Houston Lighting & Power Company
South Texas Project Electric Generating Station-Unit 1
License Application for Unirradiated Power Reactor Fuel

1.1 Reactor and Fuel

1.1.1 This application is for a license to authorize the applicant to receive, possess, return, and store unirradiated nuclear fuel assemblies for use in the South Texas Project Electric Generating Station (STPEGS), Unit 1. The applicant is Houston Lighting & Power Company (HL&P) acting for itself and for City Public Service Board of San Antonio, Texas; Central Power and Light Company, Corpus Christi, Texas; and the City of Austin. HL&P is acting as project manager for the owners and is responsible for the design, construction, and operation of STPEGS. The applicant is not controlled by any alien, foreign corporation, or foreign government. Other information concerning the applicant is set forth in the application filed in NRC Docket Nos. STN 50-498 and STN 50-499. The license will be required prior to August 15, 1986, and is to be effective until such time as an operating license is issued.

STPEGS is located in south-central Matagorda County, Texas, west of the Colorado River, 8 miles north-northwest of the town of Matagorda, Texas and about 89 miles southwest of Houston, Texas. Docket number STN 50-498 has been assigned to STPEGS Unit 1. Class 103 construction permit CPPR-128 has been issued to STPEGS Unit 1.

1.1.2 The fuel assemblies to be stored consist of uranium dioxide ceramic pellets contained in slightly cold-worked Zircaloy-4 tubing plugged and seal-welded at the ends to encapsulate the fuel. The fuel rods are supported at intervals along their length by ten Type R grid assemblies which maintain the lateral spacing between the rods in the assemblies. The grid material is Inconel-718.

The fuel assemblies are comprised of fuel rods arranged into a square array of 17 rod locations per side. Fuel rods take up 264 positions in each fuel assembly. A total of 193 fuel assemblies are required to make up the core for the STPEGS reactor. The Zircaloy-4 tubing is supported by the grid assemblies in a 0.496 inch rod pitch.

Each fuel rod contains 168 inches of slightly enriched uranium dioxide in the form of pellets 0.3225 inch in diameter and 0.530 inch in length.

The cladding for each fuel rod is 0.374 inch in outside diameter and is 0.0225 inch thick. The diametral gap between the fuel and the cladding is 0.0065 inch.

In addition to the 264 fuel rods, each fuel assembly has the center position reserved for the incore instrumentation, while

the remaining 24 positions in the array are equipped with guide thimbles joined to the grids, top nozzles, and bottom nozzles.

Figure 1 depicts the cross sectional arrangement of typical fuel assemblies while in the core. Figure 2 illustrates top, bottom, and side views of a typical fuel assembly.

1.1.3 Uranium enrichment of the initial fuel load for the STPEGS Unit 1 will vary depending on the core region. The uranium in the assemblies of the region of highest enrichment will have a nominal maximum enrichment of 2.9 w/o U-235. Each such assembly contains approximately 35 lbs (approximately 16 kg) of U-235. (The initial core load contains no U-233, plutonium, natural uranium, depleted uranium, or thorium.) The total weight of a new fuel assembly, including structural material, is approximately 1730 lbs.

1.1.4 This application is for a full core load of 193 fuel assemblies for STPEGS Unit 1. The fuel will contain approximately 5100 lbs (approximately 2300 kg) of U-235. The fuel will contain no U-233, plutonium, natural uranium, depleted uranium, or thorium. In addition, up to eight (8) spare assemblies may be obtained, for a total of 201 assemblies.

1.2 Storage Conditions

1.2.1 Figures 3 and 4 are scale drawings which give the general arrangement of the Fuel Handling Building (FHB) where the fuel assemblies will be stored and inspected.

1.2.2 All new fuel handling activities for STPEGS will be conducted in the FHB. New fuel may be stored in four locations in the FHB: spent fuel pool; new fuel storage pit; new fuel handling area; new fuel inspection laydown area on the operating floor.

At least 127 assemblies of the initial load of fuel will be stored in the spent fuel pool. The spent fuel pool is a stainless steel-lined reinforced concrete pool and is an integral part of the FHB. While intended primarily for wet storage of spent fuel assemblies, the pool is also capable of dry storage of new fuel. Fuel assemblies will be stored in fuel rack modules providing 14-inch center-to-center spacing. All rack surfaces that come into contact with fuel assemblies are made of annealed austenitic stainless steel. These materials are resistant to corrosion during normal and emergency water quality conditions.

A maximum of 66 assemblies of the initial load of fuel will be stored dry in the new fuel storage pit. The new fuel storage pit is a reinforced concrete pit and is an integral part of the FHB. The fuel is stored in racks composed of individual vertical cells fastened together to form three 2 x 11 modules

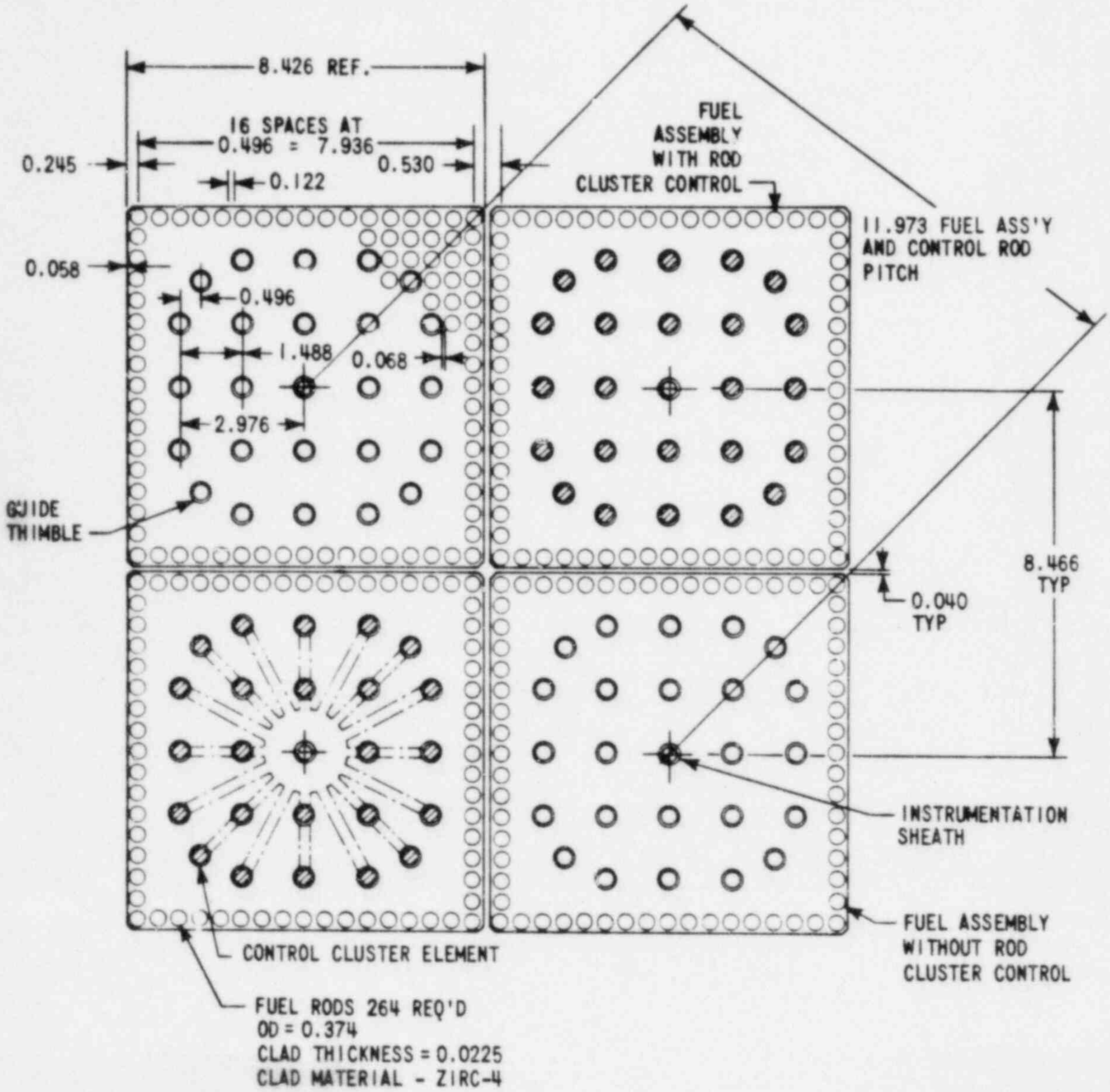


Figure 1
17 x 17 Fuel Assembly
Cross Section
(Dimensions in inches)

