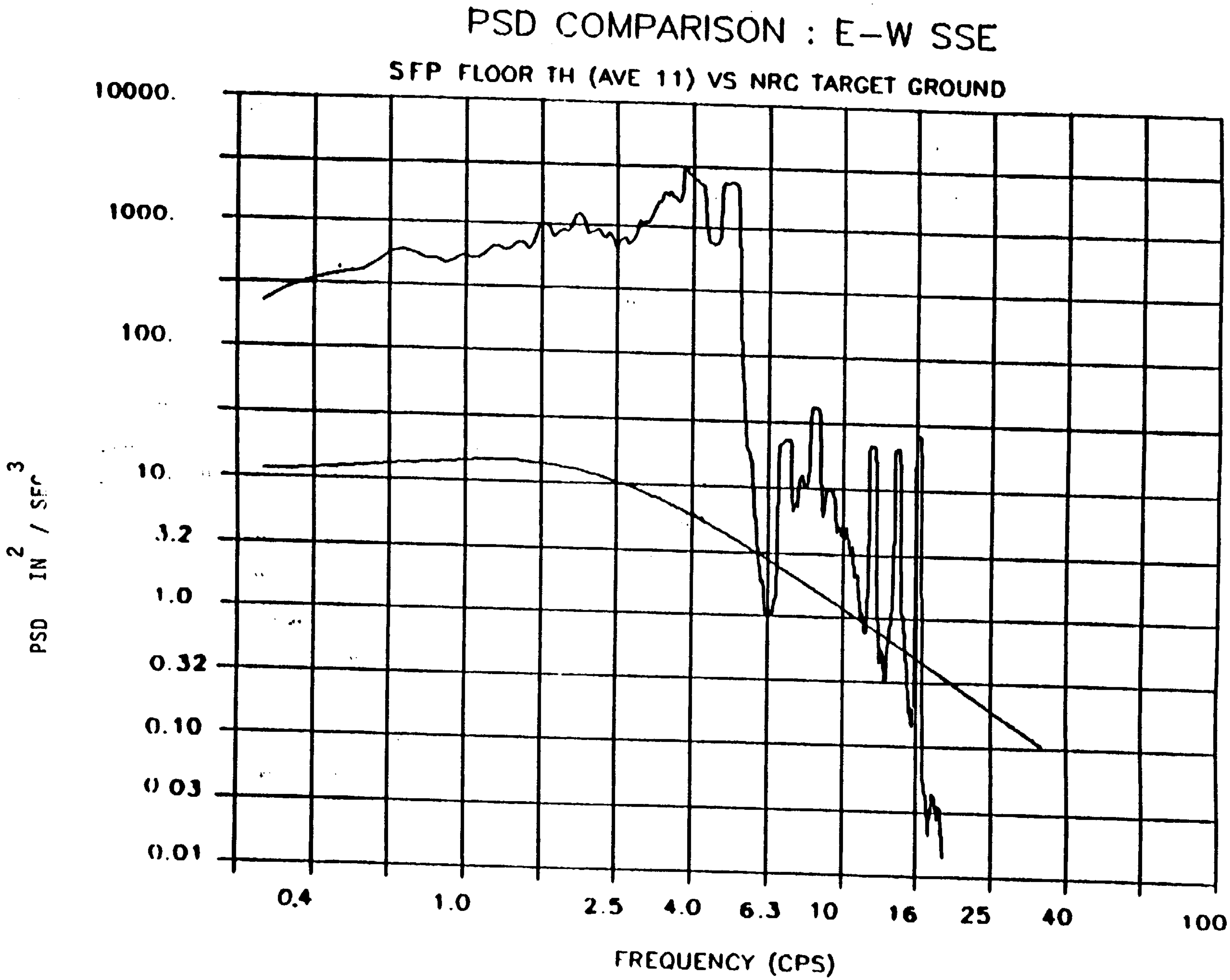
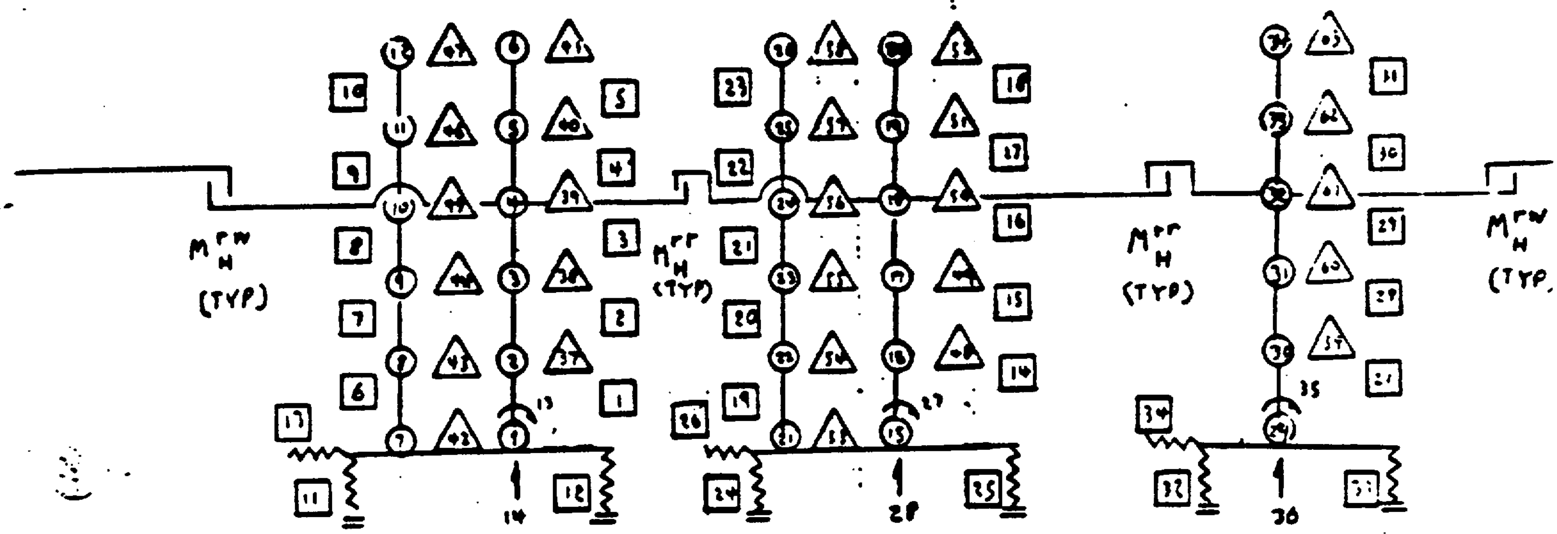

