Chapter 4
INSTALLATION DESIGN

This chapter provides descriptive information on the Surry ISFSI structures, systems and components. It also provides the bases for the design criteria presented in Chapter 3.

4.1 SUMMARY DESCRIPTION

4.1.1 Location and Layout of Installation

The Surry ISFSI is located within the Surry site, as described in Section 2.1.1.

The only components with a safety function are the SSSCs. The SSSCs are stored on three nonsafety-related concrete slabs, 230-by-32-feet, to be built one at a time, as needed, within the fenced-in area shared with the low level waste (LLW) storage installation. Each slab is designed to accommodate approximately 28 casks, each approximately 8 feet in diameter and weighing no more than 125 tons, with approximately 8 feet surface to surface distance when stored in the vertical position. The exact number of casks will depend on the specific characteristics of the particular SSSCs used.

The Surry ISFSI fenced-in area is approximately 800-by-800 feet with an entrance on the south side. An inner security fence is also provided around each slab.

The layout is shown on Figure 4.1-1.

4.1.2 Principal Features

4.1.2.1 Site Boundary

A description of the area owned and controlled by Virginia Power is provided in Section 2.1 and is shown on Figure 2.1-1.

4.1.2.2 Controlled Area

As described in Section 2.1, the controlled area for the Surry ISFSI is the same as for the Surry Power Station.

4.1.2.3 Emergency Planning Zone

The Emergency Planning Zone (EPZ) for the Surry ISFSI is the same EPZ as for the Surry Power Station, and is shown on Figure 4.1-2.

4.1.2.4 Site Utility Supplies and Systems

The only utility associated with the Surry ISFSI is electrical power for lights, communications, and monitoring instrumentation as shown on Figure 4.1-3. The source of this power is described in Section 4.3.2.

4.1.2.5 **Storage Facilities**

There are no holding ponds, chemical or gas storage vessels, or other open-air tankage within the ISFSI fenced-in area.

Hazardous materials stored at, or near, the Surry site are described in Section 2.2.

4.1.2.6 **Stack**

There are no stacks at the Surry ISFSI.

Figure 4.1-1
GENERAL SITE LAYOUT

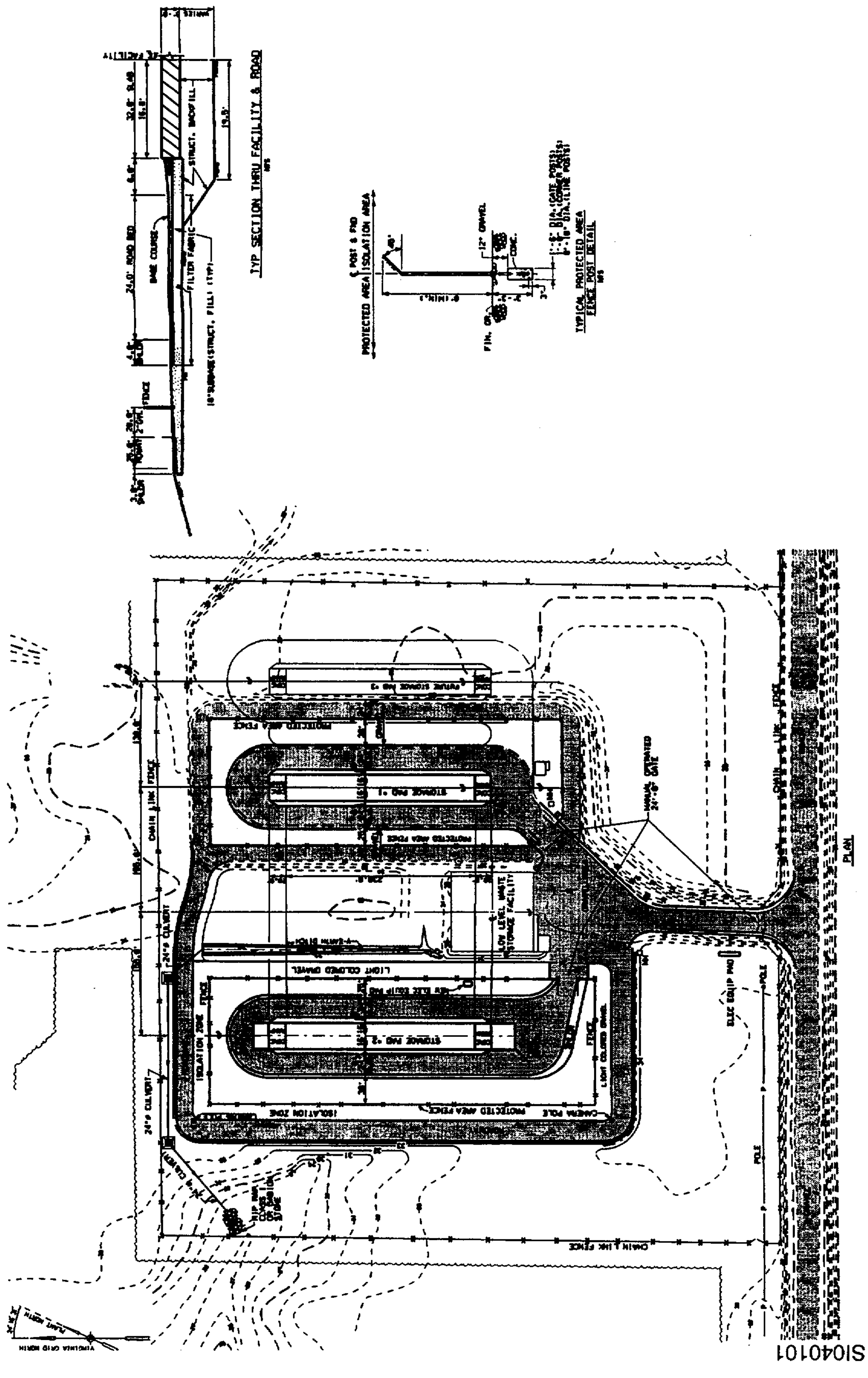


Figure 4.1-2
EMERGENCY PLANNING ZONE

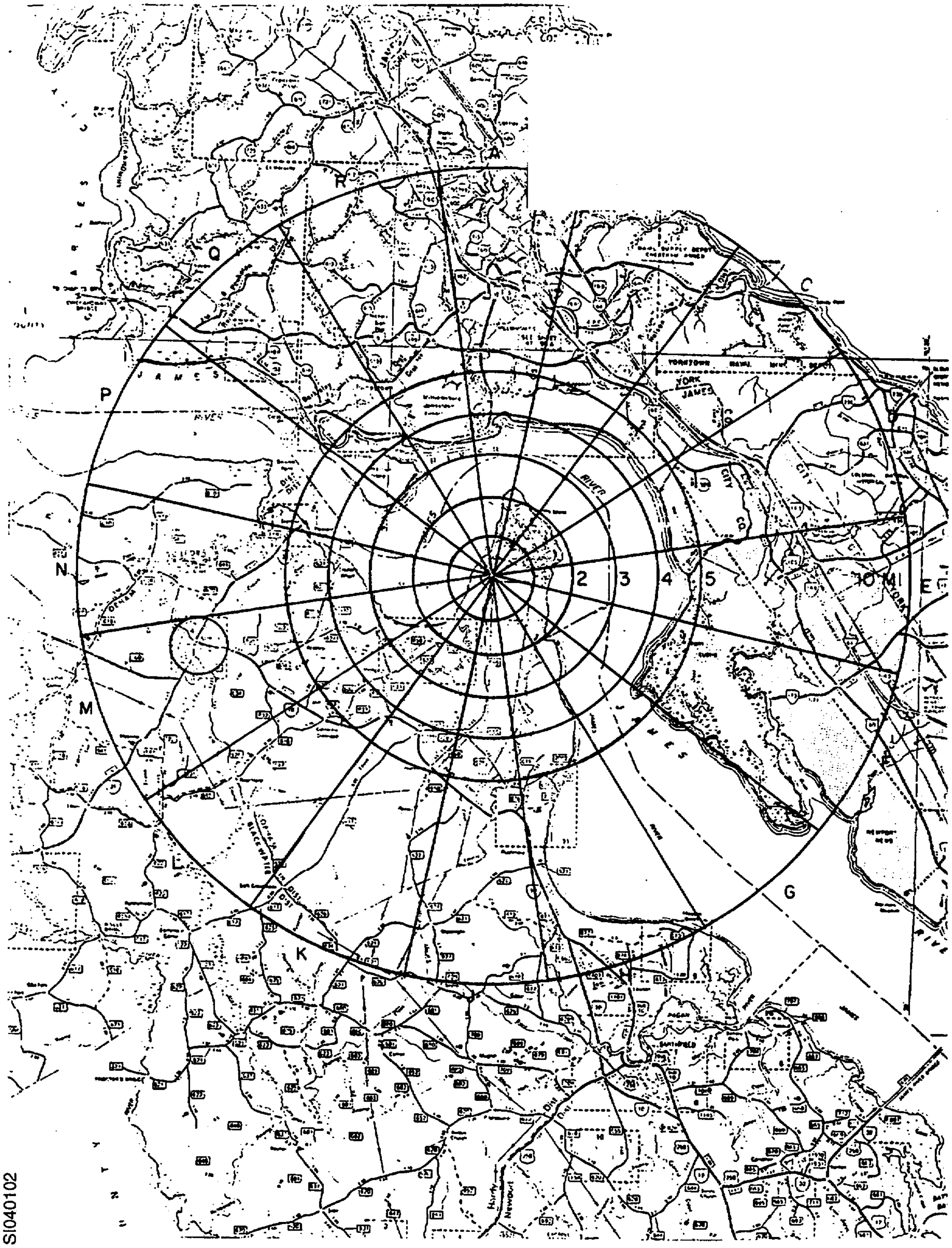
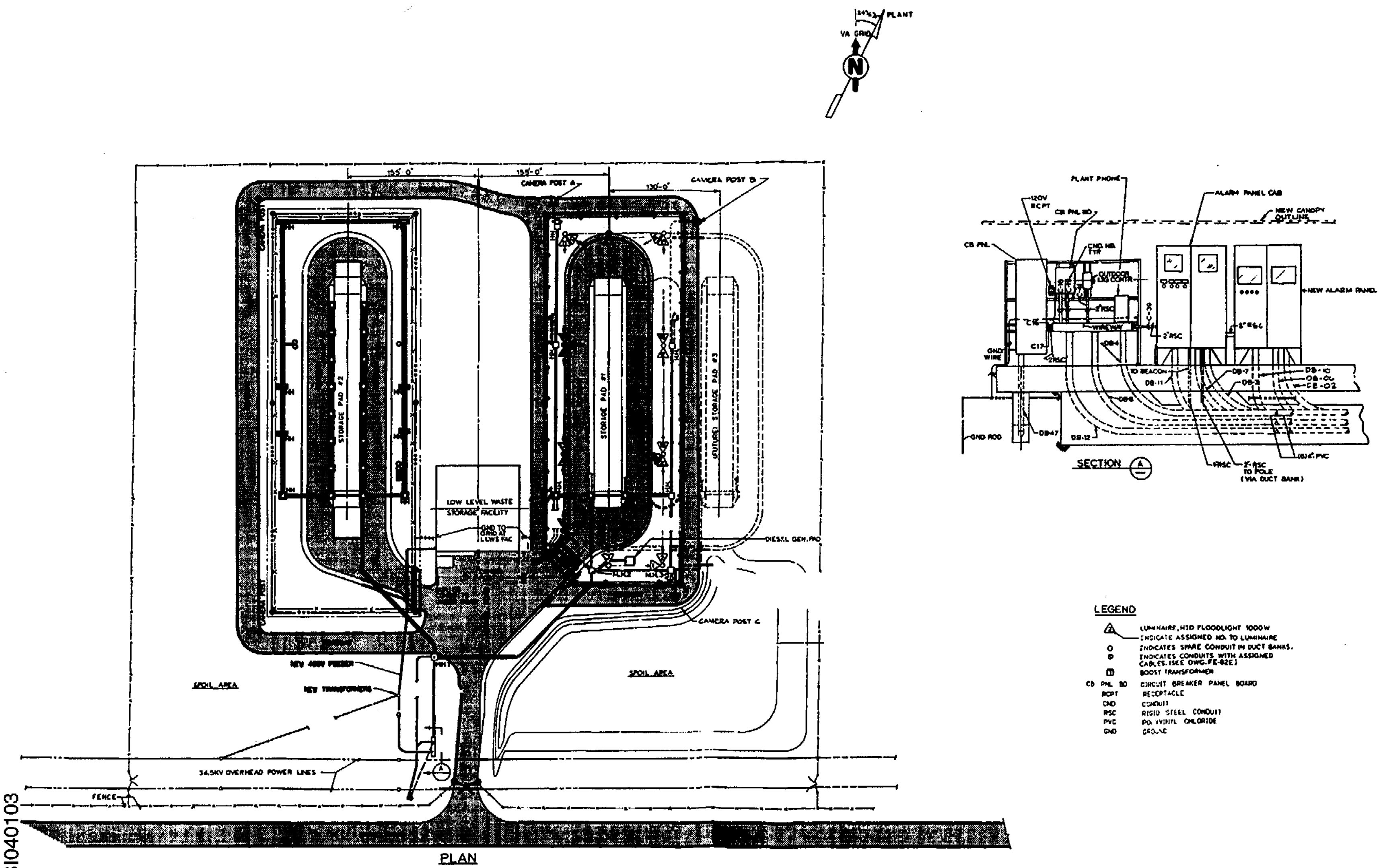


Figure 4.1-3
ISFSI ELECTRICAL EQUIPMENT LOCATION



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4.2 STORAGE STRUCTURES

The design criteria for the SSSCs are described in Chapter 3 of this SAR. These criteria are based on potential site hazards, limiting conditions for operation, and postulated accidents which the SSSCs must be able to withstand. Compliance with the specified design criteria will ensure that the requirements of 10 CFR Part 72 are satisfied.

4.2.1 Structural Specifications

The operational areas of the Surry ISFSI are the three concrete slabs and the areas immediately around them. The slabs provide a uniform level surface for storing the SSSCs. The compacted areas around the slabs allow movement and positioning of the handling equipment.

These slabs will be built in accordance with the BOCA Basic Building Code (Reference 1) and applicable American Concrete Institute codes and standards (References 2 & 3), and will be approximately 3.0-foot-thick reinforced concrete with design compressive strength of 3000 psi at 28 days. The three concrete slabs will be built within the fenced-in area adjacent to where the low level waste storage building is located. An elevation and engineering drawing of the slabs is shown on Figure 4.2-1.

The area surrounding the slabs will be compacted to properly support the haul vehicle and transporter needed for the handling of the SSSCs.

4.2.2 Installation Layout

4.2.2.1 Building Plans

The overall layout of the Surry ISFSI is described in Section 4.1.1, and is shown on Figure 4.1-1. The SSSCs are the only component of the Surry ISFSI vital to the fulfillment of its safety function. All other structures and components are of a support nature and do not perform safety functions.

The most important of these support systems are the concrete slabs, which provide a uniform level surface, slightly above grade elevation, for the SSSCs. These are described in Section 4.2.1.

4.2.2.2 Building Sections

There are no building sections as such. However, engineering drawings showing section and details of the concrete base mat are presented on Figure 4.2-1.

4.2.2.3 Confinement Features

Confinement of radioactivity is accomplished solely by the SSSCs and is not dependent upon the particular layout of the installation. Therefore, other than the SSSCs themselves, no confinement features are provided at the ISFSI. Analyses of the casks' ability to perform their confinement function are provided in the SSSC topical reports.

4.2.3 Individual Unit Description

The bases and engineering design specifications for the SSSCs are described in the SSSC topical reports. These reports also provide assurance that the applicable design criteria described in Chapter 3 are met.

In turn, compliance with the design criteria ensures that the General Design Criteria in Subpart F of 10 CFR Part 72 are satisfied. This is illustrated in Table 4.2-1 which shows a correspondence between each of the General Design Criteria and the design criteria of specifications that the SSSCs must meet as identified in Chapter 3.

4.2.3.1 **Functions**

Descriptions of the fuel loading, cask preparation, and cask placement operations are provided in Chapter 5.

Performance objectives during fuel loading are to transfer the selected fuel assemblies from their storage location to the SSSC without damaging the fuel. All operations within and outside the fuel building will be conducted in a manner that does not jeopardize the safe operation of the Surry Power Station, does not present a hazard to the stored fuel, and does not result in releases of radioactive gases in excess of the guidelines in 10 CFR Part 100.

Performance objectives for the post-loading activities are to ensure that the casks can fulfill all of their design functions, and in particular that the casks will confine the radioactive products under all credible conditions.

Cask transfer and emplacement operations will be performed according to procedures which will ensure that the design criteria are not exceeded, and that the safety of the Surry Power Station is not impaired.

4.2.3.2 **Components**

A description of the components used for loading, preparing, and handling the SSSCs is provided in Chapter 5.

4.2.3.3 **Design Bases and Safety Assurance**

The ability of the SSSCs to perform their design function is demonstrated in the SSSC topical reports or Appendix A .

Loading and handling of the casks will be done according to the applicable procedures.

As described in Chapter 8, the design and operation of the Surry ISFSI ensure that a single failure does not result in the release of significant radioactive material.

The interactions between the ISFSI and the Surry Power Station are primarily those concerning the loading and handling of the casks in the fuel handling and decontamination buildings. These are discussed in Chapter 5.

Radiation protection of operating personnel is addressed in Chapter 7.

Nuclear criticality safety for the SSSCs is addressed in the SSSC topical reports or Appendix A.

4.2.4 References

1. *BOCA Basic Building Code*, Building Officials and Code Administrations International, Inc., 1981.

2. *Building Code Requirements for Reinforced Concrete,* American Concrete Institute, ACI 318-77, and 1980 Supplement and Commentary.

3. *Manual of Standard Practice for Detailing Reinforced Concrete Structures,* American Concrete Institute, ACI 315-74.

Table 4.2-1
COMPLIANCE WITH GENERAL DESIGN CRITERIA
(SUBPART F, 10 CFR PART 72)

§72.122	(a)	Quality standards	The design criteria require that the SSSCs be designed, fabricated, delivered to the site, and sealed according to recognized commercial codes and standards and in accordance with Vepco's QA program for safety-related equipment
	(b)	Protection against environmental conditions and natural phenomena	Extreme environmental conditions are defined in Chapter 2. The design criteria require that the SSSCs be designed to withstand the Design Earthquake, high ambient temperature and humidity, exposure to sunlight, and extreme winds.
	(c)	Protection against fire and explosions	No large fire within the Surry ISFSI is considered credible. The design criteria require that SSSCs be designed to withstand extreme ambient temperatures and peak overpressure resulting from postulated nearby explosions.
	(d)	Sharing of structures, systems, and components	The ISFSI activities will be done without jeopardizing the safe shutdown capability of the Surry Power Station Units 1 and 2.
	(e)	Proximity of sites	The design and operation of the Surry ISFSI result in minimal additions of risk to the health and safety of the public.
	(f)	Testing and maintenance of systems and component	The SSSCs require minimum maintenance. The design criteria require that the SSSCs be capable of being inspected and monitored.
	(g)	Emergency capability	Scenarios requiring emergency actions are neither considered credible, nor postulated to occur. Nevertheless, all emergency facilities at the Surry Power Station would be available if needed
	(h)	Confinement barriers and systems	The design of the SSSCs will ensure that the stored fuel is maintained in a safe condition. No paths for radioactive releases are considered credible. Therefore, no ventilation or offgas systems are needed.

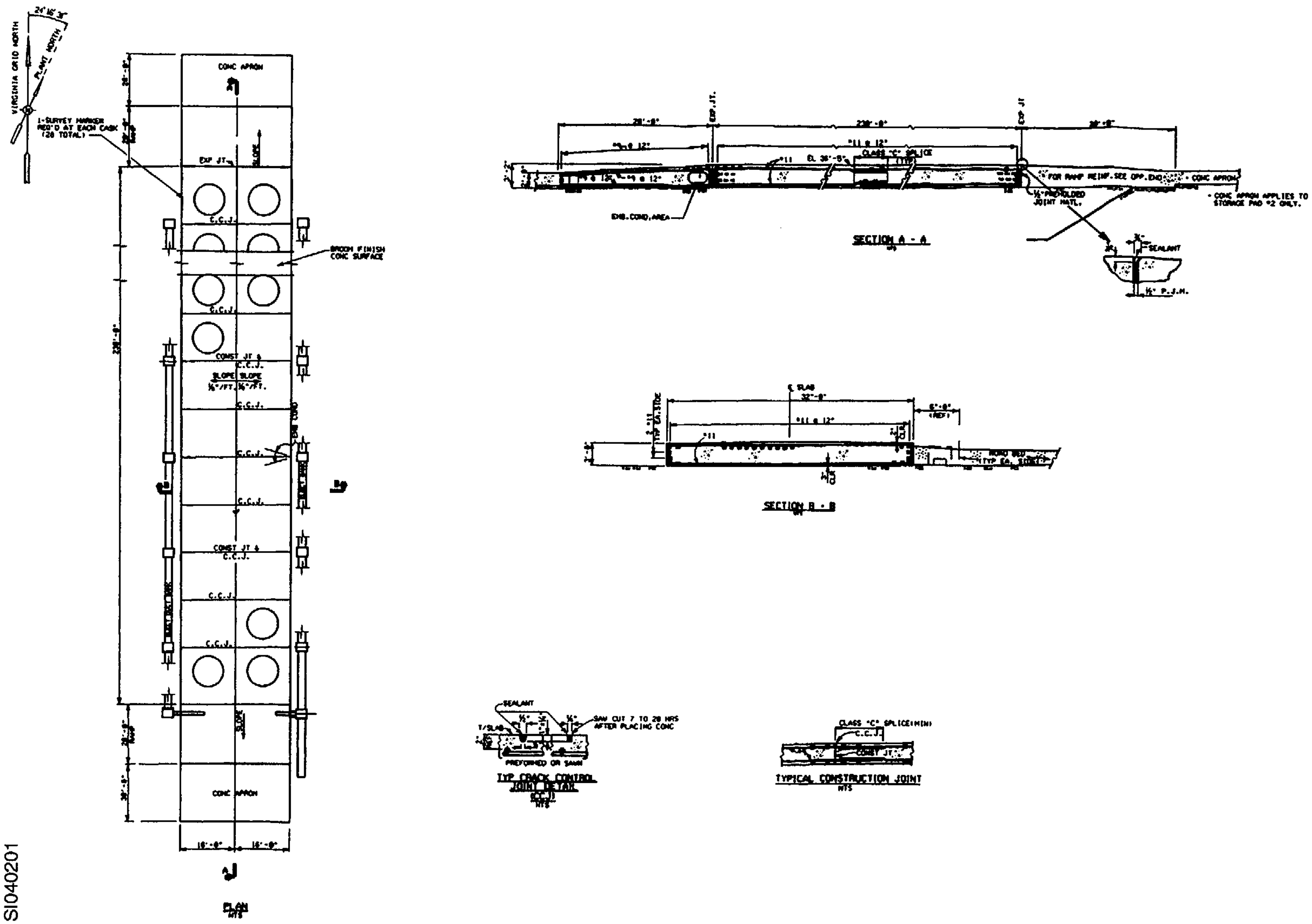
Table 4.2-1 (continued)
COMPLIANCE WITH GENERAL DESIGN CRITERIA
(SUBPART F, 10 CFR PART 72)

	(i)	Instrumentation and control systems	No instrumentation or control systems are needed for the SSSCs to perform their safety functions. Nevertheless, some monitors and alarms will be provided.
	(j)	Control room or control areas	The Surry ISFSI is a passive installation, with no need for operator actions. Thus, no control room is needed.
	(k)	Utility services	The SSSCs are the only safety-related components at the Surry ISFSI. There are no utility or emergency systems required to perform any safety functions at the Surry ISFSI.
§72.124	(a)	Design for criticality safety	The design criteria require that the SSSCs be designed to maintain subcriticality at all times, assuming a single active or credible passive failure.
	(b)	Methods of criticality control	Different SSSC designs may use different methods of criticality control. However, all designs use conservative analyses and specified error contingency criteria.
§72.126	(a)	Exposure control	Operations at the Surry ISFSI will be done according to ALARA procedures. Minimal maintenance operations are needed following SSSC emplacement at the ISFSI. SSSC loading, sealing, decontamination, and preparation are done at the fuel and decontamination buildings according to health physics procedures in effect for the Surry Power Station.
	(b)	Radiological alarm systems	No radioactive releases are considered credible at the Surry ISFSI. No safety-related alarm systems are needed.
	(c)	Effluent and direct radiation monitoring	Operation of the Surry ISFSI does not result in radioactive contamination of any effluents. No safety-related monitors are needed. Direct radiation monitors will be installed around the ISFSI.

