



PSE&G

Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038-0236

Nuclear Business Unit

APR 10 1996

LR-N96099

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

Gentlemen,

**SPENT FUEL POOL COOLING SYSTEM DESCRIPTION
SALEM GENERATING STATION
UNIT NOS. 1 AND 2
DOCKET NOS. 50-272 AND 50-311**

On March 28, 1996 Public Service Electric and Gas (PSE&G) representatives met with Mr. L. Olshan and C. Marschall of your staff to address questions relating to the Spent Fuel Pool Cooling System (SFPCS) design basis for Salem Generating Station, Units 1 and 2. As a result of that discussion, it was requested that PSE&G submit marked-up Updated Final Safety Analysis Report (UFSAR) pages to clarify certain aspects of the SFPCS design basis. Attachment 1 has been prepared to address this request.

Attachment 1 consists of those UFSAR revisions which were developed as part of the recently completed Salem Unit 1 and 2 Spent Fuel Pool (SFP) Rerack projects. Details of those projects were submitted to the NRC and approved in License Amendments 151 and 131 for Salem Units 1 and 2 respectively. These changes are scheduled for incorporation into the UFSAR as part of a June 1996 revision required by 10 CFR 50.71(e).

As we discussed in our meeting, PSE&G performed a design change package quality review in late 1995 which included a review of the Salem SFP Rerack design changes. The results of that review identified the need to reconstitute various aspects of the SFP design basis. The reconstitution project serves two purposes; 1) to capture information required to support long term safe operation of the SFP and its support systems, and 2) to support evaluation of past operation against the NRC approved Licensing and design basis. The project plan also includes development of further clarification to the design basis description contained in the UFSAR where necessary.

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The power is in your hands.

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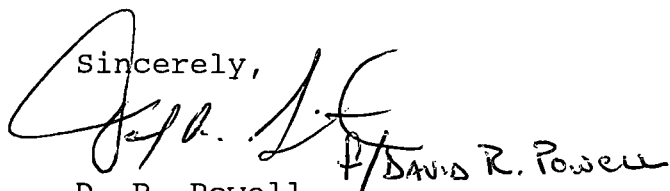
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APR 10 1996

If you have any questions regarding this information, please feel free to contact us.

Sincerely,

A handwritten signature in black ink, appearing to read 'D.R. Powell', written over a printed name.

D. R. Powell
Manager - Licensing and Regulation

Attachment

APR 10 1996

C Mr. T. T. Martin, Administrator - Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. L. N. Olshan, Licensing Project Manager - Salem
U. S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Mail Stop 14E21
Rockville, MD 20852

Mr. C. Marschall (X24)
USNRC Senior Resident Inspector

Mr. K. Tosch, Manager, IV
Bureau of Nuclear Engineering
33 Arctic Parkway
CN 415
Trenton, NJ 08625

LR-N96099
ATTACHMENT 1
44 PAGES

The safety against sliding, overturning, and flotation for all Category I structures under all loading combinations are within the limits set by the SRP 3.8.5.

Masonry Walls

For the loading criteria for non-structural masonry walls see Section 3.8.4.5.1.

3.8.4.4 Design and Analysis Procedures

The Category I structures have been designed based on ACI 318-63 "Working Stress Design," for normal operating load plus OBE and "Ultimate Strength Design," for normal loads plus DBE or tornado. In the working stress design under OBE the allowable stresses are one-third above the normal applicable code working stresses. Wind stresses are found to be less critical than those generated for an OBE. Load factors of unity have been used in the ultimate design under DBE or tornado loading. The stress of reinforcing steel under ultimate strength design has been kept under $0.9 F_y$. The capacity reduction factor " ϕ " as described in Section 3.8.1.4.1 for concrete stress is applicable for all Category I structures. A coefficient "k" of 0.85 for 3500 psi concrete has been used in addition to " ϕ " for equivalent rectangular concrete stress distribution.

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Steel members inside the Category I structures are designed in accordance with the AISC Manual of Steel Construction (Sixth Edition or later edition, as applicable).

Seismic design criteria are described in Section 3.7. Tornado and tornado generated missile design is described in Sections 3.3.2 and 3.5.2.

Two independent seismic analyses, similar to those for the Containment Building, have been performed for the Auxiliary

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During the design phase of the reracking which was implemented in 1994, the Spent Fuel Pool was reanalyzed for increased fuel storage capacity. The following is the list of items incorporated in the analysis:

- 1) The "ultimate strength" design method based on NUREG-0800, Standard Review Plan 3.8.4, Rev. 1, 1981 was used.
- 2) The plant design spectra, given in Salem UFSAR, for DBE and OBE events were broadened, per provisions of Reg. Guide 1.122 and used.
- 3) The response spectrum method was used to determine the self-excitation loading on the pool structure.
- 4) The pool structure was modelled in three dimensions via a 3-D finite element model.
- 5) The thermal gradient across the pool slab and the pool walls was computed using finite element method. Thus, the effect of interaction between the ambient, pool water, and grade temperatures is fully incorporated in the analysis.
- 6) The pressure on the lower portion of the wall during a seismic event undergoes a cyclic pulsation due to the hydrodynamic coupling between racks and the pool walls. This loading was quantified using Whole Pool Multi-Rack analysis. This loading was included in the analysis.

LIST OF FIGURES

<u>Figure</u>	<u>Title</u>	
9.1-1	Fuel Handling System	1
9.1-2	High Density Poison Spent Fuel Storage Module	← INSERT "B"
9.1-3	Poison Spent Fuel Storage Cell	
9.1-4A	Spent Fuel Cooling System - Unit 1	
9.1-4B	Spent Fuel Cooling System - Unit 2	
9.1-5	Fuel Transfer System (Typical)	
9.1-6	Reactor Vessel Head and Lifting Device	
9.1-7	Missile Shield	
9.1-8	Missile Shield Handling Fixture	
9.1-9	Internals Lifting Rig	
9.2-1A	Service Water System - Nuclear Area - Unit 1	
9.2-1B	Service Water System - Nuclear Area - Unit 2	
9.2-2	Service Water System - Turbine Area - Unit 1	
9.2-3	Service Water Intake (Arrangement)	

INSERT B

- 9.1-2A High Density Region I Spent Fuel Storage Module
- 9.1-2B Maximum Density Region II Spent Fuel Storage Module
- 9.1-3 Unit 1 Spent Fuel Pool Rack Layout
- 9.1-3A Unit 2 Spent Fuel Pool Rack Layout
- 9.1-3B High Density Region I Spent Fuel Storage Cell
- 9.1-3C Maximum Density Region II Spent Fuel Storage Rack Array
- 9.1-3D Maximum Density Region II Spent Fuel Storage Cell
- 9.1-3E Maximum Density Region II Spent Fuel Storage Cell Cross Section
- 9.1-3F Maximum Density Region II Spent Fuel Storage Rack Baseplate

9.1.2.1 Design Bases

The spent fuel storage racks are designed in a subcritical array such that k_{eff} is limited to a value of less than or equal to 0.95 even if the pool is flooded with demineralized water. The spent fuel racks are built to ensure a nominal 10.5-inch center-to-center distance between fuel assemblies stored in the racks. The storage capacity is limited to 1170 spent fuel assemblies, which will cover a period of 18 years, assuming that one-third of the core is replaced annually.

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The reactor cavity, refueling canal and spent fuel storage pool are reinforced concrete structures with seam-welded stainless steel plate liners. These Seismic Category I structures are designed to withstand the anticipated earthquake loadings and to prevent liner leakage even in the event the reinforced concrete develops cracks.

Design criteria for spent fuel storage racks assure conformance with recognized codes and applicable Regulatory Guides as follows:

1. The spent fuel storage rack design is based on the requirements of the ASME Boiler and Pressure Vessel Code, Section III Subsection NF, Class 3 Linear Supports.
2. Regulatory Guide 1.13 - The design conforms with the Regulatory Guide, except that high radiation instrumentation does not actuate the filtration system.
3. Regulatory Guide 1.29 - The spent fuel storage racks are designed as Seismic Category I Structures.
4. Regulatory Guide 1.92 - Seismic load combinations of vibrational modes and three orthogonal component motions (two horizontal and one vertical) meet the provisions of the Regulatory Guide.

The Spent Fuel Pool reracking project implemented in 1994 increased the fuel storage capacity from 1170 fuel assemblies to 1632 fuel assemblies. The reracking project retained 3 existing high density Exxon Nuclear Corporation modules containing 300 cells (Region I) and added 9 new maximum density Holtec modules containing 1332 cells (Region II) with a total storage capacity of 1632 fuel assemblies. The reracking provided an additional 10 years of storage capacity, which is expected to be sufficient up to the year 2008 for Unit 1 and 2012 for Unit 2.

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

- a. A maximum Keff equivalent of 0.95 with the storage racks filled with unborated water.
- b. A nominal 10.5 inch center-to-center distance between fuel assemblies stored in Region I (flux trap type) racks.
- c. A nominal 9.05 inch center-to-center distance between fuel assemblies stored in Region II (non-flux trap) racks.
- d. Fuel assemblies stored in Region I racks shall meet one of the following limiting conditions.
 1. Unirradiated fuel assemblies with a maximum enrichment of 4.25 w/o U-235 have unrestricted storage.
 2. Unirradiated fuel assemblies with enrichments greater than 4.25 w/o U-235 and less than or equal to 5.0 w/o U-235, that do not contain Integral Fuel Burnable Absorber (IFBA) pins, may only be stored in the peripheral cells facing the concrete wall.
 3. Unirradiated fuel assemblies with enrichments (E) greater than 4.25 w/o U-235 and less than or equal to 5.0 w/o U-235, that contain IFBA rods with nominal 2.35 mg B-10/inch surface loading, and a number of IFBA rods equal to or greater than the number determined by the equation below, have unrestricted storage.

$$N = 42.67 (E - 4.25)$$
 4. Irradiated fuel assemblies with enrichments (E) greater than 4.25 w/o U-235 and less than or equal to 5.0 w/o, that have attained the minimum burnup (BU) as determined by the equation below, have unrestricted storage.

$$BU \text{ (MWD/kg U)} = -26.212 + 6.1677E$$

INSERT C (Continued) To Section 9.1.2.1, P. 9.1-3

e. Fuel assemblies stored in Region II racks shall meet one of the following limiting conditions.

