1.2.2 Fuel Storage Facilities

The designated storage area for new fuel still in shipping containers is inside the Fuel Storage Building in the rail bay and the new fuel shipping container area (see Reference 1, FSAR Figures 1.2-16 and 1.2-18). During this storage there are no criticality concerns because the shipping containers will be stored in the configuration permitted per the requirements of the U. S. NRC Certificate of Compliance (No. 5450) for that container. This Certificate of Compliance allows the storage of up to 60 containers in any array without separation distance since analysis shows there is no criticality problem. There will be no more than 60 loaded shipping containers allowed on site at any one time. Per station procedures, the containers will be stacked no more than two high.

In the unlikely event that more containers are received than can be stored in the rail bay and new fuel shipping container area, a temporary holding area outside and adjacent to the Fuel Storage Building will be set up. This area will only be used in the event that more containers are received than can comfortably be handled inside the fuel storage building. Any containers stored outside will be moved inside as soon as space is available. The station security plan contains measures for protection of the shipping containers while stored outside. No special provisions will be taken to protect the containers from the forces of nature, unless abnormal weather conditions dictate otherwise (e.g., excessive winds, heavy snowfall, etc.). In those cases, the containers will be moved inside. It is not anticipated that outside storage of containers will be for more than 72 hours.

The designated storage area for new fuel that has been received and inspected is the spent fuel pool. The new fuel storage vault and associated racks may be used during receipt for inspection of new fuel. The new fuel storage vault will not be used for storage until it meets seismic Category I requirements. A description of the new fuel storage vault is contained in Reference 1.

Further detail is shown in Attachment C. The square shaped canisters have an inside dimension of 9 inches, a length of 169.85 inches, a thickness of 0.093 inches and are made of ASTM A240 type 304 stainless steel material.

The dust cover for the new fuel storage vault consists of removable steel plates which cover the entire vault (see Reference 1, FSAR Figure 1.2-16). There are a total of 8 steel plates covering the new fuel storage vault, each covers approximately 2-1/2 rows of storage locations. Thus, with one plate removed, a total of 10 assembly locations could be exposed and another 5 could be partially exposed. Because the fuel will be stored in a checkerboard pattern, not all of the locations will be utilized. During the receipt process, the new fuel storage vault covers will be off, thus exposing the maximum allowable storage of 12 fuel assemblies which will be in a checkerboard storage configuration. The dust cover for the assemblies stored in the spent fuel pool consists of plastic sheets placed on top of storage racks containing new fuel.

The design basis of the fuel storage facilities includes the following:

- The spent fuel pool storage facility is designed in accordance with Regulatory Guide 1.13.

- The spent fuel racks are designed for high density fuel storage, and contain neutron absorbing material to assure a $K_{eff} \leq 0.95$, even if the fuel is immersed in unborated water.

contamination and inspected. In the unlikely event that an assembly is contaminated, the vendor will be notified and the assembly will be handled in accordance with the Station Radiological Protection Manual. The inspection takes place in the new fuel storage vault and includes visual checks for bowing, twisting, foreign debris, cracks, scratches, pits and burrs and dimensional checks of nozzles, fuel rod spacing and thimble tubes. At this time other core components such as control rods will undergo a similar inspection. When the inspection is complete, the assemblies will be transferred to the spent fuel pool for storage until core load. Assemblies that do not pass inspection will be stored in the same manner as any other fuel assembly until appropriate disposition with this vendor has been made (e.g., repaired, returned, etc.).

1.2.4 Fuel Handling System Design Bases

The primary design requirement of the fuel handling equipment is reliability. A conservative design approach is used for all load-bearing parts. Where possible, components are used that have a proven record of reliable service.

The following design bases were used for fuel handling system equipment assemblies during transfer of fuel assemblies:

- Fuel handling devices have provisions to avoid dropping or jamming of fuel assemblies during transfer operation.
- Handling equipment used to raise and lower spent fuel has a limited maximum lift height.
- The Fuel Transfer System (FTS) has provisions to preserve the integrity of the containment pressure boundary. Figure 1.2.3-1 shows the Fuel Transfer System.

- Handling equipment will not fail in such a manner as to damage Seismic Category I equipment or fuel in the event of a SSE.

- The static design load for the refueling machine crane structure and all its lifting components is normal, dead and live loads, plus three times the fuel assembly weight with a rod cluster control assembly.