Table 4.2-1 (continued)
COMPLIANCE WITH GENERAL DESIGN CRITERIA
(SUBPART F, 10 CFR PART 72)

	(d)	Effluent control	No radioactive releases are considered credible at the Surry ISFSI.
§72.128	(a)	Spent fuel and radioactive waste storage and handling systems	The design criteria require that the SSSCs have adequate provisions to monitor the SSSC performance, provide sufficient shielding to lower surface doses to below prescribed levels, maintain leak tightness under all operating and credible conditions, and maintain fuel in a safe condition. Only minimal amounts of radioactive waste are generated in the decontamination of the casks.
	(b)	Waste treatment	Radioactive wastes generated in the decontamination of the SSSCs are processed by the Surry Power Station waste processing systems.
§72.130		Decommissioning	Operation of the Surry ISFSI does not result in contamination of the outside surface of the SSSCs or any other ISFSI components. Therefore, there is no need for provisions to facilitate decommissioning.

Figure 4.2-1
CONCRETE BASE MAP



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4.3 AUXILIARY SYSTEMS

The Surry ISFSI does not rely upon auxiliary systems for the performance of its safety functions. No safety-related auxiliary systems are required, and none are provided.

4.3.1 Ventilation and Offgas Systems

Ventilation and offgas systems are not required for the Surry ISFSI and none are provided.

Ventilation is not needed because the ISFSI design features the SSSC array in an open arrangement which allows cooling to take place by natural heat convection. Hence, no forced ventilation is needed.

Offgas systems are not required because the casks are double sealed, and there are no credible scenarios that could result in radioactive releases.

4.3.1.1 Major Components and Operating Characteristics

This subsection does not apply to the Surry ISFSI for reasons stated in Section 4.3.1.

4.3.2 Electrical Systems

Electric power is not required to support functions of the Surry ISFSI important to safety. A discussion of power for security equipment is provided in the Security Program (Reference 1).

Nonsafety-related electric power is supplied to the ISFSI for purposes of lighting, general utility, and instrumentation with which cask seal integrity is monitored. Cask temperature may also be monitored, depending on specific cask design. These functions are supportive in nature, and are not needed for effective SSSC function.

4.3.2.1 Major Components and Operating Characteristics

The source of electric power is obtained from a 34.5/0.48 kV transformer which also feeds the low level waste storage facility. The 34.5 kV line is normally fed from an offsite power source, but can be manually transferred to the station switchyard. The low level waste storage facility transformer provides power to ISFSI loads through a separate feeder and disconnect and distribution panel which are located near the ISFSI local annunciator. This distribution panel will provide feed to loads for all three pads. Service power for lighting and welding receptacles is 480V, 60 Hz, single or three phase.

4.3.2.2 Safety Considerations and Controls

Since only the casks are important to safety and since the casks do not require electric power to perform their functions, loss of electricity will not jeopardize the safety of the facility.

The ISFSI is a passive installation. There are no operations to control, no motorized fans, dampers, louvers, or valves, and no electrically operated cranes or lifts. Electricity is required only for monitoring equipment and convenience lighting. A power loss will result in no more than

a temporary loss of data. This is not considered of major significance because there are no conditions under which the parameters monitored will change abruptly.

4.3.3 Air Supply Systems

Since there are no airborne contaminants associated with the ISFSI, neither compressed air nor breathing air supply systems are required or provided.

4.3.3.1 **Compressed Air**

This section does not apply to the Surry ISFSI for reasons stated in Section 4.3.3.

4.3.3.2 **Breathing Air**

This section does not apply to the Surry ISFSI for reasons stated in Section 4.3.3.

4.3.4 Steam Supply and Distribution System

Steam is not required at the Surry ISFSI, and none is provided, because the SSSCs do not require steam for heat, motive power, or any other reasons. No other feature of the ISFSI requires steam.

4.3.4.1 **Major Components and Operating Characteristics**

This section does not apply to the Surry ISFSI for reasons stated in Section 4.3.4.

4.3.4.2 **Safety Considerations and Controls**

This section does not apply to the Surry ISFSI for reasons stated in Section 4.3.4.

4.3.5 Water Supply System

Water is not required at the Surry ISFSI, and none is provided because the SSSCs do not require a continuous water supply for cooling, makeup, cleaning, or any other reason.

Potable water is not required because the ISFSI is only manned on an infrequent basis by a small number of people during cask handling operations and inspections.

Cask washdown is not done while the casks are stored at the ISFSI.

Decontamination of the casks takes place at the Surry Power Station decontamination building prior to their transfer to the ISFSI.

Fire suppression water is not required because no large credible fire exists.

4.3.5.1 **Major Components and Operating Characteristics**

This section does not apply to the Surry ISFSI for reasons stated in Section 4.3.5.

4.3.5.2 Safety Considerations and Controls

This section does not apply to the Surry ISFSI for reasons stated in Section 4.3.5.

4.3.6 Sewage Treatment System

Neither sanitary nor chemical sewage is produced at the Surry ISFSI. During the infrequent periods of manning for cask transfer operation, portable sanitary facilities may be provided in the vicinity of but not directly in the ISFSI. Chemical wastes, such as small amounts of ethylene glycol (antifreeze) or drips of lubricating fluid from transport vehicles could be cleaned up manually and disposed of at appropriate facilities of the Surry Power Station. No permanent sewage treatment system is required or provided.

4.3.6.1 Sanitary Sewage

This section does not apply to the Surry ISFSI for reasons stated in 4.3.6.

4.3.6.2 Chemical Sewage

This section does not apply to the Surry ISFSI for reasons stated in 4.3.6.

4.3.7 Communication and Alarm Systems

4.3.7.1 Major Components and Operating Characteristics

The ISFSI is not manned on a continuous basis. Some SSSCs will be provided with a pressure sensing device to monitor their seal tightness. Some casks may also be monitored for temperature. This instrumentation is not required for safe operation of the ISFSI and therefore will not be safety related.

The monitoring devices will actuate a pressure or temperature switch, as applicable, at a preset alarm level. Specific recommendations for monitoring the casks are provided in SSSC topical reports. Each of the cask alarms (maximum of two per cask) will initiate an annunciator lamp in the local annunciator at the ISFSI. The alarm point will indicate the specific cask and parameter in question and will remain lit until reset.

In addition, the initiation of any alarm will energize a flashing light visible to personnel doing Surry site monitoring.

4.3.7.2 Safety Considerations and Controls

Degradation of an SSSC primary seal is considered extremely unlikely. Nevertheless, in the event that this were to occur, the pressure sensor would activate an alarm.

Upon identification of the affected SSSC, a series of actions identified in the ISFSI procedures will be taken. Depending on the exact circumstances, these may include monitoring the cask pressure with a time recorder in order to ascertain whether the failure is a progressing one, and checking for possible instrumentation failure. If a failure of a seal is ascertained,

arrangements will be made to fix the cask in place, to transfer the cask to the fuel building for repair work, if necessary, or any other action recommended by the manufacturer and included in the Surry ISFSI operating procedures.

It should be remembered that the hypothesized seal failure addressed in this section would not result in radioactive releases because of the double-seal nature of the SSSCs.

The ISFSI operating procedures will be prepared to provide step-by-step actions to be taken for all kinds of alarms. These will be prepared according to the specific SSSC manufacturers' designs and recommendations.

4.3.8 Fire Protection System

As described in Section 8.2.5, no fires other than small electrical fires are considered credible at the Surry ISFSI, and separation has been provided for security-related equipment.

Therefore, the Surry ISFSI does not include a fire protection system, other than portable fire extinguishers which will be available within the ISFSI. In addition, the fire fighting equipment and personnel present at the Surry Power Station would be available if needed.

4.3.8.1 Design Bases

This section does not apply to the Surry ISFSI for reasons stated in Section 4.3.8.

4.3.8.2 System Description

This section does not apply to the Surry ISFSI for reasons stated in Section 4.3.8.

4.3.8.3 System Evaluation

This section does not apply to the Surry ISFSI for reasons stated in Section 4.3.8.

4.3.8.4 Inspection and Testing Requirements

This section does not apply to the Surry ISFSI for reasons stated in Section 4.3.8.

4.3.8.5 Personnel Qualification and Traininq

This section does not apply to the Surry ISFSI for reasons stated in Section 4.3.8.

4.3.9 Maintenance Systems

Major maintenance operations are not required at the Surry ISFSI. Cask design features have been included to minimize or eliminate maintenance. The SSSCs are either coated with a polymer protection or are made of corrosion-resistant material such as stainless steel. Other equipment, instrumentation, etc., will be specified and selected to withstand the effects of the environment at the site. Specific maintenance recommendations for the casks are provided in the SSSC topical reports.

The Surry ISFSI does not include active components such as remotely operated equipment or hot cells, nor is an active ventilation system required or provided.

Fuel stored in the Surry ISFSI will be in its original unconsolidated form. Handling of the fuel and cask loading will be done with systems and equipment that are presently in use for this or equivalent purposes in the Surry fuel building. Since cask loading does not present unique or new handling procedures, extraordinary equipment contamination is not expected. Hence, the need for disposal of contaminated equipment is not expected.

4.3.9.1 **Major Components and Operating Characteristics**

This section does not apply to the Surry ISFSI for reasons stated in Section 4.3.9.

4.3.9.2 **Safety Consideration and Controls**

This section does not apply to the Surry ISFSI for reasons stated in Section 4.3.9.

4.3.10 Cold Chemical Systems

No chemical operations are required for the Surry ISFSI, and no chemical storage, handling, process, or other system involving chemical reactions are planned or provided.

Cask designs featuring liquid neutron shields typically use a mixture of water and ethylene glycol (a common antifreeze) in the shield volume. Ethylene glycol is not a hazardous chemical when used for the purpose stated.

New casks may be shipped under an internal nitrogen (or other inert gas) blanket and may employ desiccants such as silica-gel. These materials are not hazardous when used for this purpose.

4.3.11 Air Sampling Systems

Air sampling systems are not required at the Surry ISFSI as discussed in Section 3.3.2.1.

4.3.11.1 **Major Components and Operating Characteristics**

This section does not apply to the Surry ISFSI for reasons stated in Section 4.3.11.

4.3.11.2 **Safety Considerations and Controls**

This section does not apply to the Surry ISFSI for reasons stated in Section 4.3.11.

4.3.12 Reference

1. North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, and Independent Spent Fuel Storage Installations Physical Security Plan.

4.4 DECONTAMINATION SYSTEMS

There are no credible mechanisms which could result in contamination of the outside surface of the SSSCs, other ISFSI components, or operating personnel, after the casks leave the fuel building. Therefore, the Surry ISFSI does not include provisions for decontamination.

Decontamination of the casks after they have been loaded is done within the decontamination building of the Surry Power Station, as described in Chapters 5 and 6.

4.4.1 Equipment Decontamination

This section does not apply to the Surry ISFSI for reasons stated in Section 4.4.

4.4.1.1 Major Components and Operating Characteristics

This section does not apply to the Surry ISFSI for reasons stated in Section 4.4.

4.4.1.2 Safety Considerations and Controls

This section does not apply to the Surry ISFSI for reasons stated in Section 4.4.

4.4.2 Personnel Decontamination

This section does not apply to the Surry ISFSI for reasons stated in Section 4.4.

4.5 SHIPPING CASK REPAIR AND MAINTENANCE

Incidental mechanical operations involving the storage casks include receiving of the new casks from the supplier, their temporary (empty) storage, and transfer to the fuel building. During these operations, the casks will be inspected in detail and abnormalities corrected. The facilities and machine shops of the Surry Power Station will be made available in the event repair operations become necessary.

No repair operations are anticipated once the casks are placed into storage. Periodic maintenance is not required beyond instrument adjustments and other similar evolutions of a minor nature, such as touching up defects in outer decontamination coatings. These can all be performed within the ISFSI area, without need to move the SSSCs.

4.6 CATHODIC PROTECTION

In general, cathodic protection is not required for the SSSCs since the surrounding medium for the SSSC is air which is a poor electrolyte. Hence, protection from electrolytic decomposition of the SSSCs is not required. See the SSSC topical reports for discussions of provisions for cathodic protection systems.

4.7 FUEL HANDLING OPERATION SYSTEMS

There are no fuel handling facilities exclusively dedicated to the Surry ISFSI. Loading, preparation, decontamination, and testing of the SSSCs take place within the fuel and

decontamination buildings of the Surry Power Station. These operations are described in detail in Chapter 5.

4.7.1 Structural Specifications

This section does not apply to the Surry ISFSI for reasons stated in Section 4.7.

4.7.2 Installation Layout

This section does not apply to the Surry ISFSI for reasons stated in Section 4.7.

4.7.3 Individual Unit Description

This section does not apply to the Surry ISFSI for reasons stated in Section 4.7.

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Chapter 5
OPERATIONS SYSTEMS

This chapter describes the operations associated with the Surry ISFSI. As indicated in previous chapters, the Surry ISFSI is a totally passive installation, requiring no actions or maintenance for its proper functioning. The operations described in this chapter relate to the loading and preparation of the SSSCs and their transfer to the ISFSI.

5.1 OPERATIONS DESCRIPTION

5.1.1 Narrative Description

The loading and preparation of the SSSCs take place within the Surry Power Station fuel and decontamination buildings. These operations are essentially those which are followed for the loading of fuel into shipping casks, as described in Section 9.12 of the Surry Power Station FSAR.

A Technical Specification change request (Reference 1) to the Surry Power Station Units 1 and 2 operating license addressed the handling of the SSSCs within the fuel and decontamination buildings. As concluded in the NRC's Safety Evaluation (Reference 2), these operations are conducted in a manner that ensures that the capability to safely operate the power station is not jeopardized. Specifically, the consequences of a postulated cask drop have been evaluated and have been determined to meet the guidelines of NUREG-0612, Section 5.1 (Reference 3).

Following loading and decontamination, the SSSCs are moved to the crane enclosure where they are picked up by the transporter that transfers them to the ISFSI. The path followed by the transporter from the decontamination building to the ISFSI is shown on Figure 5.1-1. This figure also shows the location of all nearby systems and structures needed for the safe shutdown of the power station. Drop of an SSSC while in transit to the ISFSI will not result in damage to any of these systems and structures, nor in radioactive releases in excess of the guidelines in 10 CFR Part 100.

The transfer path shown on Figure 5.1-1 is a compacted gravel road capable of holding the transporter and SSSC. Other heavy equipment, including the replaced steam generators, has been moved along this road. The road is maintained clear of obstacles.

As indicated in Section 3.3.1, the design criteria require that the SSSCs maintain their integrity, preclude physical damage to the fuel, and ensure subcriticality following a cask drop onto ISFSI pad. Operating procedures will limit the lifting heights once the casks are placed onto the transporter.

Therefore, none of the operations needed to emplace the SSSCs at the Surry ISFSI will result in unacceptable damage to the Surry Power Station Units 1 and 2, or to the stored spent fuel.

5.1.2 Flowsheets

Table 5.1-1 shows a typical sequence of operations performed before the SSSCs are placed on their storage position at the ISFSI. Operations more specific to a particular vendor's casks are outlined in the vendor's SSSC topical report.

These operations are performed in accordance with procedures addressing health physics and handling of the SSSCs. They also fulfill the surveillance requirements specified in Chapter 10.

Wastes resulting from the decontamination process are handled by the Surry Power Station radioactive waste disposal systems, as described in Section 6.3.2.1.

Descriptions of equipment used in these operations are provided in Section 5.2.

5.1.3 Identification of Subjects for Safety Analysis

5.1.3.1 Criticality Prevention

The design criteria specified in Section 3.3.4 require that spent fuel stored at the Surry ISFSI be maintained subcritical at all times. The specific means by which the casks comply with this criterion are described in the SSSC topical reports or Appendix A.

5.1.3.2 Chemical Safety

Section 2.2 describes the hazardous chemicals stored at, or transported in the vicinity of, the Surry site and their potential effects on the safety of the ISFSI. As a result of these hazards, a design criterion regarding overpressure protection has been placed on the SSSCs.

The Surry ISFSI does not require operator actions for its safe operation and is not continuously manned. Therefore, the presence of chemicals in the vicinity of the Surry ISFSI does not result in an undue risk to the safe storage of spent fuel.

5.1.3.3 Operation Shutdown Modes

The Surry ISFSI is a totally passive installation with no actions needed for the fulfillment of its safety functions. Thus, this section is not applicable.

5.1.3.4 Instrumentation

Due to the totally passive and inherently safe nature of the SSSCs, there is no need for any instrumentation to perform safety functions. Nevertheless, it may be desirable to monitor the performance of some or all of the SSSCs. Accordingly, the design criteria described in Section 3.3.3.2 require that the SSSCs have adequate provisions for the installation, testing, and calibration of monitors.

The parameters to be monitored will be selected based on recommendations made by the SSSC manufacturers, experience gained with specific SSSC designs, and other engineering and

health physics considerations. Instrumentation provisions for the casks are described in the SSSC topical reports.

Although these instruments are not safety related, commitments for their installation, inspection, and calibration and replacement, if needed, are proposed in Section 10.9.

Actions to be taken when monitored parameters exceed preset levels are described in Section 4.3.7.

5.1.3.5 Maintenance Techniques

Because of their passive nature, the SSSCs require little, if any, maintenance over the lifetime of the ISFSI. No major maintenance tasks are required. Typical maintenance tasks would involve occasional replacement and recalibration of monitoring instrumentation and recoating of some casks with corrosion-inhibiting coatings. No special maintenance techniques are necessary.

Specific maintenance recommendations for the casks are provided in the SSSC topical reports.

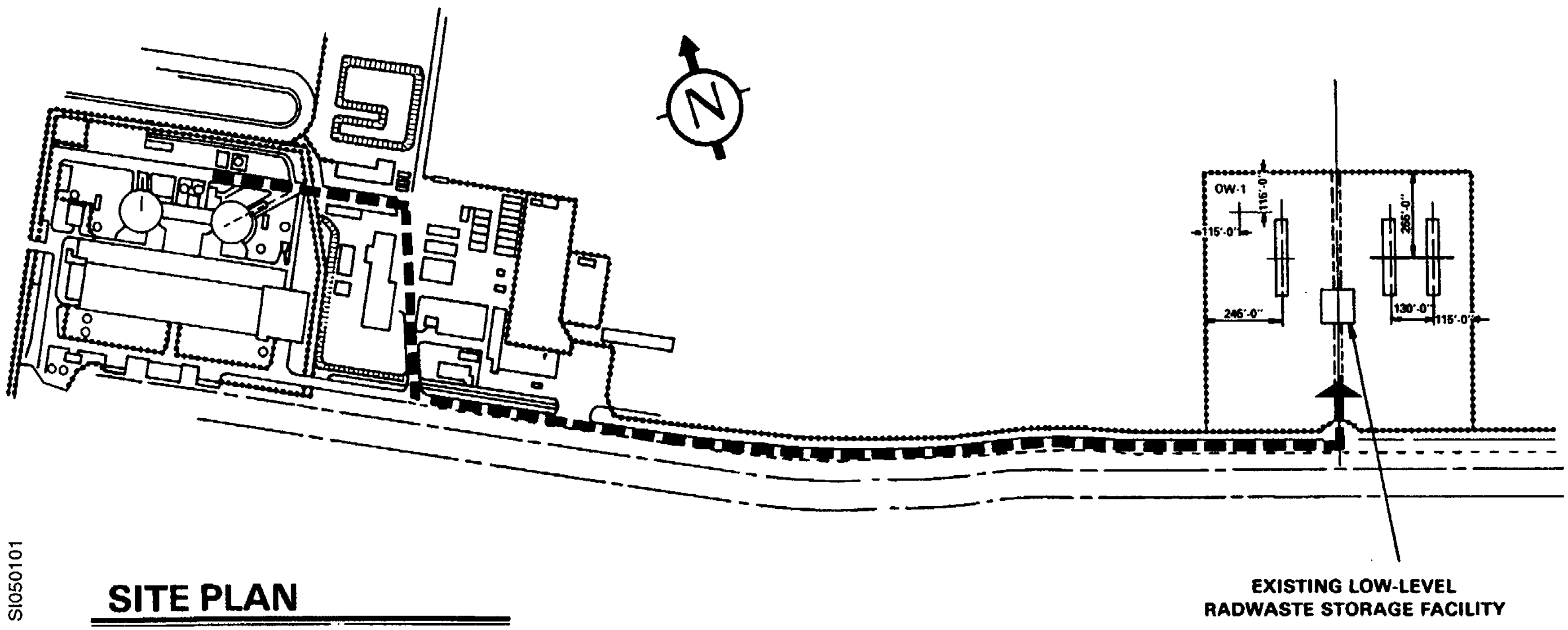
5.1.4 References

1. Letter No. 543 from R. H. Leesburg to Harold Denton, *Amendment to Operating Licenses DPR 32 and 37, Surry Power Station Units 1 and 2, Proposed Technical Specification Changes*, September 23, 1982.
2. NRC Safety Evaluation, Amendment Nos. 84 and 85 (Serial No. 131) to Facility Operating License Nos. DPR-32 and DPR-37, March 4, 1983.
3. NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants*, July 1980.

Table 5.1-1
TYPICAL SEQUENCE OF OPERATIONS

1. Unload empty SSSC outside decontamination building using fuel cask trolley.
2. Move SSSC inside decontamination building.
3. Perform visual inspection of seals.
4. Move SSSC to fuel building.
5. Lower SSSC into the spent fuel pool cask loading area.
6. Load SSSC with preselected spent fuel assemblies using spent fuel handling crane.
7. Reconfirm inventory of fuel assemblies loaded into SSSC.
8. Place primary lid on SSSC.
9. Dewater cask.
10. Lift SSSC out of spent fuel pool.
11. Wash down exterior.
12. Move SSSC to decontamination building.
13. Decontaminate outside surface of SSSC.
14. Secure primary lid or place secondary lid, as applicable.
15. Dry cask and contents.
16. Perform radiation measurements.
17. Install thermocouples or other instrumentation, as applicable.
18. Pressurize SSSC and test seals (If applicable, place secondary lid, pressurize, and test seals).
19. Move cask to crane enclosure.
20. Load SSSC on transporter.
21. Transfer to ISFSI.
22. Perform radiation measurements.
23. Connect appropriate instrumentation.

Figure 5.1-1
TRANSFER PATH



5.2 FUEL HANDLING SYSTEMS

5.2.1 Spent Fuel Receipt, Handling, and Transfer

The equipment associated with the receipt, handling, and transfer of the SSSCs (and its associated spent fuel) is described below:

1. The casks are handled in the fuel building by the fuel cask trolley crane. The crane runs on fixed rails and has a capacity of 125 tons. The rails span the east end of the spent fuel pool where the cask loading area is located and pass over the decontamination building to the roadway where the cask will be loaded onto the cask transporter.
2. The fuel handling crane is used to load the spent fuel assemblies into the cask.
3. The fuel cask trolley crane transfers the sealed casks from the fuel pool to the decontamination building and to the crane enclosure.
4. A cask transporter will be used to transfer the SSSC from the crane enclosure to the ISFSI. This transporter is an A-frame design which carries the cask in a vertical orientation. It is about 14 feet wide, 21 feet long, and 25 feet high, and it weighs approximately 56,000 pounds. It has a tow bar pivot and four steerable wheels as well as four fixed wheels with foam filled pneumatic tires to ensure maximum stability, maneuverability, and even-load distribution. A haul vehicle will be used to pull the transporter and cask.
5. The transporter is equipped with hydraulic lift cylinders which will be used to place the casks into storage at the ISFSI. A similar procedure would be used in reverse when the cask is shipped off site for final disposition.