1. Unirradiated fuel assemblies with a maximum enrichment of 5.0 w/o U-235 may be stored in a checkerboard pattern with intermediate cells containing only water or non-fissile bearing material.
2. Unirradiated fuel assemblies with a maximum enrichment (E) of 5.0 w/o U-235 may be stored in the central cell of any 3x3 array of cells provided the surrounding eight cells are empty or contain fuel assemblies that have attained the minimum burnup (BU) as determined by the equation below.

$$\text{BU (MWD/kg U)} = -15.48 + 17.80E - 0.7038E^2$$

In this configuration, none of the nine cells in any 3x3 array shall be common to cells in any other similar 3x3 array. Along the rack periphery, the non-fueled region is equivalent to 3 outer cells in a 3x3 array.

3. Irradiated fuel assemblies with a maximum enrichment (E) of 5.0 w/o U-235 that have attained the minimum burnup (BU) as determined by the equation below, have unrestricted storage.

$$\text{BU (MWD/kg U)} = -32.06 + 25.21E - 3.723E^2 + 0.3535E^3$$

4. Irradiated fuel assemblies with a maximum enrichment (E) of 5.0 w/o U-235 that have attained the minimum burnup (BU) as determined by the equation below, may be stored in a peripheral cell facing the concrete wall.

$$\text{BU (MWD/kg U)} = -25.56 + 15.14E - 0.602E^2$$

5. Design loads and load combinations meet the requirements of the Standard Review Plan, Section 3.8.4, Structural Design Criteria for Seismic Category I Structures Outside Containment and ASME Section III NF-3400.

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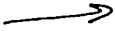
9.1.2.2 System Description

A stainless steel lined spent fuel storage pool is provided for onsite storage of spent fuel assemblies until they are shipped to a reprocessing facility.

Sufficient space is available to hold approximately 6 cores and the depth is sufficient to provide a minimum shielding depth over the top of the stored fuel of 10 feet of water.

The pool is designed to prevent inadvertent drainage below a water elevation of 124 feet-8 inches. Storage racks located in the pool are physically arranged such that the assemblies are always maintained in a subcritical condition. Adjacent to the spent fuel pool and separated by a structural wall is the transfer pool. The transfer pool serves to facilitate the fuel transfer operation between the Fuel Handling Building and containment. It is also the pool where the spent fuel cask is placed for shipment. The cask is handled by the cask handling crane which is prevented by structural restraint from moving over the spent fuel pool.

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The high density (poison) spent fuel racks' construction is shown on Figure 9.1-2. The design utilizes a stiffened module base and an upper box structure consisting of plate diaphragms and a top grid. The storage module is constructed of stainless steel, mostly Type 304. The vertical loads are carried by the module base. Horizontal seismic loads are carried to the module base through the plate diaphragms. Tipping is prevented by interconnection of adjacent racks by a bolted connection at the top grid level.

High Density

The design of the spent fuel storage cells is illustrated on Figure 9.1-3. Each cell is a square cross-section formed from an inner shroud of stainless steel, a center sheet of aluminum clad

6. During the design phase of the reracking in 1994, the following additional documents were used as a reference:
 1. OT Position paper for Review and Acceptance of Spent Fuel Pool Storage and Handling Applications, USNRC, 1978.
 2. USNRC Branch Technical Position ASB9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling," Rev. 2, 1981.

Sufficient space is available to hold approximately 8 full core offloads and the depth is sufficient to provide a minimum shielding depth over the top of the stored fuel of 10 feet of water.

Rack Modules for Region I (Retained Exxon Racks)

With the implementation of reracking in 1994, three previously installed Exxon racks were retained which are of poisoned flux trap construction to serve as Region I. Each rack module retained in the pool for this purpose consist of one hundred storage cells and has cross-sectional dimensions of 109.5" X 109.5". The locations of the existing modules are shown on Figures 9.1-3 and 9.1-3A.

boron carbide (B_4C), and an outer shroud of stainless steel. This cell acts as storage space and provides sufficient neutron absorption to allow close spacing of spent fuel. The fuel weight is carried directly on the module base. A flared guide and transition section is provided at the top of each storage cell. This transition is designed to assure ease of entry and to preclude fuel assembly hang-up and damage. Swelling of the inner stainless steel shroud has been observed in a number of the spent fuel storage cells. The swelling is the result of hydrogen gas buildup from the corrosion of the aluminum in the boral poison plates. The gas buildup has bowed or swollen the cells, thereby reducing the inner cell dimension. An ongoing program to monitor the condition of the spent fuel storage cells is being conducted. The hydrogen gas will be vented from the swollen cells to allow the shroud to return to some position closer to the original. This may allow the cell to be returned to service as an available spent fuel storage location. The hydrogen gas is radiologically stable and does not present a personnel hazard. The insignificant volume of gas released will not increase the hydrogen concentration in the area into the explosive range.

This condition was reviewed by the NRC in Supplement 4 of NUREG-0517, Safety Evaluation Report, Salem Generating Station Unit 2, April 1980. It was concluded that the minor degradation of the boral poison plates resulting from the corrosion of the aluminum would not preclude the spent fuel storage cells from performing their intended function.

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"G" → After a sufficient decay period the fuel may be removed from storage and loaded into a shipping cask for removal from the site.

9.1.2.3 Design Evaluation

Borated water is used to fill the spent fuel storage pool at a concentration to match that used in the reactor cavity and refueling canal during refueling operations. The fuel is stored in a vertical array with sufficient center-to-center distance

Rack Modules for Region II (New Holtec Racks)

With the implementation of reracking in 1994, nine existing Exxon racks were replaced with nine maximum density Holtec racks, shown on Figure 9.1-2B. The locations of new rack modules are shown on Figures 9.1-3 and 9.1-3A. These racks have a single poison panel between adjacent austenitic stainless steel surfaces and serve as Region II. The significant components of these racks are: 1) Box cell assembly, 2) Boral panel and Sheathing, 3) Formed and Periphery cells, 4) Baseplate, 5) Support legs.

See Figures 9.1-3C, 9.1-3D, 9.1-3E and 9.1-3F.

1) Box cell assembly

The box cells are fabricated from two precision formed channels by continuous seam welding. The inside (nominal) dimension of the box is 8.86 x 8.86 inches. Each box constitutes a storage location. See Figure 9.1-3C.

2) Boral panel and sheathing

Boral is used as the neutron absorber material. The boral panels are manufactured by using a homogenized particulate mixture of Boron carbide and aluminum powder sandwiched between thin aluminum sheets using a hot rolling process. The boral panels are placed in the customized flat depression region of the sheathing, which is laid on a side of the "box". The flanges of the sheathing are attached to the box on all four sides using intermittent weld. The sheathing serves to locate and position the boral panel accurately, and to preclude its movement under seismic conditions. See Figures 9.1-3D and 9.1-3E.

3) Formed and periphery cells

The boxes with integrally connected sheathing, are arranged in a checkerboard array to form the storage cell rack module. This way, Formed cells (interior boxes) are automatically created. The inter-box welding and pitch adjustment are accomplished by small longitudinal connectors. Flat plates are welded to the edges of the boxes at the outside boundary of the boxes to create the periphery cells. See Figure 9.1-3C.

4) Baseplate

The baseplate provides a continuous horizontal surface for supporting the fuel assemblies. See Figure 9.1-3F.

5) Support legs

All support legs are adjustable type. The top portion is made of austenitic stainless steel material. The bottom portion is made of SA564 type 630 age hardened stainless steel to avoid galling problems. See Figure 9.1-3F.

between assemblies to assure $k_{eff} \leq 0.95$ even if unborated water is used to fill the pool. (Based on 17 x 17 fuel with 4.05 w/o enrichment). as described in Section 9.1.2.1

The spent fuel storage pool is provided with a Spent Fuel Cooling System which is discussed in Section 9.1.3. The system maintains pool temperature below approximately 120°F .
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The design of the Fuel Handling Building is such that it is physically impossible for a load greater than 5 tons to be carried over the spent fuel pool. This is a result of both the physical arrangement of the Fuel Handling Building and limits on the fuel handling crane. Administrative controls prohibit loads greater than that of a fuel assembly to travel over the spent fuel pool. The maximum height at which a fuel assembly can be carried is restricted by limit switches on the crane to 15 inches over the top of the spent fuel racks. The spent fuel racks have been designed to absorb the energy released by a fuel assembly dropping from 15 inches above them.

except during implementation of the rerack project.

for the ~~Holtec~~ Exxon racks and 36 inches for the Holtec racks.

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The spent fuel storage pool and new fuel storage pit are outside the area over which the fuel cask may travel by design (travel restricted by a limit stop switch). The cask handling crane travels only over the truck bay, decontamination pit and fuel transfer pool, as indicated on Figure 9.1-1.

Gamma radiation is continuously monitored in the Fuel Handling Building. A high level signal is alarmed locally and is annunciated in the Control Rooms.

All fuel and waste storage facilities are contained and equipment designed so that accidental releases of radioactivity directly to the atmosphere are monitored and will not exceed the guidelines of 10CFR100.

A Controlled Ventilation System removes gaseous radioactivity from the atmosphere in fuel and waste treating areas of the Fuel

During implementation of the rerack project in 1994, temporary modification was made to the crane which increased the capacity of the crane from 5 tons to 20 tons. No permanent attachment or welding to the crane was made. This temporary modification was removed at the completion of the project and the crane was brought back to it's original capacity of 5 tons.

Special procedures were also implemented to prevent the possibility of a rack under transport from impacting stored fuel. Racks were also temporarily located in the transfer pool (two for Unit 1, one for Unit 2) to reduce the amount of fuel in the Spent Fuel Pool.

Handling and Auxiliary Buildings and discharges it to the atmosphere via the plant vent. Radiation monitors are in continuous service in these areas to actuate high-activity alarms in the Control Rooms.

^{"I"} INSERT → 9.1.3 Spent Fuel Pool Cooling System

9.1.3.1 Design Bases

The following description is for Unit 1 with Unit 2 having an identical system.