1.2.6 Fire Alarm and Fire Control Systems

The Fuel Storage Building is one large fire area and has been provided with an ionization fire detection system, manual firefighting equipment consisting of portable fire extinguishers, halon and dry chemical along with a standpipe and hose reel system. Fire fighting teams and personnel are directed to use dry chemical extinguishers as the primary method for fire suppression in the building. The standpipe and hose reel system, using water as a suppressant, is provided as a secondary source of fire suppression. Fire brigade training addresses its use as a secondary method. The fire loading is very low. The building is locked with controlled access. Administrative controls will control ignition source work and the storage and use of combustibles in the building. The Seabrook Station Fire Protection Program, Evaluation and Comparison to BTP APCSB 9.5-1 Appendix A, Revision 2, Section F.1, Tab 9 has the fire hazard analysis for this building. The fire suppression capability of the building was evaluated in Section 9.5.1 of the SER and found acceptable.

1.3 Physical Protection

A description of the Seabrook Station Physical security Plan for the Protection of Nuclear Material of Low Strategic Significance will be provided as a separate part of the application withheld from public disclosure. This plan was prepared pursuant to the requirements of 10CFR70.67, Physical Protection of Special Nuclear Material of Moderate and Low Strategic Significance.

1.4 Transfer of Special Nuclear Material

1.4.1 Transfer

Westinghouse Electric Corporation of Columbia, South Carolina, fabricator of the nuclear fuel assemblies, is responsible for shipment of the fuel to Seabrook Station. All fuel assemblies are to be delivered to the site in accordance with shipping procedures and arrangements of the Westinghouse Electric Corporation authorized for use by that company under separate nuclear material license SNM-1107. The shipping container that will be used to ship the initial core to the site is supplied by Westinghouse and is covered by U. S. NRC Certificate of Compliance 5450. Shipping quantities will be in accordance with the Certificate of Compliance.

The Station Operation Review Committee (SORC) is responsible for reviewing and recommending for approval, procedures related to fuel handling and storage operations. Their minimal qualifications will be in accordance with ANSI 3.1 1978, Section 4.2 and 4.4. The SORC is comprised of the following:

Chairman:	Station Manager
Member:	Assistant Station Manager
Member:	Operations Manager
Member:	Technical Services Manager
Member:	Maintenance Department Manager
Member:	Instrumentation and Control Department Supervisor
Member:	Reactor Engineering Department Supervisor
Member:	Health Physics Department Supervisor
Member:	Engineering Services Department Supervisor
Member:	Chemistry Department Supervisor

The Station Manager is responsible for final approval of all procedures.

Site receipt of licensed source and special nuclear material is performed in accordance with station procedures. These contamination detection procedures are implemented in the manner described below.

Preliminary radiation and contamination surveys are conducted on a new fuel shipment prior to protected area access to evaluate its radiological status. Upon arrival on-site and pursuant with 10CFR20 and 49CFR173, a comprehensive survey is performed on the transport vehicle and the exterior of the shipping containers. Radiation surveys are performed using a portable Geiger-Mueller survey instrument. Contamination levels are evaluated by performing a smear survey. Smears are counted for alpha and beta-gamma contamination. Radiological conditions exceeding the limits specified in 10CFR20.205 shall result in the establishment of appropriate radiological controls and notification of the NRC/DOT.

During the opening of shipping containers, appropriate radiological precautions are implemented. Radiation and contamination surveys are performed on the interior of the container and container contents. Periodic airborne radioactivity sampling is established during shipping container opening. As the fuel assembly protective covering is removed, smears and contact radiation readings are taken at representative locations.

All fuel handling activities are performed under a Radiation Work Permit (RWP). The RWP specifies requirements for personnel dosimetry, protective clothing, and health physics work coverage responsibilities.

The fuel handling area is posted with appropriate caution signs and boundaries as radiological conditions warrant. A contamination frisking station is provided at the exit of the area during receipt, inspection, and movement of new fuel. A portable area radiation monitor with an audible alarm is located in the vicinity of the fuel handling area during fuel movement.

The station limit for removable contamination is 20 dpm/100 cm^2 (alpha). If contamination levels exceed this limit, contamination control measures will be implemented. These measures will include establishing a contamination control point, strict control of personnel access to the area, use of appropriate protective

clothing, and re-establishment of area barriers, boundaries, and postings. As a precaution, periodic air sampling may be performed.

Decontamination of the area and equipment is initiated under the supervision of Health Physics personnel. Waste generated by the decontamination activities is handled under the requirements set forth by station procedures.

2.1.3 Contamination/Radiation Detection Equipment

The station has a counting room that is equipped with radiation detection equipment to analyze routine contamination survey smears.

This equipment is capable of detecting alpha, beta, and gamma activities at levels less than contamination control levels. This counting room equipment is used for quantitative and qualitative analysis of smear samples. Except for the Health Physics Gamma Spectroscopy System (HPGSS) calibrations will be done on a quarterly basis. For the HPGSS, calibration will be done, at a minimum, once per year.