The cask is cooled by convective and radiant heat transfer, and as described in Chapter 4, no forced cooling is required or provided.

Provisions for maintaining the fuel assemblies in a subcritical array are described in Section 5.1.3.1.

Provisions for shielding the fuel assemblies are described in Sections 4.2 and 7.3.

5.2.1.1 **Functional Description**

A functional description of the systems used to load and transport the storage casks is given in Section 5.2.1. A flow diagram of this process is shown in Table 5.1-1. No defective fuel needing special handling provisions will be placed in the storage casks.

5.2.1.2 **Safety Features**

Handling of fuel is done according to procedures in effect for the Surry Power Station, as summarized in Section 5.2.1. The proper use of this equipment limits the possibility of mishandling the fuel.

The fuel handling crane handles the spent fuel under the protective cover of the spent fuel pool water. The effects of any mishandling of the fuel, such as a fuel handling accident, is discussed in Section 8.2.6.

The cask transporter is used on a graded road (shown on Figure 5.1-1) which limits the possibility of dropping the cask. The transporter is also equipped with hydraulic lift links and a hydraulically actuated restraint system to help prevent the cask from dropping. Even if the cask should drop from the transporter, it could only drop the maximum lift height allowed by the Technical Specifications. This drop would not damage the cask nor its contents, and would not result in any radioactive releases.

5.2.2 Spent Fuel Storage

All handling of the fuel and the SSSCs within the fuel and decontamination buildings is done according to the procedures in effect at the Surry Power Station.

After the loading is completed, a final verification of the fuel assemblies loaded into the cask will be performed. The primary lid will then be put into place, and the SSSC lifted out of the spent fuel pool to be dewatered.

The cask trolley crane then moves the casks to the decontamination building where they are dried. Here the secondary lid, if applicable, is placed and the casks are sealed and decontaminated. The cask trolley crane then moves the casks out of the decontamination building and places them in the crane enclosure for pickup by the transporter.

The transporter is pulled to the cask storage location by a haul vehicle. The transporter then unloads the cask and places it in its storage position.

Once the cask is in its storage position, the cask monitoring instrumentation is connected. This is described in Sections 5.1.3.4 and 5.4.1.

5.2.2.1 **Safety Features**

The safety of the Surry ISFSI resides mainly in the multiple-barrier confinement function provided by the SSSCs and the lack of active components needed for their safety functions. In addition, the design criteria specified in Chapter 3 ensure that these safety functions are not jeopardized by possible hazardous conditions to which the SSSCs may be exposed; e.g., natural phenomema, or by postulated accidents.

The casks are designed to withstand potential conditions experienced during normal or off-normal handling as described in Sections 3.3 and 5.2. Operating procedures, where necessary, will ensure that the casks are handled within these limits. The transporter equipment, utilized in the handling of casks, was selected based on adequacy for the operations to be performed and was verified in writing to be in compliance with all applicable codes and standards prior to cask handling operations. This was established by prior engineering review of the entire movement and documented in formal procedures. Adequate supervision, engineering, and health physics

coverage will be provided to ensure that the equipment is used properly and that prewritten operating procedures are followed. These procedures will include:

1. Location and stable position on the transporter (number, type, location, and strength requirements of attachments)
2. Maximum speed of the transporter
3. Required plant support groups to be present during the move
4. Organization chart of responsible parties
5. Defined haul path
6. Allowable environment limits (high wind, etc.)
7. Maximum height above surface(s) which may be employed during all cask motions
8. Reference to proper positioning, lowering, and leveling procedures through load release
9. Check lists required for all milestone points throughout move
10. An emergency list of requirements for a dropped cask will be developed, including those described in Section 8.2.10.

These would be applicable for movement from the plant to the ISFSI, for movements of casks(s) within the ISFSI, or return trips from the ISFSI to the plant. During transport, it is not envisioned necessary but would be acceptable to stop the transporter and/or rest the cask on the ground for a short time (e.g., a day). These contingencies and associated actions such as temporary security, health physic coverage, cleaning, etc. will be included in the procedures. Therefore, no physical devices are required for the handling equipment to limit impact loads or lifts.

5.3 OTHER OPERATING SYSTEMS

5.3.1 Operating System

The SSSC is the only operating system pertinent to this section.

5.3.1.1 Functional Description

A functional description of the SSSC is provided in Section 1.3.

5.3.1.2 Major Components

The SSSCs are the only safety-related components at the Surry ISFSI.

5.3.1.3 Design Description

The Surry ISFSI uses sealed and shielded casks to hold the PWR spent fuel assemblies. The cask designs are described in Appendix A.

The ALARA aspects of the Surry ISFSI operation are discussed in Chapter 7.

5.3.1.4 **Safety Criteria and Ensurance**

The Surry ISFSI, constructed, operated, and maintained as described in this SAR, is a safe and secure method for interim storage of spent fuel.

Design criteria which the SSSCs must meet are specified in Chapter 3. Compliance with these criteria ensure that operation of the Surry ISFSI will be in accordance with all the applicable safety requirements in 10 CFR Part 72. As shown in Chapters 7 and 8, its operation and response to credible events and postulated hypothetical accidents will not result in unacceptable risks to the health and safety of the public.

5.3.1.5 **Operating Limits**

Proposed operating limits for the Surry ISFSI are described in Chapter 10. Compliance with these limits will ensure that the design criteria specified in Chapter 3 and the safety assessments in this SAR are met.

5.3.2 Component/Equipment Spares

The cask monitoring instrumentation will be inspected and tested periodically in accordance with the commitments in Chapter 10 to ensure their proper operation. Replacement instrumentation will be available in accordance with historical requirements of the type of components used.

5.4 OPERATION SUPPORT SYSTEMS

There are no chemical systems used to monitor or control any of the ISFSI functions.

5.4.1 Instrumentation and Control Systems

There are no instrumentation and control systems necessary for the safe operation of the Surry ISFSI. Nonsafety instrumentation and alarms are described in Sections 4.3.7 and 5.1.3.4.

5.4.2 System and Component Spares

As indicated in previous sections, there is no safety-related instrumentation at the Surry ISFSI.

Failure of any of the monitoring equipment provided does not have any effects on cask integrity or the safe storage of the fuel.

5.5 CONTROL ROOM AND/OR CONTROL AREAS

Local panels at the ISFSI site provide annunciator alarm which would indicate the specific parameter and cask in question. Provisions have been made to allow for two alarms per cask.

The Surry ISFSI does not require continuous surveillance or operator actions, even during postulated accidents. Therefore, a control room is not considered necessary.

Coordination and supervision of emplacement operations take place in the area surrounding the SSSC, on the handling mechanism, and on any other equipment in use. Appropriate portable communications and radiation monitoring equipment will be used at those times.

5.6 ANALYTICAL SAMPLING

Neither radioactive releases during normal operation nor events resulting in radioactive releases are considered credible. Therefore, no means exist for the contamination of the outside surface of the casks.

6. Waste Containment and Management

Chapter 6
WASTE CONFINEMENT AND MANAGEMENT

6.1 WASTE SOURCES

No radioactive wastes are generated during the storage of spent fuel in a dry cask. However, since there may be some surface contamination deposited on the casks during fuel loading in the spent fuel pool, this contamination would have to be removed prior to placing the casks in storage. This contamination would consist of impurities typically found in the spent fuel pool water.

To remove this contamination, the cask will be washed down with water over the spent fuel pool before moving it to the decontamination building. Any residual contamination will be removed in the decontamination building in a manner similar to the design processes for smaller casks used for shipping fuel. The resulting contaminated water will be piped to the existing liquid waste disposal system and processed as described in Section 6.3.

The liquid waste generated is estimated to be less than 100 gallons per cask. Since the existing system is sized to process more than 8 million gallons of letdown annually from the primary system, it is seen that this is a negligible increase in the volume of waste processed by the Surry Power Station.

The liquid wastes from the Surry Power Station are either processed and discharged or dewatered and shipped for offsite disposal in high integrity containers. The contribution to these wastes from the cask decontamination process is expected to be negligible. The solid waste such as scrubbing towels from decontamination is estimated to fill less than two 55-gallon drums per cask.

6.2 OFFGAS TREATMENT AND VENTILATION

As discussed in Section 4.3.1, ventilation and offgas treatment systems are not required for the Surry ISFSI, and none are provided. Hence, there is no radioactive waste from items such as filters or scrubbers which would need to be treated.

However, as described in Sections 9.13 and 9.14 of the Surry Power Station FSAR, ventilation systems are provided for the fuel handling and decontamination buildings where cask loading and decontamination processes take place. Ventilation air from these buildings may be exhausted through filter banks consisting of roughing, particulate, and charcoal filters in series. Since the loading and decontamination of the SSSCs entail operations similar to the operations for which these ventilation systems were designed, these operations produce no new types of radwaste. Hence, these filters, when replaced, will continue to be handled using the procedures discussed in Section 11.2.4.1.2 of the Surry Power Station FSAR.

6.3 LIQUID WASTE TREATMENT AND RETENTION

As stated in Section 6.1, no liquid waste is generated while the casks are in storage; however, some liquid waste is generated during the cask decontamination process. This waste consists of water contaminated with fission and activation products typically found in the existing spent fuel pool. It is of the same composition and quality as the waste for which the existing Surry Power Station liquid waste disposal system is designed, and is generated in a manner similar to that planned for when the existing system was designed. Hence, this waste will be treated by the existing system as described in Section 11.2.3 of the Surry Power Station FSAR.

6.3.1 Design Objectives

As stated in Section 11.2.2 of the Surry Power Station FSAR, the waste disposal system is designed to satisfy the discharge requirements of 10 CFR Part 20 and 10 CFR Part 100 so as not to endanger the health of station operating personnel.

To ensure that processes associated with waste disposal meet the above design objectives, sampling, analysis, and monitoring of the liquid waste disposal system is done. Shielding is provided to reduce radiation levels, and area radiation monitoring equipment, health physics facilities, environmental programs, and administrative controls are provided for surveillance and control of radiation and exposure levels.

6.3.2 Equipment and System Description

The equipment and systems used to handle and process the contaminated water from the cask decontamination process are part of the decontamination building and the liquid waste disposal system and are summarized in the following sections.

6.3.2.1 Decontamination Building

The decontamination building abuts the east end of the fuel building's north wall. The 125-ton cask handling trolley transfers the cask from the fuel building to the decontamination building. Once inside the decontamination building, the cask is lowered onto the pad where decontamination takes place. Decontamination is performed using the same equipment and processes already provided in the decontamination building for the similar, but smaller, shipping casks for which the building and equipment were designed, as shown in Section 9.14.2 of the Surry Power Station FSAR. Typically, this would involve washing down the casks with water and scrubbing as necessary. A contaminated solution holdup tank is provided to receive spillage from equipment, runoff from cleaning operations, and for the disposal of cleaning solutions. This tank has a pump for transferring liquid to the liquid waste disposal system.

6.3.2.2 **Liquid Waste Disposal System**

The liquid waste disposal system is used to process the contaminated water generated as a result of the cask decontamination process. As described in Section 11.2.3 of the Surry Power Station FSAR, the liquid waste disposal system may use the following processes:

1. Filtration of the waste to remove particulate matter.
2. Demineralization, to remove dissolved material.
3. Dilution, to reduce the concentration of the radioactive constituents of the waste.
4. Decay, to reduce the activity levels of the isotopes.

Since the existing system was designed to handle the wastes due to contamination of similar, although smaller, spent fuel casks, no additional processes are necessary to handle the SSSCs.

6.3.3 Operating Procedures

The cask decontamination process involves the following procedures:

1. Lower cask into position in the decontamination building.
2. Decontaminate as explained in Section 6.3.2.1.
3. Take a swipe sample of the exterior cask surface to check for contamination.
4. Count swipe sample.
5. If swipe sample shows contamination above level specified in applicable procedures, decontaminate.
6. Take a second swipe sample and count.
7. Repeat above process until specified contamination limits are met.

According to current health physics procedures for the Surry Power Station, no further decontamination is needed when the count rate of the swipe sample is less than or equal to 1,000 dis/min/100 cm^2. Requiring the casks to be decontaminated to this initial level provides assurance that, over the lifetime of the cask in storage, the contamination levels of 49 CFR 173.443 will not be exceeded.

As described in Section 9.14.3.1 of the Surry Power Station FSAR, in the event of an off-normal condition, such as leakage from piping or equipment, all areas of the decontamination building are provided with sumps to which fluids will drain. The sumps discharge to the liquid waste disposal system.

Airborne particulate matter is retained within the building because of the slightly subatmospheric pressure and is discharged in a controlled manner through the monitored ventilation vent.

6.3.4 Characteristics, Concentrations, and Volumes of Solidified Wastes

The design of the Surry Power Station assumed that there would be frequent processing and offsite shipment of spent fuel using casks similar to, although smaller than, the SSSCs. Since the surface area of an SSSC is smaller than the combined surface area of the number of smaller casks needed to transport the same amount of fuel, it is expected that the wastes generated during the cask SSSC decontamination process would be significantly smaller than the amount for which the station was designed and licensed and would not alter the physical, chemical, and thermal characteristics of the waste now being processed.

6.3.5 Packaging

The packaging of solid wastes generated by decontamination of the SSSCs will be the same as currently done at the Surry Power Station. As described in Section 11.2.4 of the Surry Power Station FSAR, the solid waste disposal system provides holdup, packaging, and storage facilities for the eventual offsite shipment and ultimate disposal of radioactive waste material.

6.3.6 Storage Facilities

Since the SSSCs wastes are processed in the same manner as other Surry Power Station wastes, no special storage facilities are required. Current wastes generated are stored in the yard storage area until they are transferred to a licensed disposal contractor or to a common carrier for delivery to a licensed disposal contractor. These are all done in accordance with existing Surry Power Station radwaste procedures.

6.4 SOLID WASTES

As discussed in Section 6.1, no solid wastes are generated during the storage of spent fuel in a dry cask. This is because the SSSCs perform a totally passive storage function, with all contamination being retained within the SSSCs. Maintenance activities on the cask would not generate solid waste because the casks are decontaminated prior to placement in the ISFSI, hence, eliminating the only source of contamination; no maintenance is required for the interior of the casks.

However, some solid waste would be generated during the cask loading and decontamination processes described in Section 6.3.

Solid wastes such as spent resins are generated as a result of the processing of contaminated water generated during cask decontamination. However, the cask decontamination process does not add a significant amount of contaminated water to the liquid waste disposal system and, therefore, will not create much solid waste.

The processing of solid waste generated from operation of the Surry Power Station Units 1 and 2 is described in Section 11.2.4 of the Surry Power Station FSAR. The small increment in solid wastes resulting from the cask decontamination process will be handled in the same manner.

In addition to the solid waste from the liquid radwaste processing, there would be minor amounts of waste generated from the disposal of items such as the scrubbing towels used during the decontamination process, if done. Since the plant was built assuming there would be frequent offsite shipment of similar but smaller casks, this waste is well within the design and licensing basis of the existing system.

6.4.1 Design Objectives

As stated in Section 11.2.2 of the Surry Power Station FSAR, the waste disposal system is designed to satisfy the discharge requirements of 10 CFR Part 20 and 10 CFR Part 100, and so as not to endanger the health of Surry Power Station operating personnel.

6.4.2 Equipment and System Description

The Surry Power Station solid waste disposal system is described in Section 11.2.4 of the Surry Power Station FSAR.

6.4.3 Characteristics, Concentrations, and Volumes of Solid Wastes

See Section 6.3.4.

6.4.4 Packaging

See Section 6.3.5.

6.4.5 Storage Facilities

See Section 6.3.6.

6.5 RADIOLOGICAL IMPACT OF NORMAL OPERATIONS—SUMMARY

During normal operation, the ISFSI does not produce any radioactive wastes. The casks are sealed and will not release any of their radioactive contents.

As described in Sections 6.1, 6.2, 6.3, and 6.4, radioactive wastes are only generated during cask decontamination. These wastes comprise a small fraction of the total amount of radioactive wastes generated at the Surry Power Station Units 1 and 2 and are part of the original design and licensing basis of the plant. As such, their contribution to the total dose received by the nearest area resident, while minimal, has already been accounted for in the Surry Power Station Operating Licenses.

For a description of the Surry Power Station waste disposal design bases, see Chapter 11 of the Surry Power Station FSAR.

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Chapter 7
RADIATION PROTECTION

7.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS REASONABLY ACHIEVABLE (ALARA)

7.1.1 Policy Considerations and Organization

A radiological protection program will be implemented at the Surry ISFSI in accordance with the requirements of 10 CFR 72.126. The program will be based on policies in existence at the Surry Power Station, which are described below.

The management policies, organizational structure, and program criteria for maintaining exposures ALARA at the Surry ISFSI are the same as for the Surry Power Station, and are collectively referred to as the Virginia Power ALARA Program. The Virginia Power ALARA Program is an important part of the Surry Power Station radiation protection program. The basic principles of the Virginia Power ALARA Program are described in Virginia Power administrative procedures and are implemented by health physics technical procedures.

The Surry Power Station ALARA program complies with 10 CFR 20.1101, Radiation Protection Programs, and is consistent with the guidance of Regulatory Guide 8.8 (June 1978), *Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Reasonably Achievable.* The station ALARA program includes the following aspects:

- Specific individuals are assigned responsibility for, and authority to implement the station ALARA program consistent with Virginia Power policy. These individuals include the station ALARA coordinator and department ALARA coordinators.
- A Station ALARA Committee has been established with the responsibility for overall coordination of the station ALARA program and for advising the station management on matters relating to ALARA. A member of station management chairs the Committee.
- Pre-job measures are required to implement the station ALARA philosophy. These include ALARA evaluations of proposed work, pre-job meetings, and tiered levels of review based on projected expended person-rem.
- Monitoring and control of ongoing work is accomplished by the establishment of an exposure tracking system, ALARA hold points, and ALARA Radiation Work Permit (RWP) re-evaluation meetings.
- Completed work is evaluated via post-job reviews, maintenance of job history files, and periodic process reviews of selected work evolutions.
- A temporary shielding program has been established.

- An ALARA suggestion system is maintained to solicit, evaluate, and reward employee ideas that save person-rem.
- Engineering design change packages receive an ALARA review prior to implementation.
- A system has been established to actively involve, guide, and monitor the performance of the station and individual departments toward meeting ALARA objectives.
- The location of the ISFSI within the Surry Power Station site allows the health physics facilities, equipment, and personnel to be readily available at all times to ensure that ALARA considerations are met. The ISFSI is located a sufficient distance from buildings and occupied spaces to minimize total personnel exposure.

Regulatory Guide 8.10 (May 1977), *Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Reasonably Achievable,* Regulatory Position 1, concerning management commitment to minimizing exposures, is addressed by Virginia Power administrative procedures. Regulatory Position 2, concerning radiation protection staff vigilance in ALARA matters, is addressed and implemented by Virginia Power administrative procedures and health physics technical procedures.

The health physics organization is described in an administrative procedure. The organization to maintain exposures ALARA is also described in Virginia Power administrative procedures. The Station ALARA Committee is a major part of that organization. An administrative procedure also lists Surry Power Station health physics administrative procedures by functional grouping. These procedures are also applicable to the Surry ISFSI.

The guidance of Regulatory Position 2 of Regulatory Guide 8.8 (June 1978) is followed as described in Section 7.1.2. The guidance of Regulatory Guide 8.10 (May 1977) is followed as described in this section.

Virginia Power personnel qualifications and experience are considered more than sufficient for operation of the Surry ISFSI since these personnel have gained considerable experience at the Surry Power Station. An administrative procedure provides the functional responsibilities and reporting relationships of the members of the health physics organization and the personnel qualification requirements for positions in the station health physics organization.

Health physics equipment, instrumentation, and facilities for the Surry ISFSI will be those of the Surry Power Station. Radiation surveys with portable instruments will be performed during surveillance of the SSSCs and other activities at the ISFSI. Portable instruments required for measuring dose rates and radiation characteristics are maintained in accordance with approved health physics procedures.

As indicated in Section 7.2.2, respiratory protection equipment will not be needed at the Surry ISFSI.

Radiation protection facilities, instrumentation, and equipment available at the Surry Power Station are similar to that described by Regulatory Position 4 of Regulatory Guide 8.8 (June 1978). These include count room equipment, portable instruments, personnel monitoring instruments, protective equipment, and their associated support facilities.

The guidance for testing, rejection criteria, and use in mixed radiation fields being followed for the dosimeters at the Surry Power Station will be used at the Surry ISFSI.

The bioassay program in use for personnel at the Surry Power Station will also apply to the Surry ISFSI.

The methods and procedures for conducting radiation surveys at the Surry ISFSI will comply with the approved health physics procedures in effect at the Surry Power Station.

This section describes the health physics and ALARA procedures and planning for the Surry Power Station which will be used at the ISFSI. The complete details are in the applicable Virginia Power station and departmental administrative procedures and health physics technical procedures.

The radiological respiratory protection program is outlined in Virginia Power administrative procedures and implemented by health physics technical procedures.

Access control will be accomplished by means of a perimeter fence with a locked gate surrounding the Surry ISFSI. Control of the keys will be in accordance with security and health physics policies and procedures.

The bases and methods for monitoring and controlling personnel, equipment and surface contamination control, and radiation protection training program content are described in Virginia Power administrative procedures and health physics technical procedures.

The guidance provided by Regulatory Guide 8.10 (May 1977) will be followed as described in this section and Section 7.1.3. The guidance of Regulatory Guide 8.15 (October 1976), *Acceptable Programs for Respiratory Protection,* will be followed as described in this section. There should be no need for bioassay of personnel after surveillance activities at the Surry ISFSI. There should also be no need for radiological respiratory protection equipment.

Personnel dosimetry used at the Surry ISFSI will be controlled by the external dosimetry program approved for the Surry Power Station. An ALARA feedback mechanism using dosimeter results for preplanning future tasks is included in Virginia Power administrative procedures and their implementing documents.

The criteria for performing routine and non-routine whole-body counting and bioassay are contained in Virginia Power administrative procedures and health physics technical procedures for the Surry Power Station. Methods and procedures for evaluating and controlling airborne radioactive material are also given in these procedures.

Respiratory protection program requirements, equipment use and maintenance guidance, and fit testing protocol are delineated in Virginia Power administrative procedures and health physics technical procedures. Radiological respiratory protection training is conducted in accordance with Nuclear Training Department procedures.

7.1.2 Design Considerations

The ISFSI has been located in an area adjacent to the existing Surry Low Level Waste Storage Facility (LLWSF). This location was chosen based on several considerations, including ALARA, as follows:

1. The ISFSI and LLWSF are centrally located within the Surry site boundary, thus minimizing offsite exposures.
2. The centralized location of the ISFSI and LLWSF is of sufficient distance from the Surry Power Station such that the increased dose to Surry Station personnel is not significant.
3. The LLWSF is a facility that has limited occupancy, and, as such, represents a low exposure potential for personnel. In addition, the dose rates to workers from sources within the LLWSF are much greater than those that will result from ISFSI operations.
4. A proven heavy load route has been built past the LLWSF and a perimeter fence has already been built. Both of these are also utilized by the ISFSI.

The layout of the ISFSI is designed to minimize exposures since the casks will be stored with sufficient separation between them to allow adequate personnel access between the casks for surveillance and handling operations.

The equipment design considerations are ALARA since the fuel will be stored dry, inside sealed, heavily-shielded casks. The heavy shielding will minimize personnel exposures. To avoid personnel exposure, the casks will not be opened nor fuel removed from the casks while at the ISFSI. Storage of the fuel in dry sealed casks eliminates the possibility of leakage of contaminated liquids, and gaseous releases are not considered credible. The exterior of the casks will be decontaminated before leaving the Surry Power Station decontamination building, thereby avoiding exposure to surface contamination. There will be no other radioactive equipment at the ISFSI so that there will be no exposures from surface contamination associated with maintenance of equipment. The required maintenance and surveillance of the casks will be minimal and therefore ALARA. This method of spent fuel storage is also considered ALARA because it minimizes direct radiation exposures and eliminates the potential for contamination incidents.