The Spent Fuel Pool Cooling System is designed to remove from the spent fuel pool the heat generated by stored spent fuel elements. The system serves the spent fuel pool which is located in the Fuel Handling Building adjacent to the Containment Building. A secondary function is to clarify and purify spent fuel pool, transfer pool, and refueling water. The system design considers the possibility that during the life of the plant it will become necessary to totally unload a reactor at the time when spent fuel is in the fuel pool.

The system design incorporates redundant active components. System piping is arranged so that failure of any pipe line does not drain the spent fuel pool below the top of the stored fuel elements.

^{"J"} INSERT → The spent fuel pool water is normally limited to 120°F except in the unloading of a full core, in which case the temperature is limited to 150°F with one pump in operation. Boron concentration in the pool fluid is maintained at a minimum of 2,000 ppm.

9.1.3.2 System Description

The schematic diagram for the Spent Fuel Pool Cooling System is shown on Figures 9.1-4A and B. The Spent Fuel Pool Cooling System

A nuclear criticality accident due to a fuel assembly misloaded in the Spent Fuel Pool has been analyzed. This relates to fuel assemblies loaded in a wrong region as specified in Section 9.1.2.1. One mislocated fuel bundle has been found to be acceptable as long as the soluble boron concentration is maintained above 600 ppm in the Spent Fuel Pool.

A nuclear criticality accident due to the installation of maximum density spent fuel storage racks has been analyzed. Reracked pool design factors that could affect the Spent Fuel Pool neutron multiplication factor have been addressed conservatively. It was concluded that the maximum Spent Fuel Pool neutron multiplication, with the addition of the maximum density racks, will not exceed the subcriticality limit of Keff less than or equal to 0.95 with unborated water.

A rack-to-rack and rack-to-wall impact during a postulated seismic event was analyzed for the spent fuel storage racks for the as-built configuration. The analysis concluded that the rack configuration does not result in rack-to-rack impact in the cellular region for either unit, or rack-to-wall impact for Unit 1 during postulated seismic events. For Unit 2, a rack-to-wall impact is predicted in one location for the DBE case. The calculated value of the impact force is very small and will not cause any damage to the fuel cells, the rack, the wall or pool liner.

The Spent Fuel Pool water is normally limited to 149 degrees F except in the unloading of a full core, in which case the temperature is limited to 180 degrees F with one pump in operation.

Boron may then be added to the pool from the Chemical and Volume Control System (CVCS). Borated water from the plant sources may be supplied from the RWST via the refueling water purification pump connection, or by placing a temporary line from the boric acid blender, located in the CVCS directly into the pool. Demineralized water is also added to the pool for makeup purposes by a connection in the recirculation return line.

The pool water may be separated from the water in the transfer pool by a sluice gate. The gate is installed so that the transfer pool may be drained for maintenance on the fuel transfer equipment. The draining is accomplished by pumping transfer pool water into the spent fuel pool with a portable pump. The excess water from the spent fuel pool is directed to a holdup tank in the CVCS or to the decontamination for temporary storage.

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An evaluation has been performed to determine the capability of the Spent Fuel Cooling System to provide the cooling capacity required for both the annual discharge of 65 fuel assemblies and for a full core discharge of 193 fuel assemblies into the spent fuel pool after 15 years' accumulation of spent fuel. ANS Standard 5.1 was used for decay heat load calculations. It has been determined that the Spent Fuel Pool Cooling System can provide the necessary cooling for the normal annual discharge as early as 100 hours after reactor shutdown. A full core discharge can be accomplished with an appropriate time delay after reactor shutdown, which is dependent on the number of regions stored in the spent fuel pool at the time. For example, it has been calculated that with one pump running at design capacity and with 150 hours of decay heat (after reactor shutdown) at the 18th refueling, the maximum spent fuel pool outlet water temperature will be 134°F. For the full core addition to the spent fuel pool that fills the pool (15 prior annual refuelings), the required decay cooling time in the reactor vessel that will be needed to keep the pool water temperature below 150°F with only one pump running will be approximately 570 hours (24 days) after reactor shutdown.

With the implementation of reracking in 1994 the spent fuel storage capacity in the pool was increased from 1170 fuel assemblies to 1632 fuel assemblies. The decay heat load calculation was conservatively performed in accordance with the provisions of USNRC Branch Technical Position ASB9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling," Rev. 2, July 1981. Three discharge Cases were considered in Cycle 25, 37 years' accumulation of spent fuel:

- Case 1) The reactor is shutdown and is cooled for 168 hours. Then, a batch of 88 assemblies with 1642.5 days full power operation is discharged to the pool at a rate of 7 assemblies per hour.
- Case 2) Same as case 1 except instead of 88 assemblies, 193 assemblies (Full Core Offload) is discharged.
- Case 3) The reactor is back to operation after normal refueling shutdown in Cycle 24. Thirty days later, the reactor experiences unplanned shutdown. The full core of 193 assemblies is transferred to the pool 168 hours after the reactor shutdown. Sixty eight assemblies in the core are assumed to have 30 days full power operation and 125 assemblies are assumed to have 1642.5 days full power operation.

The results of the above cases are as follows:

- Case 1) 195 hours after the reactor shutdown, the pool will experience a maximum temperature (bulk pool) of 149 degrees F, with one pump running.
- Case 2) 205 hours after the reactor shutdown, the pool will experience a maximum temperature of 180 degrees F, with one pump running.
- Case 3) 204 hours after the reactor shutdown, the pool will experience a maximum temperature of 180 degrees F, with one pump running.

positioned to take water from any elevation from the water surface to 4 inches below the surface. The elevation of the skimmers' head can be manually adjusted over a total range of 2 feet.

Spent Fuel Pool Skimmer Pump

The spent fuel pool skimmer pump circulates surface water through a strainer, a filter, and returns it to the pool.

Spent Fuel Pool Skimmer Strainer

The spent fuel pool skimmer strainer is designed to remove debris from the skimmer process flow.

Spent Fuel Pool Skimmer Filter

The spent fuel pool skimmer filter is designed to remove insoluble particles which are not removed by the strainer.

9.1.3.3 Design Evaluation

The most serious failure of this system would be complete loss of water in the spent fuel pool. To protect against this possibility, the spent fuel pool cooling suction connection enters near the normal water level so that the pool cannot be gravity-drained. The cooling water return lines contain anti-siphon holes to prevent the possibility of gravity draining the pool. There are no drains or permanently connected systems to the spent fuel pool (Seismic Class I) which, in the event of failure, could cause loss of coolant from the pool that would uncover the fuel. Also, provisions have been made to supply makeup to the spent fuel pool as noted below.

The rate of pool heatup with cooling interrupted for one-third of the core removed for refueling is approximately 6°F per hour at 150 hours after shutdown. The rate of heatup for a full core at the end of an operating cycle plus the one-third core removed at

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In the event of complete loss of Spent Fuel Pool cooling at the instant when peak pool water temperature occurs, the time-to-boil after loss of cooling would be 4.61 hours during a batch refueling discharge (88 assemblies) and would be 1.28 hours during a full core offload and would be 1.38 hours after an emergency full core discharge 30 days after a normal refueling discharge (68 assemblies).

the previous refueling is approximately 12°F per hour with cooling interrupted. An interruption in the operation of the Spent Fuel Cooling System for an extended period is not considered to be a credible occurrence. Maintenance will be scheduled when the decay heat loads are light. Two fully redundant, spent fuel pool pumps are provided, each receiving power from individual vital bus sections. As noted below, a number of makeup water sources are available and are capable of providing emergency cooling.

The use of the cross connect is controlled by appropriate procedures. Four manual valves, two per unit, have to be opened to cross connect the heat exchangers. Prior to placing both heat exchangers in parallel to minimize the temperature transient for a full core unload, the heat load in the spent fuel pool to be isolated is evaluated to ensure it can tolerate the temporary loss of cooling. Similarly, before taking one heat exchanger out of service for maintenance, the heat loads in both spent fuel pools are evaluated to verify that the remaining heat exchanger can be used in an alternating fashion to cool both spent fuel pools.

Water loss from the spent fuel pool due to the accidental opening of a sluice gate when the transfer pool is empty will not occur due to the redundancy in the sluice gates. Two sluice gates separate the spent fuel pool from the transfer pool.

A heavy load handling accident would not result in water leakage severe enough to uncover the spent fuel. The maximum load carried over the spent fuel pool is that of a fuel assembly; however, it is not possible to drop a fuel assembly on the spent fuel pool liner plate. In addition, integrity of the spent fuel pool will not be breached due to a fuel cask drop in the fuel transfer pool since each of the pool structures are separate and distinct.

Pool water level indication is provided by individual high and low water level alarms. The alarms are actuated by deviation from normal water level (Elevation 128 feet-8 inches) of plus or minus

Up to 100 gallons per minute of makeup is also available from the RWST via the refueling water purification loop.

If a leaking fuel assembly is stored in the spent fuel pool, a small quantity of fission products may enter the cooling water. Fission products and other contaminants are removed by the spent fuel pool purification loop.

A failure analyses of system pumps, heat exchangers and valves is presented in Table 9.1-3.

The spent fuel pool water is normally limited to ¹⁴⁹120°F except in unusual circumstances as previously described. Boron concentration in the pool fluid is maintained at a minimum of 2,000 ppm.