Portable radiation detection equipment consists of low- and high-range ion chamber dose rate meters, Geiger-Mueller count rate meters, scintillation alpha counters, neutron rate meters, and air samplers. Sufficient quantities of each type of instrument are available to permit calibration, maintenance, repair, and handling of peak loads without diminishing the radiation protection program. Information about instrumentation is presented in Table 2.1.1.

Equipment required to support fuel handling operations is normally stored at the fuel building control point where it is easily accessible. Extra equipment not intended for daily use is stored in a Health Physics storage area.

Personnel external contamination detection equipment consists of hand held friskers with a range of 0 to 5 x 10^4 cpm. This equipment is designed for routine use by all personnel that exit a Radiologically Controlled Area (RCA).

Bioassay services are provided whenever an internal intake of radioactive material is known or suspected to have occurred. The Seabrook Health Physics Department may provide in vivo (whole body) counting either on-site or through

documented. Applicable shipping documents are completed and maintained on file at the station with a copy provided to the driver. All waste is appropriately labeled and special instructions are given to drivers of "exclusive use" vehicles.

2.2 Nuclear Criticality Safety

2.2.1 Qualification of Personnel

The recommendations of Regulatory Guide 1.8, "Personnel Selection and Training," Revision 1-R, have been used as the basis for establishing minimum qualifications for all management, supervisory and professional-technical personnel in the Station organization, with the exception that ANSI/ANS 3.1-1978 will be used as the standard in lieu of ANS 3.1/ANSI 18.1-1971 (Reference 4).

The education, training and experience requirements for operators, technicians and mechanics will equal or exceed the qualifications for the positions stated in ANS 3.1-1978 and Regulatory Guide 1.8. Established company training programs include documented academic and on-the-job training plus comprehensive qualification examinations applicable to the skill level of the position assignment. Where desirable, off-site facilities may be used for specialized training. Records of the scope, general content and level of accomplishment for each person attending off-site training are retained at the Station.

Prior to receipt of fuel, a dummy fuel assembly and dummy control rod will be shipped to the site. At the time of dummy assembly receipt the fuel vendor will provide a training seminar for personnel involved in fuel receipt. The seminar will include a presentation on fuel receipt as well as actual removal of the dummy assembly from the shipping container and insertion into storage locations. In addition, all personnel involved with fuel receipt will have received appropriate health physics training.

2.2.2 Restonsibilities of Key Personnel

The Nuclear Quality Manager at the station has overall responsibility for assuring that the Seabrook Operational Quality Assurance Program is effectively implemented by all organizations performing work on safety-related systems and equipment at Seabrook Station. This individual has education, training and experience which equals or exceeds that of ANSI/ANS 3.1-1978, Paragraph 4.4.5.

The Reactor Engineering Department Supervisor is responsible for fuel movement sequences, fuel accountability, fuel storage, fuel inspection, and fuel receipt and training. This individual has education, training and experience which equals or exceeds that of ANSI/ANS 3.1-1978, Paragraph 4.4.1. The Fuel Management Department of Yankee Atomic Electric Company, Nuclear Services Division provides the independent nuclear accountability function, maintains in-process nuclear material inventory, and provides assistance to the Reactor Engineering Department Supervisor, as requested.

The Joint Test Group (JTG) is responsible for review and approval of all preoperational test procedures. The members of the JTG will meet the requirements of their respective positions as given by Regulatory Guide 1.58 (Rev. 1, 9/80), ANSI 45.2.6 (1978), and Regulatory Guide 1.8 (Rev. 1-R, 5/77) with ANSI 3.1 (1978) in lieu of ANSI 18.1 (1971), which apply to review and approval of test procedures and results. The JTG is comprised of the following:

Chairman:	Startup Test Department Manager
Member:	Seabrook Station Operations Manager
Member:	YAEC Engineering Manager

2.2.3 Storage of Fuel Elements

Fuel elements will be stored temporarily in the fuel receipt area in shipping containers. The shipping containers will be taken to the Fuel Storage Building and the fuel removed and stored in the spent fuel pool in a timely manner.

2.2.4 Nuclear Safety Analysis

Fuel spacing is maintained by the fuel storage racks which are Seismic Category I equipment. The racks consist of individual vertical cells. Administrative limitations on placement of fuel in the racks is not necessary according to criticality calculations. Placement and design of racks is such as to preclude insertion of fuel in places other than the vertical cells.