Guidance provided by Regulatory Position 2 of Regulatory Guide 8.8 which concerns design considerations is being followed as described in the following paragraphs:

1. Regulatory Position 2.1 on access control is met by use of a fence with a locked gate that surrounds the ISFSI and prevents unauthorized access.

2. Regulatory Position 2.2 on radiation shielding is met by the heavy shielding of the casks which minimizes personnel exposures.
3. Regulatory Position 2.3 on process instrumentation and controls is met since there are no radioactive systems at the ISFSI. No process controls are required for the cask; however, there will be minor exposure attributed to calibration of instrumentation.
4. Regulatory Position 2.4 on control of airborne contaminants is met because no gaseous releases are expected. No significant surface contamination is expected either because the exterior of the casks will be decontaminated before they leave the decontamination building.
5. Regulatory Position 2.5 on crud control is not applicable to the ISFSI because there are no radioactive systems at the ISFSI that could transport crud.
6. Regulatory Position 2.6 on decontamination is met because the exteriors of the casks are decontaminated before they are released from the decontamination building.
7. Regulatory Position 2.7 on radiation monitoring is met because the casks are sealed. There is no need for airborne radioactivity monitoring since no airborne radioactivity is anticipated. Area radiation monitors will not be required because the ISFSI will not normally be occupied; however, TLDs will be installed along the controlled access fence. Portable survey meters will normally be used. Personnel dosimetry will be used at all times.
8. Regulatory Position 2.8 on resin treatment systems is not applicable to the ISFSI because there will not be any radioactive systems containing resins.
9. Regulatory Position 2.9 concerning other miscellaneous ALARA items is not applicable because these items refer to radioactive systems not present at the Surry ISFSI.

7.1.3 Operational Considerations

The ALARA procedures for the ISFSI will be the same as those used in the health physics program for the Surry Power Station. Section 7.1.1 describes the policy and procedures that ensure that ALARA occupational exposures and contamination levels are achieved. Section 7.1.2 describes how the design considerations are ALARA.

Storage of spent fuel in SSSCs is expected to involve lower exposures than alternative methods or designs for onsite storage. For example, storage in a fuel pool would involve use of radioactive water cooling and cleanup systems and filtered HVAC that would result in higher operator exposures during pump, valve, and motor maintenance of these systems, and filter and resin replacement. This alternative would also lead to additional airborne and liquid releases that will not be present at the Surry ISFSI.

The order of cask placement in the ISFSI has been developed based on ALARA considerations. Figure 7.3-1 shows the slabs numbered in the order of their use. Slab 2 will not be used until slab 1 is filled and, likewise, slab 3 will not be used until slab 2 is filled. Casks will be placed on a slab in rows of two starting at the northern end and finishing at the southern end. In

this manner, personnel placing casks on the next available slab are closer to the older spent fuel and further from the younger spent fuel, thus minimizing the amount of radiation exposure from previously filled slabs.

The guidance provided by Regulatory Position 4 of Regulatory Guide 8.8 is being followed. That section of the Regulatory Guide concerns radiation protection facilities, instrumentation, and equipment. The counting room, portable instruments, personnel monitoring instruments, protective equipment, and support facilities for the ISFSI called for in Regulatory Position 4 will be provided by the health physics facilities and personnel at the Surry Power Station Units 1 and 2. The procedures and methods that ensure that occupational radiation exposures at the ISFSI are ALARA have been described in Sections 7.1.1, 7.1.2, and 7.1.3. The procedures and methods of operation to ensure ALARA exposures given in Regulatory Position 4 of Regulatory Guide 8.8 and in Regulatory Guide 8.10 will be followed as described in Sections 7.1.1, 7.1.2, and 7.1.3.

Operational requirements for surveillance are incorporated into the design considerations in Section 7.1.2 in that the casks are stored with adequate spacing to allow ease of surveillance. The operational requirements are incorporated into the radiation protection design features described in Section 7.3 since the casks are heavily shielded to minimize occupational exposure.

The criteria and conditions under which certain ALARA techniques are implemented to ensure ALARA exposures and contamination levels are described in Section 7.1.1. ALARA techniques will be implemented at all times.

As the number of potential man-rem per task increases, the ALARA techniques employed become more stringent as described in the Virginia Power ALARA Program.

The ISFSI does not contain any systems that process liquids or gases or contain, collect, store, or transport radioactive liquids or solids other than the stored fuel. Therefore, the ISFSI is ALARA since there are no such systems to be maintained, be repaired, or be a source of leaks.

7.2 RADIATION SOURCES

7.2.1 Characterization of Sources

Shielding of the spent fuel is provided by the casks. Physical characteristics of the fuel used at the Surry Power Station are summarized in Table 3.1-1. Typical fuel assembly sources are given in Tables 7.2-1 through 7.2-4. These tables were generated by Westinghouse using ORIGEN II. Descriptions of the fuel which the SSSCs are designed to store are provided in the SSSC topical reports or Appendix A. The exterior surfaces of the casks will be decontaminated prior to transfer to the ISFSI. The fuel will not be removed from the casks or the casks opened while at the ISFSI. The only source of radioactivity on the ISFSI pads will be the direct radiation from the fuel stored inside the SSSCs. Located within the ISFSI perimeter fence, but outside the security fences for the ISFSI pads, is a Low Level Waste Storage Facility (LLWSF). Section 7.3.2.2 provides a discussion of the contribution from the LLWSF on radiological doses.

7.2.2 Airborne Radioactive Material Sources

Respiratory protection is not needed at the ISFSI because of the lack of airborne radioactivity.

Table 7.2-1

AVERAGE NEUTRON SOURCE[a] FOR WESTINGHOUSE 15x15 FUEL

Nominal Burnup (MWd/MtU)	Time After Discharge 150 Days	Time After Discharge (years)									
		1	2	3	4	5	6	7	8	9	10
35,000	1.26+6	1.07+6	9.42+5	8.89+5	8.53+5	8.22+5	7.92+5	7.63+5	7.36+5	7.09+5	6.84+5
45,000	1.98+6	1.71+6	1.53+6	1.45+6	1.39+6	1.34+6	1.29+6	1.24+6	1.20+6	1.15+6	1.11+6

a. Neutrons/second/cm active fuel length

Table 7.2-2
AVERAGE PHOTON SOURCES[a] 150 DAYS AFTER DISCHARGE
FOR WESTINGHOUSE 15x15 FUEL

Feed Enrichment (wt.% U235)	3.09	4.13
Average Burnup (MWd/MtU)	35,000	45,000

E Mean (Mev)	Twelve Group Energy Release Rates (Mev/sec)	
3.00-1	1.48+12	1.74+12
6.30-1	5.69+13	7.20+13
1.10+0	1.61+12	2.06+12
1.55+0	9.62+11	1.28+12
1.99+0	6.18+11	7.21+11
2.38+0	2.31+9	2.72+9
2.75+0	4.01+7	4.72+7
3.25+0	1.58+7	1.86+7
3.70+0	0	0
4.22+0	0	0
4.70+0	0	0
5.25+0	0	0
Total	6.15+13	7.78+13

a. Basis is 1 cm of active fuel length

Table 7.2-3

PHOTON SPECTRUM AS A FUNCTION OF TIME FOR FISSION PRODUCTS

FEED ENRICHMENT FUEL - 3.09 wt.% U-235 - W 15X15 ASSY 35,000 MWd/MtU 4 CYCLES

Average Power = 0. Mw Average Burnup = 4.715E+01 MWd Average Flux = 0. N/cm^2-sec

Twelve Group Energy Release Rates (Mev/sec) Basis = One cm of Active Fuel Length

E Mean		Time After Discharge (Yr)									
(Mev)	Initial	1.00E+00	2.00E+00	3.000E+00	4.000E+00	5.000E+00	6.000E+00	7.000E+00	8.000E+00	9.000E+00	1.000E+01
3.00E-01	3.49E+14	7.60E+11	3.66E+11	2.05E+11	1.36E+11	1.05E+11	9.03E+10	8.24E+10	7.74E+10	7.38E+10	7.09E+10
6.30E-01	1.42E+15	2.35E+13	1.47E+13	1.10E+13	8.53E+12	6.84E+12	5.65E+12	4.80E+12	4.18E+12	3.72E+12	3.38E+12
1.10E+00	8.76E+14	1.15E+12	8.37E+11	6.50E+11	5.30E+11	4.46E+11	3.87E+11	3.41E-11	3.05E+11	2.74E+11	2.49E+11
1.55E+00	8.04E+14	6.52E+11	3.98E+11	2.58E+11	1.73E+11	1.19E+11	8.38E+10	5.96E+10	4.28E+10	3.10E+10	2.26E+10
1.99E+00	2.05E+14	3.61E+11	1.48E+11	6.09E+10	2.50E+10	1.03E+10	4.30E+09	1.82E+09	7.96E+08	3.76E+08	2.02E+08
2.38E+00	2.00E+14	1.98E+04	5.00E-05	1.26E-13	3.19E-22	8.05E-31	2.03E-39	5.13E-48	1.30E-56	3.26E-65	8.26E-74
2.75E+00	1.00E+14	3.43E+02	8.67E-07	2.19E-15	5.52E-24	1.39E-32	3.52E-41	8.89E-50	2.24E-58	5.67E-67	1.43E-75
3.25E+00	2.43E+14	1.35E+02	3.41E-07	8.62E-16	2.18E-24	5.49E-33	1.39E-41	3.50E-50	8.84E-59	2.23E-67	5.67E-76
3.70E+00	1.72E+13	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
4.22E+00	4.33E+13	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
4.70E+00	2.27E+13	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
5.25E+00	5.97E+12	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
Total	4.28E+15	2.64E+13	1.64E+13	1.21E+13	9.40E+12	7.52E+12	6.22E+12	5.29E+12	4.61E+12	4.10E+12	3.72E+12
Gamma Watts	6.86E+02	4.23+00	2.63E+00	1.95E+00	1.51E+00	1.21E+00	9.97E-01	8.47E-01	7.38E-01	6.58E-01	5.97E-01

Table 7.2-4

PHOTON SPECTRUM AS A FUNCTION OF TIME FOR FISSION PRODUCTS

FEED ENRICHMENT FUEL - 4.13 wt.% U-235 - W 15X15 ASSY 45,000 MWd/MtU 3 CYCLES

Average Power = 0. Mw Average Burnup = 5.943E+01 MWd Average Flux = 0. N/cm^2-sec

Twelve Group Energy Release Rates (Mev/sec) Basis = One cm of Active Fuel Length

E Mean (Mev)	Initial	Time After Discharge (Yr) 1.00E+00	2.00E+00	3.000E+00	4.000E+00	5.000E+00	6.000E+00	7.000E+00	8.000E+00	9.000E+00	1.00E+01
3.00E-01	4.10E+14	8.97E+11	4.36E+11	2.48E+11	1.68E+11	1.32E+11	1.14E+11	1.05E+11	9.96E+10	9.42E+10	9.06E+10
6.30E-01	1.68E+15	3.14E+13	1.99E+13	1.49E+13	1.15E+13	9.18E+12	7.52E+12	6.33E+12	5.47E+12	4.83E+12	4.35E+12
1.10E+00	1.04E+15	1.49E+12	1.10E+12	8.56E+11	7.01E+11	5.92E+11	5.13E+11	4.51E+11	4.02E+11	3.62E+11	3.27E+11
1.55E+00	9.50E+14	8.82E+11	5.47E+11	3.58E+11	2.43E+11	1.68E+11	1.19E+11	8.45E+10	6.07E+10	4.39E+10	3.20E+10
1.99E+00	2.43E+14	4.20E+11	1.73E+11	7.09E+10	2.92E+10	1.21E+10	5.02E+09	2.13E+09	9.38E+08	4.48E+08	2.46E+08
2.38E+00	2.37E+14	2.33E+04	5.89E-05	1.49E-13	3.75E-22	9.47E-31	2.39E-39	6.04E-48	1.52E-56	3.85E-65	9.72E-74
2.75E+00	1.19E+14	4.04E+02	1.02E-06	2.58E-15	6.50E-24	1.64E-32	4.15E-41	1.05E-49	2.64E-58	6.67E-67	1.68E-75
3.25E+00	2.83E+14	1.59E+02	4.02E-07	1.01E-15	2.56E-24	6.47E-33	1.63E-41	4.12E-50	1.04E-58	2.63E-67	6.64E-76
3.70E+00	2.07E+13	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
4.22E+00	5.20E+13	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
4.70E+00	2.72E+13	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
5.25E+00	7.18E+12	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
Total	5.06E+15	3.15E+13	2.22E+13	1.64E+13	1.27E+13	1.01E+13	8.27E-12	6.97E+12	6.03E+12	5.33E+12	4.80E+12
Gamma Watts	8.11E+02	5.62E+00	3.56E+00	2.64E+00	2.03E+00	1.62E+00	1.33E+00	1.12E+00	9.66E-01	8.54E-01	7.70E-01

7.3 RADIATION PROTECTION DESIGN FEATURES

7.3.1 Installation Design Features

A description of the Surry ISFSI, including layout and characteristics is provided in Section 4.1.

The ISFSI has a number of design features which ensure that exposures are ALARA.

1. [Deleted]
2. The casks are loaded, sealed, and decontaminated prior to transfer to the ISFSI.
3. The fuel is not unloaded nor are the casks opened at the ISFSI.
4. The fuel is stored dry inside the casks, so that no radioactive liquid is available for leakage.
5. The casks are sealed airtight.
6. The casks are heavily shielded to minimize external dose rates.

Also, the ISFSI will not normally be occupied. Therefore, no personnel areas, equipment decontamination areas, contamination control areas, or health physics facilities need be located at the ISFSI. These types of facilities are available at the Surry Power Station Units 1 and 2.

7.3.2 Shielding

Details on the SSSC shielding designs are provided in the SSSC topical reports. No shielding other than that afforded by the SSSCs themselves is required.

Except during cask placement and scheduled surveillance, the ISFSI will not be normally occupied. A fence with a locked gate surrounds the ISFSI to control access. If the dose rate beyond the ISFSI fenced-in area exceeds 5 mrem per hour at any time during ISFSI operation, additional control measures such as extending the fence as illustrated in Figure 7.3-1 will be enacted.

7.3.2.1 Cask Surface Dose Rates

The gamma dose rate on the cask surface with its photon energy spectrum, and the neutron dose rate on the cask surface with its neutron energy spectrum are dependent on the cask design. The cask surface gamma and neutron dose rates are also dependent on the burnup and initial enrichment of the fuel stored in the casks. Therefore, cask-specific analyses have been performed for representative Surry Power Station fuel. See Appendix A. The assumptions used in the cask-specific analyses for cask surface dose rates and energy spectra are provided in the SSSC topical reports or Appendix A.

The TN-32 cask (Appendix A.5) loaded with fuel with an initial enrichment of 3.5 weight percent U-235, burnup of 45,000 MWD/MTU and cooling time of 7 years has been chosen as the base case for analysis purposes. Using an enrichment lower than the 4.05 weight percent U-235

approved for the TN-32 yields a bounding isotope inventory, and is in accordance with NUREG-1536 and NRC Interim Staff Guidance.

Source terms for the fuel were calculated using the SAS2H/ORIGEN-S module of SCALE4.3 as described in Section 5.1 of Reference 1. These source terms are then passed through a SAS2H cask shield model for a 1-dimensional dose assessment. Section 5.2 (Reference 1) describes the source specification and Section 5.3 (Reference 1) describes the shielding analyses performed for the TN-32 cask.

In addition to the spent fuel, the TN-32 is capable of storing BPRAs and TPAs. BPRAs and TPAs with combinations of cumulative exposures and cooling times are permissible for storage in the TN-32 cask. The source evaluation of the BPRAs and TPAs is described in Section 5.2 (Reference 1).

Virginia Power conducted an independent analysis of the TN-32 surface dose rate. This analysis was used to form the basis for the cask surface dose rate limit in the ISFSI Technical Specifications. The surface dose rates calculated for the TN-32 base case cask were 224 mrem/hr (neutron and gamma) for the side surface and 76 mrem/hr (neutron and gamma) for the top surface.

Appendix A provides the cask-specific analyses for surface dose rates.

7.3.2.2 **Dose Rate Versus Distance**

Analyses have been completed to determine dose rates at the ISFSI perimeter fence, the site boundary and the nearest permanent resident. These analyses were performed using the MCNP Monte Carlo transport code (Reference 2) and the following conservative inputs.

1. Isotope inventories were based on 32 fuel assemblies with enrichment of 3.5 weight percent U-235, burnup of 45,000 MWD/MTU and seven years decay.

2. The three storage pads were filled with 84 base case TN-32 casks, each pad having 28 casks. This input is conservative, since the first storage pad is filled with CASTOR V/21, CASTOR X/33, MC-10 and NAC-I28 storage casks, all of which have maximum surface dose rates that are lower than the base case TN-32. In addition, using 84 TN-32 cask results in an amount of fuel stored on the pads which exceeds the current licensed limit of 811.44 TeU, providing additional conservatism to the analysis.

3. The analyses assume no decrease in the gamma and neutron emission rates as a result of decay beyond the initial seven-year requirement. That is, all 84 casks were assumed to be placed simultaneously at the ISFSI.

4. The effects of irradiated insert components were included in the MCNP analyses. Each cask was assumed to contain 32 irradiated insert components with the source spectrum and source strength identified in Reference 1.

Figure 7.3-1 shows the layout of the ISFSI. The MCNP analysis of the dose rate at the ISFSI perimeter fence using base case TN-32 casks resulted in peak dose rates that range from 2.9 to 12.2 mrem/hr when all three pads were full. Dose rate measurements at the ISFSI perimeter fence will be used to ensure that the requirements of 10 CFR 20 are met.

The MCNP analysis for the nearest site boundary indicated that the maximum dose rate at this location was less than 100 mrem/yr, which meets the requirements of 10 CFR 20.1301.

The licensing basis for the annual dose to the nearest permanent resident was based on 84 GNSI CASTOR V/21 casks, adjusted for decay, and air and distance attenuation of neutron and gamma rays. The annual dose to the nearest permanent resident (1.53 miles away) for this case was 6.0E-05 mrem, based on Section 2.3 of the NRC's Safety Evaluation Report for the Surry Dry Cask Independent Spent Fuel Storage Installation and Section 6.2 of the NRC's Environmental Assessment Related to the Construction and Operation of the Surry Dry Cask Independent Spent Fuel Storage Installation. The MCNP analysis using 84 base case TN-32 casks resulted in an annual dose to the nearest permanent resident from normal ISFSI operation that is bounded by the ISFSI licensing basis.

7.3.3 Ventilation

As indicated in Section 3.3.2.2, the ISFSI does not require a ventilation system. The ALARA provisions of 10 CFR 20 and of appropriate regulatory guides will be satisfied since no exposure will be incurred in ventilation system maintenance or filter changing.

7.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

As indicated in Section 3.3.5, area radiation and airborne radioactivity monitors are not needed at the Surry ISFSI; however, TLDs will be used to record dose rates along the ISFSI perimeter fence.

7.3.5 References

1. *TN-32 Safety Analysis Report,* Revision 0, Transnuclear Inc., January 2000.
2. MCNP Version P01.3, *Monte Carlo N-Particle Transport Code System,* CCC-660, Los Alamos National Laboratory.
3. [Deleted]
4. [Deleted]
5. [Deleted]
6. [Deleted]
7. [Deleted]
8. [Deleted]

9. [Deleted]
10. [Deleted]

Table 7.3-1
[DELETED]

Table 7.3-2
[DELETED]

Table 7.3-3
[DELETED]

Table 7.3-4
[DELETED]

Table 7.3-5
[DELETED]

Figure 7.3-1
ISFSI LAYOUT

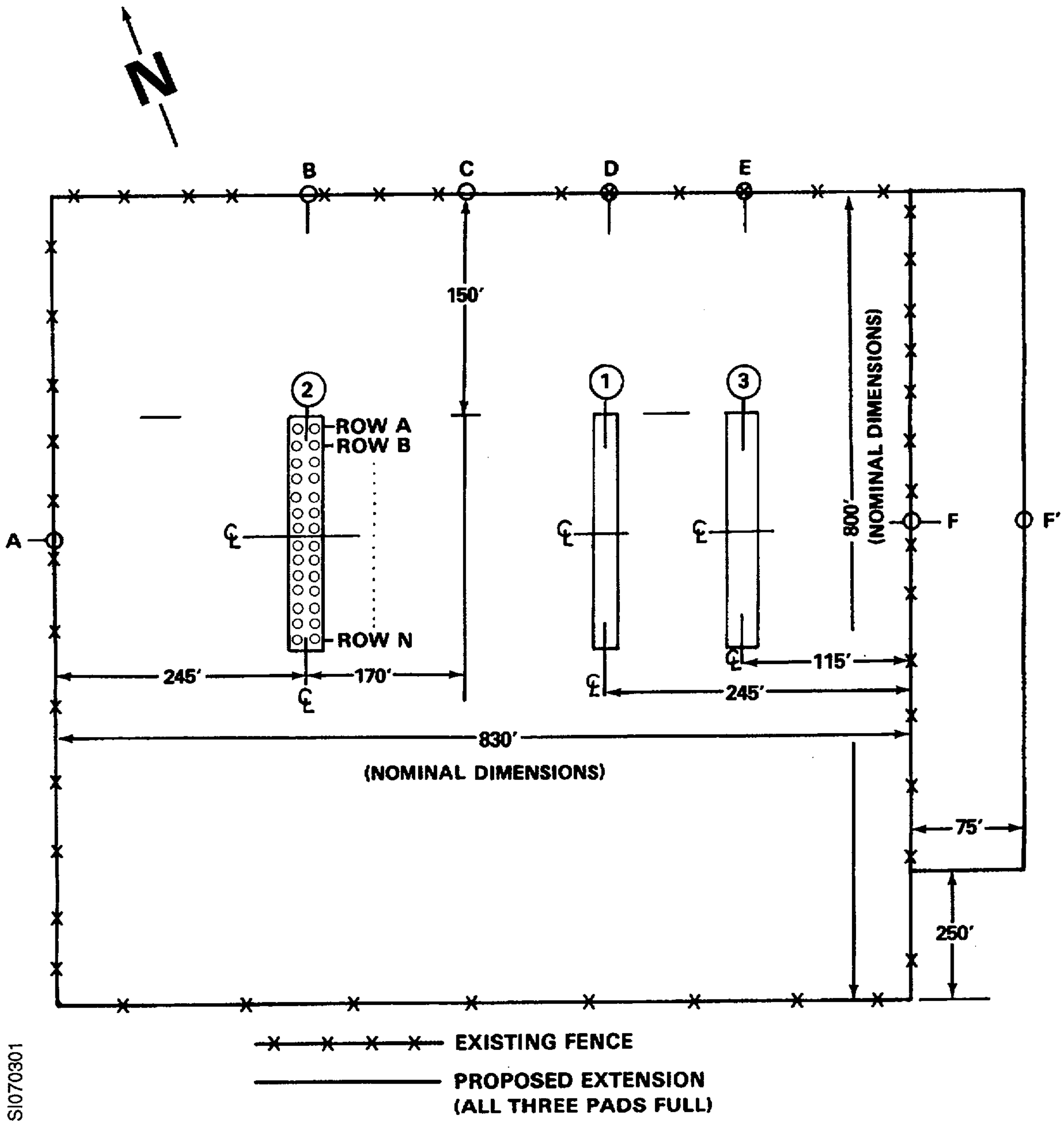


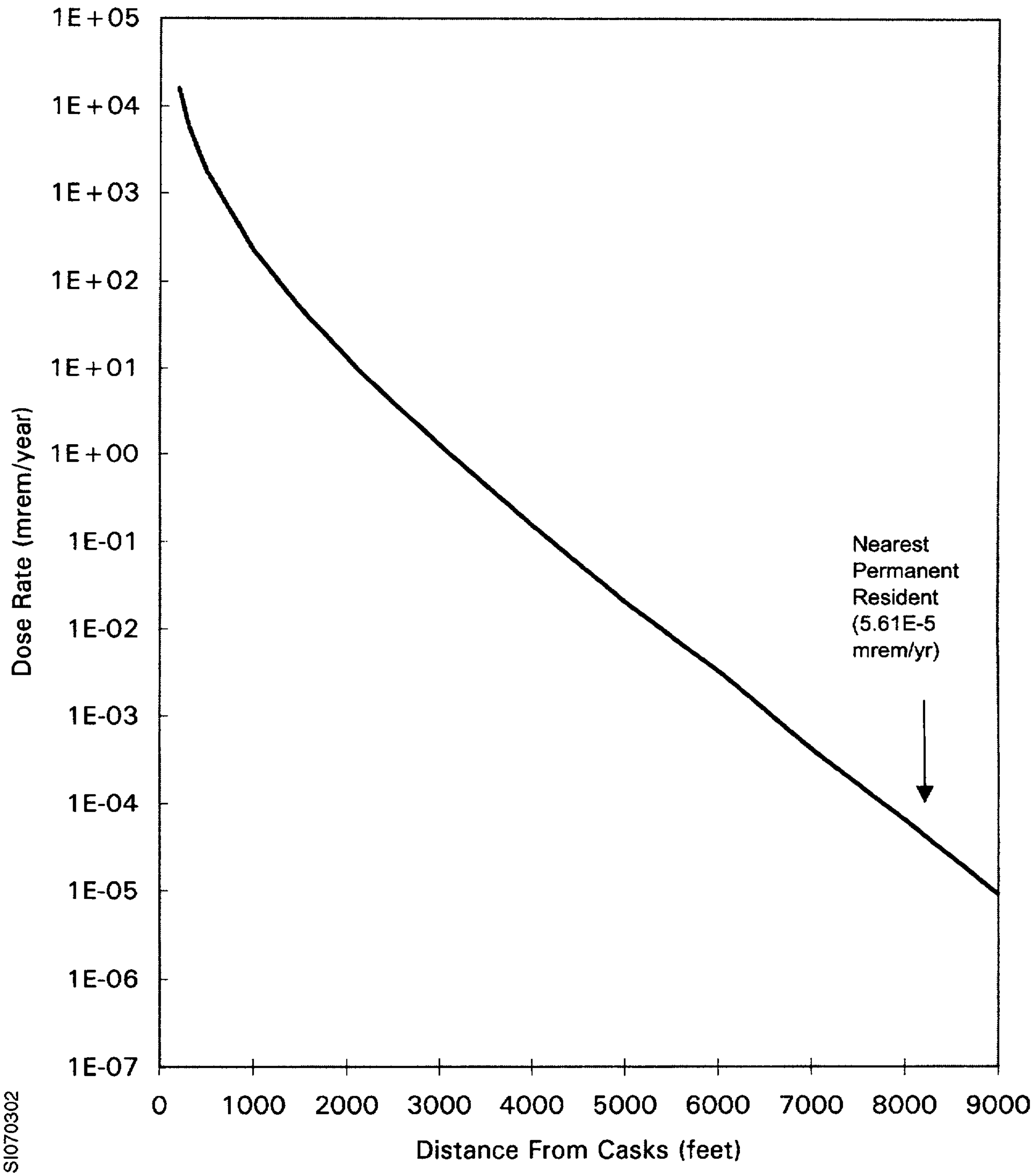
Figure 7.3-2
[DELETED]