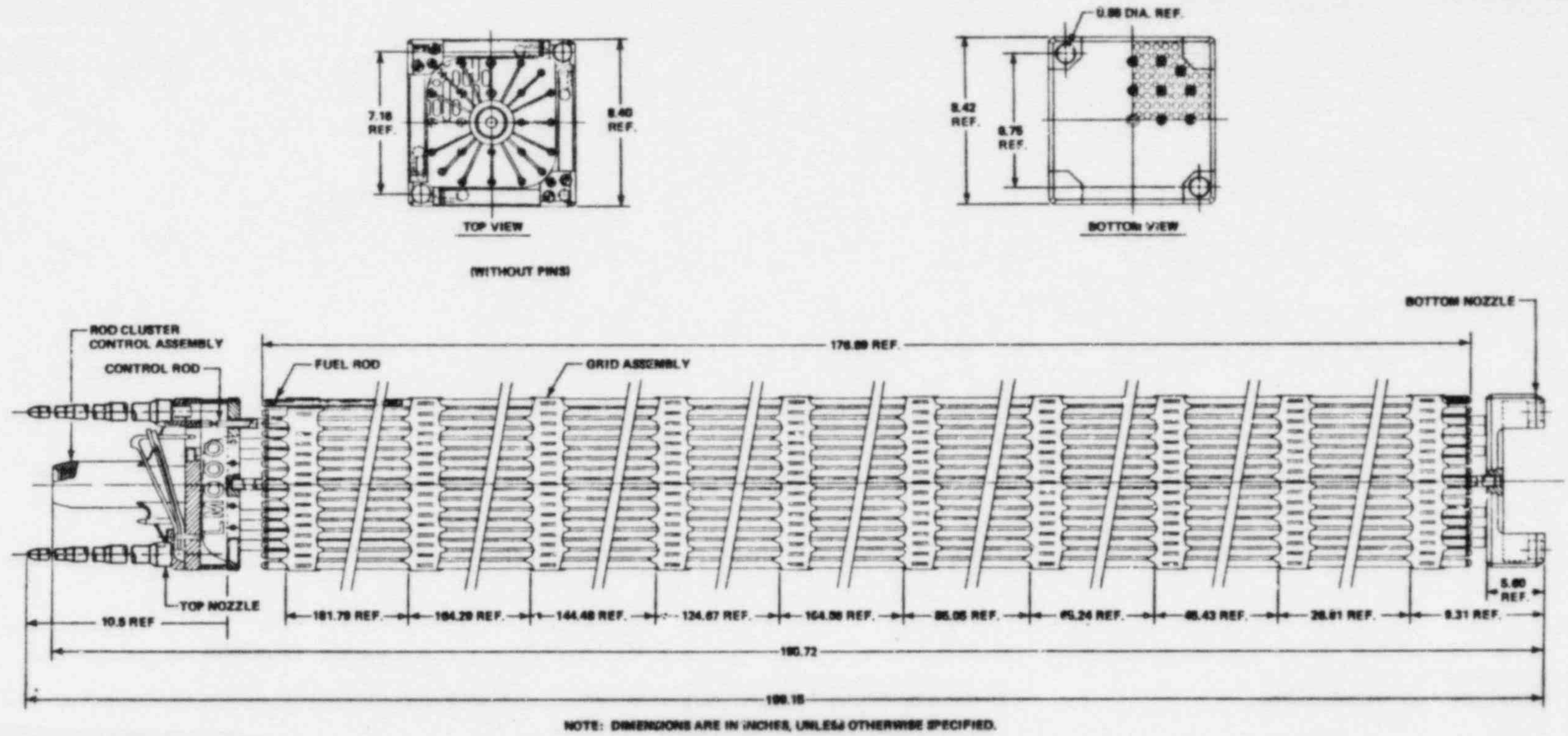
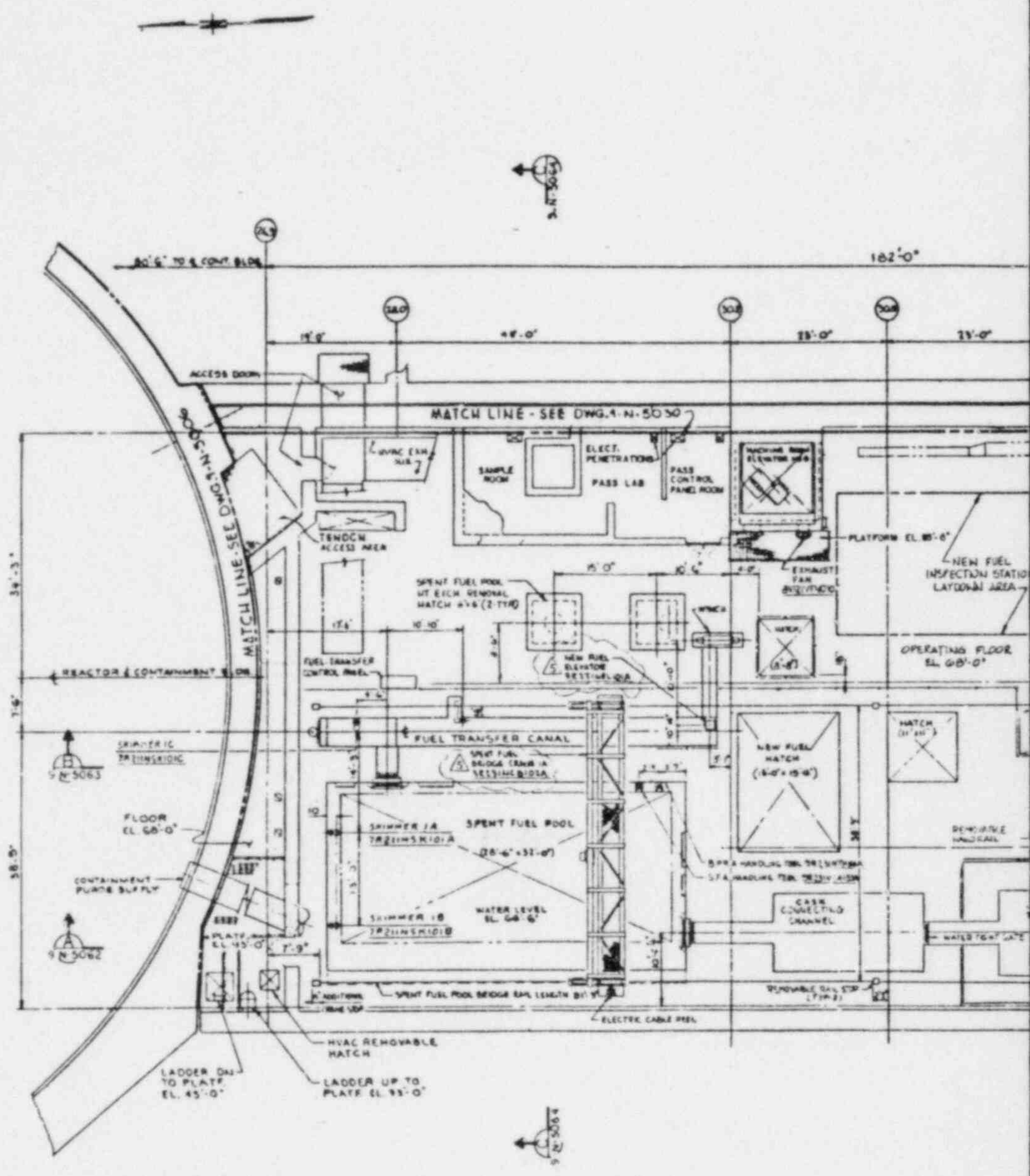
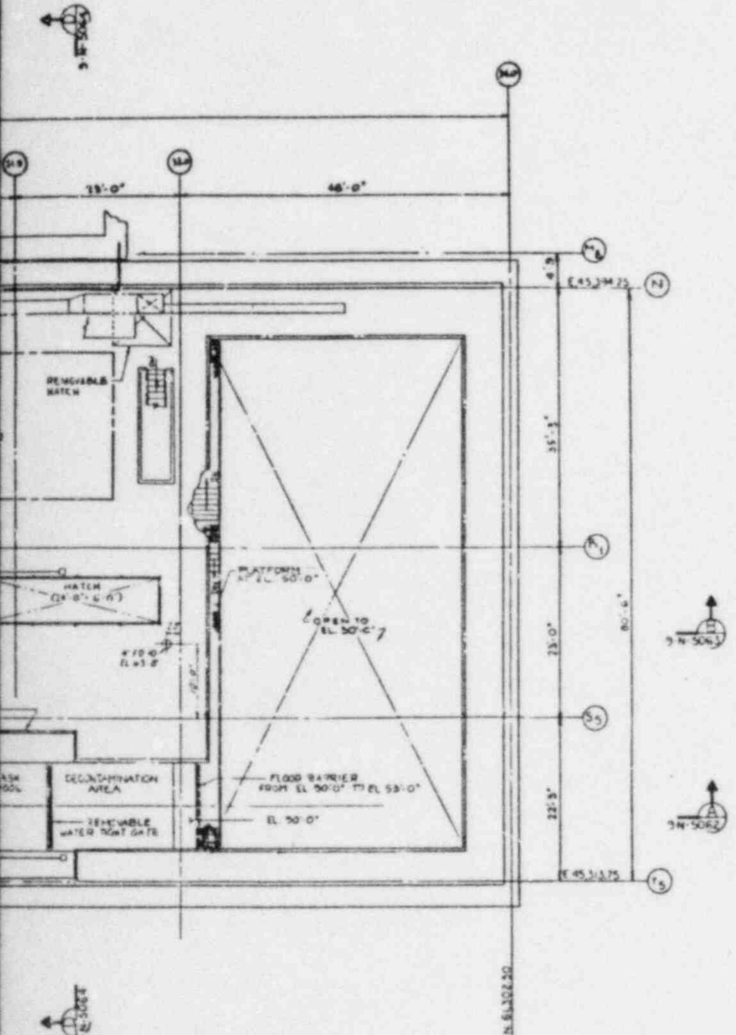


Figure 2: 17 x 17 XL Fuel Assembly Outline (Conceptual)



182'-0"
ACCESS DOOR
MATCH LINE - SEE DWG.9-N-5050
SAMPLE ROOM
ELECT. PENETRATIONS
PASS LAB
PASS CONTROL PANEL ROOM
TENDON ACCESS AREA
PLATFORM EL. 85'-0"
NEW FUEL INSPECTION STATION LAYDOWN AREA
OPERATING FLOOR EL. 68'-0"
FUEL TRANSFER CONTROL PANEL
REACTOR & CONTAINMENT BLDG
FUEL TRANSFER CANAL
NEW FUEL HATCH
SPENT FUEL POOL
WATER LEVEL
FLOOR EL. 68'-0"
CONTAINMENT PURGE SUPPLY
CASK CONNECTING CHANNEL
HVAC REMOVABLE HATCH
LADDER DN TO PLATE EL. 45'-0"
LADDER UP TO PLATE EL. 93'-0"
ELECTRIC CABLE REEL