A design description of the fuel storage racks is provided in Subsection 1.2.2. Loading, shock, fire, and corrosion are not credible methods of achieving criticality since a moderator such as water would have to be present. This would constitute a double fault situation since the fuel storage condition is without moderation. Additionally, loading was analyzed as part of the Seismic I category design. The Station Fire Protection Program Manual describes the administrative controls and permit system for the control of ignition sources and the combustible materials permit system. The racks are designed for impact loading of one fuel assembly (17 x 17), 8.426 inches square, 167 inches long, weighing 1,467 pounds, and falling at the worst possible orientation, 18 inches to the spent fuel racks and 30 feet to the new fuel racks. Racks are constructed of steel and should not be affected by credible corrosive agents.

leaching of the soluble species from the boron carbide. The soluble contaminants in the boron carbide are not considered for certified Boraflex. Boraflex is inserted on all sides of fuel holding cells which are not adjacent to the spent fuel pool liner. For dimensional information refer to Table 2.2.6-1 and FSAR Figure 9.1-14 (Reference 1).

By design the poison material is encapsulated in stainless steel for structural support. The material is not sealed, since it is compatible with the environment. Poison verification holes are included in each cavity wall for visual inspection to assure the presence of poison during fabrication.

The presence and effectiveness of the poison material is assured since:

- Controls were imposed during manufacture to ensure material installation;
- The poison material is encapsulated and not designed for removal;
- The poison material has been shown to be compatible with its environment.

Westinghouse provided the spent fuel racks, which included the encapsulated boraflex, to Seabrook. The manufacturer's (Westinghouse Nuclear Component Division) quality program complies with Westinghouse Water Reactor Division QA Plan WCAP-8370. QA Plan WCAP-8370 has been reviewed by the NRC staff.

The installation of the completed spent fuel racks and control of the racks after receipt at the site is the responsibility of the applicant although Westinghouse may provide criteria and procedures in regards to installation. Fuel racks are Category 1 QA items; therefore, inspection and installation are performed in accordance with the Station QA Program.

The site QA on the spent fuel racks involve the following:

- o Receipt Inspection: When the racks were received they were checked for shipping damage and the associated paperwork was verified to be acceptable.

- o Cleanliness Inspection: Prior to rack installation the racks were verified to be clean.

- o Installation Inspection: The rack installation process was witnessed to verify proper rack installation. This included proper rack orientation and rack level. Minimum B-10 content is assured by Westinghouse.

2.2.7 Moderator Control

The calculations performed on the spent fuel pool considered fresh fuel at an enrichment of 3.5 w/o U-235. The pool water was considered to be at a temperature of 68°F and contained no boron. Under these conditions, an infinite array of canisters with a 10.35-inch center-to-center spacing gave a K_{eff} below the NRC limit of 0.95.

The NITAWL-KENO-IV method with the 123-group library has been validated against experiment and detailed Monte Carlo calculations for spent fuel rack geometries with poison. However, NITAWL-KENO-IV criticality calculations for new fuel storage with fire-fighting foam or mist have not been benchmarked against experiment or detailed calculations. Because of this, it was suggested that an independent calculation be performed with continuous energy Monte Carlo and modern cross section data. Brookhaven National Laboratory performed the same new fuel storage rack criticality calculations with the SAM/CE Monte Carlo code and modern ENDF/B-V cross section data. Their calculation represents the highest level of computer benchmarking short of an actual experiment.

For a flooded (0% void) new fuel vault, the SAM/CE value of K_{eff} was 0.8546 $\pm$ .0044 compared to the KENO value of 0.8587 $\pm$ .0060 using a three-dimensional model of the new fuel vault. For 97.5% void, the SAM/CE value of K_{eff} was 0.8090 $\pm$ .0048 compared to a KENO value of .8414 $\pm$ .0051. As can be seen, in these cases, the KENO calculation is more conservative than SAM/CE.

Documentation of verification is contained in Yankee Atomic Electric Company's Reports 1224 and 1343. Details of the calculations and assumptions used are contained in Yankee Atomic Electric Company's Report 1343.

2.2.9 Fuel Removal from Storage

As discussed in more detail in Sections 1.2.3 and 1.2.4, fuel assemblies are removed from storage in the shipping containers and moved to the new fuel storage vault/racks for inspection and precharacterization. Upon completion of these activities, the assemblies are moved to the spent fuel pool storage racks for storage until core load.

At no time can there be more than two (2) assemblies out of shipping containers or out of the new/spent fuel racks because of crane limitations. These assemblies will be maintained a safe distance from each other (i.e., cannot achieve a critical configuration) due to the location of the cranes and the crane travel. Specifically, one assembly could be on the overhead crane which is used for removal of fuel from shipping containers and for placement of fuel in the new fuel storage vault or the new fuel elevator. The other assembly could be on the spent fuel handling crane which is used to remove fuel from the new fuel elevator and place it in the spent fuel racks.