Figure 7.3-3
[DELETED]

Figure 7.3-4
[DELETED]

Figure 7.3-5
[DELETED]

Figure 7.3-6
DOSE RATE FOR 84 BASE CASE CASKS VERSUS DISTANCE



7.4 ESTIMATED ONSITE COLLECTIVE DOSE ASSESSMENT

7.4.1 Exposure to ISFSI Personnel

Table 7.4-1 shows the estimated occupational exposures to ISFSI personnel during the loading, transport, and emplacement of the SSSCs. Base case TN-32 surface dose rates were utilized for all cases except cask transfer, when individuals will typically be at least 10 feet away from the cask.

Table 7.4-2 shows the estimated annual man-rem for surveillance and maintenance activities. Base case TN-32 surface dose rates were utilized assuming all storage pads were filled with casks. To estimate the dose rates for operability tests and calibration, the worker was assumed to be located at the control panel at the perimeter fence entrance. Visual surveillance was based on a walkdown of each of the three pads at a distance no closer than 2 meters to the casks. During surface defect repairs, the worker was assumed to be positioned next to a cask. The five surrounding casks (all within 16 feet of the worker) are the predominant dose contributors during repair work.

Both Table 7.4-1 and 7.4-2 provide for each task the estimated time required for the task, number of personnel required, the dose rates, and man-rem.

The total annual occupational dose for ISFSI operations is given in Table 7.4-4.

7.4.2 Exposure to Power Station Personnel

To evaluate the additional dose to station personnel from ISFSI operations, a conservative analysis has been performed using the assumptions given in Section 7.3.2.2. The occupational dose calculation considers all workers at the Surry Power Station to be in offices, nonshielded buildings, or in the plant yard. This population includes a normal work force of utility and contractor personnel as well as the increased staffing required during outages. As a bounding estimate, the total number of workers assumed was 600 spending a total of 1,248,000 man-hours per year in the Surry yard area and in offices.

The minimum distance between the Surry Units 1 and 2 perimeter fence and the nearest cask is approximately 2100 feet. The dose rate from the ISFSI to a yard location 2100 feet away is 1.00E-3 mrem/hr. The annual exposure for station workers due to the ISFSI is calculated to be 1.3 man-rem per year.

7.4.3 Exposure to LLWSF Personnel

The dose to workers at the LLWSF due solely to LLWSF operations is calculated to be in the range of 3.6 to 7.1 man-rem per year. This is based on a typical, historical, LLWSF occupancy time of 712 man-hours per year. Depending on exactly what operations are taking place (package

handling, movement, monitoring, etc.), these 712 man-hours are assumed to be spent in radiation areas corresponding to the LLWSF average values of 5 to 10 mrem/hr.

The dose to workers at the LLWSF due to the ISFSI is calculated to be 1.3 man-rem per year. Credit was taken for air attenuation of neutrons and gammas; however, no credit was taken for the shielding effect of one cask behind another and the shielding provided by the LLWSF building to the personnel. This dose is calculated from 84 base case TN-32 casks.

Table 7.4-1
OCCUPATIONAL EXPOSURES FOR CASK LOADING, TRANSPORT, AND EMPLACEMENT [a]
(ONE TIME EXPOSURE)

Task	Time Required (hr)	No. of Persons	Dose Rate (rem/hr)	Man-Rem
Placement in pool	1	3	0.005	0.015
Loading process	3	2	0.005	0.030
Removal from pool	2	3	0.028	0.168
Processing of cask	6	2	0.056	0.672
Helium leak test	4	2	0.056	0.448
Decontamination and inspection	3	2	0.056	0.336
Transfer from preparation area	1	3	0.028	0.084
Preparation for transport	1	3	0.028	0.084
Transfer to ISFSI	1	3	0.028	0.084
Emplacement on pad	1	2	0.028	0.056
Installation of monitoring equipment	3	2	0.112	0.672
Total				2.649

a. Dose rates are from the base case TN-32 cask.

Table 7.4-2
SURRY ISFSI MAINTENANCE OPERATIONS ANNUAL EXPOSURES

Task	Time Required (hr)	No. of Persons	Dose Rate[a] (mrem/hr)	Man-Rem
Visual Surveillance of Casks[b]	1	1	224	0.224
Monitoring System Operability Tests[c]	1	2	20	0.040
Monitoring System Alarm Response and Repairs[d]	2	2	20	0.080
Cask Surface Defect Repairs[e]	3	1	336[f]	1.008
Total				1.352

a. Dose rates are from the base case TN-32 cask. Assumes ISFSI is full.
b. Based on four surveys per year, 15 minutes each.
c. Based on two tests per year, 30 minutes each.
d. Based on two responses per year, one hour each.
e. Based on repair of three casks per year, one hour each.
f. Based on base case dose rate (224 mrem/hr) plus 50%.

Table 7.4-3
[DELETED]

Table 7.4-4
ANNUAL DOSES FROM ISFSI OPERATIONS

	Man-Rem
LLWSF [a]	1.3
Surry Power Station [a]	1.3
ISFSI Operations -	
Cask Preparation and Placement [b]	7.9
Maintenance and Surveillance	1.4
Total	11.9

a. Assumes completed ISFSI (84 design basis casks).
b. Assumes 3 TN-32 casks per year.

7.5 HEALTH PHYSICS PROGRAM

The current health physics organization and the health physics equipment associated with operation of the Surry Power Station are considered sufficient for the operation of the ISFSI. The health physics technical procedures directing routine surveys include ISFSI activities.

7.6 ESTIMATED OFFSITE COLLECTIVE DOSE ASSESSMENT

Figure 1.2-1 illustrates the plant site boundary, which is also the boundary of the restricted area. This restricted area will remain the same after addition of the ISFSI. It is the controlled area as defined in 10 CFR 72.

There are 48 permanent residents located within the 2-mile radius. The nearest permanent resident is located at 1.53 miles from the site. Based on the dose rate versus distance curve (Figure 7.3-6) and the conservative assumption that all of the residents within 2 miles are located at the same distance from the ISFSI as the nearest resident at 1.53 miles, the collective annual dose from ISFSI operations would be 2.69E-6 man-rem per year. This dose assumes a total of 84 TN-32 casks and no adjustment for fuel source decay. Considering the conservatisms in the above calculation and the rapid attenuation of neutron and gamma dose rates with distance, the collective dose for the more distant population would be negligible.

7.6.1 Effluent and Environmental Monitoring Program

The environmental monitoring program to be followed at the ISFSI is the same in effect at the Surry Power Station, but will be augmented by additional TLDs along the ISFSI restricted area fence. Since no effluents are expected from the ISFSI, the operation of the ISFSI will have minimal impact on the monitoring program.

7.6.1.1 Gas Effluent Monitoring

The Surry ISFSI does not require gaseous effluent monitoring.

7.6.1.2 Liquid Effluent Monitoring

The Surry ISFSI does not require liquid effluent monitoring.

7.6.1.3 Solid Waste Monitoring

The Surry ISFSI does not require solid waste monitoring.

7.6.1.4 Environmental Monitoring

The environmental sampling program at the ISFSI will be the same as that in effect at the Surry Power Station Units 1 and 2. The specific details of the program are described in the Surry Offsite Dose Calculation Manual (ODCM).

In addition to the TLDs maintained in areas around the ISFSI as part of the environmental and radiation monitoring program for the Surry Power Station as described above, area radiation

monitoring will also be performed routinely by extra TLDs located on the ISFSI area fence. To provide continuous monitoring capability, at least 2 gamma-sensitive TLDs will be placed at the fence on each side of the ISFSI area. For cask surveillance, portable neutron and gamma survey meters will normally be used. A correlation between gamma measurements from portable survey meters and the TLDs will first be established in preparation for assessing the neutron dose at the fence. Neutron dose rates at the ISFSI area fences will be measured by the neutron survey meters. The integrated neutron dose at the ISFSI area fence can be estimated by using the ratio of the integrated gamma dose and gamma dose rate measured in the same location. By following this procedure, the neutron dose to the environment from the ISFSI can be determined.

No individual cask radiation monitoring is necessary.

7.6.2 Analysis of Multiple Contribution

For the purpose of determining offsite exposure from the ISFSI, the design basis total dose rate versus distance curve is shown in Figure 7.3-6. Using Figure 7.3-6, the annual dose to the nearest permanent resident (1.53 miles away) due to ISFSI operations would be 5.61E-5 mrem.

The annual dose to the nearest permanent resident from the LLWSF has been estimated to be 4.4E-2 mrem. Using the whole-body dose guidelines from 10 CFR 50 Appendix I, the maximum annual dose to the nearest permanent resident from the Surry Power Station would be 3 mrem due to liquid effluents and 5 mrem due to gaseous effluents for each unit. The maximum total annual dose to the nearest permanent resident would be:

5.61E-5 mrem (ISFSI) + 4.4E-2 mrem (LLWSF) + 16 mrem (normal operation Units 1 and 2) = 16 mrem

As shown in the above equation, the dose to the nearest permanent resident from the ISFSI and LLWSF operations, in combination with the maximum permissible dose from the Surry Power Station, will not exceed the 25 mrem per year limit specified in 10 CFR 72.104(a). The above calculation is conservative, since the actual Surry Power Station effluent doses are below the 10 CFR 50 Appendix I guidelines. This is shown in Appendix 11A to the Updated FSAR for Surry Power Station Units 1 and 2.

The general population dose is not expected to increase by a detectable amount, due to the addition of the ISFSI, and will be well within the limits specified by 10 CFR 72.67(a).

7.6.3 Estimated Dose Equivalents

No radioactive effluents are expected at the Surry ISFSI.

7.6.3.1 Identification of Sources

This section does not apply for reasons stated in Section 7.6.3.

7.6.3.2 **Analysis of Effects and Consequences**

This section does not apply for reasons stated in Section 7.6.3.

7.6.4 Liquid Release

This section does not apply for reasons stated in Section 7.6.3.

7.6.4.1 **Treated Process Effluent (from Waste Treatment Area)**

This section does not apply for reasons stated in Section 7.6.3.

7.6.4.2 **Sewage**

There will be no sewage systems at the ISFSI.

7.6.4.3 **Drinking Water**

There will be no drinking water at or in the vicinity of the ISFSI.

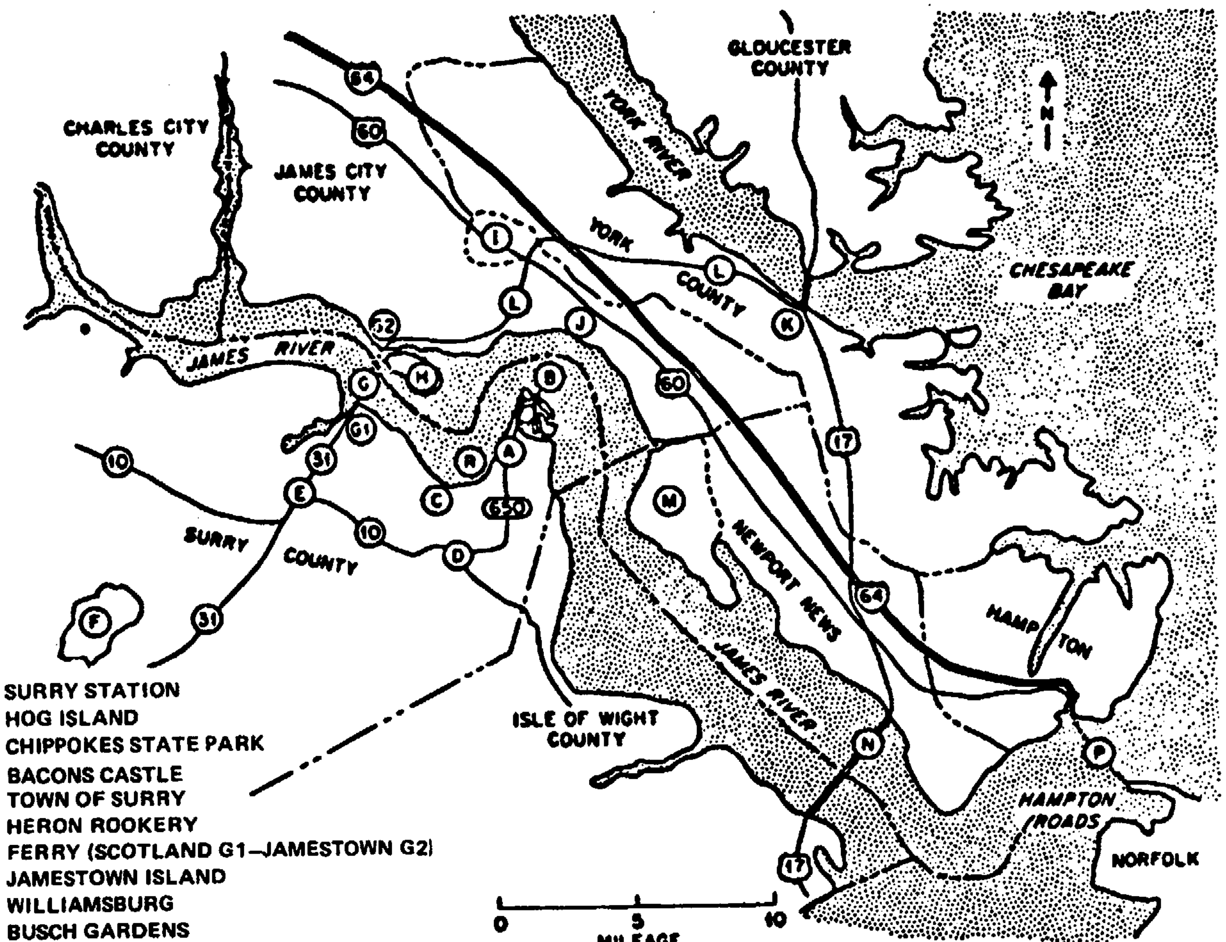
7.6.4.4 **Rain Runoff**

There are no sources of contamination at the Surry ISFSI. Therefore, rain runoff at the ISFSI will not be contaminated.

7.6.4.5 **Laundry Waste**

There will be no laundry at the ISFSI.

Figure 7.6-7
ENVIRONS OF SURRY ISFSI SITE



A SURRY STATION
B HOG ISLAND
C CHIPPOKES STATE PARK
D BACONS CASTLE
E TOWN OF SURRY
F HERON ROOKERY
G FERRY (SCOTLAND G1–JAMESTOWN G2)
H JAMESTOWN ISLAND
I WILLIAMSBURG
J BUSCH GARDENS
K YORKTOWN BATTLEFIELD
L COLONIAL PARKWAY
M FORT EUSTIS (DECOMMISSIONED)
N JAMES RIVER BRIDGE
P TUNNEL
R COBHAM BAY

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Chapter 8
ACCIDENT ANALYSES

An evaluation of the safety of the Surry ISFSI with respect to postulated accident events is presented in this chapter. The facility response is analyzed in terms of event causes and precursors, recognition and quantification, and consequence mitigation for the spectrum of postulated occurrences.

Four categories of design events have been considered. Design event categories are designated as:

I. Events that are expected to occur regularly or frequently in the course of normal operation

II. Events which can be expected to occur with moderate frequency as on the order of once per year

III. Events anticipated to occur infrequently or, at most, once during the lifetime of the installation

IV. Events which are not considered credible, but nevertheless are postulated in order to bound the consequences.

8.1 OFF-NORMAL OPERATIONS

The design and operation of the Surry ISFSI include features intended to minimize or preclude the compromise of safety functions due to off-normal conditions. These features are described in Chapters 4 and 5. Nevertheless, design events have been postulated and analyzed to demonstrate the inherent safety of the facility.

Design events in Category I (normal operations) have been previously discussed in Chapters 4, 5, and 6 and are not presented further here. A loss of electric power design event has been included as a Category II event and is discussed in the following section. The SSSC topical reports postulate additional off-normal events for the casks. The topical reports analyze the effects of these additional events and identify the corrective actions.

8.1.1 Loss of Electric Power

A total loss of ac power is postulated to occur in the feeder cabling which supplies power to the ISFSI. The failure could be either an open or a short to ground circuit, or any other mechanism capable of producing an interruption of power.

8.1.1.1 **Postulated Cause of the Event**

A loss of power to the ISFSI may occur as a result of natural phenomena, such as lightning or extreme wind, or as a result of undefined disturbances in the nonsafety-related portion of the electric power system of the Surry Power Station.

If electric power is lost, the following systems would be de-energized and rendered nonfunctional:

1. Area lighting.
2. Cask monitoring instrumentation (pressure, temperature, etc.).

8.1.1.2 **Detection of Events**

A loss of ac power at the Surry site would be indicated and/or alarmed in the main control room of the Surry Power Station. If the loss of power were localized solely at the ISFSI, this would be indicated at the local annunciator.

8.1.1.3 **Analysis of Effects and Consequences**

This event has no safety or radiological consequences. None of the systems whose failure could be caused by this event are necessary for the accomplishment of the safety function of the ISFSI. The lighting system functions merely for convenience and visual monitoring, and the instrumentation monitors the long-term performance of the SSSCs with respect to heat transfer and leakage. None of these parameters are expected to change rapidly and their status is not dependent upon electric power.

8.1.1.4 **Corrective Actions**

Following a loss of electric power to the ISFSI, plant maintenance forces will be informed and will isolate the fault and restore service by conventional means. Such an operation is straightforward and routine for the maintenance crews of an electric utility.

A loss of power will not affect the integrity of the SSSCs, jeopardize the safe storage of the fuel, nor result in radiological releases.

8.1.2 Reference

1. *Topical Safety Analysis Report for the CASTOR V/21 Cask Independent Spent Fuel Storage Installation (Dry Storage),* GNSI, January 1985.

8.2 ACCIDENTS

This section addresses more serious occurrences which are expected to happen on an extremely infrequent basis, if ever, during the lifetime of the facility (Event Category III). In addition, a maximum hypothetical accident, which is not considered credible (Event Category IV), is identified and analyzed.

8.2.1 Earthquake

8.2.1.1 Cause of Accident

The design earthquake (DE) is postulated to occur as a design basis extreme natural phenomenon. As described in Sections 2.6.2 and 3.2.3, the DE (0.07 g) is expected to occur less than once in 500 years.

8.2.1.2 Accident Analysis

Seismic response characteristics of the SSSCs are provided in the SSSC topical reports. Results of these analyses show that cask leak-tight integrity is not compromised and that no damage will be sustained.

8.2.1.3 Accident Dose Calculations

As demonstrated in the SSSC topical reports, the DE is not capable of producing leakage from the cask and hence, no radioactivity is released. There is no associated dose from this event.

8.2.2 Extreme Wind

8.2.2.1 Cause of Accident

The extreme winds due to passage of the design tornado as defined in Section 3.2.1 are postulated to occur as an extreme natural phenomenon.

8.2.2.2 Accident Analysis

The effects and consequences of extreme winds on the casks are presented in Appendix A and the SSSC topical reports.

8.2.3 Flood

As shown in Section 3.2.2, the Surry ISFSI is considered flood-dry. Therefore, floods are not considered as design bases events.

8.2.4 Pipeline Explosion

8.2.4.1 Cause of Accident

An explosion is postulated to occur as a result of a failure of the natural gas pipeline at a point approximately 400 yards from the ISFSI. This occurrence is described in detail in Section 2.2.3. A pressure wave of less than 1 psi is estimated.

8.2.4.2 Accident Analysis

The SSSC topical reports describe the response of the casks to a gas cloud explosion.

8.2.4.3 Accident Dose Calculations

As shown in the analyses referenced above, the potential cask tip over due to a gas cloud is not capable of producing leakage from the cask. Since no radioactivity is released, no resultant doses would occur.

8.2.5 Fire

The only combustible materials in the ISFSI slabs are in the form of insulation on instrumentation wiring, and coating of the outside surface of the SSSCs. No other combustible or explosive materials are allowed to be stored on the ISFSI slabs. As described in Section 2.2.3.2.3, the ISFSI area will be cleared of trees and seeded with grass. In addition, other equipment in the area have been provided with adequate separation from the ISFSI slabs. Therefore, no fires other than small electrical fires are considered credible at the ISFSI slab. The ability of the casks to withstand postulated fires and the consequence of postulated fires are addressed in Appendix A and the SSSC topical reports.

The fire protection capabilities available at the ISFSI are described in Section 4.3.8. These include portable fire extinguishers within the ISFSI and the availability of the fire protection system for the Surry Power Station.

8.2.6 Dropped Fuel Assembly

8.2.6.1 Cause of Accident

Notwithstanding the multiple layers of safeguards against a fuel handling accident, it is postulated that an assembly is dropped in the worst possible orientation while being loaded into the cask.

8.2.6.2 Accident Analysis

The dropped fuel assembly accident is the limiting fuel handling accident analyzed in Section 14.4.1 of the Surry Power Station FSAR.

8.2.6.3 Accident Dose Calculation

The FSAR analysis has been modified to reflect the age of the fuel to be stored in the SSSCs. In this analysis, it is assumed that all 204 fuel rods in the assembly rupture and there is a sudden release of the gaseous fission products held in the voids between the pellets and cladding of the fuel rods. The low temperature of the fuel during handling operations precludes further significant release of gases from the pellets themselves after the cladding is breached. After a year's decay period, the only gas of significance is Kr-85. I-131 with its 8-day half-life would have decayed to an insignificant level within 1 year out of core. Table 8.2-1 gives the inventory for Kr-85 for various decay times for an average assembly with assumed fuel enrichments and burnup. Only the Kr-85 released to the water would escape from the pool.

Since the fuel assembly is postulated to be dropped in the spent fuel pool in the fuel building, the escaping Kr-85 mixes with the fuel building air and is exhausted through the fuel building exhaust. For the purpose of site boundary dose calculations, it is conservatively assumed that all the Kr-85 activity released to the pool becomes airborne, exhausted from the fuel building, and is transported to the nearest site boundary instantaneously. It is further assumed that Pasquill "F" meteorology conditions exist with a 1 meter per second wind speed yielding a dispersion coefficient, χ/Q, equal to 8.14×10^{-4} sec/m^3, at the nearest site boundary. This χ/Q is consistent with the value used to evaluate the radiological consequences of the fuel handling accident presented in Section 14.4.1 of the Surry Power Station FSAR.

Table 8.2-2 gives the assumptions used to determine the radiological consequences. Where applicable, assumptions from Regulatory Guide 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* (Safety Guide 25, 3/23/72), are employed in the analysis.

Table 8.2-3 gives the amount of activity released from the breached assembly and the resulting exposure to an individual at the closest site boundary and to the population out to 50 miles from the facility. Refer to Section 8.2.11 for a discussion on the methodology to determine population exposures. Dose models and conversion factors are taken from Regulatory Guide 1.109, *Calculations of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix 1,* Rev. 1, October 1977.

8.2.7 Inadvertent Loading of a Newly Discharged Fuel Assembly

The possibility of a premature assembly (one with a heat generation rate greater than the maximum allowable) being erroneously selected for storage in an SSSC has been considered.

8.2.7.1 Cause of Accident

The cause of this accident is postulated to be an error during the loading operations, e.g., wrong assembly picked by the fuel handling crane, or a failure in the administrative controls governing the fuel handling operations.

8.2.7.2 Accident Analysis

The maximum allowable heat generation rate for fuel assemblies to be stored in the SSSCs is provided in the SSSC topical reports and the Surry ISFSI Technical Specifications. The fuel assemblies require several years of storage in the spent fuel pool before the heat generation decays to an acceptable rate. This accident scenario postulates the inadvertent loading of an assembly not intended for storage in the SSSC, and possibly with a heat generation rate in excess of that specified for the particular SSSC.

In order to preclude this accident from going undetected, and to ensure that appropriate rectification actions can take place prior to the sealing of the casks, a final verification of the assemblies loaded into the casks and a comparison with fuel management records are performed to ensure that the loaded assemblies do not exceed any of the specified limits.

These administrative controls and the records associated with them are included in the procedures described in Chapter 9 and in the proposed license requirements described in Chapter 10, and will comply with the applicable requirements of the Quality Assurance Program described in Chapter 11.

Therefore, appropriate and sufficient actions will be taken to ensure that an erroneously loaded fuel assembly does not remain undetected. In particular, the storage of a fuel assembly with a heat generation in excess of the maximum allowable for an SSSC is not considered credible in view of the multiple administrative controls.

8.2.7.3 **Accident Dose Calculations**

The inadvertent loading of a fuel assembly not intended for storage in the SSSC will not result in unsafe fuel conditions or releases of radioactive products.

8.2.8 Loss of Neutron Shield

The design of some SSSCs includes neutron absorbing material both internal and external to the cask body. For those casks to be stored at the Surry ISFSI which feature an outer shell of neutron absorbing material, a solid shield material is used. None of the casks at the Surry ISFSI utilize a liquid neutron shield.

As applicable to the particular SSSC design, Appendix A and the SSSC topical reports discuss a postulated loss of neutron shield. As concluded in these documents, a total loss of neutron shield is not a credible event for the Surry ISFSI.

8.2.9 Cask Seal Leakage

The SSSCs feature redundant seals in conjunction with extremely rugged body designs. Additional barriers to the release of radioactivity are presented by the sintered fuel pellet matrix and the zircaloy cladding. Furthermore, the casks are not artificially pressurized above the small amount due to heating of the air or due to the inert gas (helium) in the cask. As a result, no credible mechanisms that could result in leakage of radioactive products have been identified. Nevertheless, a complete loss of the SSSCs confinement capability is postulated in Section 8.2.11, and the results found to be negligible.

Discussions of postulated cask seal malfunctions or loss of confinement barrier are presented in the SSSC topical reports or Appendix A.

8.2.10 Cask Drops

Cask handling and drop accidents postulated to occur within the fuel and decontamination buildings are addressed as part of the Surry Power Station operating license.

The SSSCs are designed to withstand drops onto the ISFSI pads without compromising the cask integrity. Technical Specifications limit the lift height for each cask. Cask drops in excess of these heights at the ISFSI, or enroute to it, are not considered credible because of procedures that preclude the lifting of the casks any higher. Analyses of cask drop accidents are presented in the SSSC topical reports.

If an off-normal handling accident were to occur, the following steps will be taken:

1. Health Physics personnel will perform radiation surveys of the cask.
2. A visual inspection of the cask body will be performed with particular attention to the area of the lifting trunnion. The trunnion will be removed and replaced if required.
3. The cask will be moved to the Surry Power Station decontamination building where the lid seals will be leak tested.
4. A gas sample will be obtained from the interior of the cask body to check for an unusual amount of Kr-85.
5. If there is no lid seal damage and no Kr-85 present at a level indicating fuel failure, the cask will be resealed using normal procedures and moved back to the ISFSI and placed on the storage pad.
6. If it is determined that the fuel must be removed from the cask, the interior of the cask will be flooded and then the water in the interior will be sampled prior to placing in the pool. If the water sample shows unacceptable levels, the water will be drained and processed as radwaste. The cask will then be reflooded and resampled prior to removing the lid to prevent uncontrolled releases of contamination to the fuel pool water.
7. The cask will then be moved into the fuel pool; the primary lid and the fuel will be removed.
8. If the fuel was removed due to the detection of potential fuel damage, the fuel will be inspected and any fuel assemblies containing rods with clad damage will be identified as being damaged and these assemblies will be stored in the fuel pool.
9. If the fuel was removed due to seal damage, the cask will be removed from the pool and repaired prior to further use.
10. The cask will then be reloaded with fuel using normal procedures and will be moved back to the ISFSI and placed on the storage pad.

8.2.11 Loss of Confinement Barrier

The following postulated accident scenario is not considered to be credible. It is hypothesized solely to demonstrate the inherent safety of the Surry ISFSI by subjecting it to a set of simultaneous multiple failures, any one of which is far beyond the capability of natural phenomena or man-made hazards to produce.

8.2.11.1 Cause of Accident

A simultaneous failure of all protective layers of confinement is postulated to occur by unspecified nonmechanistic means in an SSSC.

8.2.11.2 Accident Analysis

In this accident, the confinement function is nonmechanistically removed for the noble gas Kr-85. Heat removal and radiation shielding functions operate in the normal passive manner.

This is equivalent to breaking the cask seal barriers (no release), removing the closure lids (no release), failing all the cladding in all the loaded fuel assemblies (gap activity release), and finally, failing the fuel pellets themselves such that matrix confinement is no longer operable (remaining Kr-85 release).

8.2.11.3 Accident Dose Calculations

An analysis has been performed to determine the radiological consequences of a release of the entire gaseous inventory in a cask. The resulting dose at the nearest site boundary to an individual is well within the 5 rem criteria given in 10 CFR 72.68(b). The assumptions are given in Table 8.2-4. The dose models and dose conversion factors given in Regulatory Guide 1.109, Rev. 1, are used in this analysis. The resulting doses are given in Table 8.2-5.

To evaluate the impact upon the general population due to this postulated failure of a cask, the population exposure from this postulated event is compared to the population exposure resulting from background radiation sources. The plume of gaseous radioactivity is conservatively assumed to remain within the sector which would result in the highest population exposure. No credit is taken for the meandering of the plume which would greatly decrease the gaseous concentration in the plume and the fraction of the plume that would approach a given sector. Using Figure 2.1-3, which shows the 0- to 50-mile population distribution, and Figure 2.3-14 of Surry Power Station Units 3 and 4 PSAR, which shows χ/Q values as a function of distance from the site, to obtain the appropriate accident χ/Q at the midpoint of each annular sector, the population exposure doses are estimated. The sector receiving the highest estimated population exposure is the east-southeast sector. This sector receives an estimated exposure of approximately 153 man-rem, which is a small fraction of the annual population dose estimated to be approximately 49,000 man-rem from exposure to background radiation. In other words, the exposure due to a hypothetical incredibly severe accident at the Surry ISFSI would result in a general population dose approximately equal to 3 tenths of 1 percent of background.

8.2.12 Reference

1. *Topical Safety Analysis Report for the CASTOR V/21 Cask Independent Spent Fuel Storage Installation (Dry Storage),* GNSI, January 1985.

Table 8.2-1

Kr-85 INVENTORY[a] FOR WESTINGHOUSE 15x15 FUEL

Nominal Burnup (MWd/MtU)	Time After Discharge 150 days	Time After Discharge (years)									
		1	2	3	4	5	6	7	8	9	10
35,000	4.65+3	4.46+3	4.21+3	3.91+3	3.69+3	3.45+3	3.23+3	3.03+3	2.84+3	2.66+3	2.50+3
45,000	5.89+3	5.67+3	5.30+3	4.97+3	4.68+3	4.39+3	4.10+3	3.84+3	3.61+3	3.38+3	3.17+3

a.Curies/assembly

Table 8.2-2
ASSUMPTIONS USED TO EVALUATE RADIOLOGICAL CONSEQUENCES FROM A FUEL HANDLING ACCIDENT DURING ISFSI OPERATIONS

Spent Fuel Characteristics	
U-235 Enrichment	4.13 wt.%
Burn-Up	45,000 MWd/MtU
Time Out of Core	5 years
Number of Assemblies Damaged	1
Number of Failed Rods	204
Radial Assembly Peaking Factor	1.65
Kr-85 Inventory in Average Assembly	4.39E+3 Ci
Percent Assembly Inventory in Fuel Rod Gaps	30
Percent Gap Activity Released to Pool	100
Percent Activity Released to Pool Becoming Airborne	100
χ/Q at Nearest Site Boundary from Surry Power Station	8.14E-4 sec/m^3
Duration of Release	Instantaneous

Table 8.2-3
RADIOLOGICAL CONSEQUENCES FROM A FUEL HANDLING ACCIDENT DURING ISFSI OPERATIONS

Kr-85 Activity Released	2.17E+3 Ci
Total Body Dose at Site Boundary	0.90 mrem
Population Exposure (0 to 50 miles, ESE sector)	3.2 man-rem

Table 8.2-4
ASSUMPTIONS USED FOR LOSS OF CONFINEMENT BARRIER ANALYSIS

Activity Release Assumptions	
Spent Fuel Assembly Characteristics	
U-235 Enrichment	4.13 wt.%
Burn-up	45,000 MWd/MtU
Time out of core	5 yr
Kr-85 Inventory Per Assembly	4.39E+3 Ci
No. Assemblies per Cask	24
Gaseous Inventory Released	100%
Duration of Release	Instantaneous
Dose Model Assumptions	
Nearest Site Boundary from ISFSI χ/Q	1.56E-3 sec/m^3

Table 8.2-5
RADIOLOGICAL CONSEQUENCES FROM
LOSS OF CONFINEMENT BARRIER ANALYSIS

Kr-85 Activity Released	1.05E+5 Ci
Total Body Dose from Cloud Immersion at Nearest Site Boundary	84 mrem

8.3 SITE CHARACTERISTICS AFFECTING SAFETY ANALYSIS

Site characteristics have been considered in the formation of the bases for these safety analyses. Conditions of meteorology were used in the determination of χ/Q values as well as the characteristics of extreme winds and their contribution to maximum flood level. Regional and site seismology and geology were used to help define the design earthquake acceleration value. Population distribution and other demographic data were used to determine radiation doses.

Other site characteristics affecting safety analyses include the natural gas pipeline located 400 meters from the ISFSI which was used to develop the bases for the pipeline explosion (Section 8.2.4).

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Chapter 9
CONDUCT OF OPERATIONS

9.1 ORGANIZATIONAL STRUCTURE

9.1.1 Corporate Organization

The ISFSI will be operated under the same corporate management organization responsible for operation of the Surry Power Station. This organization is depicted in the Virginia Electric and Power Company Operational Quality Assurance Program Topical Report, VEP-1-5A (QA Program Topical Report).

9.1.1.1 Corporate Functions, Responsibilities, and Authorities

Corporate functions, responsibilities, and authorities for the Surry ISFSI are discussed in the QA Program Topical Report.

9.1.1.2 Applicant's In-House Organization

A discussion of Virginia Power's in-house organization is provided in the QA Program Topical Report

9.1.1.3 Relationships with Contractors and Suppliers

Bechtel Power Corporation was contracted for the engineering design of the Surry ISFSI, excluding the casks, and for the preparation of the license application.

The SSSC suppliers are responsible for the fabrication and testing of the SSSCs, and for recommending SSSC handling procedures. The Nuclear Analysis and Fuel Department is the primary interface with the SSSC supplier and other equipment vendors.

Site preparation and construction will be performed by Virginia Power, using specialty subcontractors, as required.

9.1.1.4 Applicant's Technical Staff

Virginia Power's technical staff is described in the QA Program Topical Report. Radiation dose assessment support services are being provided to the Nuclear Analysis and Fuel Department by Bechtel Power Corporation.

9.1.2 Operating Organization, Management, and Administrative Control System

9.1.2.1 Onsite Organization

The Surry Power Station onsite organization is described in the QA Program Topical Report.

9.1.2.2 Personnel Functions, Responsibilities, and Authorities

Personnel functions, responsibilities, and authorities are described in the QA Program Topical Report.

9.1.3 Personnel Qualification Requirements

9.1.3.1 Minimum Qualification Requirements

Each member of the Surry Power Station staff is required to meet or exceed the minimum qualifications specified in the QA Program Topical Report.

9.1.3.2 Qualifications of Personnel

The qualification requirements for the managerial and technical positions are described in the QA Program Topical Report.

9.1.4 Liaison with Other Organizations

Bechtel Power Corporation provides technical expertise on the design and licensing of the facility, and in the development of computer models to assess radiation doses. SSSC vendors provide technical expertise in the design, fabrication and use of the SSSCs.

9.2 STARTUP TESTING AND OPERATION

9.2.1 Administrative Procedures for Conducting Test Program

The administrative procedures and instructions for the Surry ISFSI are the same as those used for the Surry Power Station. Any changes to, or deviations from, these procedures and instructions are reviewed and approved in accordance with the QA Program Topical Report.

9.2.2 Test Program Description

The objectives of the startup testing program are to ensure that the SSSCs perform their safety functions as intended and that the means to fulfill the commitments made in Chapter 10 are available.

9.2.2.1 Physical Facilities

Before or during operation of the ISFSI, the SSSC monitoring instrumentation, the electrical system, the communications system, and the security system are tested to ensure their proper functioning. The ISFSI security system is tested after completion of its installation. Details on the security system are provided in the Security Plan.

The SSSC monitoring instrumentation alarms are tested to ensure that individual alarm signals annunciate at the local annunciator enclosure at the ISFSI location.

The electrical system is tested to ensure that power is available for the SSSC monitoring instrumentation and the local annunciator. The lighting and service receptacles are also tested for proper operation.

The communications system is tested to ensure that the telephone at the local annunciator is properly connected into the station telephone system.

9.2.2.2 **Operations**

Testing of SSSC operations, i.e., loading, drying, sealing, and unloading, shall be conducted prior to the first use of each SSSC design. This simulation shall include all SSSC loading and unloading operations, with the exception of loading actual fuel assemblies in the SSSC. SSSC loading will instead be tested with a dummy fuel assembly to ensure that fuel assemblies will fit properly into the SSSC. All SSSCs are tested for fuel assembly fit by the vendor at the fabrication facility. The SSSCs are also tested by the vendor to ensure that they seal properly. New seals are installed prior to and tested following fuel loading.

The function of the transporter is tested prior to its first use with each new SSSC design using an empty SSSC for a transport simulation to and from the ISFSI, including placement of the SSSC at a storage location.

9.2.3 Test Discussion

The pre-operational test purposes, responses, acceptance criteria, margins, and corrective actions are discussed in the Technical Specifications.

Instrumentation, electrical, and communications equipment shall be functionally tested to confirm operability. Acceptance criteria for the SSSC seal testing shall be as specified in Section 3.3.

9.3 TRAINING PROGRAM

9.3.1 Program Description

The training program has the objective of providing and maintaining a well-qualified work force for the safe and efficient operation of the ISFSI. All personnel working in the fuel storage area receive radiation and safety training. Those personnel actually performing SSSC and fuel handling functions are given additional training in specific areas as required by the radiological protection program in effect at the Surry Power Station.

All personnel working at the Surry ISFSI receive training and indoctrination geared toward providing and maintaining a well-qualified work force for the safe and efficient operation of the ISFSI. The existing Surry training programs are INPO accredited and are directly applicable to the Surry ISFSI, and provide this training and indoctrination. Additional training requirements specific to the ISFSI will address the following subjects:

- ISFSI Licensing Basis and Technical Specifications

- ISFSI Layout and Function
- ISFSI Security
- ISFSI Communications Systems
- ISFSI Operation, Emergency, Maintenance, and Administrative Procedures
- SSSC Loading and Unloading, Handling and Onsite Transportation
- SSSC Decontamination Techniques

Following completion of the ISFSI training program, trainees are given a written and practical exam to ensure they understand the important aspects of the information described above. Retention of training records and certifications of proficiency is consistent with that for personnel involved in fuel handling operations.

ISFSI retraining is consistent with the retraining requirements in effect at the Surry Power Station for personnel involved in fuel handling operations.

Training records are maintained in accordance with the QA Program Topical Report. Such records include dates and hours of training and other documentation on training subjects, information on physical requirements, job performance statements, copies of written examinations, information pertaining to walk-through examinations, and retesting particulars.

9.4 NORMAL OPERATIONS

9.4.1 Procedures

Written procedures for all normal operating, maintenance, and testing at the ISFSI will be prepared and will be in effect prior to operation of the Surry ISFSI. These procedures are briefly described in Sections 9.4.1.1 through 9.4.1.8.

These procedures, and any subsequent revisions, will be reviewed and approved in accordance with the QA Program.

The Nuclear Oversight Department periodically reviews the procedures to ensure revisions are made promptly and that obsolete material is deleted.

9.4.1.1 Administrative Procedures

Administrative procedures will provide a clear understanding of operating philosophy and management policies to all ISFSI personnel. These procedures include instructions pertaining to personnel conduct and control, including consideration of job-related factors which influence the effectiveness of operating and maintenance personnel, e.g., work hours, entering and exiting the ISFSI, organization, and responsibility, etc.

9.4.1.2 **Annunciator Procedures**

Operating procedures for Post-Alarm testing provide information relative to each alarm annunciator which monitors SSSC parameters. Alarm setpoints are provided in the Technical Specifications. The procedures provide appropriate corrective action.

9.4.1.3 **Health Physics Procedures**

Health physics procedures are used to implement a radiation protection plan. The radiation protection plan involves the acquisition of data and provision of equipment to perform necessary radiation surveys, measurements, and evaluations for the assessment and control of radiation hazards associated with the operation of the ISFSI. Procedures have been developed and implemented for monitoring exposures of employees, utilizing accepted techniques, radiation surveys of work areas, radiation monitoring of maintenance activities, and for records maintenance demonstrating the adequacy of measures taken to control radiation exposures of employees and others within prescribed limits and as low as practicable. These procedures will be revised as needed to address ISFSI operations prior to operation of the ISFSI. The revised procedures will ensure the safety of personnel performing loading, transport and unloading operations, and surveillance and maintenance at the ISFSI. Entrance to the ISFSI and all work performed inside will require a radiation work permit and will be controlled by health physics and security personnel.

9.4.1.4 **Maintenance Procedures**

Maintenance procedures will be established for performing preventative and corrective maintenance on ISFSI equipment and the SSSCs. Preventative maintenance will be performed on a periodic basis to preclude the degradation of ISFSI systems, equipment, and components. Corrective maintenance will be performed to rectify any unexpected system, equipment, or component malfunction, as the need arises.

9.4.1.5 **Operating Procedures**

The operating procedures will provide instructions for handling, loading, sealing, transporting, storing, and unloading the SSSCs.

9.4.1.6 **Test Procedures**

Periodic test procedures will be established to verify operability of the ISFSI systems, equipment, and components on a routine basis.

9.4.1.7 **Startup Test Procedures**

Startup test procedures will be established to ensure that ISFSI structures, systems, and components satisfactorily perform their required functions. These test procedures will further ensure that the ISFSI has been properly designed and constructed and is ready to operate in a manner that will not endanger the health and safety of the public.

9.4.1.8 Procedures Implementing the QA Program

Procedures will be established to ensure that the operation and maintenance of the ISFSI is performed in accordance with the QA program described in Chapter 11.

9.4.2 Records

Records for decommissioning of the ISFSI that will be retained under 10 CFR 72.30(d) are described below. Records Management will maintain these records until the site is released for unrestricted use.

- Records of spills or other unusual occurrences involving the spread of contamination in and around the ISFSI.
- As-built drawings and modifications of structures and equipment at the ISFSI
- A list contained in a single document and updated no less than every two years of (a) all areas designated and formerly designated as restricted areas as defined by 10 CFR 20.1003, and (b) all areas outside of restricted areas that require documentation under item 1.
- Records of the cost estimate performed for the decommissioning funding plan and the funding method used for assuring funds.

Records on the spent fuel stored at the ISFSI that will be retained under 10 CFR 72.72(a) are described below. These records will be maintained by Records Management for the period that spend fuel is stored at the ISFSI plus five years after transfer.

- Fuel manufacturer
- Date of delivery to the Station
- Reactor exposure history
- Burnup
- Calculated special nuclear material content
- Inventory control number
- Pertinent data on discharge and storage at the reactor, transfer to the ISFSI, storage at the ISFSI and disposal
- Other information needed to verify compliance with ISFSI Technical Specifications

A record of the current physical inventory of spent fuel at the ISFSI required by 10 CFR 72.72(b) will be retained by Records Management until the ISFSI license is terminated by the US NRC. The current material control and inventory procedures required by 10 CFR 72.72(c) will be retained by Records Management until the ISFSI license is terminated by

the US NRC. Records of spent fuel transferred out of the ISFSI will be preserved for a period of five years after the date of transfer.

9.5 EMERGENCY PLANNING

The Surry Emergency Plan (SEP) describes the organization, assessment actions, conditions for activation of the emergency organization, notification procedures, emergency facilities and equipment, training, provisions for maintaining emergency preparedness, and recovery criteria used at the Surry Power Station. This emergency plan will also be used for any radiological emergencies that may arise at the Surry ISFSI.

Portions of SEP Section 4 and Appendix 10.8 and the applicable implementing procedure reflect the conditions and indications that require entry into the Emergency Plan. Appropriate response actions and notifications have been established in the Emergency Plan. SSSC seal leakage and SSSC drop or other handling mishap requires declaration of a Notification of Unusual Event. Loss of all SSSC and fuel containment barriers requires declaration of an Alert.

9.6 DECOMMISSIONING

Decommissioning considerations are discussed in Section 3.5, and in the Decommissioning Plan attached to the License Application.

9.6.1 Decommissioning Program

The dry cask design concept utilized at the Surry ISFSI features inherent ease and simplicity of decommissioning. At the end of its service lifetime, cask decommissioning could be accomplished by one of the following options:

1. The ISFSI cask, including the spent fuel stored inside, could be shipped to an offsite facility for temporary or permanent storage. Depending on licensing requirements existing at the time of shipment offsite, placement of the entire ISFSI cask inside a supplemental shipping container or overpack would be considered

2. The spent fuel could be removed from the ISFSI cask and shipped in a licensed shipping container to a suitable fuel repository. If desirable, cask decontamination could be accomplished through the use of conventional high pressure water sprays to further reduce contamination on the cask interior. The sources of contamination on the interior of the cask would be crud from the outside of the fuel pins and the crud left by the spent fuel pool water. The expected low levels of contamination from these sources could be easily removed with a high pressure water spray. After decontamination, the ISFSI cask could either be cut-up for scrap or partially scrapped and any remaining contaminated portions shipped as radioactive waste to a disposal facility.

Cask activation analyses have been performed to quantify specific activity levels of cask materials after years of storage. These activation calculations and the assumptions under which they were performed are described in the SSSC Topical Reports. Based on the results of the

analyses, the cask materials will be only slightly activated by the low level neutron flux emanating from the stored spent fuel. Consequently, it is expected that after application of the surface decontamination process as described above, the radiation level due to activation products will be negligible and the cask could be scrapped. A detailed evaluation will be performed at the time of decommissioning to determine the appropriate mode of disposal.

Due to the zero-leakage design of the SSSCs, no residual contamination is expected to be left behind on the concrete base pad. The base pad, fence, and peripheral utility structures are de facto decommissioned when the last cask is removed.

The spent fuel pool at Surry Power Station will remain functional until the ISFSI is decommissioned. This will allow the pool to be utilized to transfer fuel from the storage casks to licensed shipping containers for shipment offsite if this decommissioning option is chosen.

9.6.2 Cost of Decommissioning

Virginia Power presently owns and operates four nuclear power generating units. In view of the financial qualifications represented by this fact, it is anticipated that decommissioning costs of the Surry ISFSI will not be an issue. It is expected that decommissioning costs will represent a small fraction of the costs of decommissioning the Surry Power Station Units 1 and 2.

9.6.3 Decommissioning Facilitation

The volume of waste material produced incidental to ISFSI decommissioning will be limited to that necessary to accomplish surface decontamination of the casks once the spent fuel elements are removed. Furthermore, it is estimated that the cask materials will be only very slightly activated as a result of their long-term exposure to the relatively small neutron flux emanating from the spent fuel, and that the resultant activation level will be well below allowable limits for general release of the casks as noncontrolled material. Hence, the casks may be decommissioned from nuclear service by surface decontamination alone.

Chapter 10
OPERATING CONTROLS AND LIMITS

This chapter provides safety limits, limiting conditions for operation, and surveillance requirements for the Surry ISFSI which were incorporated into the ISFSI operating license.

Part of the evaluation of the ISFSI is the evaluation of the "Independence" of an ISFSI on an existing reactor site such as is the case with the Surry ISFSI. This evaluation has been performed using the definition for "Independent" contained in 10 CFR Part 72.

The results of this evaluation are as follows:

1. The ISFSI can operate independently without affecting the safety and operation of the nuclear units at Surry Power Station as there are no physical connections between the reactor units and the ISFSI other than connections which serve no safety-related functions (power for ISFSI lighting and security equipment) and the ISFSI security alarm indications.
2. The ISFSI can recover from normal or off-normal incidents or accidents without affecting the safety and operation of the nuclear units at Surry Power Station.
3. The nuclear reactor units at the Surry Power Station do not affect the safety and operation of the ISFSI.
4. No changes to the Surry Power Station 10 CFR Part 50 operating licenses are required as a result of the ISFSI.

In conclusion, the Surry Dry Cask ISFSI is "Independent" as defined in 10 CFR Part 72.

10.1 TECHNICAL SPECIFICATIONS

The Surry ISFSI Technical Specifications govern the safety of the receipt, possession, and storage, of irradiated nuclear fuel at the Surry Dry Cask. Independent Spent Fuel Storage Installation and transfer of such irradiated nuclear fuel to and from the Surry Nuclear Power Station and the Surry Dry Cask Independent Spent Fuel Storage Installation.

10.2 RECORDS

10.2.1 Records

10.2.1.1 Records shall be kept identifying the spent fuel assemblies stored in each SSSC, manufacturer, date of delivery, their storage location within the SSSC basket, initial enrichment, reactor exposure history, estimated burnup, time since discharge from the core, and the estimated heat rate.

10.2.1.2 Records shall be kept of the radiation measurements specified in Technical Specifications.

10.2.1.3 Records shall be kept of fuel transferred out of the ISFSI.

10.2.2 Retention of Records

10.2.2.1 Records specified in Section 10.2.1.1 shall be retained as long as the stored fuel remains within the Surry site.

10.2.2.2 Records specified in Sections 10.2.1.2 and 10.2.1.3 shall be retained for 5 years.

10.3 REPORTS OF ACCIDENTAL CRITICALITY OR LOSS OF SPECIAL NUCLEAR MATERIAL

Any case of accidental criticality or any loss of special nuclear material at the ISFSI shall be reported immediately to the appropriate NRC authorities, as specified in 10 CFR 72.74.

10.4 MATERIAL STATUS REPORTS

Material Status Reports shall be completed and submitted to the NRC, as specified in 10 CFR 72.53.

10.5 NUCLEAR MATERIAL TRANSFER REPORTS

Nuclear material stored in the ISFSI will not be transferred from Virginia Power to other ownership. Hence, assuming the existing Reporting Identification Symbol (RIS) for Surry fuel remains the same, Nuclear Material Transaction Reports (DOE/NRC Form-74) required by 10 CFR 72.54 will not be needed for operation of the ISFSI.

10.6 FINANCIAL REPORTS

A copy of the Virginia Power annual financial report, including certified financial statements shall be submitted to the NRC, as specified in 10 CFR 72.55(b).

10.7 ISFSI ACTIVITIES REPORTS

The Monthly Operating Reports for the Surry Power Station Units 1 and 2 shall include pertinent information regarding operation of the ISFSI.

10.8 ADMINISTRATIVE CONTROLS

10.8.1 The Surry Power Station Units 1 and 2 Site Vice President shall be responsible for the safe operation of the ISFSI. In his absence or unavailability, the Director Station Operations and Maintenance shall be responsible for the safe operation of the ISFSI. During the absence or unavailability of both, the Site Vice President shall delegate in writing the succession to this responsibility.

10.8.2 The station and offsite organization for management and technical support of the ISFSI, and their functions, shall be the same as for the Surry Power Station, as applicable.

10.9 MONITORING AND SURVEILLANCE COMMITMENTS

Monitoring and surveillance commitments are provided in the Surry ISFSI Technical Specifications.

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Chapter 11
QUALITY ASSURANCE

11.1 QUALITY ASSURANCE PROGRAM DESCRIPTION—VIRGINIA POWER

10 CFR 72.140 requires that a quality assurance program be established and implemented for the structures, systems, and components of an ISFSI that are important to safety, commensurate with their importance to safety. However, 10 CFR 72.140 provides for the use of previously approved programs.

Since Virginia Power is currently licensed under 10 CFR Part 50 to operate nuclear power facilities, a quality assurance (QA) program meeting the requirements of 10 CFR Part 50, Appendix B, is already in place. The governing document for this program is the Virginia Electric and Power Company Operational Quality Assurance Program Topical Report (QA Program Topical Report), VEP-1-5A (updated) which has been reviewed and approved by the NRC. (See Section 1.5.) The document is updated in accordance with 10 CFR 50.54(a). The NRC is periodically notified of changes to the document. This program is implemented through the Virginia Power administrative and technical procedures. The objective of the company Quality Assurance Program for operating nuclear power stations is to comply with the criteria as expressed in 10 CFR 50, Appendix B, as amended, and with the quality assurance program requirements for nuclear power plants as referenced in the Regulatory Guides and ANSI standards referenced in Table 17.2-0 of the QA Program Topical Report. This program will be applied to those activities associated with the Surry ISFSI that are important to safety. No changes to this program are required for the ISFSI activities.

As indicated in previous chapters, the SSSCs are the only components with a safety function. As such, Virginia Power procedures delineate the requirements for the engineering, procurement, fabrication, and inspection of this equipment. The procurement documents (specifications, requisitions, etc.) are reviewed technically prior to use to ensure that the proper criteria have been specified. During the SSSC design phase, vendor information (drawings, specifications, procedures, etc.) are reviewed to ensure compliance with Virginia Power's technical requirements. During SSSC fabrication, Virginia Power's vendor surveillance representative will visit the vendor's shop to ensure compliance with Virginia Power's requirements and to witness parts of the cask fabrication and testing.

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QUESTIONS AND RESPONSES

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Question 1.1.1

It is stated in Section 2.1.3.1 that the population projections (0-10 mile radii) are the same as those presented in the environmental report on the Surry Power Station Units 3 and 4, and they are used in this SAR because there was little change in this population between 1970 and 1980. This seems inconsistent with the projections of population for the 1980-1990 period for the 0-5 mile radii showing a decrease of over 60 percent and for the 5-10 mile radii which more than doubled. Explain the difference in the population trends between these two decades.

Response

Surry ISFSI ER Section 2.1 and SAR Section 2.1 have been updated to reflect revised 0- to 10- and 10- to 50-mile estimates and projections of population.

Question 1.1.2E

The large increase in population within 10 miles between 1980 (61,711) Figure 2.1-3 and 1990 (161,454) Figure 2.1-4 seem unreasonable. Subsequent rates of growth to 2020 are much smaller. Please justify or correct these numbers.

Response

Surry ISFSI ER Section 2.1.2 and SAR Section 2.1.3 have been updated to reflect revised 0- to 10- and 10- to 50-mile estimates and projections of population.

Question 1.1.3E

Attachment 1 to Figure 2, page 4-C-13 of the Virginia Radiological Emergency Plan, revised August 1981, indicates a 1980 population within the 10 mile EPZ of 79,991. This is considerably higher than the 61,711 reported in 2.1.2.1. Please explain the basis of the difference.

Response

The population estimates in ISFSI ER Section 2.1 and SAR Section 2.1 have been revised.

Question 1.1.4

The projected population distributions in Figures 2.1-3 through 7 should be checked for errors. For example, the NE sector in Figure 2.1-3 for 40 to 50 miles has an indicated population of 4,000; yet, this area is totally within the Chesapeake Bay. How were the population distributions for 10-50 mile area estimated.

Response

Surry ISFSI ER Section 2.1 and SAR Section 2.1 have been updated to reflect revised 0- to 10- and 10- to 50-mile estimates and projections of population.

Question 1.1.5E

Why are 10-year old transient population estimates incorporated by reference in Section 2.1.2.3 when more current estimates are available in the Virginia Radiological Emergency Plan, referred to above? Does VEPCO still consider the population estimates and projections from the Surry 3 and 4 Environmental Report to be valid? If not, furnish the most and valid current population data.

Response

Surry ISFSI ER Section 2.1 and SAR Section 2.1 have been updated to reflect revised 0- to 10- and 10- to 50-mile estimates and projections of population.

Question 1.2.1

It is stated in Sections 2.2.1, 2.2.3, and 2.2.3.1.4 that the Commonwealth Natural Gas Corporation and the Colonial Pipeline Company own pipelines which cross the southeast corner of the Surry property. How many pipelines are there? Table 2.2-6 indicates six and notes that the two Commonwealth National Gas Corporation lines lie four feet beneath the river bed. How about the remaining four? Once these pipelines emerge from the river are they buried or aboveground?

Response

Table 2.2-6 of the Surry ISFSI SAR has been revised to include the information requested above.

Question 1.2.2

Section 2.2.3.1.2 of the SAR states the dredged channel in the river is 2.5 miles from the ISFSI at the closest point (this distance is also used in the accompanying accident analysis), but in Table 2.2-2, the distance is given as 1.5 miles. Which is correct?

Response

Section 2.2.3.1.2 of the Surry ISFSI SAR states that the dredged channel in the James River is 2.5 miles from the ISFSI site at its closest point. Inspection of Figure 2.3-25 of the Surry ISFSI ER confirms this statement. The 2.5-mile distance refers to the separation between the site and the mid-river channel.

The 1.5-mile distance given in the SAR Table 2.2-2 (referenced from the Surry Onsite Toxic Chemical Analysis, Vol. II, NUS, June 1981) refers to the minimum separation of the Surry Power Station control room and the James River. Note that the NUS analysis does not specify what part of the James River is used to calculate the minimum separation. The Surry ISFSI site is farther from the James River than the control room.

Question 1.3.1

Section 3.2.1.1 identifies the design basis extreme ambient temperatures of -20°F and 115°F. These temperatures were selected because they exceed the extreme temperatures recorded at the Norfolk and Richmond National Weather Service Stations as reported in Section 2.3.2.1.1. The conservatism of the upper temperature extreme should be clearly established in Section 2.3.2 because, for a truly passive spent fuel storage system, the unaided atmosphere will serve as the principal heat sink. Provide a rigorous basis for establishing these temperatures. The discussion could answer the following questions:

1. Are these temperatures reported from nearby weather service stations representative of the region?
2. Did the onsite meteorological station temperature records correlate with the nearby water service stations records?
3. Does the site have peculiar micrometeorological conditions that could cause a difference between its readings and the nearby weather stations readings?
4. What is the probable length and frequency of occurrence of excessively hot periods?
5. What is the worst combination of climatology conditions which would adversely affect the ability of the ambient air to remove heat from the cask surface?
6. What is the recurrence intervals and duration for the selected extreme temperatures?

The following references may be of help in developing the statistical bases for the discussion:

> "Extreme Meteorological Events in Nuclear Power Plants, Excluding Tropical Cyclones," IAEA Safety Guide No. 50-SG-S11A, and "Probability Estimates of Temperature Extremes for the Contiguous United States," NUREG/CR-1390, May, 1980.

Response

The response to this question has been incorporated into ISFSI SAR Section 2.3 and ER Section 2.3.

Question 1.3.2

Also in Section 2.3.2 the basis for insolation design parameters should be provided. The conversatism of the solar heat load burden at the ISFSI site should be substantiated in a discussion that justifies the selection of the 90 percent transmissivity factor and the 100-hour exposure period, or these should be changed to the more severe conditions.

Response

The response to this question has been incorporated into ISFSI SAR Section 2.3.

Question 1.3.3

Provide a discussion of the potential for lightning strikes at the ISFSI. This discussion could include the following topics:

1. Onsite experience with lightning strikes on Surry Power Station structures and switchyard facilities.
2. A correlation of the frequency and intensity of both single and multiple lightning strikes associated with regional thunderstorms.
3. The expected frequency of thunderstorms at the ISFSI site.
4. The limiting case for energy release associated with a lightning strike.

Response

ISFSI ER Section 2.3 and SAR Section 2.3 have been revised to reflect the response to this question.

Question 1.3.4

Provide a correlation between the Surry site specific data, developed over the years the onsite meteorological program has been established at the Surry site, and the Richmond and Norfolk data provided in Tables 2.3-1 through 2.3-4.

Response

Surry ISFSI ER Section 2.3 and SAR Section 2.3 have been revised to reflect the response to this question.

Question 1.3.5E

Describe the methodology for obtaining the χ/Q values in Table 7.1-3 and Figure 7.1-1. Include the models and input data used.

Response

Surry ISFSI SAR Section 2.3 has been revised to reflect the response to this question.

Question 1.3.6

Provide an analysis of the χ/Q values based on onsite meterological data and appropriate atmospheric diffusion models.

Response

Surry ISFSI SAR Section 2.3 has been revised to reflect the response to this question.

Question 1.3.7

Since the ISFSI will be located close to the primary and back-up meteorological towers, and the casks provide a continuing heat source; provide an analysis of the impact of the meteorological measurements at these towers.

Response

Surry ISFSI SAR Section 2.3 has been revised to reflect the response to this question.

Question 1.3.8E

In Section 5.6.2, Climatological Impact, it is stated that the cask surface temperature may reach 260°F, concluded that the affected area for atmospheric heating and fogging during precipitation would be small and any enhancement of fog beyond the site boundary would be negligible. Provide the calculations and bases for these conclusions. Include input parameters and equations used in the calculation.

Response

Surry ISFSI ER Section 5.6.2 has been revised to reflect the response to this question.

Question 1.4.1

The historical earthquake data presented and the development of the ISFSI design earthquake (0.07g) utilizing Trifunac and Brady's 1975 study *The Correlation of Seismic Intensity Scales with Peaks of Recorded Strong Ground Motion* does not correspond to the information presented for the selection of 0.15g safe shutdown earthquake for the Surry Power Station. For example, the 1927 Coastal Plain earthquake occurring near the central New Jersey coast is not identified in the historical earthquake data for the ISFSI. Adopt the corresponding g value developed for the nuclear power plant, or justify this new analysis in terms of the criteria of Appendix A of 10 CFR Part 100. In any case the g value should not be less than 0.10g per paragraph 72.66 (a)(b)(iii) or 10 CFR Part 72.

Note: Section 2.6.2.5 uses Applied Technology Council's seismic zonation map in their 1978 publication, *Tentative Provisions for the Development of Seismic Regulations for Buildings* NBS special publications 510, for additional justification. This reference is based on USGS Open File Report published in 1978, and it is not suitable for determining design earthquakes for structures. This was discussed in supplementary information published in the *Federal Register*, Volume 45, No. 220 on November 12, 1980 accompanying the promulgation of 10 CFR Part 72. In addition, there is an update on the USGS report. It is *Probabilistic Estimates of Maximum Acceleration and Velocity in Rock in the Contiguous United States,* by S. T. Algermissen, D. M. Perkins, P. C. Thenhaus, S. L. Hauson and B. L. Bender, U.S.G.S., Open-File Report 82-1033, 1982. This report includes preliminary maps of horizontal acceleration (expressed as percent of gravity) with a 90 percent probability of not being exceeded in 10, 50 and 250 years. On the 10-year map, the maximum g value for site area is 0.04, on the 50-year map, the maximum g value for the site area is 0.10 and on the 250-year map, the maximum g value for the site area is 0.20. The differences between the results of this study and those presented in Surry ISFSI SAR are significant relative to both g values and recurrence interval. This report is more recent than any of the references cited in connection with Section 2.6 of the SAR.

Response

1. NRC Questions 1.4.1, 3.2.1, and 3.2.2 presented NRC concerns with respect to the seismological, geological, and structural design bases for the Surry Dry Cask ISFSI. A meeting was held between Virginia Power and the NRC on March 8, 1984 to discuss the issues raised in the NRC questions. Due to the nature of the questions, and the fact all three deal with the determination of the design seismic event and its effect on the structural slab, cask, and stored spent fuel, no individual responses are made to these questions. Instead, a single response that answers the concerns raised by the NRC, as related to all three questions, was prepared and is presented below. This approach was discussed and agreed upon at the March 8, 1984 meeting.

   The Surry Dry Cask Independent Spent Fuel Storage Installation (ISFSI) is designed to store spent fuel resulting from the operation of Surry Power Station Units 1 and 2. The spent fuel will be stored in dry sealed surface storage casks (SSSCs), which provide shielding and confinement of the radioactive fission products. The ISFSI facility will consist, in its final stage, of three separate reinforced concrete slabs. A general site layout for the ISFSI is shown in Figure 4.1-1 of the Safety Analysis Report. Each concrete slab, which will have overall dimensions of 32 feet in width, 230 feet in length, and 3 feet in thickness, is designed to support 28 SSSCs. The slab will be supported on a 7-foot-thick bed of compacted backfill material, which is then underlain by the naturally occurring site soils.

Separate investigations and analyses, outside those previously performed for Surry Units 1 and 2 and the once proposed Surry Units 3 and 4, were conducted for the Surry ISFSI. These analyses were performed as part of the response to the NRC questions and later discussions with the NRC regarding these questions. At the March 8, 1984 meeting between Virginia Power and the NRC, specific criteria for the resolution of NRC concerns were discussed. These criteria are as follows:

a. A Design Earthquake (DE) that is developed based on criteria of 10 CFR 72.66(b).

b. A cask tip over must be assumed regardless of analyses that demonstrate that the integrity of the structural pad is maintained and that the cask will not tip over. This criteria is independent of a specific seismic acceleration. Assuming a cask tip over, the analysis must demonstrate that: 1) criticality is within acceptable limits, 2) there is no loss of confinement, and 3) fuel is removable after a tip over.

c. Analyses of the slab under DE must demonstrate that the design function of the ISFSI is not adversely affected and that there is no impact on the public health and safety.

In order to meet these criteria, the additional analyses and investigations which were performed included the determination of a Design Earthquake (DE) based on 10 CFR 72.66(b), a site specific investigation, a soil stability analysis (static and dynamic), and design analyses of the structural slab. Additional analyses in support of their licensing efforts are being performed by General Nuclear Systems, Inc. (GNS), the cask vendor, to determine the criticality, cask integrity, and basket integrity based on a hypothetical cask overturning event.

The extent of the information required to be submitted to address the NRC's concerns with respect to seismicity and stability of subsurface materials has been included in revised SAR Section 2.6 and new SAR Appendix 3A.

A summary of revised SAR Section 2.6 and new Appendix 3A is contained in Parts 2 and 3 of this response. A discussion of the structural analysis as requested by the NRC, the response of the cask due to a overturning event, and the conclusions of these additional analyses are provided in new Appendix 3A.

2. SEISMOLOGY

The requirements of 10 CFR 72.66(b) stipulate that for determining the seismic design level of a dry cask facility, a site specific investigation, must be performed to establish site suitability commensurate with the specific requirements of the ISFSI. Due to the inherent safety of the SSSCs and the fact that the structural slabs are not important to safety, the approach that was taken to determine the proper seismic design level was based on the use of a building code type seismic design level. Determination of this type of seismic design level depends principally on historic site intensity or probabilistic site

acceleration at approximately the 500-year return period and not on a maximum credible site intensity. Applicable studies, which are referenced in revised Section 2.6 of the Safety Analysis Report, indicate that the appropriate probabilistic acceleration is 5 percent of gravity or less and that the historic intensity is VI Modified Mercalli (MM) or less for the Surry ISFSI site. This intensity can be related to a peak horizontal acceleration of 6.6 percent of gravity.

Based on the results of this site specific investigation, a conservative value for the seismic design level or Design Earthquake (DE) of 7 percent of gravity at the foundation level was adopted for the Surry ISFSI. The details of the site specific investigation and the development of the Design Earthquake are contained in revised Section 2.6 of the Safety Analysis Report.

3. STABILITY OF SUBSURFACE MATERIALS

A site specific subsurface investigation and laboratory testing program was conducted for the ISFSI in April and May 1982. The investigation included the drilling of nine test borings, the installation of an observation well, and taking of both undisturbed and disturbed soil samples. The boring logs resulting from the investigation are shown in Figures 2.6-32 through 2.6-42 of the Safety Analysis Report. The boring location plan is shown in Figure 2.6-43. All field work was monitored by a geotechnical engineer. Select recovered samples obtained during the investigation were sent to a testing laboratory for determination of the engineering properties of the site soils. The results of both the field and laboratory investigations were then used to determine the static and dynamic stability of the site soils.

The static analyses, which were performed, included a bearing capacity and settlement analysis. Based on the results of these analyses, it was determined that it will be necessary to excavate and replace the upper soil with a compacted backfill material. In order to obtain the required minimum factor of safety of 3.0 for bearing capacity, 7 feet of soil below the bottom of the slab will be excavated and replaced with fill. The fill will be placed to a minimum density of 95 percent of optimum modified proctor density (ASTM D 1557). The bearing capacity factor of safety with the structural backfill in place is greater than 3.0. The calculated settlement due to static loading is less than 2.0 inches.

The dynamic analysis of the site soils was performed to determine the soil response to dynamic loading. The dynamic loading considered in the analysis was the Design Earthquake (DE) with a maximum ground acceleration of 7 percent of gravity at the foundation level. The analysis indicated that the stability of the site soils would not be adversely affected by the level of dynamic loading. The magnitude of subsidence under dynamic loading can be considered insignificant and will have no adverse effect on the structural slab. A liquefication analysis was also performed on all soil layers below the

maximum ground water level. The analysis, which was based on the "Simplified Procedure" developed by Seed and Idriss (References 1 and 2), indicated that the minimum factor of safety against liquefication occurring is 1.5. The calculations using the simplified procedure do not incorporate any adjustment factors for the silt content of any specific soil layer. In addition, the dynamic stresses induced by the DE in the cohesive soil layer are considerably less than the shear strength of these layers. Therefore, no reduction of shear strength will result.

In summary, it can be concluded that the site soils at the ISFSI site will provide a safe and stable foundation under both static and dynamic loading conditions. The details of the soil stability analysis are contained in revised Section 2.6 of the SAR.

4. REFERENCES

1. Seed, H. B., I. M. Idriss, "Simplified Procedure for Evaluating Soil Liquefication Potential," Journal of the Soil Mechanics and Foundation Division ASCE, Vol. 97, No. SM9, September 1971.

2. Seed, H. B., I. M. Idriss, I. Arrango, "Evaluation of Liquefaction Potential Using Field Performance Data," Journal of Geotechnical Engineering ASCE, Vol. 109, No 3, March 1983.

Question 1.5.1

Provide a summary of all factors developed in the site characteristics chapter that are deemed significant to the selection of design bases for the ISFSI. For each factor, identify if it was newly developed in the ISFSI SAR, or if it was referenced from a document previously submitted to the NRC. If it is referenced, provide the specific reference, its date and revision, and the applicable page numbers.

Response

Table 2.7-1 of the ISFSI SAR has been added to include the response to this question.

Question 2.1.1

Section 3.1.1 should be expanded to include the allowable limits on all pertinent characteristics of the spent fuel that can affect the design and operation of any portion of the ISFSI system. An allowable limit should be in terms of maximum, minimum, or a range of values, as appropriate and not an average or typical value. These limits provide the basis for assessing the compatibility of the ISFSI system with the spent fuel to be stored. Confirmatory analyses and performance requirements of the design and operation of the ISFSI system and its components will be required to envelope these limits. It is expected that most of these limits will be included in the limiting conditions for operation of the ISFSI and as such should be readily verifiable.

NOTE: If no limit is identified for a pertinent characteristic, then it will be assumed that the design and operation of the ISFSI will accommodate all possible values of that particular characteristic. For example, Section 10.1.1.1 Fuel To Be Stored At ISFSI states "The fuel shall be stored unconsolidated and shall not be extensively damaged". It further identifies damage relative to maintaining cooling geometry and the ability to insert and remove the fuel from the storage cask. There is no mention of the initial fuel pin cladding integrity relative to its function as the primary confinement barrier to the release of radioactive material. Therefore, if that is the intention, a substitute barrier for the fuel pin cladding would have to be provided in the ISFSI system.

Response

Surry ISFSI SAR Sections 3.1 and 10.1 have been revised to reflect the response to this question.

Question 2.1.2

For each pertinent characteristic identified in 2.1.1, provide the method by which the characteristic will be verified.

NOTE: A verification method need not directly measure a pertinent characteristic. Another characteristic, more amenable to verification, could be used to assure the existence of the pertinent characteristic. Also a verification method can accommodate more than one pertinent characteristic.

Response

Surry ISFSI SAR Sections 9.1 and 10.1 have been revised to reflect the response to this question.

Question 3.1.1

Elaborate on Table 3.3-1 by providing the design criteria and performance specifications to be imposed on cask designers and suppliers. The following paragraphs present examples of topics that should be addressed as an indication of the breath of coverage needed. They should not be considered as a comprehensive listing of cask design requirements, nor should any example requirements be considered as mandatory for the Surry ISFSI system. The discussion of each topic should include the basis for any requirement.

Spent Fuel Storage Environment

Identify the required storage environment for the spent fuel inside the cask and the range of external conditions under which the environment must be maintained. This could include the following characteristics:

- Maximum Clad Temperature
- Cask Internal Pressure
- Storage Atmosphere and its Allowable Impurities (e.g., moisture)
- Corrosion Protection
- Fuel Element Spacing, Support and Protection

Physical Constraints

These are the limits placed on the physical characteristics of the cask due to pre-established ISFSI interfaces:

- Size limits due to at-reactor handling and decontamination facilities;
- Weight limits due to at-reactor crane constraints and design parameters of onsite transporter, roadways, storage pad and placement crane;
- Cask appendages required to match any existing cask handling; monitoring, and servicing hardware and subsystems.

Material Considerations

In order to assure material compatibility with its use and environment, consider the following topics:

- Limits on degradation due to radiation damage, weathering, and temperature extremes;
- Corrosion resistance and compatibility with fuel element materials; and
- Qualification of uncoded materials.

Mechanical Requirements

These following features are required for the proper operation of the casks:

- Support structure design constraints to preclude spent fuel loading damage, control criticality, and to assure post-storage and recovery operation fuel element removal;
- Cask appendages required to interface with cask handling, monitoring, servicing, closure, testing, onsite transport, and placement hardware and subsystem design;
- Cask lid closure and sealing requirements to control releases.

Structural Requirements

The following requirements identify the forces the cask must be able to resist:

- Identify all individual site and system related environmental and operational loads to be considered in the cask's structural analysis. (These loads should be quantified, or identified in such a manner that they can be readily quantified by the cask designer.)
- Specify the required variance for each load to be considered in the structural analysis.
- For each different operational or environmental condition to be analyzed, provide the applicable combined loading equations with the site and system defined loads inserted.
- List operational and environmental conditions not considered in the loading equations and the ISFSI design feature that precludes their consideration. (For example: "Impact loads due to casks overturning during the design earthquake are not considered. The casks are placed on a Seismic Category 1 concrete pad, and the following analysis shows that casks of a L/D ratio of less than 4 to 1 will not tip over due to the motion of the concrete pad during by the design earthquake. All cask having a L/D ratio in excess of 4 to 1 will be anchored to the concrete pad to prevent overturning due to the design earthquake." The mentioned analysis is then presented.)
- Specify any special structural code requirements.

Thermal Characteristics

Thermal performance specification for individual casks may include:

- Amount of heat to be used dissipated to the atmosphere under limiting operational and environmental conditions;
- The required heat capacity available to accommodate thermal fluctuations beyond the limiting conditions, during transient operational modes, and for accident recovery operations; and
- Allowable peak temperatures within the cask.

Nuclear and Radiological Characteristics

Provide the limiting specification for:

- Neutron and gamma shield requirements based on ALARA studies for all operational modes;
- Allowable leak rates for gaseous and particulate material; and
- Criticality considerations.

Special Features

Provide the requirements for special features associated with the cask. These could include:

- Cask Atmosphere Purging Appendages
- Monitoring Instrumentation for Stored Spent Fuel Condition, Cask Internal Atmosphere, and Cask Condition
- Material Accountability Seals
- Decontamination and Recovery Features
- Lightning Protection

Response

The information requested by this question is contained in the GNSI Topical Report(1).

Reference

1. *Topical Safety Analysis Report for the CASTOR V/21 Cask Independent Spent Fuel Storage Installation (Dry Storage),* GNSI, January 1985.

Question 3.1.2

The various design criteria and performance specification identified in response to Question 3.1.1 will be reflected in actual design and manufacture of the storage casks. For each requirement identified in the response to Question 3.1.1, specify the actions that VEPCO will take to assure that the requirement is properly executed by the cask supplier. These actions could include: Specific quality control requirements in a quality assurance program, code usage and stamping, analytical verifications, acceptance tests, and prototype testing. Keep in mind that a performance characteristic or a design requirement need not be verified directly, it can be verified by qualifying a related characteristic or feature that is a good indicator for the basic requirement.

Response

Surry ISFSI SAR Section 11.1 has been revised to reflect the response to this question.

Question 3.2.1

Provide a detailed analysis of the stresses in stored spent fuel elements due to the forces caused by a cask tip over during the design earthquake. This analysis should include the basis for all assumptions and the factor of safety between the resultant stresses and the threshold stress for fuel damage. The verifiable post-reactor condition of the fuel should be quantified. That is, what allowance is made for the difference in structural integrity between irradiated and unirradiated fuel. How is this difference verified in terms of accepting individual spent fuel elements for storage in a cask.

Response

See response to Question 1.4. 1.

Question 3.2.2

In lieu of responding to Question 3.2.1, or if the resultant factor of safety identified from the analysis is too low; the concrete storage pad can be upgraded to a Seismic Category 1 item. If this is done, provide an appropriate structural analysis of the concrete slab and its supporting soil.

Response

See response to Question 1.4.1.

Question 3.3.1

What are the physical devices used on the handling equipment to limit impact loads during normal and off-normal operating conditions?

Response

The response to this question has been incorporated into Surry ISFSI SAR Section 5.2.

Question 3.3.2

What are the physical devices used to prevent lifts in excess of these specified for handling equipment?

Response

The response to this question has been incorporated into Surry ISFSI SAR Section 5.2.

Question 3.3.3

Provide the details of requalification activity for the cask and spent fuel following an off-normal handling accident.

Response

ISFSI SAR Section 8.2.10 has been revised to reflect the response to this question.

Question 4.1.1E

What was the neutron spectrum and flux-to-dose response used to estimate the neutron surface dose rate and dose vs. distance (Figures 3.5-2 and 3.5-3) for a cask?

Response

Surry ISFSI SAR Section 7.3 and Surry ISFSI ER Section 3.5 have been revised to reflect the response to this question.

Question 4.1.2E

What were the assumptions used in postulating the steel-lead-water wall used to shield the fuel for calculating the gamma surface dose rate and energy spectrum?

Response

The GNSI cask utilizes steel and polyethylene instead of steel-lead-water to shield the neutron and gamma radiation. The GNSI Topical Report (Reference 1) provides detailed discussion of the design features as well as the structures of the cask.

The assumptions used in calculating the cask surface dose rates and energy spectrum are discussed in Sections 3.3.5 and 7 of the GNSI Topical Report. A discussion of the calculation of neutron and gamma dose rates has been provided in revised Surry ISFSI SAR Section 7.3 and revised Surry ISFSI Section 3.5.

REFERENCE

1. *Topical Safety Analysis Report for the CASTOR V/21 Cask Independent Spent Fuel Storage Installation (Dry Storage),* General Nuclear Systems, Incorporated, January 1985.

Question 4.1.3E

What computer code was used to generate the data in Tables 3.5-1 and 3.5-2 of the ER and Tables 7.2-1, 7.2-2, 7.2-3 and 7.2-4 of the SAR?

Response

The response to this question has been incorporated in Surry ISFSI SAR Section 7.2 and Surry ISFSI ER Section 3.5.

Question 4.1.4

How will vendors demonstrate compliance with the cask surface dose rate criteria? What other computer codes or calculational techniques will be acceptable to VEPCO to demonstrate compliance? What are the key input parameters needed to demonstrate compliance?

Response

Sections 3.3.5 and 7 of the GNSI Topical Report (Reference 1) provide a discussion of the standard computer codes and calculational techniques used in determining the GNSI cask surface dose rates. The use of standard, industry-accepted computer codes is acceptable to Virginia Power for demonstrating compliance of the SSSCs with the design criteria. The key input data to be used in the calculations are the spent fuel design information (source term data) and the cask design information. Westinghouse source data, representative of the fuel to be stored in the SSSCs, have been utilized in the shielding calculations.

REFERENCE

1. *Topical Safety Analysis Report for the CASTOR V/21 Cask Independent Spent Fuel Storage Installation (Dry Storage),* General Nuclear Systems, Incorporated, January 1985.

Question 4.2.1E

Section 7.3.2 of the SAR says, "Except during cask placement and scheduled surveillance the ISFSI will not be normally occupied." What about operations at the LLWSF collocated with the ISFSI? No occupational exposure from the ISFSI to workers involved in operations at the LLWSF have been provided. Provide estimated occupational exposures to workers at the LLWSF from the ISFSI and the basis for your calculations.

Response

The response to this question has been incorporated into Surry ISFSI SAR Section 7.4 and ER Section 4.4.

Question 4.2.2

Provide an ALARA justification for collocating the LLWSF and the ISFSI.

Response

The response to this question has been incorporated into Surry ISFSI SAR Section 7.1.

Question 4.2.3E

Section 7.3.3.2 of the SAR indicates that the dose rate analysis at the restricted area fence does not include the contribution from the LLWSF. Revise Table 4.4-1 of the ER and Table 7.3-1 of the SAR to include the contribution from the LLWSF.

Response

Surry ISFSI SAR Section 7.3 and ER Section 4.4 have been revised to include the contribution from the LLWSF.

Question 4.2.4E

In Section 7.4 of the SAR, no total collective occupational dose is provided. How many workers at the Surry Power Station will receive the additional 56 mrem yr from the ISFSI? What is the additional occupational dose to workers at the LLWSF from the ISFSI? What are the bases for the dose rate estimates used in SAR Tables 7.4-1, 7.4-2 and 7.4-3 (ER Tables 4.4-2, 4.4-3 and 4.4-4)?

In SAR Table 7.4-3 (ER Table 4.4-4), what is the occupational dose due to excavation and construction? Provide a total collective occupational dose for ISFSI operations and assess how it affects the collective occupational dose at the Surry Power Station.

Response

The response to this question has been incorporated into Surry ISFSI SAR Section 7.4 and ER Section 4.4.

Question 4.3.1E

SAR Section 7.6 is titled "Estimated Offsite Collective Dose Assessment," yet no collective offsite dose is evaluated. Only a maximum dose to an individual located at 1.53 miles is given. Figure 2.1-3 in the SAR indicates that 3 people reside within the 0-1 mile annulus and 49 people in the 1-2 mile annulus of the Surry Power Station. Explain the discrepancy about the location of the nearest individual and provide an offsite collective dose assessment.

Response

Surry ISFSI SAR Figure 2.1-3 has been revised and a collective offsite dose has been calculated accordingly and is reported in revised SAR Section 7.6 and ER Section 5.2.

Question 4.3.2E

Relative to SAR Section 7.6.1: the principle contributor to dose from the ISFSI will be neutrons, and the information about environmental monitoring does not provide enough information about capabilities for measuring doses from neutrons around the restricted area fence. Describe the type, number and locations of the TLD's to be placed around the ISFSI restricted area fence.

Response

The response to this question has been incorporated into Surry ISFSI SAR Section 7.6 and ER Section 6.2.

Question 4.3.3E

Relative to SAR Section 7.6.2: If the maximum dose to an individual from ISFSI operations were added to the design objective doses specified in 10 CFR Part 50, Appendix I, for releases of radioactive material from reactor operation, the limits of 10 CFR 72.67 and 40 CFR Part 190 could be exceeded. Considering the uncertainty in calculating the neutron dose rate at distances of 1.5 miles, and the difficulty of measuring such low doses from neutrons; how will you ensure that the dose to an individual from ISFSI operations when combined with doses from reactor operations does not exceed these limits? Provide your method of demonstrating, by calculational procedures based on models and data, that the actual dose of an individual from ISFSI operations when combined with the doses from Surry Power Station Units 1 and 2 and other Uranium Fuel Cycle facilities does not exceed the 25 mrem/yr limit specified in 10 CFR 72.67 and 40 CFR Part 190.

Response

The response to this question has been incorporated into Surry ISFSI SAR Section 7.6 and ER Section 5.2.

Question 4.4.1

The procedures for the decommissioning of the ISFSI should be addressed conceptually for the purposes of demonstrating that it is a manageable task. To the extent possible, Vepco should identify: specific levels of contamination and activation products expected at the end of useful cask life, the specific procedures anticipated for clean-up of the cask, and the expected disposition of the cask.

Response

Surry ISFSI SAR Section 9.6 and ER Section 5.8 have been revised to reflect the response to this question.

Question 4.4.2E

Provide the basis and supporting analysis for your conclusion in ER Section 5.8 that "the cask materials will be only very slightly activated as a result of their long-term exposures to the relatively small neutron flux..."

Response

Surry ISFSI SAR Section 9.6 and ER Section 5.8 have been revised to reflect the response to this question.

Question 4.5.1E

Other radiation from the uranium fuel cycle is included in 10 CFR 2.67(a)(3). Provide an assessment of the combined effects of the ISFSI and the Surry Power Station.

Response

The response to this question has been incorporated into Surry ISFSI SAR Section 7.6 and ER Section 5.2.

Question 4.6.1E

More information is needed about your capabilities for assessing neutron dose in the environment.

Response

The response to this question has been incorporated into Surry ISFSI SAR Section 7.6 and ER Section 6.2.

Question 4.6.2

Provide a complete description of any radiation monitoring program to be established for the ISFSI related activities. This discussion should address the following issues as appropriate:

- Location and type of instrumentation for ISFSI storage area radiation monitoring;
- Type and location of instrumentation for individual cask radiation monitoring;
- Monitoring schedule; and
- High radiation alarm system.

Response

The response to this question has been incorporated into Surry ISFSI SAR Section 7.6 and ER Section 6.2.

Question 4.7.1

The radiation protection procedures related to a high radiation situation should discuss the following topics:

- Safety precautions
- Personnel required (by skilled/specialty level)
- Repair equipment/material required
- Provisions for workers protection
- Provisions for protection of other personnel

Response

Surry ISFSI SAR Section 9.4 has been revised to reflect the response to this question.

Question 5.1.1E

Report separately the capital cost, and operation and maintenance cost components of the total lifetime cost reported in Section 9.1.3 and 9.1.4.

Response

Surry ISFSI ER Section 9.1 has been revised to reflect the response to this question.

Question 5.2.1.

What special provisions will be added to your Quality Assurance Program to accommodate the ISFSI activity?

NOTE: During the development of the specific information for ISFSI Quality Assurance Program, it is suggested that you review your existing program for attributes described in the attachment "QA Checklist for Dry Storage Casks".

Response

Surry ISFSI SAR Section 11.1 has been revised to reflect the response to this question

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Question 5.2.2

What special provisions will be added to your Emergency Plan to accommodate the ISFSI activity?

NOTE: It is suggested that an adjunct to the existing Emergency Plan adds the following events:

For the Notification of Unusual Events Category,

•Loss of Cask Neutron Shield

•Cask Seal Leakage

•Cask Drop or Other Handling Mishap

For the Alert Category,

•Loss of All Fuel Confinement Barriers From Some Undefined Cause

Response

Surry ISFSI SAR Section 9.5 has been revised to reflect the response to this question.

Question 5.2.1.

What special provisions will be added to your Quality Assurance Program to accommodate the ISFSI activity?

NOTE: During the development of the specific information for ISFSI Quality Assurance Program, it is suggested that you review your existing program for attributes described in the attachment "QA Checklist for Dry Storage Casks".

Response

Surry ISFSI SAR Section 11.1 has been revised to reflect the response to this question

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Question dated October 1, 1984

The proposed revision to Section 2.6.4.8, "Liquefaction Potential," of the Surry ISFSI Safety Analysis Report presents the procedure, including mathematical relations, used to predict soil liquefaction potential at various depths below the proposed ISFSI. This procedure consists of evaluating the shear stresses expected during the design earthquake loading. It is stated in Subsections 2.6.4.8.2 and 2.6.4.8.3 that the calculated factors of safety are 2.5 and 1.5, respectively, for the Pleistocene Sand and Miocene Silty Sand layers.

We understand that these liquefaction potential calculations were performed at a variety of depths below the water table. Please provide the calculational results for the range of depths examined in the two sand layers mentioned above. Please specify what specific parameters (e.g., effective overburden pressures, standard penetration field values, etc.) were used in making the above calculations. A tabular presentation form, which clearly identifies the values of the various parameters used in the calculation, as well as the calculational result, would be very useful.

Response

The response to this question has been incorporated into SAR Section 2.6.

Question 1 (November 14, 1984)

Regarding your response to Question 4.1.1E, what are the neutron energy flux response functions that were used to calculate the neutron dose rate versus distance from the cask flux leakage spectra supplied by GNS?

Response

The response to this question has been incorporated into Surry ISFSI SAR Sections 7.3 and 7.4 and Surry ISFSI ER Sections 3.5 and 4.4.

Question 2 (November 14, 1984)

In your response to Questions 4.2.4E and 4.3.1E, neutron and gamma dose rates were cited for distances greater than shown in ER Figure 3.5-2 and SAR Figure 7.3-4. Please provide additional figures showing neutron and gamma dose rates, from one cask of five-year-old spent fuel, versus distance out to the nearest permanent resident (1.53 mi). ER Figure 3.5-4 and SAR Figure 7.3-5 should also show the neutron and gamma components of the total dose rate.

Response

Surry ISFSI SAR Figures 7.3-5 and 7.3-6 and ER Figures 3.5-4 and 3.5-5 have been added to reflect the requested distance.

Question 3 (November 14, 1984)

Please provide revised figure showing the neutron and gamma dose rate at the cask surface versus time that was used to calculate dose rates from a full 84-cask configuration, adjusted for decay.

Response

Surry ISFSI ER Figure 3.5-3 and Surry ISFSI SAR Figure 7.3-2 have been revised to show the normalized GNSI cask surface dose rates versus time.

Question 4 (November 14, 1984)

Your response to Questions 4.2.1E and 4.2.2 discusses occupational exposures received at the LLWSF due to the ISFSI. In order to better assess the impact the ISFSI has on the LLWSF occupational exposures, please provide an estimate of the occupational exposure at the LLWSF without the collocated ISFSI.

Response

The response to this question has been incorporated into Surry ISFSI SAR Section 7.4 and ER Section 4.4.

Question 5 (November 14, 1984)

In your response to Question 4.2.3E, Table 4.4.1 does not include the dose rate contribution from the LLWSF as indicated in the accompanying narrative. Please revise this table to include the dose rate contribution from the LLWSF assuming that it was filled to design capacity.

Response

Surry ISFSI ER Table 4.4-l and SAR Table 7.3-l have been revised to include the dose rate contribution from the LLWSF assuming that it was filled to design capacity.

Question 6 (November 14, 1984)

In reviewing the ER, SAR and your response to Question 4.2.3E, some confusion has arisen about your use of the term "restricted area". Please review the use of this term as applied to the ISFSI to ensure that it is consistent with health physics practices at the Surry Power Station.

Response

The Surry Updated FSAR Figure 2.1-12 illustrates the existing Surry Power Station restricted area boundary. The restricted area boundary for the ISFSI will be identical with the one shown on the figure. The Surry ISFSI SAR and ER have been revised accordingly. Designation of this boundary for the ISFSI is not in conflict with health physics practices at the Surry Power Station.

Question 7 (November 14, 1984)

The License Application (Chapters 1 and 3) indicate 82 casks at the ISFSI. Yet, the ER (Chapter 3) and the SAR (Chapters 4 and 7) imply 84 casks at the site. Please clarify.

Response

The response to this question has been incorporated into the License Application, ISFSI SAR Section 3.1 and ISFSI ER Section 3.5.

Appendix A
SSSC Specific Information

Appendix A
SSSC SPECIFIC INFORMATION

This appendix provides:

- A list of topical reports issued by cask manufacturers (Table A/1.5-1)
- A subappendix for each cask type that provides specific references to the SSSC topical reports (in tables) and specific information not contained in the SSSC topical reports

Table A/1.5-1
TOPICAL SAFETY ANALYSIS REPORTS ISSUED
BY CASK MANUFACTURERS

A.1 *Topical Safety Analysis Report for the CASTOR V/21 Cask Independent Spent Fuel Storage Installation (Dry Storage),* Revision 2A, General Nuclear Systems, Inc., June 1987.

A.2 *Topical Safety Analysis Report for the Westinghouse MC-10 Cask for an Independent Spent Fuel Storage Installation (Dry Storage),* Revision 2A, Westinghouse Nuclear Energy Systems, November 1987.

A.3 *Topical Safety Analysis Report for the NAC Storage/Transport Cask Containing 28 Intact Fuel Assemblies for Use at an Independent Spent Fuel Storage Installation,* Revision 1A, Nuclear Assurance Corporation, June 1990.

A.4 *Topical Safety Analysis Report for the CASTOR X Cask for an Independent Spent Fuel Storage Installation (Dry Storage),* Revision 4, General Nuclear Systems, Inc., September 1990.

A.5 *TN-32 Dry Storage Cask Topical Safety Report,* Revision 9A, Transnuclear, Inc., December 1996.

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