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SOUTH TEXAS PROJECT
UNITS 1 & 2

GENERAL ARRANGEMENT
FUEL HANDLING BUILDING
PLAN EL. 68'-0" AREA M

Dwg. No. 6F-18-9-N-5060 Rev. 5

Figure 3

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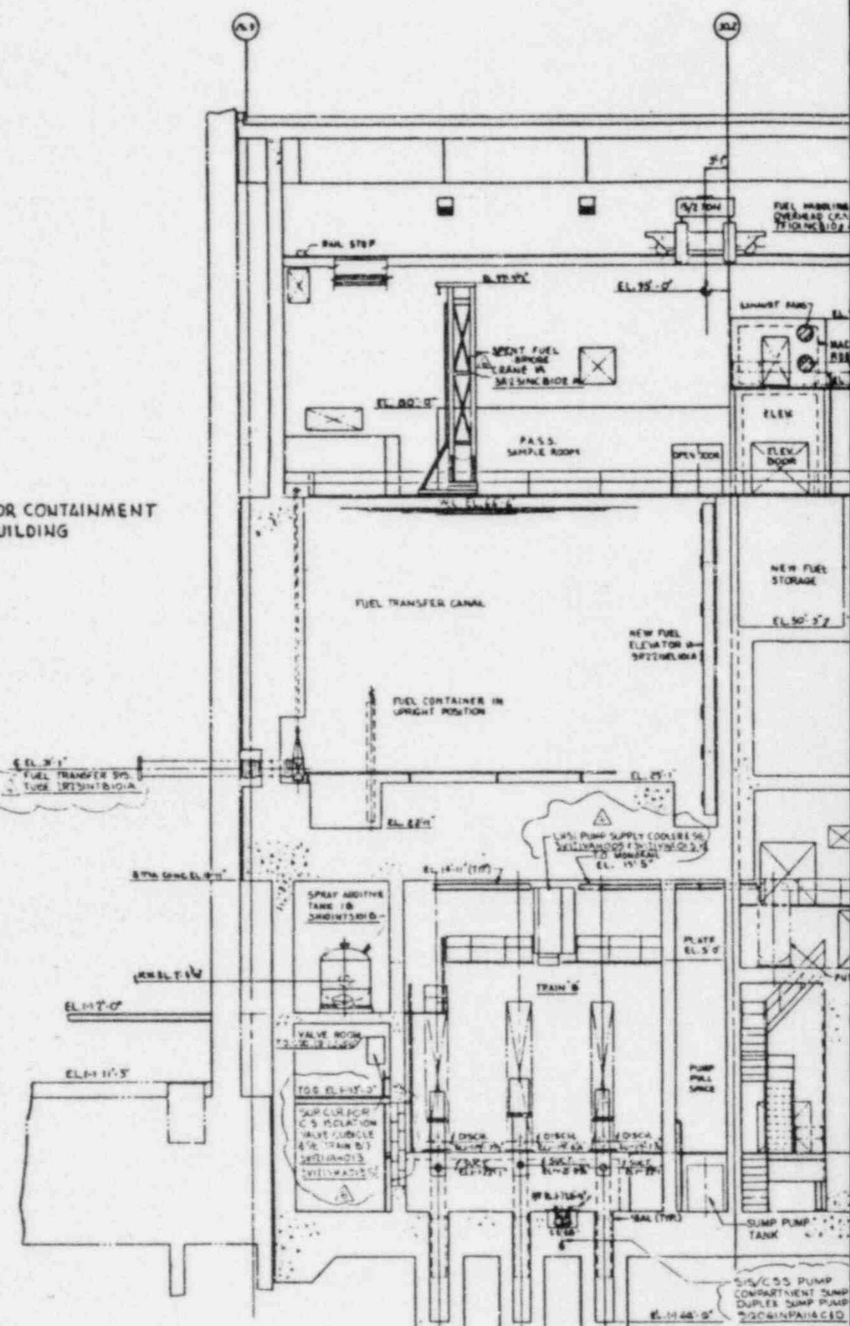
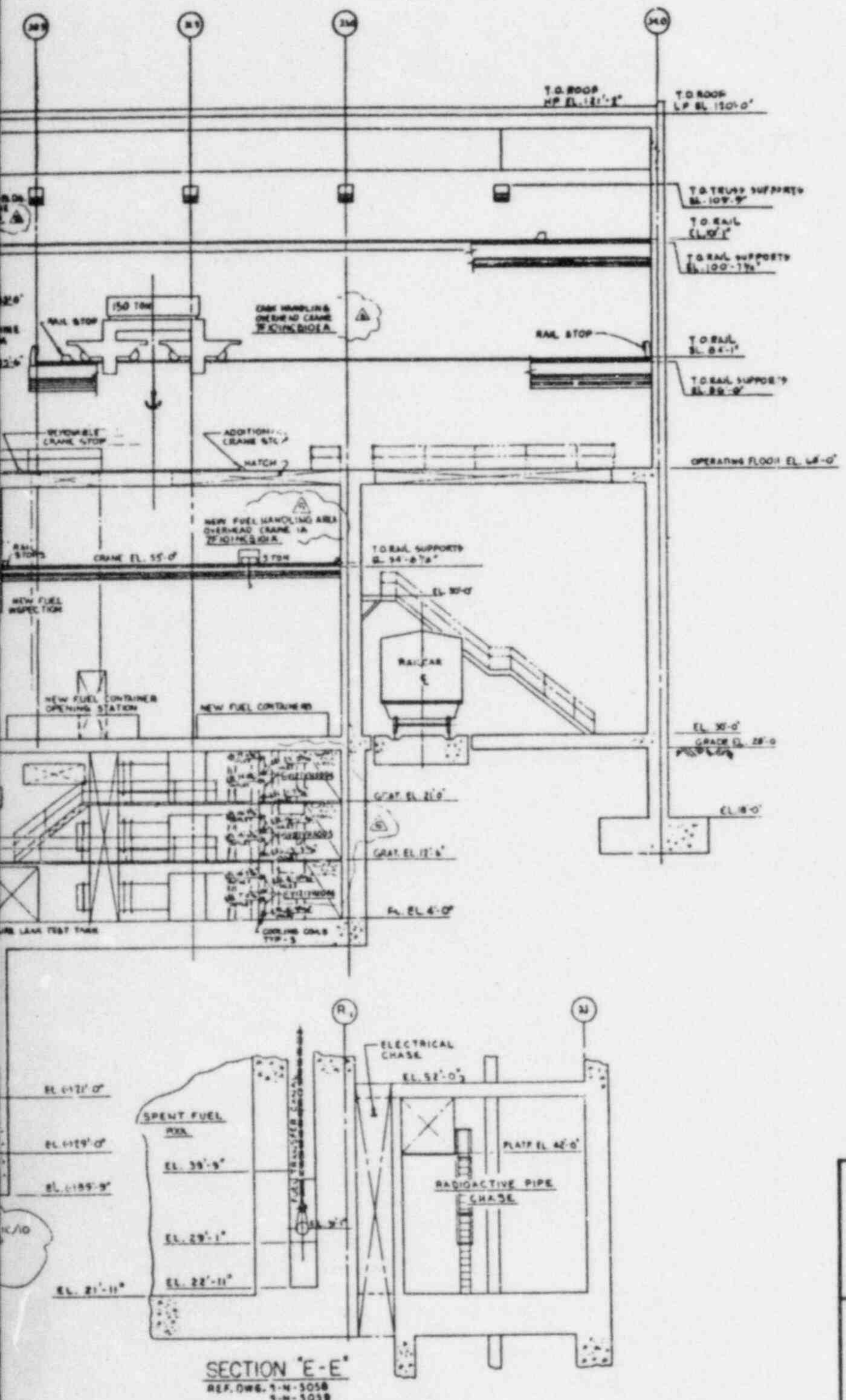


SECTION "B-B"
REF. DWG 9-N-5057, 9-N-5058, 9-N-5059, 9-N-5060.



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SOUTH TEXAS PROJECT UNITS 1 & 2

GENERAL ARRANGEMENT
FUEL HANDLING BUILDING
SECTION "B-B" & "E-E" AREA M

Dwg. No. 6F-18-9-N-5063 Rev. 6

Figure 4

8507150065 - 02

bolted to anchors in the floor and walls of the new fuel storage pit. The new fuel racks are designed with a center-to-center spacing of 21 inches, providing a minimum of 12 inches between adjacent fuel assemblies. All rack surfaces that come into contact with the fuel assemblies are made of annealed austenitic stainless steel, and the support structure is painted carbon steel. In addition, a three section cover is provided as the new fuel storage pit access hatch. This cover will minimize the introduction of dust and debris into the pit. The cover is designed to withstand the impact force of a new fuel assembly dropped from the maximum elevation allowed by the 2-ton hoist of the FHB overhead crane.

Following delivery, unloading, and examination for shipping damage, the shipping containers are then lifted to the operating floor by the FHB overhead crane. The shipping containers can be placed for temporary storage either directly in the new fuel inspection laydown area on the operating floor or lowered through one of the equipment hatches in the operating floor to the new fuel handling area below by means of the FHB overhead crane. The new fuel handling area is a room located below the operating level (68-ft elevation) of the FHB and adjacent to the new fuel storage pit. The storage area is connected to the operating level by two equipment hatches. The shipping containers are placed horizontally on the floor. The containers are stacked using the overhead crane servicing the area. One by one, these shipping containers are unstacked, their covers removed, and the pivotal, fuel support structure within the shipping container is elevated from horizontal to vertical by the overhead crane. The fuel assembly is removed from the shipping container support structure and then inspected for acceptability.

Following inspection, unacceptable new fuel assemblies are set aside for repackaging and return to the fuel fabrication plant. Acceptable assemblies are lifted by the FHB overhead crane and either: 1) inserted into the fuel storage racks in the new fuel storage pit or spent fuel pool, or 2) placed into the new fuel elevator, which is located in the fuel transfer canal. The elevator lowers the assemblies to the bottom of the fuel transfer canal from where they are engaged by the spent fuel handling tool suspended from the Fuel Handling Machine, then transferred to the storage racks in the spent fuel pool.

Equipment associated with fuel handling are as follows:

> <u>FHB Overhead Crane</u> - This 15/2-ton capacity crane is to be used for general handling operations in the FHB. It runs over the entire FHB area. The crane is designed to maintain its structural integrity under the dynamic loading of the SSE (safe-shutdown earthquake). The crane will retain its load under such dynamic loadings. The main hoist of this (15-ton capacity) crane is also provided with a redundant reeving system to preclude the

dropping of its load. With this redundancy, the main hoist can withstand a single failure without dropping its load and therefore meets the intent of RG 1.104.

The 2-ton hoist of this crane is designed to handle new fuel assemblies. Use of the 2-ton hoist ensures that the design uplift of the new fuel racks will not be exceeded. This hoist is used to move new fuel assemblies from the new fuel handling area and new fuel inspection laydown area to either the new fuel storage pit, the spent fuel pool, or the new fuel elevator.

New Fuel Handling Area Crane - The 5-ton new fuel handling area overhead crane is used for movement of new fuel assemblies and their shipping containers within the new fuel handling area. Dropping of new fuel assemblies due to SSE-induced dynamic loading of the crane will not result in an offsite radiological hazard; only localized contamination may result. The crane travels over no safety-related equipment. It is therefore not designed to accommodate SSE-induced dynamic loading without failure.

New Fuel Elevator - The new fuel elevator consists of a box-shaped elevator assembly with its top end opened and sized to house one fuel assembly. The new fuel elevator is used to lower a new fuel assembly to the bottom of the transfer canal.

Fuel Transfer Canal - The fuel transfer canal leads from the new fuel elevator to the spent fuel pool. It also connects to the refueling area of the containment.

Fuel-Handling Machine - The fuel-handling machine consists of an electric monorail hoist carried on the spent fuel pit bridge which is a wheel-mounted walkway serving the spent fuel pool and fuel transfer canal. The fuel handling machine is used exclusively for handling fuel assemblies and core components. The hoist has a maximum capacity of 2 tons.

Cask Handling Crane - This 150 ton crane is provided for movement of spent fuel shipping casks which are not moved over new or spent fuel storage areas.

Fuel Handling Tools -

New Fuel Assembly Handling Tool: The new fuel assembly handling tool is used to lift and transfer fuel assemblies from the new fuel shipping containers to the new fuel storage racks, and to transfer new fuel assemblies from the new fuel storage racks to the new fuel elevator. The tool weighs approximately 100 lbs. and is preoperationally tested at 125 percent of the weight of one fuel assembly and RCCA.

Spent Fuel Assembly Handling Tool: The spent fuel assembly handling tool is used to handle new and spent fuel assemblies in the spent fuel pool. The tool weighs approximately 400 lbs. and is preoperationally tested at 125 percent of the weight of one fuel assembly and RCCA.

Telescoping Spent Fuel Assembly Handling Tool: The telescoping spent fuel handling tool is a backup to the spent fuel handling tool. This tool is qualified for the same use as the spent fuel assembly handling tool.

1.2.3 The activities that will be conducted in the areas adjacent to where new fuel is being stored can be grouped into five categories. The categories and the potential effects on the safety of storage are as follows:

a. Preventive and corrective maintenance on equipment/systems in the Fuel Handling Building.

Potential Effects on the Safety of Storage - Minimal. Work will be performed by or under the supervision of HL&P Maintenance personnel. Any work to be performed within the security control area established to protect the fuel will require maintenance personnel to be authorized for entry to the area.

b. Construction completion activities to resolve discrepancy items on equipment/systems turned over to the Startup Group or to complete minor work on equipment/systems in preparation for turnover.

Potential Effects on the Safety of Storage - Minimal. The security control area has been established to ensure that any work in adjacent areas will not damage the fuel. In addition, work to resolve discrepancy items on equipment/systems turned over to the Startup Group will be controlled by the Startup Work Request Form. This form requires approval by the Startup Engineer prior to starting work. Personnel entering the security control area shall be required to have authorization for entry to the area.

c. Startup testing on equipment/systems that could include checkout, flushing, hydrostatic testing, and preoperational testing.

Potential Effects on the Safety of Storage - Minimal. Testing will be performed by or under the supervision of assigned system Startup Engineers who will ensure that the testing activities have no impact on the stored fuel. Testing support for the Startup Engineer will be provided by maintenance technicians, maintenance mechanics, operators, chemistry technicians, or construction personnel as appropriate for the testing being performed. Personnel performing work within the security control area established to protect the fuel shall be required to have authorization for entry.

d. Routine housekeeping.

Potential Effects on the Safety of Storage - Minimal. Housekeeping activities will be planned and supervised to ensure that the activities have no detrimental effect on the stored fuel. Chemicals used in the vicinity of the new fuel will be reviewed for compatability with fuel materials prior to use.

e. New fuel receipt and inspection.

Potential Effects on the Safety of Storage - None. Receipt and inspection of new fuel will be conducted in accordance with approved procedures and under the direction of a qualified individual.

1.2.4 A description of structures and equipment used in storing and handling fuel is provided in Section 1.2.2. This section provides the design criteria for these structures and equipment.

The Fuel Handling Building is designed as a controlled-leakage Seismic Category I structure. The building enclosure consists of reinforced-concrete walls and roof slab capable of resisting tornado-generated missiles. The wall thickness is designed to be at least equal to the missile penetration depth.

Seismic Category I structures are designed to withstand the maximum flood levels and associated effects by:

1) Having external walls and slabs of structures designed to resist the hydrostatic and hydrodynamic forces associated with surge-wave runup and steady-state water level.

2) Ensuring the overall stability of the total structure against overturning and sliding due to the hydrostatic and hydrodynamic forces associated with the surge-wave runup and steady-state water level.

3) Ensuring that the total structure will not float due to buoyancy forces.

All exterior FHB openings are located above the maximum steady-state flood level or are equipped with waterproof, interlocking double doors when located below this elevation. The exception is the opening for the rail car in the spent fuel cask loading area of the FHB. The door is not protected from flooding because the area does not have any safety-related systems and components. In addition, the area is separated from the remainder of the building by walls which do not contain openings below the maximum surge-wave runup height for the location.

The bases for the seismic analyses of the FHB are the foundation motions developed from the finite element method for soil/structure interaction (SSI) analyses. Seismic analysis of the FHB

has been performed using the modal time-history method. The seismic response within the FHB is determined by using the motions calculated at the base of the foundations from the finite element SSI analyses (first-step analysis) as input to the detailed three-dimensional lumped-parameter mathematical model of the building (second-step analysis). Time-history acceleration records are obtained from the second step analysis at all major floor levels and other locations necessary for the seismic analyses of systems and components. Response spectra are calculated for each of these time-history records for subsequent use in modal response spectrum analyses for subsystems. Seismic analyses of the FHB have been performed using the computer program STRUDL DYNAL.

Design of the FHB is in accordance with Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," Revision 1, which ensures adequate safety under both normal and postulated accident conditions. Spent fuel racks are designed to withstand handling, normal operating loads, and SSE and OBE seismic loads meeting Safety Class 3 and American Institute of Steel Construction (AISC) requirements. The spent fuel racks are also designed to meet Seismic Category I requirements of RG 1.13. The new fuel racks are designed to withstand operating loads as well as SSE and OBE seismic loads meeting Safety Class 3 and AISC requirements. The new fuel racks are designed to withstand maximum uplift force of 5,000 lbs.

Codes and standards used in the design of fuel-handling equipment are as follows:

1) Cranes: Crane Manufacturers Association of America (CMAA) specification no. 70, Class A-1.

2) Structural: ASME Code, Section III, Appendix XVII.

3) Electrical: Applicable standards and requirements of:

   National Electric Code

   National Fire Protection Association no. 70

   National Electrical Manufactures Association standards MGI and ICS

4) Materials: Materials conform to the specifications of the American Society for Testing Materials.

5) Safety: Applicable requirements of Section 1910.179 of Subpart N of the Occupational Safety and Health Act Code.

6) Others:

   American Institute of Steel Construction

   American National Standards Institute

American Society of Testing Materials

Institute of Electrical & Electronic Engineers

National Electric Manufacturers Association

Occupational Safety & Health Administration

American Welding Society

Expansion Joint Manufacturers Association

ASME B&PV Code Sections VIII and XI

American Concrete Institute

Hydraulic Institute Standards

1.2.5 Early detection of fire in the FHB is made possible by the fire detection system. The primary operation of this system is automatically governed by a series of local control panels located throughout the plant. These control panels monitor the general plant area fire detectors and monitor and/or control all of the special hazard fire protection systems to ensure their continuous availability. System design and installation complies with the requirements of NFPA 72E, "Automatic Fire Detectors," and NFPA 72D, "Proprietary Protective Signaling System," except that the underground ring main system post indicator valves are locked open and administratively supervised.

The primary fire suppression facility is a standpipe and hose system, supplied from the underground ring main and the internal ring main in the Mechanical Auxiliary Building. The number of hose stations provided is the optimum required, and their locations are determined by the physical arrangement of the FHB and the accessibility of the hazards. The system design is such that all locations in the building are within reach of at least one hose stream. Backup fire protection is provided by portable fire extinguishers. Their type, size, number, and locations are governed by the nature of the hazards.