It would be physically possible to place the two assemblies adjacent to each other, one assembly on either crane and one in the new fuel elevator. However, this is not considered a credible configuration because it would require an intentional act on the part of the crane operator. To preclude this configuration from happening, administrative controls will be used to assure a minimum separation of four (4) feet.

If, at the time of fuel receipt, the new fuel racks are not seismically qualified using the current Amplified Response Spectra (ARS) not more than twelve (12) assemblies shall be placed in the new fuel racks for purposes of inspection and characterization. The placement of assemblies into the new fuel racks shall be administratively controlled to preclude any criticality concerns (i.e., placement of fuel in racks in at least a checkerboard fashion). Procedural controls provide adequate assurance that the checkerboard pattern is adhered to. In addition, the selected new fuel storage vault locations will be clearly marked. Administrative controls provide adequate measures to assure nuclear criticality safety because:

A. The new fuel storage racks were seismically qualified using an ARS which is approximately 67% lower than the current ARS. However, this qualification analysis was based on fully loaded racks (i.e., 90 assemblies). Therefore, limiting the number of assemblies in the fuel racks to 12 assemblies (approximately an 87% reduction) would assure the structural integrity of the racks. That is, as a minimum, it is not credible to consider gross failure of the racks due to a seismic event with the reduced fuel loading.

B. The checkerboard spacing, which would be the limiting spacing condition used under administrative control, provides greater spacing between the fuel assemblies than the as-designed spacing used in the criticality analysis for the racks with 3.1% w/o U-235 (reference Section 2.2.5).

C. The as-designed spacing is also conservative since under optimum conditions for moderation, the racks were shown to be subcritical with 4.0% w/o U-235.

D. In order to obtain optimum conditions for moderation, a double fault must have occurred as discussed in Section 2.2.4.

E. Approximately 2/3 of the fuel assemblies to be received contain either 2.4% or 1.6% U-235.

2.3 Accident Analysis

A new fuel assembly accident in which the necessary geometry and moderator are provided for criticality is highly improbable. Subsections 1.2.2, 1.2.3, and

3.0 OTHER MATERIALS REQUIRING NRC LICENSE

3.1 Irradiation Test Capsules

The Seabrook Station Unit 1 reactor vessel material surveillance program will utilize radioactive materials as dosimeters. Six test capsules will be provided for the reactor vessel, and will contain radioactive dosimeters (Uranium 238 and Neptunium 237 in a single capsule) in the following concentrations:

Uranium 238

Approximately 4.0×10^{-3} microcuries (per capsule)
Approximately 24.0×10^{-3} microcuries (total per 6 capsules)

Neptunium 237

Approximately 12.1 microcuries (per capsule)
Approximately 72.6 microcuries (total per 6 capsules)

The Uranium 238 (approximately 12 milligrams per capsule) will be supplied as U_3O_8 powder encapsulated in stainless steel capsules, and the Neptunium 237 (approximately 20 milligrams per capsule) will be supplied as NpO_2 powder encapsulated in stainless steel capsules (sealed sources). The capsules containing these radioactive materials are sealed in steel blocks which are prepared by Westinghouse and are inserted in each reactor vessel irradiation surveillance test capsule.

3.2 Material of Any Form

In addition to other special nuclear material, the Licensee will possess the following radionuclides in any chemical or physical form for use in calibration or chemical analysis:

- Plutonium-238 (100 microcuries)

- Plutonium-239 (100 microcuries)

TABLE 1.1.4-1

Fuel Assembly Design

Materials of Construction

Component	Material
Cladding	Zircaloy-4
Grid assembly	Inconel-718
Guide thimbles	Zircaloy-4
Bottom nozzle	304-SS
Top nozzle	304-SS
Top nozzle springs and bolts	Inconel-718

Core Mechanical Design Parameters

Design	RCC canless, 17 x 17
UO_2 rods per assembly	264
Rod pitch (in.)	0 496
Overall dimensions (in.)	8 426 x 8.426
Number of grids per assembly	8 - Type R
Active Fuel Length (in.)	144

Fuel Rods

Outside diameter (in.)	0.374
Diametral gap (in.)	0.0065
Clad thickness (in.)	0.0225
Clad material	Zircaloy-4

Fuel Pellets

Material	UO_2 sintered
Density (% of Theoretical)	95
Diameter (in.)	0 3225
Length (in.)	0.530