9.1.3.4 Tests and Inspections

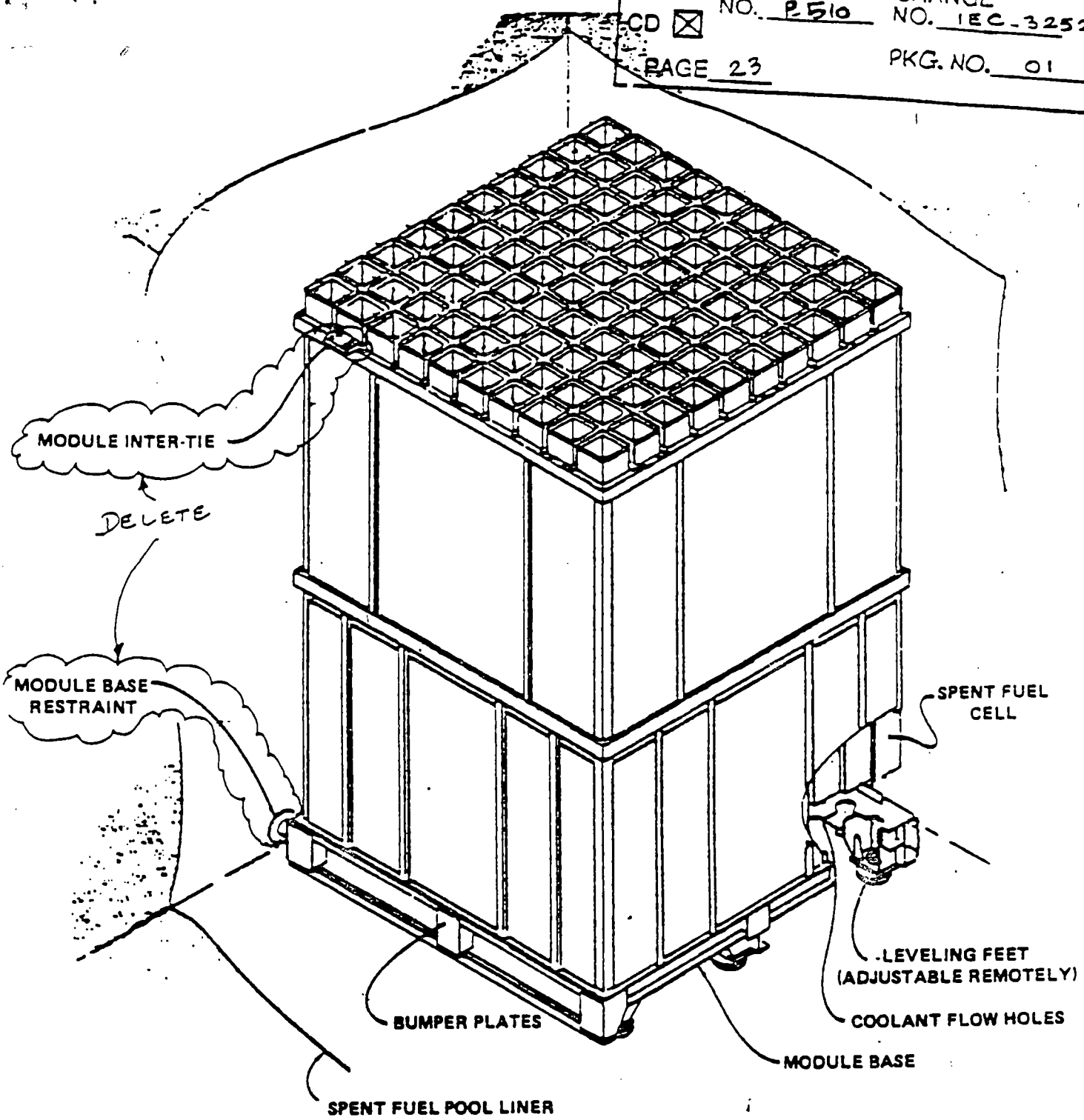
The active components of the system are in continuous use during normal plant operation and no additional periodic tests are required. Periodic visual inspections and preventive maintenance are conducted following normal industrial practice.

9.1.4 Fuel Handling System

The Fuel Handling System consists of equipment and structures utilized for handling new and spent fuel assemblies in a safe manner during refueling and fuel transfer operations. The Fuel Handling System is shown on Figure 9.1-5.

The design of the Fuel Handling System conforms to the recommendations of Regulatory Guide 1.13.

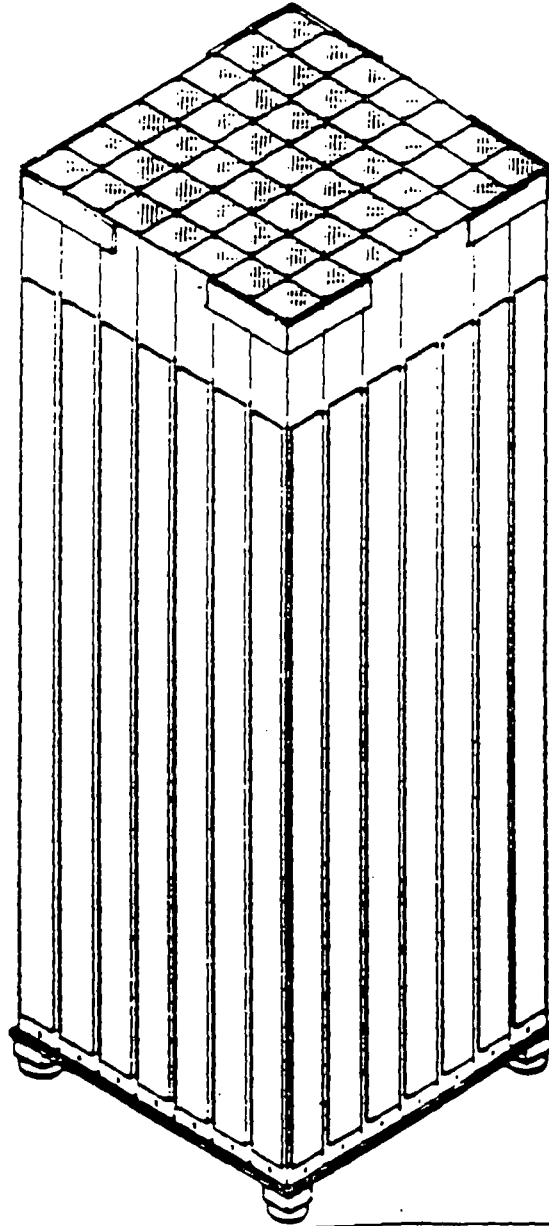
The Fuel Handling System components are not generally designed to Class I (seismic) requirements because they do not fit within the definition of Class I (seismic) structures. Those components are designed to Class III requirements. The spent fuel racks, spent



REVISION 6
 FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	REGION I High Density Poison Spent Fuel Storage Module Updated FSAR Figure 9.1-2A
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Ref
 4/2/94