PSD COMPARISON : E-W SSE
SFP FLOOR TH (AVE 11) VS NRC TARGET GROUND
10000.
1000.
100.
10.
3.2
1.0
0.32
0.10
0.03
0.01
PSD IN^2 / SEC^3
0.4
1.0
2.5
4.0
6.3
10
16
25
40
100
FREQUENCY (CPS)

Figure 6



NOTE: NOT SHOWN ARE THE

1) FUEL-RACK GAP SPRINGS

2) FUEL-RACK FLUID COUPLINGS

Figure 7

MULTI-RACK ANALYSIS MODEL

**TABLE 1**

**SPENT FUEL RACK DATA**

| Region | Rack Module Number | Storage Cells Per Module | Array Size* |
|---|---|---|---|
| 1 | 1 | 48 | 8 x 6 |
| 1 | 2 | 48 | 8 x 6 |
| 1 | 3 | 48 | 8 x 6 |
| 1 | 4 | 48 | 8 x 6 |
| 1 | 5 | 48 | 8 x 6 |
| 1 | 6 | 48 | 8 x 6 |
| 2 | 7 | 110 | 10 x 11 |
| 2 | 8 | 110 | 10 x 11 |
| 2 | 9 | 110 | 10 x 11 |
| 2 | 10 | 121 | 11 x 11 |
| 2 | 11 | 132 | 12 x 11 |
| 2 | 12 | 132 | 12 x 11 |
| 2 | 13 | 121 | 11 x 11 |
| 2 | 14 | 121 | 11 x 11 |
| 2 | 15 | 132 | 12 x 11 |
| 2 | 16 | 132 | 12 x 11 |
| 2 | 17 | 110 | 11 x 10 |
| 2 | 18 | 110 | 11 x 10 |
| 2 | 19 | 120 | 12 x 10 |
| 2 | 20 | 120 | 12 x 10 |

* The array size indicates the number of storage cells in the N-S direction x the number of cells in the E-W direction.

Note: This is the same table as Table 3.2 in Reference 1.

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**TABLE 2**

**RACK MODULE DIMENSIONS AND WEIGHTS**

| Rack Module Number | Nominal Cross-Section Dimensions (inches) N-S | E-W | Estimated Dry Weight (lbs) Per Module | Estimated Dry Weight (lbs)per Module with Single Density Fuel |
|---|---|---|---|---|
| 1 | 88 | 66 | 26,100 | 114,516 |
| 2 | 88 | 66 | 26,100 | 114,516 |
| 3 | 88 | 66 | 26,100 | 114,516 |
| 4 | 88 | 66 | 26,100 | 114,516 |
| 5 | 88 | 66 | 26,100 | 114,516 |
| 6 | 88 | 66 | 26,100 | 114,516 |
| 7 | 91 | 101 | 23,040 | 225,660 |
| 8 | 91 | 101 | 23,040 | 225,660 |
| 9 | 91 | 101 | 23,040 | 225,660 |
| 10 | 101 | 101 | 25,220 | 248,102 |
| 11 | 110 | 101 | 27,400 | 270,544 |
| 12 | 110 | 101 | 27,220 | 270,364 |
| 13 | 101 | 101 | 25,400 | 248,282 |
| 14 | 101 | 101 | 25,400 | 248,282 |
| 15 | 110 | 101 | 27,600 | 270,744 |
| 16 | 110 | 101 | 27,420 | 270,564 |
| 17 | 101 | 91 | 23,200 | 225,820 |
| 18 | 101 | 91 | 23,200 | 225,820 |
| 19 | 110 | 91 | 25,200 | 246,240 |
| 20 | 110 | 91 | 25,040 | 246,080 |