The Fire Protection System is designed utilizing the basis of NRC guidelines of Appendix A to Branch Technical Position APCSB 9.5.1, the recommendations of the American Nuclear Insurers (ANI), the applicable standards of the National Fire Protection Association (NFPA), and the requirements of GDC 3. For further discussion of the Fire Protection System, refer to the Fire Hazards Analysis Report.

1.2.6 HL&P shall implement a physical security program at STPEGS that will minimize the possibilities of unauthorized removal of special nuclear material. The program calls for:

1) Storing the unirradiated fuel within an area to which access is controlled.

2) Ensuring the controlled access area is monitored in accordance with security procedures to detect unauthorized penetrations or activities.

3) Providing a member of the security force to respond to all unauthorized penetrations or activities.

4) Providing detailed response procedures for dealing with threats of theft or the actual theft of such material.

5) Providing detailed response procedures for dealing with threats of sabotage or the actual sabotage of such material.

1.3 Physical Protection

The quantity of U-235 to be stored at STPEGS is greater than the quantity specified in 10CFR73.1(b). The material is classified as special nuclear material of low strategic significance under 10CFR73.2(y)(3); i.e., 10,000 grams or more of U-235 contained in uranium enriched above natural but less than 10 percent in the U-235 isotope. Consequently, HL&P shall comply with the terms of 10CFR73 regarding physical protection of the plant and licensed materials. A description of the fuel storage security program is given in Section 1.2.6.

1.4 Transfer of Special Nuclear Material

The responsible shipper of the fuel for STPEGS is Westinghouse.

1.4.2 Packaging of new fuel for shipment will be the responsibility of the fuel fabricator, Westinghouse. However, should HL&P have need to package and transport new fuel, this will be done in accordance with the provisions of 10CFR71.

1.4.3 Shipping container handling, removal of fuel from containers, and fuel inspection for shipping damage will be performed according to STP procedures which will be based on specifications supplied by Westinghouse.

Records will be maintained showing receipt, inventory, disposition, and transfer of all special nuclear material in HL&P's possession. As appropriate, the DOE and NRC will be notified via the DOE/NRC Form 741 as to the amount of special nuclear material shipped or received, the date of the shipment or receipt, and the names of the shipper and receiver.

Written procedures and/or instructions shall delineate the material control and accountability measures for the new fuel. Basic

requirements are detailed in 10CFR70. As a minimum, procedures shall address the following:

a. Organization and personnel responsibilities and authorities.

b. Formal designation of a Special Nuclear Materials Custodian responsible for administration of the special nuclear materials license provisions at the STPEGS plant site.

c. Designation and description of areas wherein special nuclear materials are to be kept.

d. Preparation, distribution, and use of records or reports which are received, transmitted, or prepared by STPEGS in support of the STPEGS special nuclear materials license provisions.

e. Special nuclear materials receipt and shipment.

f. Special nuclear materials internal transfers.

g. Physical inventories which shall be conducted at intervals not to exceed twelve months. Procedures shall require the establishment of an inventory system that permits ready identification, by type and location, of all special nuclear materials on hand at any time.

As a minimum, established internal accountability records shall:

a. Have a cross-check capability.

b. Readily yield information regarding material receipts internal transfers, shipments, and special nuclear material balances on hand.

1.5 Financial Protection and Indemnity

HL&P will apply for nuclear energy liability insurance with the American Nuclear Insurers. Insurance will be in the amount of $1,000,000 as stipulated in 10CFR140.13. This coverage will be effective from the time of receipt of the new fuel at STPEGS until the fuel is loaded into the reactor. Confirmation of such protection will be provided to the NRC under separate submittal.

2.1 Radiation Control

2.1.1 The Health and Safety Services Manager (HSSM) shall be responsible for radiation safety at STPEGS. The HSSM will have qualifications in accordance with RG 1.8, Rev. 1-R, except that experience and training are substituted for the degree.

2.1.2 HL&P has established a committee to administer the HL&P radiation safety program. The group performing this function is the Radiation Safety Committee. The Chairman of the Radiation Safety Committee is the HSSM.

The Radiation Safety Committee will review the program for compliance with all license requirements, and assure that proposed operations involving licensed radioactive material are carried out safely in a manner that will protect the health of the workers and minimize the hazard to property.

Meetings of the Committee will be held at least quarterly. Meetings may be held more frequently as required. A quorum consists of the Committee Chairman (or co-Chairman) and two members.

The Committee will review the experience, training, and past performance of potential users of radioactive material. Interim approval may be granted by the HSSM provided the individual meets the qualifications required by the full Committee. Before the HSSM's interim approval is confirmed, the Committee must determine that the individual has had adequate training and experience in the handling of radioactive materials to assure that the sources will be used in a safe manner that will protect health and minimize the hazard to property.

Records of proceedings of the Radiation Safety Committee and safety evaluations of proposed uses of radioactive material will be maintained by the HSSM at the STPEGS.

The safety program, including records required to be maintained, will be periodically reviewed under supervision of the Radiation Safety Committee.

At STPEGS, the Radiation Safety Officer (RSO) is the HSSM or his designated alternate. The RSO has overall responsibility for the day-to-day aspects of health physics (radiation protection) at STPEGS facilities.

The RSO is responsible to the Radiation Safety Committee to see that their recommendations, as they apply to the use of licensed radioactive material, are implemented. The RSO has the responsibility to stop any operation which he has reason to believe will lead to injury of personnel or property damage if continued. In addition, if in the opinion of the RSO an action by the Committee may be detrimental to the overall radiological protection program, the RSO has the option of requesting a review of the decision by higher management.

2.1.3 The resume of the Health and Safety Services Manager is provided as Attachment 1.

2.1.4 Each unirradiated fuel assembly will be tested during receipt inspection for contamination using a standard smear technique and counted for contamination with an alpha scintillation survey meter, or a low-background alpha-beta gas flow proportional counter. If a fuel assembly reveals the presence of smearable alpha contamination in excess of 20 dpm/100 cm^2, that assembly will be isolated and an inspection will be conducted to determine the cause of the contamination.

A gamma survey using a beta-gamma survey instrument will be made on each new fuel shipping container upon receipt. Additionally, each container will be tested for smearable beta-gamma contamination. If any container reveals abnormally high radiation levels or smearable contamination in excess of 1000 dpm/100 cm^2, the container will be isolated and an investigation initiated.

2.1.5 Procedures for calibrating radiation detection equipment are documented in the South Texas Project Plant Procedures Manual. The procedures describe the methods to be used to maintain current calibrations for all portable survey meters, radiation detection equipment, and other measuring, sampling, and detection equipment used by the Health and Safety Services Division. Radiation survey instruments are to be inspected for physical damage, checked for battery strength, and response checked prior to each use. Instruments with any obvious damage will be removed from service and tagged for repair. Weak batteries will be replaced in accordance with the battery replacement procedure for the specific instrument. The instrument shall be source-checked prior to use with the exception of neutron survey instruments. Instrument calibration/standardization shall be performed at the frequency specified for each instrument in the instrument file computer system. Survey instruments that are to be used for quantitative measurements will be supplemented at least every 6 months with at least a two-point calibration on each scale of each instrument.

Two of the calibration points chosen will be separated by at least 50% of the scale.

Radiation detection equipment will be calibrated semi-annually when in use. The calibration normally is performed in the calibration room using the calibrator and/or other smaller sources. (A shielded instrument calibrator for gamma exposure is used to calibrate most ranges of the portable gamma and beta-gamma portable survey instruments.) With the exception of neutron survey instruments, the instrument response is checked with a source at least daily prior to use to verify that the instrument is functioning properly. Neutron survey instrument response will be checked prior to use. Neutron survey instruments will be calibrated semi-annually by a vendor.

Instruments may require calibration/standardization more frequently if any of the following occurs:

1. Maintenance, repair, or replacement of detectors or associated electronics.

2. Abnormal or erratic response or operation.

3. Periodic checks indicate operation outside acceptance limits (i.e., survey instrument pre-use source check deviates more than ± 20% from post-calibration value).

When an instrument is calibrated, the following data will be recorded:

1. Initial instrument condition.

2. As found response to test input or standard.

3. Acceptance criteria, or space provided to insert acceptance range calculated from variable dose rate or input levels.

4. Final response, whether or not adjustment was made.

5. Calculation of any correction factors.

6. After-calibration response to a source or standard.

7. Date and signature of individual(s) performing the calibration/standardization.

If during calibration an instrument is found to be outside the designated acceptance range, a review will be conducted to determine:

1. Whether the instrument error is significant and/or non-conservative.

2. Whether exposure estimates were made using data from the instrument.

3. If any repeat surveys need to be performed.

4. If any radiation work permit requirements must be modified.

Calibration data records shall be retained to provide a history of instrument maintenance and performance.

Upon successful completion of calibration/standardization, a sticker shall be affixed to the instrument indicating the following:

1. Date of calibration/standardization.

2. Date next calibration/standardization is due.

3. Initials of individual(s) performing calibration/ standardization.

4. Any correction factors required to yield correct response.

5. Check source or standard used and reading achieved in after-calibration response check.

Instrumentation is normally repaired by HL&P or by a vendor. Maintenance may be performed in place (for fixed instruments) at HL&P facilities, or at the vendor facilities, if necessary. Contractors other than the vendor will be employed at the discretion of the HSSM.

Personnel monitoring devices will be provided to individuals working near or around radioactive sources that have a potential of delivering doses specified in 10CFR20.202.

The t'pe of personnel dosimeter provided shall be at the discret' ' the Health and Safety Services Division; generally, . r, a direct reading dosimeter and/or a beta-gamma-sensitive th. oluminescent dosimeter (TLD) are to be worn.

Monitoring services will be provided primarily by HL&P Health and Safety Services Division staff, although contractors approved by the Radiation Safety Committee may also provide such services. TLD's shall be exchanged for routine evaluation on a monthly basis for individuals badged as described above.

Pocket dosimeters will be available in various ranges. The ranges worn are as specified in various emergency plans or as specified in the STPEGS Plant Procedures Manual.

Pocket dosimeters and other integrating dosimeters should be evaluated daily when in use.

TLD's are calibrated at least annually, and direct-reading dosimeters are calibration-checked at least semi-annually to provide accurate personnel monitoring.

Employees shall have whole body counting performed at least annually while performing permanent assignments at nuclear facilities.

A whole-body counter will be available to check on internal exposure. Data from the counter will be processed by a multi-channel analyzer. The processed data will be analyzed to determine the radionuclides detected and the percent body burden. Calibration of the whole body counter system shall be current prior to an individual being counted. Individuals being counted should

be free of external contamination. A background count of the whole body counter shall be performed at least once during any day in which personnel are body counted. Only qualified personnel shall operate the whole body counting equipment.

2.1.6 The Health and Safety Services Division has prepared a procedures manual that specifies procedures to be followed in health physics activities associated with STPEGS. All of the procedures necessary for proper conduct of the radiological protection program will be in place prior to initiation of activities related to radioactive materials under this license. Summaries of the procedures and lists of equipment to be used to meet applicable sections of 10CFR Part 20, "Standards for Protection Against Radiation," are given in Section 2.1.5 of this application. Radiation zones will be posted in accordance with 10CFR20.

2.1.7 Solid radioactive waste will be transferred to a commercial waste disposal firm. The commercial waste disposal firm will be selected by the HSSM.

A current set of DOT and NRC regulations concerning the transfer, packaging, and transport of radioactive waste material shall be maintained in the Document Control Center at the STP site.

When radioactive waste is shipped, it will be collected by a contractor for packaging and shipment to a burial site. Verification of the contractor's documentation will be obtained before any waste is collected.

The HSSM is responsible for the safe transfer, packaging, and transport of radioactive material commensurate with the division of responsibilities between HL&P and the waste collection contractor.

Transfer, packaging, and transport of radioactive material will be performed in accordance with approved procedures.

HL&P will provide training in DOT and NRC regulatory requirements, the waste burial license requirements, and HL&P operating procedures for all personnel involved in the transfer, packaging, and transport of radioactive material. Initial training and periodic retraining will be provided. The HSSM will maintain a record of training dates, attendees and subject materials.

Processing of radioactive waste will be provided by the waste contractor.

A management-controlled audit of all transfer, packaging and transport activities will be established and implemented once such activities begin.

2.2 <u>Nuclear Criticality Safety</u>

2.2.1 The Reactor Performance Supervisor will be responsible for nuclear

criticality safety and fuel-handling during receipt, inspection, storage, and return of new fuel. The minimum qualifications for this position are specified in RG 1.8, Rev. 1-R.

2.2.2 The Reactor Performance Supervisor will coordinate the development of procedures and direct the activities involving the receipt, inspection, storage, and return of nuclear fuel.

2.2.3 New fuel for STPEGS will be shipped in XLR new fuel shipping containers provided by Westinghouse. The new fuel may also be temporarily stored in the shipping containers.

The design of the XLR container complies with the structural requirements of 10CFR 71.35. This is accomplished through the application of design criteria which permit no yielding of any component of the package under loadings six times the weight of the loaded package. The XLR container design has also been demonstrated to comply with the requirements of 10CFR 71.36. A fully loaded package of similar design was subjected to the drop conditions for the hypothetical accident of 10 CFR 71, Appendix B. The drop test did not produce a configuration more limiting than that analyzed in the criticality analysis. Furthermore, the overall design, fabrication, and use of these containers comply with the provisions of WCAP-8370/8700, Revision 10/6A, "Westinghouse Water Reactor Divisions Quality Assurance Plan."This plan meets the requirements of 10 CFR 71, Subpart H. The NRC issued Certificate of Compliance USA/5450/AF, Revision 19, to Westinghouse dated March 1, 1984, certifying the XLR containers meet 10CFR 71 in full.

Each container can hold a maximum of two assemblies. The storage array will be limited to stacking no more than two containers, or three if additional support is provided.

The fuel enrichment limit currently licensed for the XLR containers with STPEGS fuel is 3.55 w/o U-235 (with 0.19 in. carbon steel absorber plate). This limit is used in evaluating the maximum credible accident condition. The maximum credible accident is defined as an accident in which two shipping containers are crushed together such that there is a four-inch separation between fuel assemblies in adjacent containers. The containers and surrounding areas are considered to be flooded with full density water. The criticality evaluation that determines the maximum enrichment that can be shipped in the container is performed using the maximum credible accident condition geometry as a basis. In the criticality analysis, reactivity credit is taken for: fuel rod cladding; shipping container internals support frame skin; and absorber plates (2).

2.2.4 Following removal from the shipping containers, the new reactor fuel will be stored in either the spent fuel pool or the new fuel storage pit. At least 127 assemblies of the initial core load will be stored in the spent fuel pool, preferably in a dry condition,

while up to 66 assemblies can be stored in the new fuel storage pit.

The spent fuel pool is a stainless-steel-lined, reinforced concrete pool and is an integral part of the Seismic Category I FHB. The spent fuel pool is designed in accordance with General Design Criterion (GDC) 62 and Regulatory Guide 1.13, Revision 1. The new fuel is to be stored in modules positioned on the floor of the spent fuel pool. The modules are designed with a minimum center-to-center spacing of 14 inches. The 14-inch spacing provides sufficient separation between fuel assemblies to maintain a subcritical array assuming optimum moderation. In accordance with ANSI N18.2, the design of the normally dry fuel storage racks is such that the effective multiplication factor will not exceed 0.98 with fuel enriched to a nominal maximum of 2.9 w/o, assuming optimum moderation. For the unborated flooded condition, assuming new fuel enriched to a nominal maximum of 2.9 w/o, the effective multiplication factor does not exceed 0.95. Credit is taken for the inherent neutron-absorbing effect of the materials of construction.

Storage racks in the spent fuel pool are designed to withstand handling, normal operating loads as well as SSE and OBE seismic loads meeting Safety Class 3 and AISC requirements. The racks are designed with adequate energy absorption capabilities to withstand the impact of a dropped spent fuel assembly from the maximum lift height of the Fuel Handling Machine. The racks are also designed to withstand a maximum uplift force equal to the uplift force of the Fuel Handling Machine. The racks meet the requirements of ASME Boiler and Pressure Vessel Code, Section III, Appendix XVII. The spent fuel racks are classified as Seismic Category I, as defined by Regulatory Guide 1.29, and as ANS Safety Class 3. All rack surfaces that come into contact with fuel assemblies are made of annealed austenitic stainless steel. These materials are resistant to corrosion. Fire in the spent fuel pool that could affect the racks is precluded by keeping flammable materials in the area to a minimum. Fire-fighting equipment will be maintained nearby.

The design of the 14-inch racks in the spent fuel pool is such that fuel assemblies can only be inserted in the designated storage locations. They can not be inserted between racks or between the storage cells. They also can not be inserted between the racks and the north wall (approximately 6-inch clearance). Exceptions are on the west, south, and east sides where the spacing is 27-inches or greater.

Clearance between the cell containing the fuel assembly and the floor of the spent fuel pool is 4 1/2-inches minimum. Spacing between fuel assemblies and the walls is greater than 6-inches.

The new fuel storage pit is a reinforced concrete pit and is an integral part of the FHB. The new fuel pit is designed in accordance with GDC 62 and RG 1.13, Revision 1. The new fuel is to

be stored in racks composed of individual vertical cells fastened together to form three 2 x 11 modules which may be bolted to anchors in the floor and walls of the new fuel storage pit. The new fuel racks are designed with a center-to-center spacing of 21 inches. This spacing provides a minimum of 12 inches between adjacent fuel assemblies. This separation is sufficient to maintain a subcritical array. In accordance with ANSI N18.2, the design of the normally dry new fuel storage racks is such that the effective multiplication factor will not exceed 0.98 with fuel of the highest anticipated enrichment in place assuming optimum moderation. For the unborated flooded condition, assuming new fuel of the highest anticipated enrichment, the effective multiplication factor does not exceed 0.95. (Credit is taken for the neutron-absorbing effect of the construction materials.)

Storage racks in the new fuel storage pit are designed to withstand normal operating loads, as well as to remain functional with the occurrence of a Safe Shutdown Earthquake. The new fuel racks are designed to withstand a maximum uplift force of 5,000 lbs and to meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Appendix XVII. The new fuel racks are classified as Seismic Category I components, as defined by Regulatory Guide 1.29, and as ANS Safety Class 3. All rack surfaces that come into contact with the fuel assemblies are made of annealed austenitic stainless steel, and the support structure is painted carbon steel. The new fuel storage pit access hatch is a three-section cover. This cover will minimize the introduction of dust and debris into the pit. The cover is designed to withstand the impact force of a new fuel assembly dropped from the maximum elevation allowed by the 2-ton hoist of the FHB overhead crane. Fire in the new fuel storage pit that could affect the racks is precluded by keeping flammable materials in the area to a minimum. Fire-fighting equipment will be maintained nearby.

The design of the 21-inch racks in the new fuel storage pit is such that fuel assemblies can only be inserted in the designated storage locations. They cannot be inserted between the storage cells or between the storage cells and the pit walls (approximately 8-inch clearance). Exceptions are between the racks where sufficient space is available for insertion.

Clearance between the fuel pins and the pit floor is 5.6-inches minimum. Spacing between storage cell walls and the pit walls is approximately 8-inches.

Drawings of representative spent fuel racks and new fuel racks are provided as Figures 5 and 6.

2.2.5 The racks in the new fuel storage pit and the spent fuel pool are designed to prevent accidental criticality by maintaining adequate spacing between fuel assemblies. The initial fuel load will be enriched to a nominal maximum of 2.9 w/o, which is low enough that

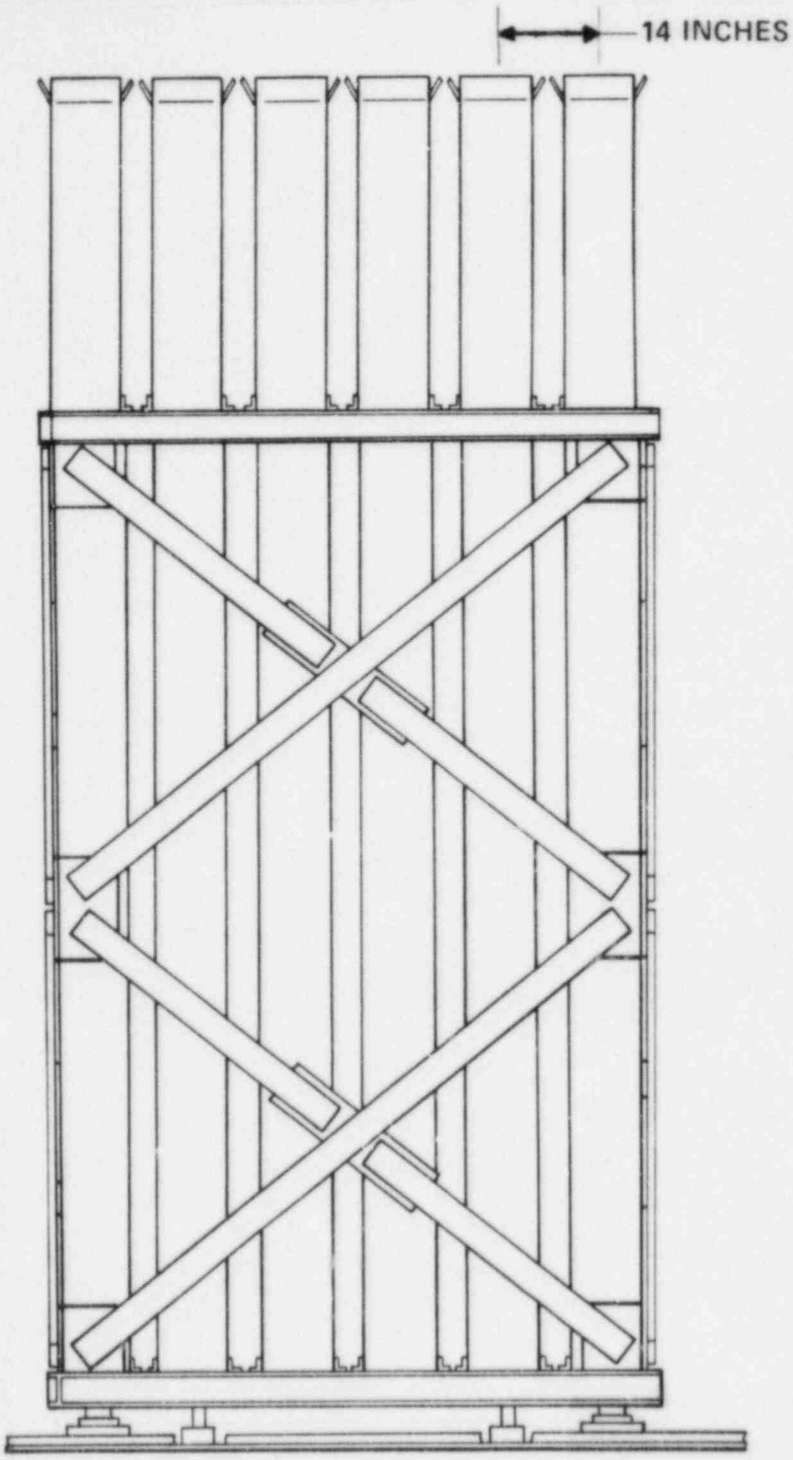

Figure 5
14-in. Center-to-Center
Spent Fuel Racks

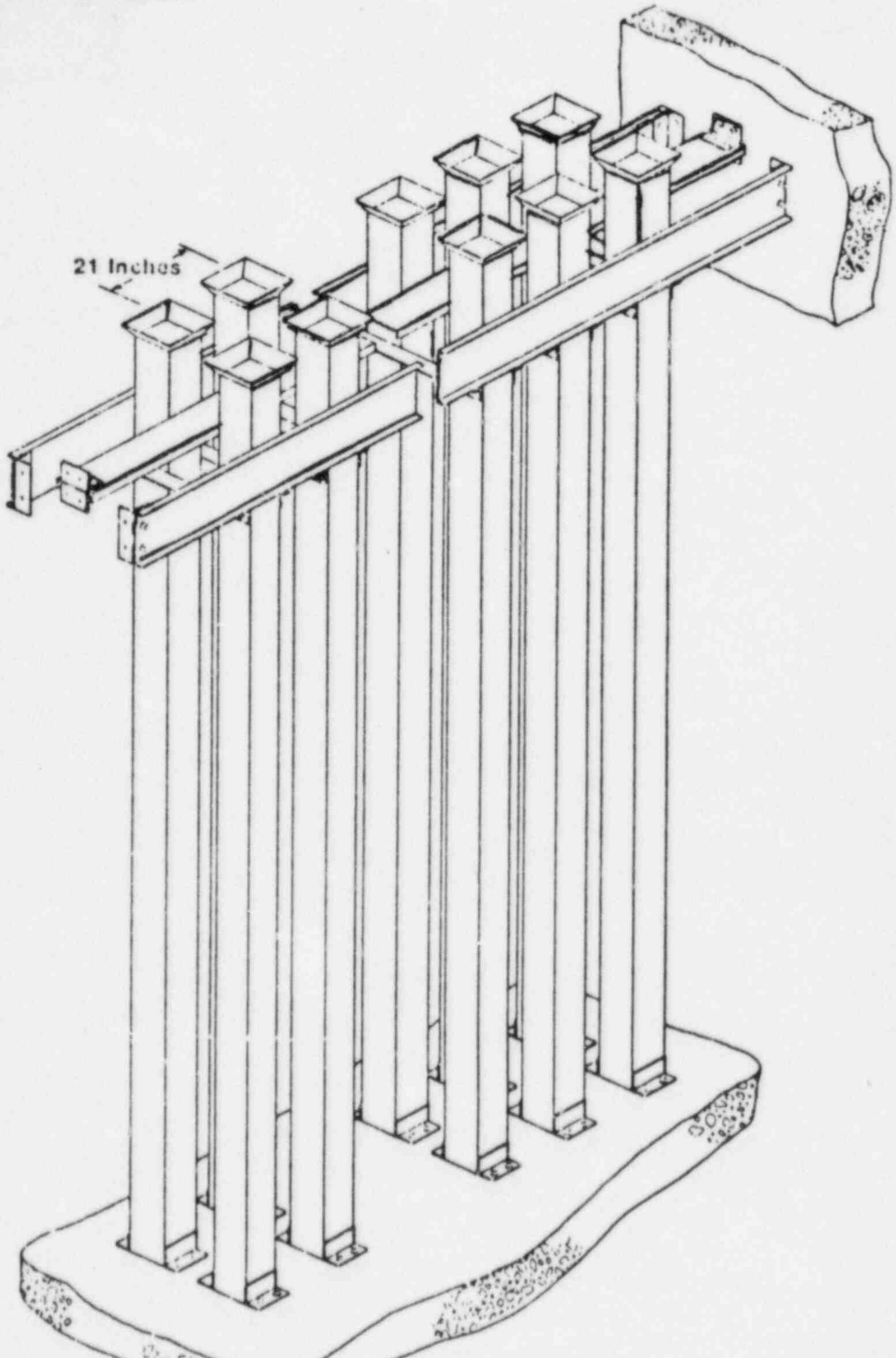

Figure 6
New Fuel Storage Racks

accidental criticality will not occur even under conditions of optimum moderation.

2.2.6 The fuel assemblies are assumed to be in their most reactive condition, fresh or undepleted with no control rods or removable neutron absorbers present. Credit is taken for the inherent neutron-absorbing effect of the construction materials of the racks. Assemblies will not be closer together than the design separation provided by the storage facility, except in special cases such as in fuel shipping containers where analyses are carried out to establish the acceptability of the design. The mechanical integrity of the fuel assembly is assumed.

No additional neutron absorber material is added to the structural material. However, the inherent neutron-absorbing effects of the structural materials are considered in reactivity calculations.

2.2.7 Nuclear safety of the stored fuel is not based on moderation control. Separation of the fuel assemblies and the absorption characteristics of the storage racks have been determined to provide sufficient protection against criticality regardless of moderation.

2.2.8 The criticality calculation method and cross-section values are verified by comparison with critical experiment data for assemblies similar to those for which the racks are designed. This benchmarking data is sufficiently diverse to establish that the method bias and uncertainty will apply to rack conditions which include strong neutron absorbers, large water gaps and low moderator densities.

The design method which insures the criticality safety of fuel assemblies in the spent fuel storage rack uses the AMPX system of codes (Ref. 1,2) for cross-section generation and KENO IV (Ref. 3) for reactivity determination.

The 218 energy group cross-section library (Ref. 1) that is the common starting point for all cross-sections used for the benchmarks and the storage rack is generated from ENDF/B-IV data. The NITAWL program (Ref. 1) includes, in this library, the self-shielded resonance cross-sections that are appropriate for each particular geometry. The Nordheim Integral Treatment is used. Energy and spatial weighting of cross-sections is performed by the XSDRNPM program (Ref. 2) which is a one-dimensional Sn transport theory code. These multi-group cross-section sets are then used as input to KENO IV which is a three dimensional Monte Carlo theory program designed for reactivity calculations.

A set of 27 critical experiments have been analyzed using the above method to demonstrate its applicability to criticality analysis and to establish the method bias and variability. The experiments range from water moderated, oxide fuel arrays separated by various

materials (Boral, steel, water) that simulate LWR fuel shipping and storage conditions (Ref. 4,5) to dry, harder spectrum uranium metal cylinder arrays with various interspersed materials (Ref. 6) (Plexiglas, steel and air) that demonstrate the wide range of applicability of the method.

The average Keff of the benchmarks is 0.9998 which demonstrates that there is essentially no bias associated with the method. The standard deviation of the bias value is 0.0014 delta-k. The 95/95 one-sided tolerance limit factor for 27 values is 2.26. Thus, there is a 95 percent probability with a 95 percent confidence level that the uncertainty in reactivity, due to the method, is not greater than 0.0032 delta-k.

These methods conform with ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," Section 5.7, Fuel Handling System; ANSI 57.2-1973, "Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations," Section 5.1.12; ANSI N16.9-1975, "Validation of Calculational Methods for Nuclear Criticality Safety," NRC Standard Review Plan, Section 9.1.2, "Spent Fuel Storage"; and the NRC guidance, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications."

2.2.9 New fuel assemblies may be removed from storage for purposes of inspection and for initial core loading. All fuel handling activities will follow approved fuel handling procedures. When fuel assemblies are removed from storage, no more than two will be allowed out of storage at one time. Conditions that could lead to accidental criticality will be avoided by having the fuel assemblies that are out of approved storage locations in physically separate areas. Single fuel assemblies may be out of storage in the new fuel handling area/new fuel storage pit/new fuel inspection laydown area and in the new fuel elevator/transfer canal/spent fuel pool. This arrangement will ensure that a subcritical configuration is maintained at all times.

2.2.10 10CFR70.24 requires the following:

a) A criticality monitoring system. Coverage of all areas shall be provided by two detectors.

b) Emergency procedures for each area in which such licensed special nuclear material is handled, used or stored.

HL&P requests exemption from the requirements of 10CFR70.24 as provided in Subsection 70.24(d). As described in this application, the fuel assemblies will be stored in critically safe storage racks. In addition, procedural controls also described in the application will preclude situations in which critical conditions may occur.

2.3 Accident Analysis

The new fuel assemblies are exposed to only four types of accidents. These accidents are:

1) Dropping an assembly in the spent fuel pool or new fuel storage pit.

2) Dropping an assembly other than in the spent fuel pool or new fuel storage pit.

3) Dropping an object other than an assembly in the spent fuel pool or new fuel storage pit.

4) Flooding the storage area.

Dropping an assembly in the storage areas raises two concerns: 1) breaking fuel rods; and 2) accidental criticality. Should a rod or rods be broken, the safety consequences are minimal. The radiological effects will be negligible because there will be no fission products present in the unirradiated fuel. Criticality will not occur as a result of a dropped assembly. The spent fuel racks are designed to withstand the impact of a fuel assembly dropped from the Fuel Handling Machine. The new fuel racks are designed to withstand the impact of a fuel assembly dropped from the FHB overhead crane. Fuel assemblies may be carried over the spent fuel pool by the 2-ton hoist of the FHB overhead crane. However, the lift height when over the spent fuel racks will be administratively controlled so that the fuel assembly is no higher than one that would be carried by the Fuel Handling Machine.

If an assembly is dropped in an area other than in the spent fuel pool or new fuel storage pit, the safety consequences are also minimal. The radiological effects of broken rods will be negligible because no fission products will be present in the unirradiated fuel. Accidental criticality will not occur because no more than two assemblies will be out of storage at a time. See Section 2.2.9. The assemblies outside the storage areas are protected from criticality by their shipping containers.

Objects other than fuel assemblies can be dropped into the fuel assembly storage areas from the following:

1. FHB Overhead Crane

2. Fuel-Handling Machine

3. New Fuel Handling Area Overhead Crane.

The FHB Overhead Crane passes over both the spent fuel pool and the new fuel storage pit. This 15/2-ton capacity crane is to be used for general equipment-handling operations in the FHB. The 15-ton

main hoist of the crane is designed to withstand a single-failure without dropping its load and therefore meets the intent of RG 1.104. An assessment of potential consequences of dropping a load from the main hoist is not necessary; however, administrative precautions, such as requiring designated safe load paths, will be taken.

Administrative controls will be imposed so that loads capable of damaging the spent fuel racks will not be carried over the spent fuel pool by the 2-ton hoist. Furthermore, the 2-ton hoist will not be used to carry loads heavier than a fuel assembly over the new fuel storage pit.

The Fuel Handling Machine passes over the spent fuel pool. This device is used exclusively for handling fuel assemblies and core components by means of handling tools suspended from the hoist. The storage racks can absorb the impact of these loads without any significant radiological consequences.

The New Fuel Handling Area Overhead Crane passes over the new fuel handling area. This 5-ton crane is used exclusively to handle new fuel assemblies and their shipping containers in the new fuel handling area. (It does not pass over the new fuel storage pit.) The radiological consequences of a dropped assembly are insignificant, both in terms of the dropped assembly itself, and the assemblies in their protective shipping containers.

Unirradiated fuel assemblies are ordinarily stored dry. However, the storage racks and storage containers are designed to prevent criticality so that if flooding should occur there will be no radiological consequences.

Handling of fuel assemblies will be conducted so as to minimize potential for handling accidents. Administrative controls, physical limitations on the fuel handling operations and experience and training of personnel performing fuel movement assure that damage to the new fuel assemblies will not occur. Procedures for responding in the event a new fuel assembly is dropped will be approved and in place prior to receipt of fuel onsite. The requisite equipment for this will be maintained in the FHB during fuel handling activities.

3.0 Other Materials Requiring NRC License

HL&P has received a Type A License of Broad Scope for STPEGS. The license number is 42-23140-01, docket number 030-20301. The expiration date is February 28, 1990. The license includes small amounts of source and special nuclear material in addition to radioactive byproduct material. This license addresses all anticipated types and amounts of such materials.

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6. J. T. Thomas, "Critical Three-Dimensional Arrays of U (93.2) -- Metal Cylinders," Nuclear Science and Engineering, Volume 52, pages 350-359 (1973).

Gene L. Jarvela
Health and Safety Services Manager

Education:

a. U.S. Navy Schools
   1) X-ray Technician Course
   2) Radioisotope Course
   3) Basic Nuclear Power School
   4) Specialized Radiation Control and Health Physics Course

b. Public Health Service Courses
   1) Basic Radiological Health
   2) Occupational Radiation Protection
   3) Radionuclide Analysis by Gamma Spectroscopy

c. Other Short Courses
   1) Respiratory Protection - LASL
   2) Packaging and Transportation of Radioactive Materials - NEWC
   3) Electronic Training - Eberline

Experience:

1950 - 1970 - U.S. Navy
While stationed at the Oakland Naval Hospital, Mr. Jarvela served as an x-ray technician instructor at the x-ray Technician School and as a technician and instructor in deep and superficial x-ray therapy. He was a nuclear submarine trained Hospital Corpsman for 12 years assigned to two submarines through their entire construction phase and through 2 to 3 years of their subsequent operational phase. His duties included first aid, radiation protection, chemistry, photo dosimetry and radiation monitoring onboard submarines. He also qualified and served as Diving Officer on both submarines. Mr. Jarvela left the Navy as a Chief Hospital Corpsman.

1970 - 1980 - Wisconsin Public Service Corporation
Mr. Jarvela was employed as Assistant Radiological-Chemistry Supervisor at Wisconsin Public Service Corporation. In this position, which he held for 12 months, he was involved in preparing the Kewaunee Radiation Protection Manual, Emergency Plan, Security Manual and the first Environmental Report. He also collaborated on departmental staffing activities,

Gene L. Jarvela
Health and Safety Services Manager

equipment purchase, procedure development, set up of the environmental monitoring program, resolution of design problems from a radiological aspect, and plant training of both licensed and non-licensed personnel in the areas of radiation protection, rad systems, safety, and first aid. He was then promoted to the position of Radiological Specialist as a member of the Nuclear Engineering Group. He served in this capacity for six months where he was responsible for ensuring the implementation of the Kewaunee Radiation Protection Manual, Security Plan, and Emergency Plan, and for coordinating the assistance of outside agencies and groups in relation to these plans and manuals. In addition, he served as a consultant to the Kewaunee Plant Superintendent in the areas of startup testing involving radiation monitoring systems, industrial and radiation safety areas involving training and retraining and training of local outside groups in radiation protection/safety procedures. Mr. Jarvela was then promoted to the position of Health Physics Supervisor on the Kewaunee Plant staff. He served in this position through initial plant startup and 7 years of plant operation including five refueling outages. In this capacity he supervised and was responsible for all site personnel associated with work or evolutions involving radiological controls, safety and certain facets of the Emergency Plan. He was responsible for the immediate supervision of a Rad Protection Leadman and seven Technicians assigned the responsibility for training of personnel in the area of radiological controls; respiratory protection; first aid; security; health physics procedure writing and updating; personnel dosimetry; radiological surveys; operation, maintenance and calibration of portable and laboratory radiation counting equipment; radioactive material receipt, storage and transfer; and personnel and material decontamination.

1980 - Present - Houston Lighting and Power Company
Mr. Jarvela joined HL&P as the Radiation Protection Supervisor at the South Texas Project. He is now the Health and Safety Services Manager.