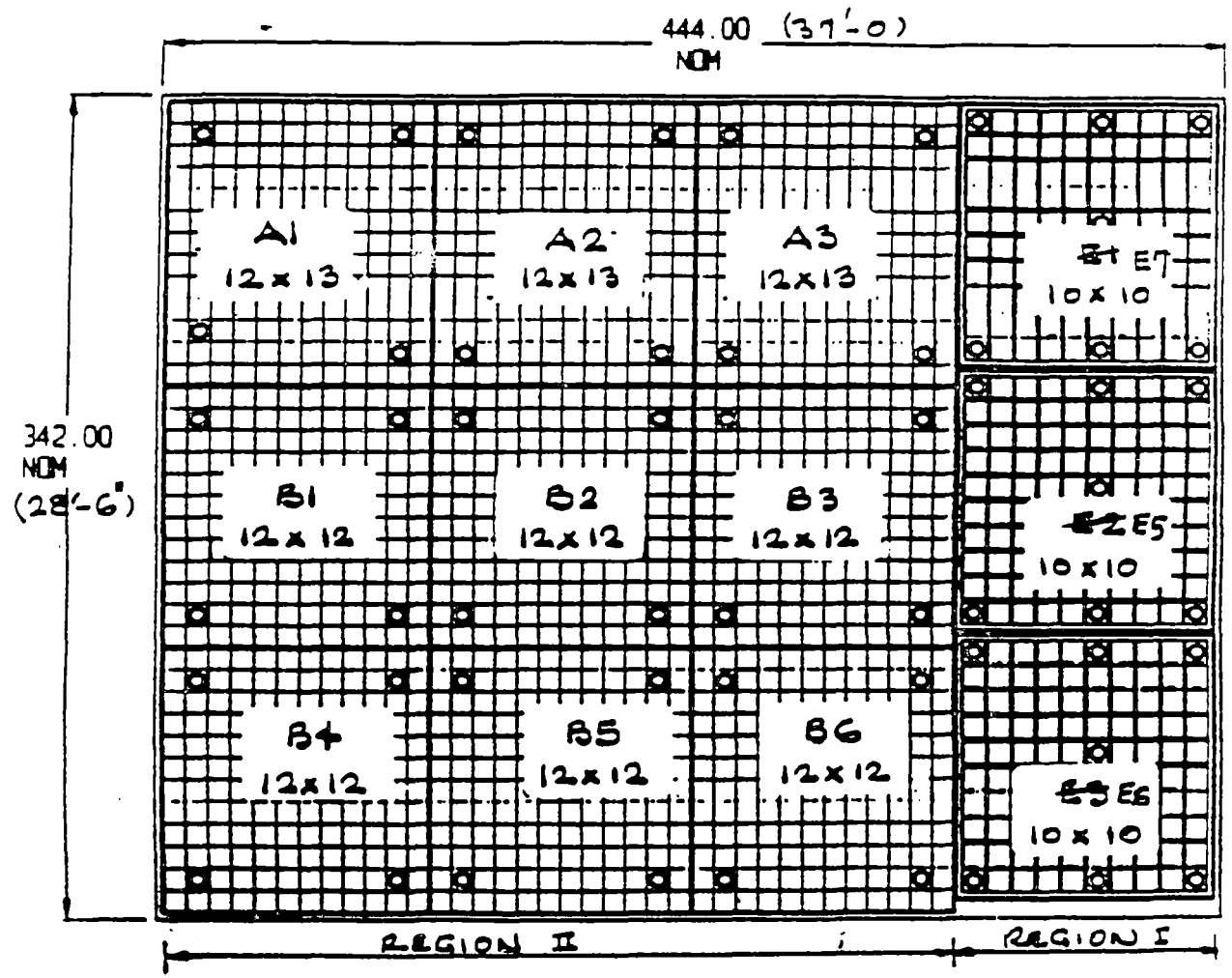
NEW UFSAR FIGURE 9.1-2B
MAXIMUM DENSITY SPENT FUEL STORAGE MODULE
REGION II

NEW (HOLTEC) RACK
PICTORIAL VIEW OF RACK STRUCTURE

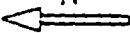
Feb 22-96

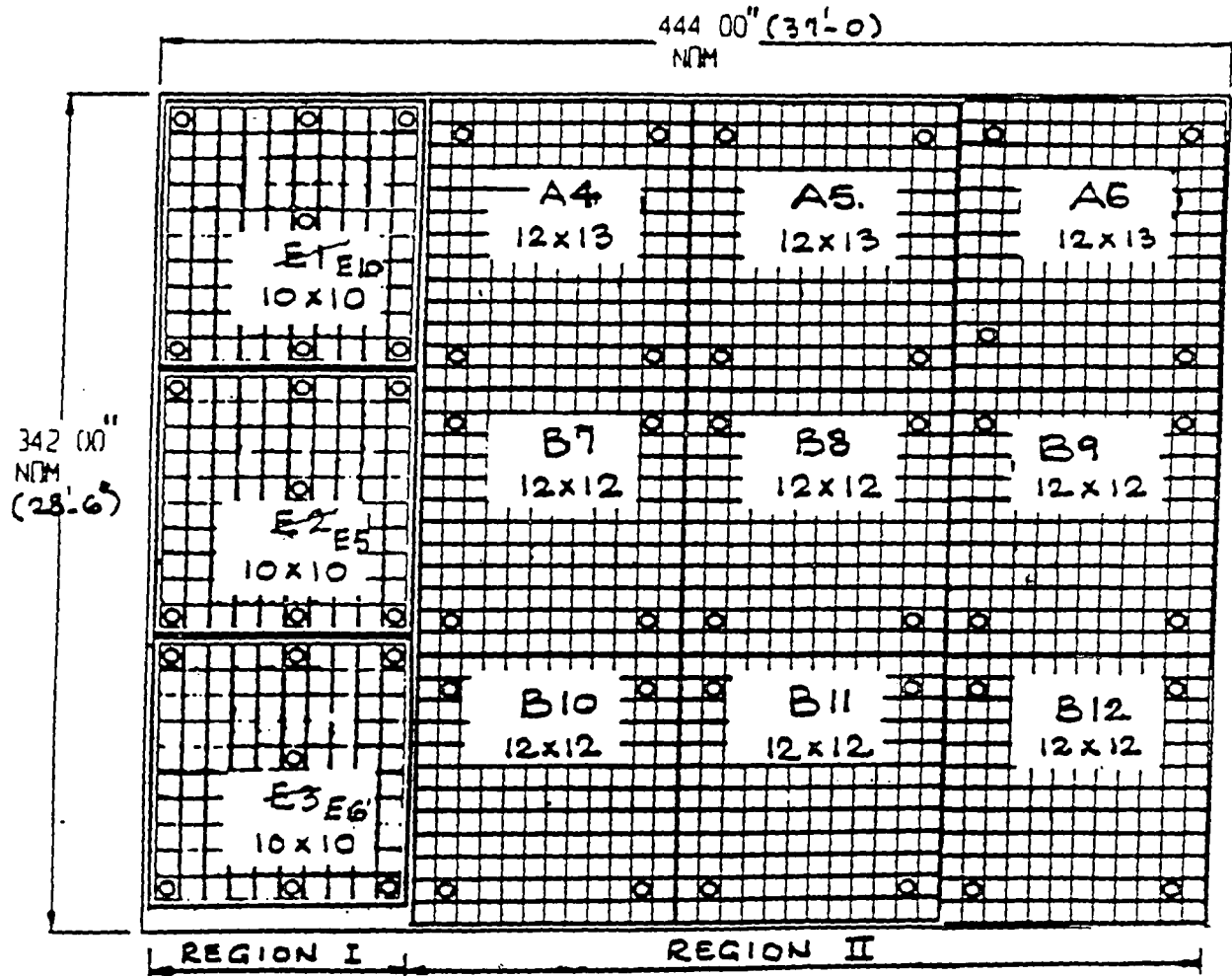
MD NO. P 510 CHANGE NO. IEC-3252
 CD PAGE 17, Rev. 1 PKG. NO. 01 REV. 3

N
 ←
 UNIT 1



INCREASED CAPACITY POOL LAYOUT (1632 CELLS)
 SALEM UNITS 1

N -

 UNIT 2



INCREASED CAPACITY POOL LAYOUT (1632 CELLS)
 SALEM UNIT. 2

~~3 x 10 x 10 = 300 CELLS~~
~~5 x 12 x 12 = 864 CELLS~~
~~3 x 13 x 12 = 468 CELLS~~
 = 1632 CELLS

NEW WFSAR FIGURE 9.1-3A

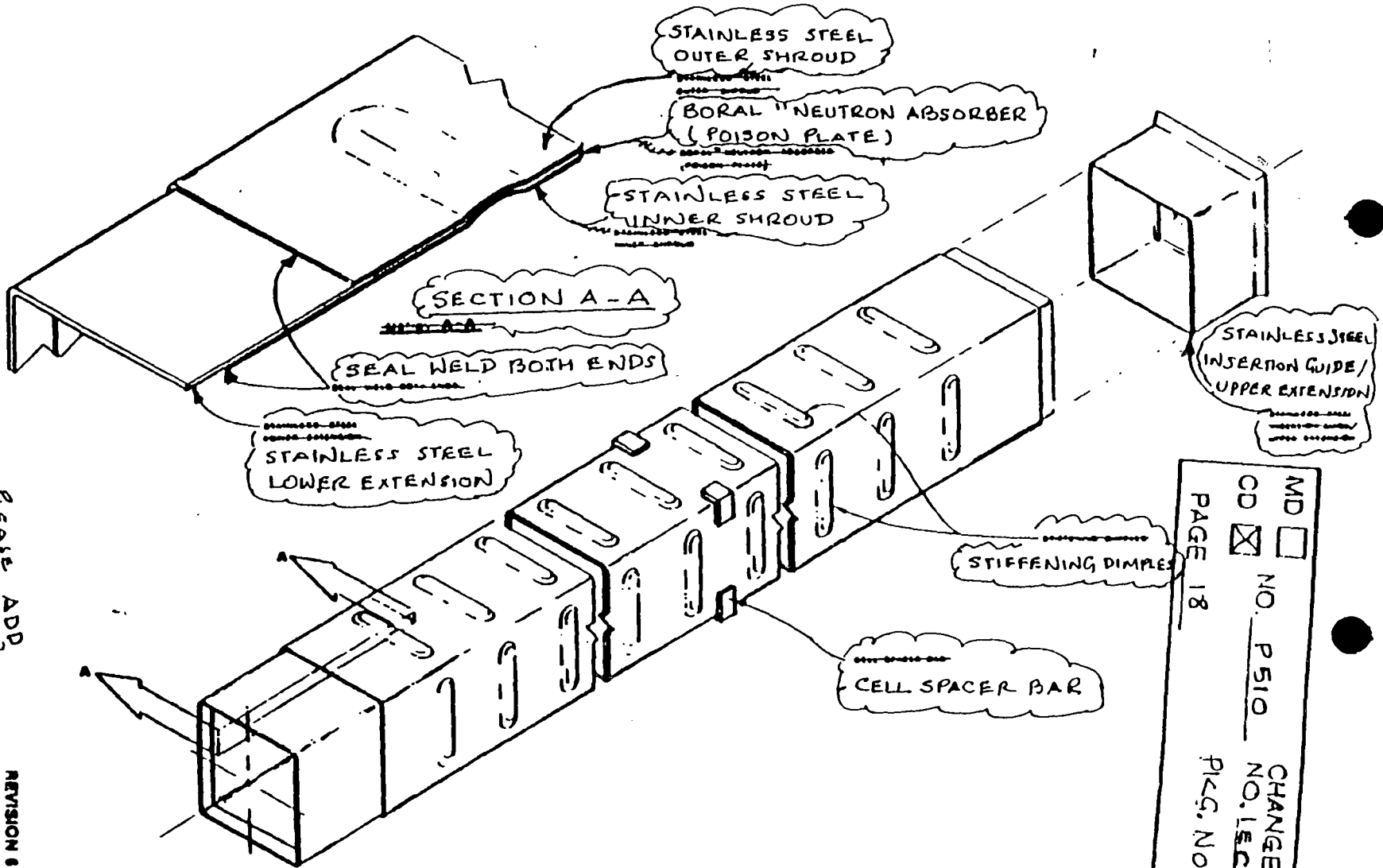
* UNIT 2 SPENT FUEL POOL RACK LAYOUT "

HIGH DENSITY REGION I
Poison Spent Fuel Storage Cell

ADD

PLEASE ADD

REVISION 8
FEBRUARY 15, 1987



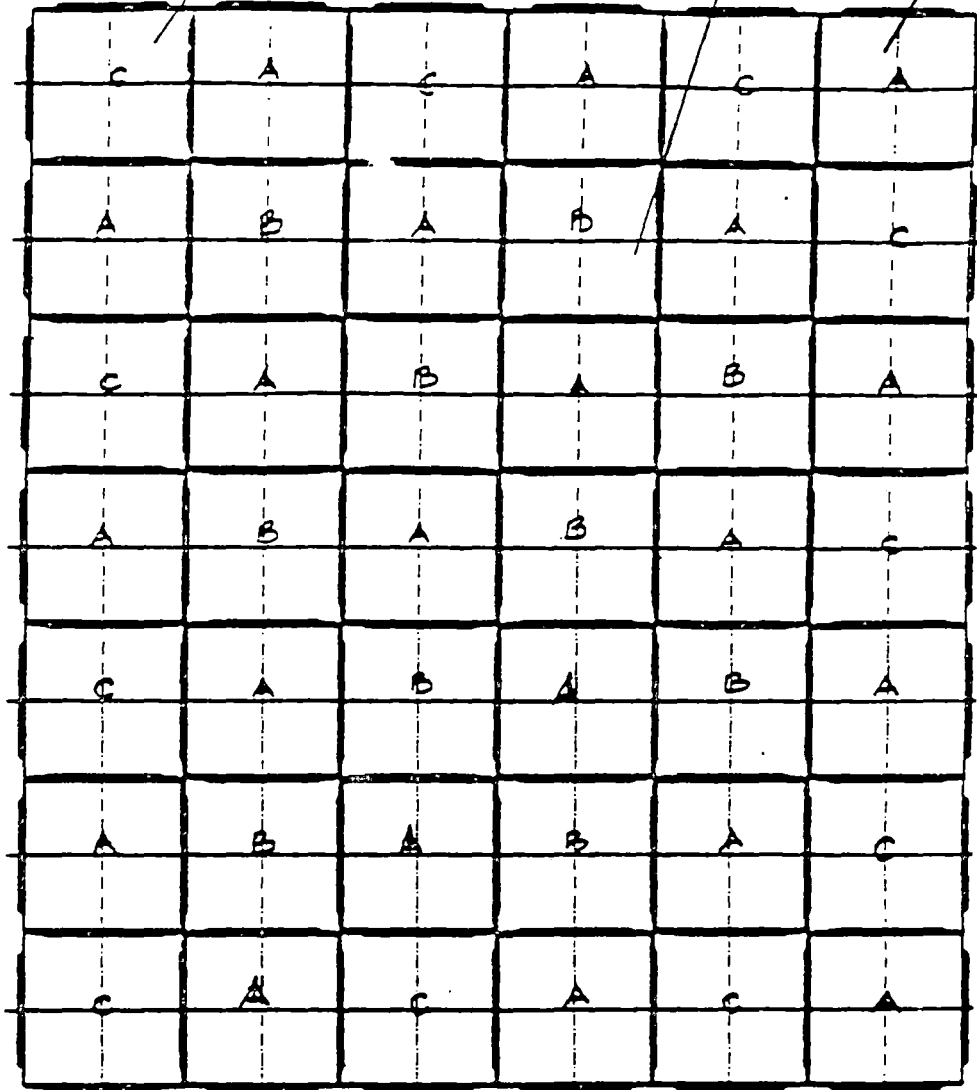
MD
CD NO. P510 CHANGE NO. 1 E.C.-3252
PAGE 18 PKG. NO. 01

(C) PERIPHERY
CELL
(TYP)

(B) FORMED CELL
(TYP)

(A) BOX CELL
(TYP)

MD CHANGE
CD NO. P 510 NO. IEC-3252
PKG. NO. 01
PAGE 19



TYPICAL ARRAY OF REGION II CELLS
(NON-FLUX TRAP CONSTRUCTION)

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
SALEM NUCLEAR GENERATING STATION

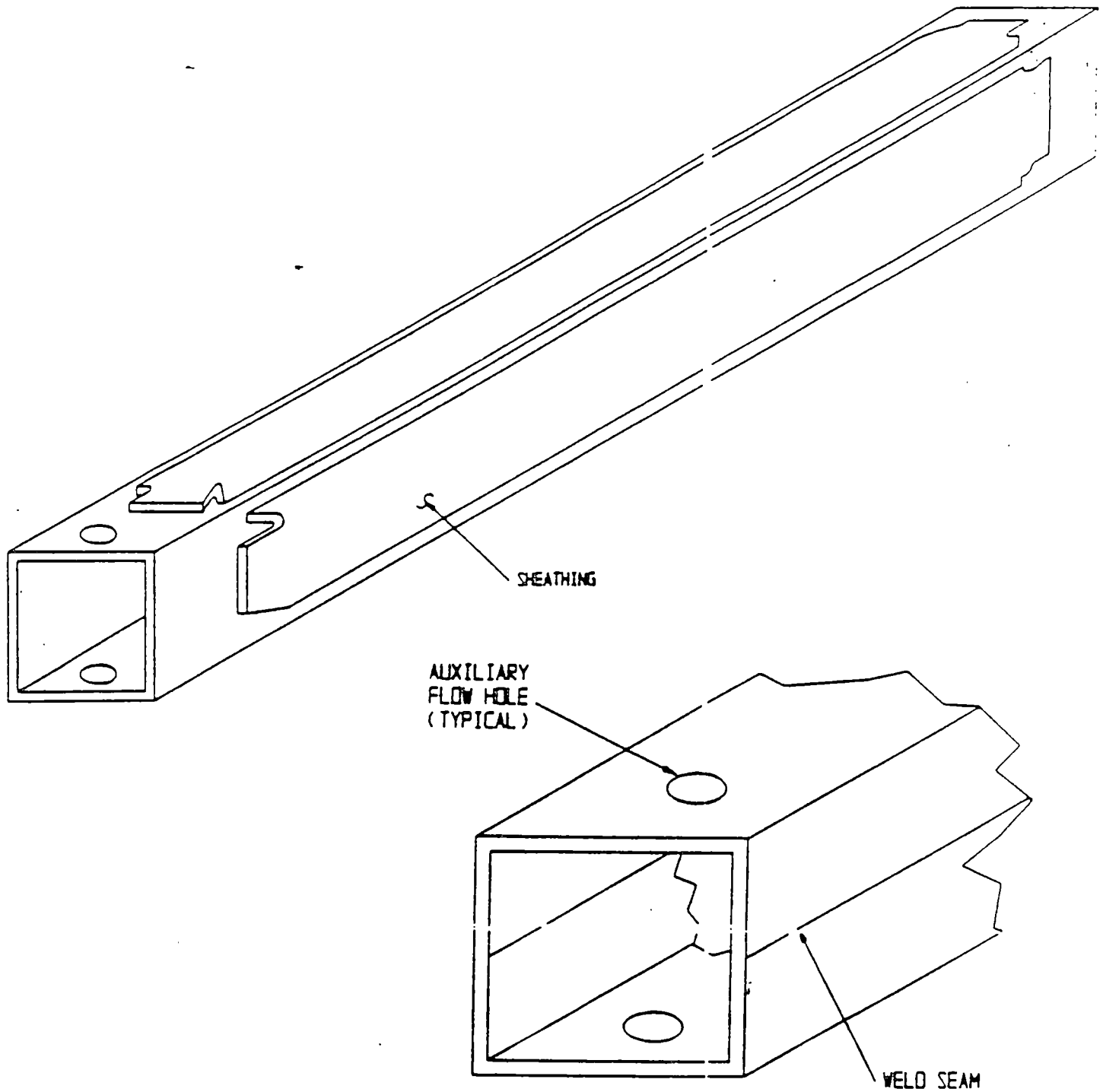
MAXIMUM DENSITY REGION II
SPENT FUEL STORAGE RACK ARRAY

Updated FSAR

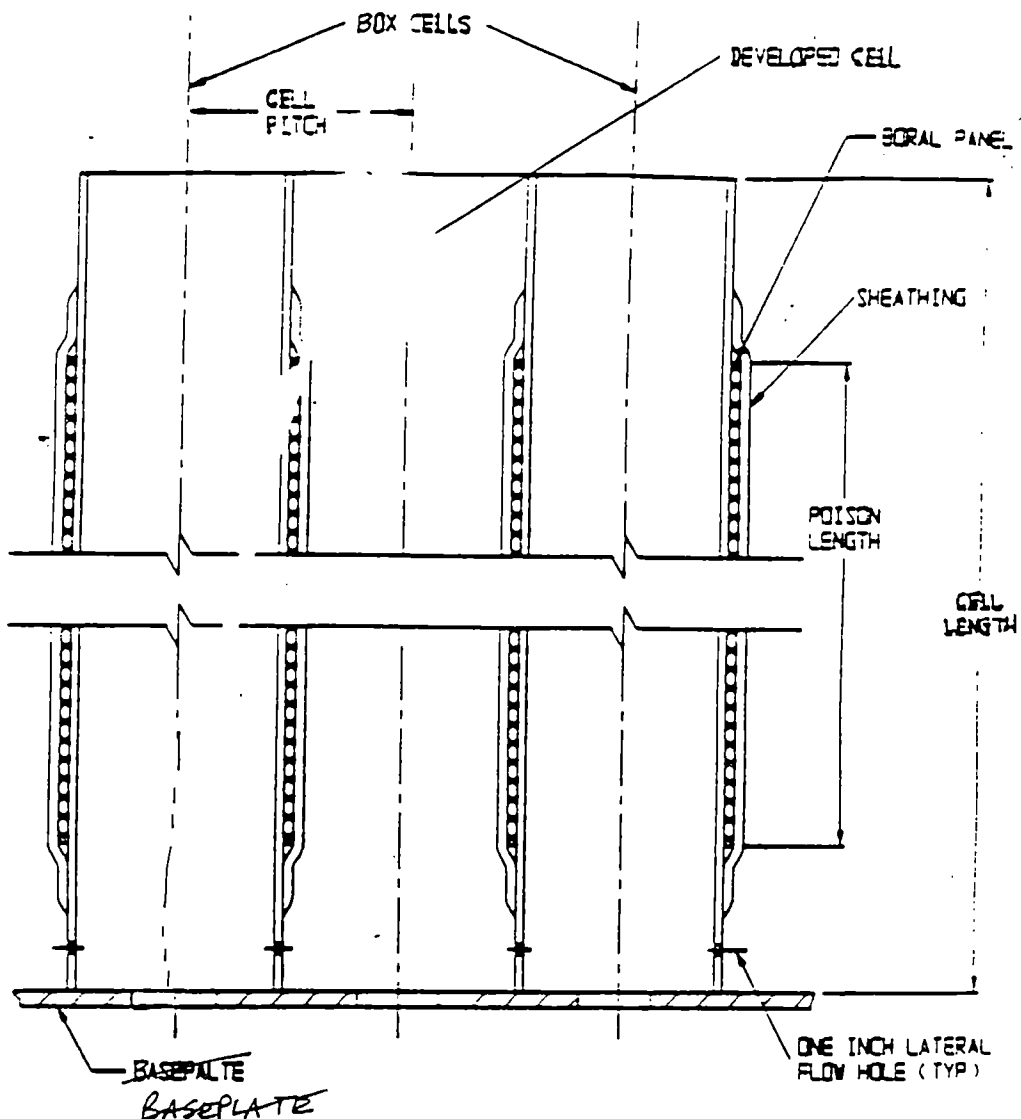
Figure 9.1-3C

led
4/2/96

MD <input type="checkbox"/>	NO. P510	CHANGE NO. I.E.C.-3252
CD <input checked="" type="checkbox"/>		
PAGE 20		PKG. NO. 01



MD
 CD
 CHANGE NO. P 510
 NO. IEC-3252
 PKG. NO. 01
 PAGE 21



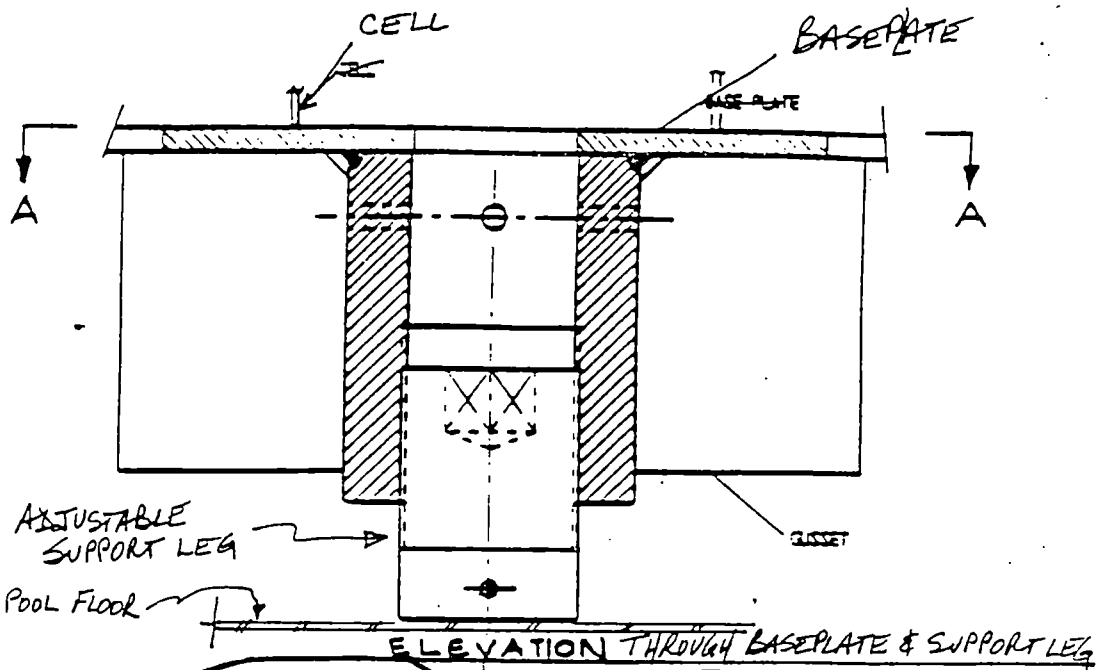
ELEVATION VIEW OF REGION II RACK MODULE

lil
4/19/96

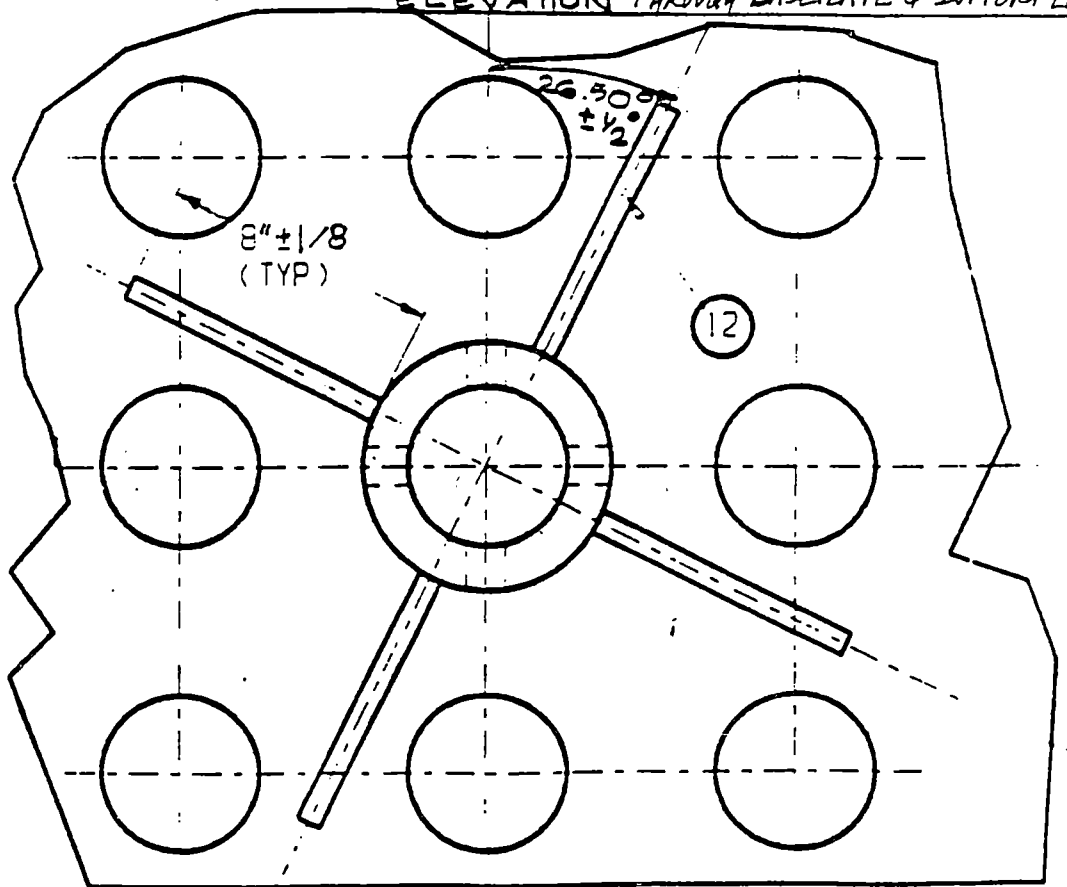
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
 SALEM NUCLEAR GENERATING STATION

MAXIMUM DENSITY REGION II
 SPENT FUEL STORAGE CELL Cross Section
 Updated FSAR Figure 9.1-3E

MD	<input type="checkbox"/>	NO. <u>P510</u>	CHANGE NO. <u>IEC-3252</u>
CD	<input checked="" type="checkbox"/>		PKG. NO. <u>01</u>
PAGE <u>22</u>			



Ref A-296



PLAN A-A

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	MAXIMUM DENSITY REGION II SPENT FUEL STORAGE BACK	<u>BASEPLATE</u>
	LOCATION PSAR	FIGURE 9.1-3F

LIST OF TABLES (Cont)

<u>Table</u>	<u>Title</u>
15.4-10	Nuclear Characteristics of Rated Discharged Assembly <i>Activity Released in the Fuel Handling Accident</i>
15.4-11	Activities in Highest Rated Discharge Assembly Curies at Time of Reactor Shutdown <i>Properties of Radionuclides Included in Fuel Handling Accident Analysis</i>
15.4-12	Parameters Used in the Analysis of the Rod Cluster Control Assembly Ejection Accident <i>Accident Analysis</i>
15.4-13	Summary of Heat Transfer Correlations Used to Calculate Steam Generator Heat Flow in the SATAN Code
15.4-14	Sensitivity of Core Stored Energy to Power Level, No. Nodes in Pellet and Fuel Densification
15.4-15	Mass and Energy Release to the Containment during Blowdown
15.4-16	Mass and Energy Release to the Containment during Reflood
15.4-17	Mass and Energy Release to the Containment during Reflood
15.4-18	Resistances Used in Froth Analysis
15.4-19	Post Reflood Mass and Energy Release
15.4-20	Passive Heat Sink
15.4-21	Structural Heat Sinks
15.4-22	Peak Pressure

15.4.6 Fuel Handling Accident

15.4.6.1 Identification of Causes and Accident Description

The accident is defined as dropping of a spent fuel assembly onto the spent fuel pit floor resulting in the rupture of the cladding of all the fuel rods in the assembly despite many administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor.

15.4.6.2 Analysis of Effects and Consequences

Method of Analysis

The following assumptions are postulated in the calculation of the radiological consequences of a fuel handling accident:

1. The accident occurs at 100 hours following the reactor shutdown, i.e., the minimum time at which spent fuel would be first moved into the spent fuel pit.
2. The accident results in breakage of all fuel rods in an assembly.
3. The damaged assembly is, coincidentally, the one operating at the highest power level in the core region to be discharged.
4. The power in this assembly and corresponding fuel temperatures establish the total fission product inventory and the fraction of this inventory which is present in the fuel pellet-cladding gap at the time of reactor shutdown.

INSERT

M

During the design phase of the reracking which was implemented in 1994, the potential radiological consequences resulting from a Fuel Handling Accident were evaluated. In performing this evaluation the following documents were used as a reference.

- 1) NUREG 1.25, Assumptions used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in The Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors, 1972.
- 2) NUREG/CR-5009, Assessment of the Use of Extended Burnup Fuel in Light Water Reactors, 1988.
- 3) ORNL Isotope GENERation and Depletion, ORNL/TM-7175, Oak Ridge National Laboratory, 1980.

Method of Analysis

Evaluation of this accident was based on the following data and assumptions:

- 1) The reactor was assumed to have been operating at 3600 Mw thermal power prior to shutdown, with an average specific power of 40.45 kw/kgU.
- 2) Initial enrichment of fuel considered is 4.5 wt% and burned to 65,000 Mwd/mtU.
- 3) The failed fuel cooling time considered prior to accident was 168 hours.
- 4) The fuel handling accident was assumed to result in the release of the gaseous fission products contained in the fuel/cladding gaps of all the 264 fuel rods in a peak-power fuel assembly.
- 5) Gap inventories of fission products available for release were estimated using the release functions identified in NUREG 1.25 except for Iodine-131. The release fraction for I-131 increased 20% in accordance with NUREG/CR-5009.
- 6) Core specific fission product inventories (Curies per metric ton of uranium) were estimated using the ORIGEN-2 Code. See Table 15.4-10 and 15.4-11.
- 7) The fission product gap inventory in a fuel assembly used in the thyroid dose calculation is Iodine-131, 12%; Other Iodines, 10%; Krypton-85, 30%; Other Krypton, 10%; Xenons, 10%.
- 8) The pool decontamination factor for iodine used is 500 and noble gases is 1.
- 9) The filter decontamination factor for iodine used is 100 and noble gases is 1.

- 10) The atmospheric diffusion factor used is $5.00E-4$ sec/m and breathing rate used is $3.47E-4$.

15.4.6.3 Conclusions

15.4.6.3.1 Radiation Doses

The doses at the Salem Exclusion Area Boundary (EAB) from the specified fuel handling accident are listed below. The doses are based on the release of all gaseous fission product activity in the gaps of all 264 fuel rods in the highest-power assembly.

Thyroid dose, rad	=	0.307
Whole-body dose, rem	=	0.483

These potential doses are "well within" the exposure guideline values of 10CFR part 100, paragraph 11.

15.4.6.3.2 Solid Radwaste

A significant increase in the volume of solid radioactive wastes is not expected as a result of expanded storage capacity.

15.4.6.3.3 Gaseous Releases

Gaseous releases from the fuel storage area are combined with other plant exhausts. Normally, the contribution from the fuel storage area is negligible compared to the other releases; therefore, significant increases are not expected as a result of the expanded storage capacity.

15.4.6.3.4 Personnel Exposures

No increase in radiation exposure to operating personnel is expected as a result of the expanded storage capacity; thus, neither the current radiation protection program nor the area monitoring system requires modification.

5. The fuel pellet-cladding gap inventory of fission products is released to the spent fuel pit water at the time of the accident.
6. The spent fuel pit water retains a large fraction of the gap activity of halogens by virtue of their solubility and hydrolysis. Noble gases are not retained by the water as they are not subject to hydrolysis reactions.

Fission Product Inventories

The fission product gap inventory in a fuel assembly is dependent on the power rating of the assembly and the temperature of the fuel. The parameters used for the calculation of the fission product inventory of the history rated assembly to be discharged are summarized in Table 15.4-10 while the associated activities at the time of shutdown are given in Table 15.4-11 for expected and conservative cases.

Iodine Decontamination Factors

An experimental test program was conducted to evaluate the extent of removal of iodine released from a damaged irradiated fuel assembly.

Iodine removal from the released gas takes place as the gas rises through the body of solution in the spent fuel pit to the pool surface. The extent of iodine removal is determined by mass transfer from the gas phase to the surrounding liquid and is controlled by the bubble diameter and contact time of the bubble in the solution.

In order to obtain all the necessary information regarding this mass transfer process, a number of released scale tests were conducted, using trace iodine and carbon dioxide in an inert carrier gas. Iodine testing was performed at the design basis solution conditions (temperature and chemistry) and data was

collected for various bubble diameters and solution depths. This work resulted in the formulation of a mathematical expression for iodine decontamination factor in terms of bubble size and bubble rise time.

Similar tests were conducted with carbon dioxide in an inert carrier, except that the solution temperature and chemistry were patterned after that of a deep pool where large scale tests were also performed with carbon dioxide. The small scale carbon dioxide tests also resulted in a mathematical expression for decontamination factor in terms of bubble size and bubble rise time through the solution.

To complete the experimental program, a full size fuel assembly simulator was fabricated and placed in a deep pool for testing, where gas released would be typical of that from the postulated damaged assembly. Tests were conducted with trace carbon dioxide in an inert carrier gas and overall decontamination factors were measured as a function of the total gas volume released. These measurements, combined with the analytical expression derived from small scale tests with carbon dioxide, permitted an in situ measurement of both the effective bubble diameter and rise time, both as a function of the volume of gas released. Having measured the characteristics of large scale gas releases, the decontamination factor for iodine was obtained, using the analytical expression from small scale iodine testing.

$$\text{Decontamination factor} = 7.3 e^{0.312 t/d}$$

where:

t = rise time

d = effective bubble diameter

The overall test results clearly indicate that iodine will be readily removed from the gas rising through the spent fuel pit

solution and the efficiency of removal will depend on the volume of gas released instantaneously from the full void space.

The pool decontamination factor for iodine is indicated to be a minimum of 760 for the 26-foot depth. Thus, a decontamination factor of 760 constitutes one parameter of the expected case.

In the conservative case, a lower decontamination factor is selected to provide for reasonable deviation in the factors which control iodine absorption by the pool water. For this analysis, a decontamination factor value of 500 is used, which is a reduction to 66 percent of the expected value. This factor of conservatism is selected on the basis that more than 90 percent of the experimental data for iodine would support such a value.

15.4.6.3 Conclusions

The activities available for release from the gap of the highest rated discharged assembly are given in Table 15.4-11 for expected and conservative cases. The resulting offsite doses from this accident may be calculated considering an expected and conservative pool decontamination factor of 760 and 500, respectively.

15.4.7 Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

15.4.7.1 Identification of Causes and Accident Description

This accident is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a RCCA and drive shaft. The consequence of this mechanical failure is a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

TABLE 15.4-10

NUCLEAR CHARACTERISTICS OF RATED
DISCHARGED ASSEMBLY

*DELETE THIS
TABLE.
REFER TO WITH
NEW TABLE 15.4-10
fil
4-2-96*

<u>Reactor Power</u>		<u>Expected Case (1)</u>
Rating (MWt)		3558
10 ² rating		3629.2
<u>Number of Assemblies</u>		193
Array		17 x 17
<u>Core Average Assembly Power</u>		
At 10 ² rating (MWt)		18.8
<u>Discharged Assembly (Highest Power)</u>		
Axial peak to average (ratio)		1:2
Peak power (kW/ft)		14.38
Maximum temperature (°F)		3600
Radial peak to average (ratio)		1:26
<u>Temperature/Power Distribution Local</u>		
<u>Temperature (°F)</u>	<u>Fuel Volume (%)</u>	<u>Power (MWt in Volume)</u>
>3400	0.4	0.1
3400 - 3200	1.8	0.5
3200 - 3000	3.2	0.9
3000 - 2800	3.7	1.0
2800 - 2600	5.0	1.4
2600 - 2400	6.0	1.7
2400 - 2200	6.1	1.7
2200 - 2000	6.4	1.8
<2000	67.4	18.8

NOTE:

(1) Use Regulatory Guide 1.25 for conservative case.

NEW TABLE 15.4-10

Ref
A-2-96

Table 9.1

(FIGURE TITLE)

RESULTS OF ORIGEN-2 CALCULATIONS FOR RADIONUCLIDES
OF IODINE, KRYPTON, AND XENON AT 168-HOURS COOLING TIME;
ACTIVITY RELEASED IN THE FUEL-HANDLING ACCIDENT

<u>Radionuclide</u>	<u>Curies per mtU</u>	<u>Activity Released, Ci</u>
I-131	6.231×10^5	5.860×10^4
I-132	3.586×10^5	2.810×10^4
I-133	8.077×10^3	6.330×10^2
I-134	-----	-----
I-135	4.515×10^{-2}	3.538×10^{-3}
Kr-85	1.646×10^4	2.276×10^3
Kr-85m	1.083×10^{-6}	8.487×10^{-8}
Kr-87	-----	-----
Kr-88	8.124×10^{-13}	6.367×10^{-14}
Xe-131m	1.135×10^4	8.895×10^2
Xe-133	1.033×10^5	8.096×10^4
Xe-133m	1.180×10^4	9.248×10^2
Xe-135	1.575×10^1	1.234×10^0
Xe-135m	7.232×10^{-3}	5.668×10^{-4}

TABLE 15.4-11

ACTIVITIES IN HIGHEST RATED
DISCHARGE ASSEMBLY CURIES AT
TIME OF REACTOR SHUTDOWN

*DELETE
THIS TABLE
REPLACE WITH
NEW TABLE
15.4-11*

4-2-96

Isotope	Expected Case (1)		
	Total Curies (x 10 ⁵)	Fraction in Fuel-Cladding Gap	Fuel-Cladding Gap Curies (x 10 ³)
I-131	6.75	0.0131	8.82
I-132	10.2	0.00144	1.48
I-133	15.1	0.00433	6.55
I-134	17.7	0.000893	1.58
I-135	13.7	0.00246	3.38
Kr-83m	1.26	0.00132	0.166
Kr-85	0.0767	0.218	1.67
Kr-85m	3.03	0.00198	0.601
Kr-87	5.82	0.00107	0.624
Kr-88	8.29	0.00158	1.31
Kr-89	10.7	0.00022	0.236
Xe-131m	0.0512	0.0159	0.0814
Xe-133	15.6	0.0106	16.5
Xe-133m	0.396	0.00697	0.276
Xe-135	4.26	0.00288	1.23
Xe-135m	4.19	0.000486	0.203
Xe-138	13.7	0.000507	0.696

NOTE:

(1) Use Regulatory Guide 1.25 for conservative case.

NEW TABLE 15-4-11

Table 9.2

Ref
A-296

PROPERTIES OF RADIONUCLIDES INCLUDED IN FUEL HANDLING
ACCIDENT ANALYSES PERFORMED BY HOLTEC
[Rev. 0, 1-93]

Radionuclide	Thyroid Dose Conversion, Rads/Curie ⁽¹⁾	E_p (Mev/dis)	E_r (Mev/dis)
Iodine-131	1.48×10^6	0.191 ⁽²⁾	0.381 ⁽²⁾
Iodine-132	5.35×10^4	0.392 ⁽³⁾	2.285 ⁽³⁾
Iodine-133	4.0×10^5	0.409 ⁽²⁾	0.612 ⁽²⁾
Iodine-134	2.5×10^4	0.455 ⁽³⁾	1.840 ⁽³⁾
Iodine-135	1.24×10^5	0.308 ⁽³⁾	1.775 ⁽³⁾
Krypton-85	-----	0.251 ⁽²⁾	0.002 ⁽²⁾
Krypton-85m	-----	0.213 ⁽³⁾	0.151 ⁽³⁾
Krypton-87	-----	1.050 ⁽³⁾	1.374 ⁽³⁾
Krypton-88	-----	0.341 ⁽³⁾	1.745 ⁽³⁾
Xenon-131m	-----	0.142 ⁽²⁾	0.020 ⁽²⁾
Xenon-133	-----	0.136 ⁽²⁾	0.047 ⁽²⁾
Xenon-133m	-----	0.185 ⁽²⁾	0.040 ⁽²⁾
Xenon-135	-----	0.303 ⁽³⁾	0.246 ⁽³⁾
Xenon-135m	-----	0.095 ⁽²⁾	0.428 ⁽²⁾

(1) Regulatory Guide 1.25 (Safety Guide 25), Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors, USNRC, 3/23/72.

(2) Personal communication, C. W. Reich, INEL, to S. E. Turner, Holtec International, August 16, 1991.

(3) C. M. Lederer, et al., Table of Isotopes, Sixth Edition, John Wiley & Sons, New York, 1967.