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**TABLE 3**

**RACK MODEL PARAMETERS ***

| Rack Module | | 11 x 10 | 12 x 11 | 8 x 6 |
|---|---|---|---|---|
| $K_I$ (lb/in) | E-W | $.099 \times 10^6$ | $.0871 \times 10^6$ | $.0527 \times 10^6$ |
| | N-S | $.099 \times 10^6$ ** | $.0873 \times 10^6$ ** | $.0788 \times 10^6$ *** |
| $K_d$ (lb/in) | E-W | $1.77 \times 10^7$ | $1.84 \times 10^7$ | $1.40 \times 10^7$ |
| | N-S | $1.87 \times 10^7$ | $1.75 \times 10^7$ | $1.40 \times 10^7$ |
| h (in) | | 13.12 | 13.12 | 13.12 |
| H (in) | | 201.31 | 201.31 | 201.31 |
| $W_y$ (lb) | | 22938 | 27366 | 26047 |
| $W_F$ (lb) | | 386936 | 463094 | 88416 |
| $L_x$ (in) | | 91.50 | 100.65 | 65.70 |
| $L_y$ (in) | | 100.65 | 109.80 | 87.60 |

* Rack model parameters are for the consolidated fuel except the 8 x 6 size which only will store single density spent fuel.
** Nominal gap between cell wall & fuel assembly = 0.185"
*** Nominal gap between cell wall & fuel assembly = 0.237"

$K_I$ - Fuel assembly-to-cell wall impact spring rate
$K_d$ - Vertical axial spring rate for concrete, pedestal, base plate and cell deformation.
h - Length of support leg
H - Height of rack above base plate
$W_y$ - Weight of rack without fuel
$W_F$ - Weight of fuel
$L_x$ - Platform dimension (x - Direction = East)
$L_y$ - Platform dimension (y - Direction = North)

**TABLE 4**

**SUMMARY OF SAFETY FACTORS IN CRITICAL FUEL RACK LOCATIONS**

| Item/Location | Safety Factor * * | Comments |
|---|---|---|
| Support Footing (Pedestal) to Baseplate Weld Stress | 1.97 | Table 6.6* |
| Cell to Baseplate Weld Stress | 1.06 | Table 6.6* |
| Cell to Cell Weld stress | 1.12 | Thermal Plus Seismic Stress Due to Effects of Isolated Hot Cell. |
| Impact Load Between Fuel Assembly and Cell Wall | 1.20 | Standard Fuel |
| Shear Load on Baseplate Near a Support Footing | 1.07 | Table 6.6* |
| Compressive Stress in Cell Wall | 4.23 | Based on Local Buckling Considerations (Standard Fuel) |
| Rack to Wall Impact Loads | - | No Impact with Pool Walls occur at any Location |

* Table 6.6, Licensing Submittal, ST-HL-AE-2417; See Table 6.6 for other related Safety Factors.

** All Safety Factors are for consolidated fuel unless otherwise noted.

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UNITED STATES NUCLEAR REGULATORY COMMISSION

HOUSTON LIGHTING & POWER COMPANY

DOCKET NO. 50-498

NOTICE OF ISSUANCE OF AMENDMENT

TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 2 to Facility Operating License No. NPF-76, issued to the Houston Lighting & Power Company, (the licensee), which revised the Technical Specifications for operation of the South Texas Project, Unit 1, located in Matagorda County, Texas. The amendment was effective as of the date of its issuance.

The amendment allows the expansion of the spent fuel pool storage capacity form the current 196 fuel assemblies to 1969 fuel assemblies. The expansion is to be achieved by removing the existing racks and installing new, high density ones.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter 1, which are set forth in the license amendment.

The Notice of Consideration of Issuance of Amendment was published in the Federal Register on June 23, 1988 (53 FR 230707) and amended on September 14, 1988 (53 FR 35570). No request for a hearing or petition for leave to intervene was filed following the notices.

The Commission has prepared an Environmental Assessment related to the action and has concluded that an environmental impact statement is not warranted because there will be no environmental impact attributable to the action beyond that which has been predicted and described in the Commission's Final Environmental Statement related to the Operation of South Texas Project, Units 1 and 2 dated August 1986.

For further details with respect to the action, see: (1) the application for amendment dated March 8, 1988, as supplemented by letter dated March 26, 1988; (2) additional information supplied in the licensee's letters of August 9, 10, 19, 30, and September 21, 22, and 29, 1988; (3) Amendment No. 2 to License No. NFP-76; and (4) the Commission's related Safety Evaluation and Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, 2120 L Street N. W., Washington, D.C. 20555; at Wharton Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, Texas 77488; and the Austin Public Library, 810 Gaudalupe Street, Austin, Texas 78701. A copy of items (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects - III, IV, V and Special Projects.

Dated at Rockville, Maryland this 1st day of November, 1988.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick Jr.

George F. Dick, Jr., Project Manager
Project Directorate - IV
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation