**CHAPTER 7**

**RADIATION PROTECTION**

**LIST OF EFFECTIVE PAGES**

| PAGE | REVISION | PAGE | REVISION |
|---|---|---|---|
| LEP 7-1 | 22 | | |
| 7-i | 15 | | |
| 7-ii | 15 | | |
| 7-iii | 20 | | |
| 7-iv | 22 | | |
| 7.1-1 | 8 | | |
| 7.1-2 | 15 | | |
| 7.1-3 | 15 | | |
| 7.1-4 | 15 | | |
| 7.2-1 | 15 | | |
| 7.2-2 | 22 | | |
| 7.2-3 | 22 | | |
| 7.3-1 | 15 | | |
| 7.3-2 | 12 | | |
| 7.3-3 | 15 | | |
| 7.4-1 | 20 | | |
| 7.4-2 | 22 | | |
| 7.4-3 | 22 | | |
| 7.5-1 | 8 | | |
| 7.6-1 | 8 | | |
| 7.7-1 | 12 | | |
| 7.7-2 | 20 | | |
| Figure 7.2-1 | 20 | | |
| Figure 7.4-1 | 20 | | |
| Figure 7.4-2 | 20 | | |

CHAPTER 7

RADIATION PROTECTION

TABLE OF CONTENTS

**CHAPTER 7**

**RADIATION PROTECTION**

**LIST OF TABLES**

## CHAPTER 7

## RADIATION PROTECTION

## LIST OF FIGURES

**CHAPTER 7**

**RADIATION PROTECTION**

**LIST OF ACRONYMS**

| | |
|---|---|
| ALARA | As Low As Reasonably Achievable |
| BGE | Baltimore Gas and Electric Company |
| CCNPP | Calvert Cliffs Nuclear Power Plant |
| CFR | Code of Federal Regulations |
| DSC | Dry Shielded Canister |
| GS-RS | General Supervisor-Radiation Safety |
| HSM | Horizontal Storage Module |
| HSM-HB | High Burnup Horizontal Storage Module |
| ISFSI | Independent Spent Fuel Storage Installation |
| NUHOMS | Nutech Horizontal Modular Storage® |

## 7.0 RADIATION PROTECTION

## 7.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS REASONABLY ACHIEVABLE

### 7.1.1 POLICY CONSIDERATIONS

Baltimore Gas and Electric Company (BGE) Radiation Safety and As Low As Reasonably Achievable (ALARA) policies are described in the Quality Assurance Manual for Nuclear Power Plants, the Calvert Cliffs Radiation Safety Manual, and the ALARA Program. These policies are applied to the Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI). Baltimore Gas and Electric Company is committed to a strong ALARA program in the design and operation of its nuclear facilities. The ALARA program follows the general guidelines of Regulatory Guides 1.8, 8.8, 8.10 and Title 10, Code of Federal Regulations (CFR) Part 20. Plant and design personnel are trained and updated on ALARA practices and dose reduction techniques. Design and implementation of systems and equipment are reviewed to ensure ALARA exposure on all new and modification projects. The basic ALARA program consists of:

A. The Calvert Cliffs Radiation Safety Manual, ALARA Program, and Radiation Safety (Implementation) Procedures.

B. Continued surveillance and evaluation of in-plant radiation and contamination conditions, as well as the monitoring and control of the exposure of personnel by radiation safety professionals and technical personnel.

C. The Radiation Safety-ALARA Unit comprised of radiation safety technical personnel, whose primary function is to perform ALARA reviews of operations, maintenance, and modifications.

D. Responsible Engineers who are responsible to assure ALARA considerations are accounted for in the design process.

Although upper level management is vested with the primary responsibility and authority for administering the Calvert Cliffs ALARA program, the responsibility for ALARA is extended through lower management to the individual employee and contractor.

The Vice President-Nuclear Energy Division has overall responsibility for all health and safety matters.

The Plant General Manager-Calvert Cliffs Nuclear Power Plant (CCNPP) Department is responsible for the protection of all persons against radiation and for compliance with Nuclear Regulatory Commission regulations and license conditions. This responsibility is, in turn, shared by all supervisors. Furthermore, all personnel are required to work safely and follow the regulations, rules, policies, and procedures that have been established for their protection.

The General Supervisor-Radiation Safety (GS-RS) is responsible for the administering and reviewing of the Calvert Cliffs ALARA program at the staff level. The GS-RS has direct access to the Vice President-Nuclear Energy Division on all matters vital to the radiation protection and ALARA programs.

The Supervisor of Radiation Safety-ALARA, who reports to the Assistant General Supervisor of Radiation Safety, is responsible for the ALARA program at the implementation level.

The Health Physics Consultant-Radiation Protection Manager establishes the Radiation Safety Program, including the program for handling and monitoring radioactive material for Calvert Cliffs, that is designed to assure compliance with applicable regulations, technical specifications, and regulatory guides. He also provides technical guidance and support for conducting this program, reviews the effectiveness and the results of the program and modifies it as required, based on experience and regulatory changes, to assure that occupational radiation exposures and exposure to the general public are maintained ALARA.

The GS-RS is responsible for conducting the radiation protection program for CCNPP, including the ISFSI. The GS-RS has the responsibility and authority to:

A. measure and control the radiation exposure of personnel
B. evaluate and review the radiological status of the plant
C. make recommendations for control or elimination of radiological hazards
D. assure that all personnel are trained in radiation protection
E. assist all personnel in carrying out their radiation responsibilities
F. protect the health and safety of personnel both on-site and in the surrounding area

To fulfill these responsibilities, radiological monitoring, survey, and personnel exposure control activities are performed on a continuing basis for plant operations and maintenance, including the ISFSI.

### 7.1.2 DESIGN CONSIDERATIONS – NUHOMS-24P

Section 12.7 contains a description of the radiation protection design features associated with the use of Nutech Horizontal Modular Storage® (NUHOMS)-32P dry shielded canisters (DSCs). The design considerations which ensure that occupational exposures for the NUHOMS-24P ISFSI are ALARA are discussed in Reference 7.1. The following paragraphs, which are numbered to correspond with Section 7.1.2 of Reference 7.1, discuss differences in the Calvert Cliffs implementation of the NUHOMS-24P generic design which affect the shielding design considerations.

1-7. Same as Reference 7.1.

8. The water used to fill the DSC cavity prior to immersion in the spent fuel pool will be borated. The shielding analyses were performed assuming that pure water was used to fill the cavity. The impact on the shielding calculation results is negligible (Reference 7.11).

   The cavity of the DSC will be submerged in the spent fuel pool for about 12 hours and, on removal from the pool, will contain borated water from the spent fuel pool for less than 50 hours. There is a substantial body of industry experience with exposure of austenitic stainless steels to borated water since that condition exists in most pressurized water reactor spent fuel pools. A literature search did not reveal any journal articles referring to corrosion of austenitic stainless in pools, from which one could infer that none has been observed since anecdotal experiences have all been very good. A Combustion Engineering, Inc. study on the effects of borated water on corrosion of low alloy steels reports a complete absence of corrosion in a very aggressive environment (dripping borated water and wet borated steam) for Type 304 and other corrosion resistant materials. The author concludes that "…corrosion resistant alloys such as … Types 304 … are not susceptible to borated water corrosion. Furthermore, stressed samples of these materials did not exhibit any localized forms of corrosion, such as stress corrosion cracking, hydrogen embrittlement, etc." (Reference 7.12)

After the DSC cavity has been drained, about 2 to 4 gallons of residual borated water will remain due to surface tension. As the borated water evaporates during the vacuum drying process, the ortho-boric acid crystals will precipitate out of the solution at concentrations substantially higher then 2000 ppm. After the free water has evaporated, a small amount of dehydrated $B_2O_3$ crystals will remain in the cavity. A literature search did not uncover any reference to corrosive or aggressive behavior of anhydrous boric acid and discussions with chemists at an industry supplier (U.S. Borax) revealed that it will only become corrosive in the molten state. The melting temperature for anhydrous boric acid is about 450°C, well above the peak DSC material temperatures and the peak fuel cladding temperatures. Even if corrosion of the DSC shell or basket is postulated, the extremely small quantity of borate and the expected corrosion mechanism of general surface oxidation is not likely to lead to degradation of the structural integrity of the DSC shell or basket.

9-13. Same as Reference 7.1.

14. Same as Reference 7.1 except that the shielding calculations were performed assuming that water would be present in the annular gap when the DSC is flooded, and that the annular gap would be drained right before the transfer cask cover plate is installed.

15. Same as Reference 7.1.

During the design phase of the Calvert Cliffs NUHOMS ISFSI, the NUHOMS-07P demonstration project was successfully completed at the H. B. Robinson plant. Comparison between predicted dose rates and those measured during the first fuel load at Robinson confirmed that the ALARA design considerations employed in the NUHOMS ISFSI design are sound and effective. The NUHOMS design incorporates certain improvements in the design and analysis of the radiation shielding as compared to the NUHOMS-07P system (References 7.1 and 7.2). Furthermore, lower design criteria for the horizontal storage module (HSM) average surface dose rates have been specified for the Calvert Cliffs ISFSI. Successful demonstration of the NUHOMS-07P system, design improvements in the generic NUHOMS-24P system, and lower design criteria for the HSM surface dose rates all ensure that the Calvert Cliffs ISFSI shielding design and occupational radiation exposures will be ALARA.

### 7.1.3 OPERATIONAL CONSIDERATIONS

Consistent with BGE's overall commitment to keep occupational radiation exposures ALARA, specific plans and procedures are followed by plant personnel to ensure that ALARA goals are achieved. Operational ALARA policy statements are formulated for the Nuclear Energy Division through the issuance of the Quality Assurance Manual for

Nuclear Power Plants, the Radiation Safety Manual, and the ALARA Program and are implemented by means of procedures. These statements and procedures are consistent with the intent of Section C.1 of Regulatory Guides 8.8 and 8.10.

Since the ISFSI is a passive system, no maintenance is expected on a normal basis in the facility. Maintenance operations on the transfer cask, transfer trailer and other ancillary equipment is performed in a very low-dose environment when fuel movement is not occurring. Planned maintenance activities are preventive in nature and include motor oil changes, hydraulic oil filter replacement and the like.

## 7.2 RADIATION SOURCES – NUHOMS-24P

Section 12.7 discusses the radiation sources associated with the use of NUHOMS-32P DSCs.

### 7.2.1 CHARACTERIZATION OF SOURCES

The source terms used in the shielding analyses for the Calvert Cliffs ISFSI were developed in the same manner as for the NUHOMS-24P Topical Report (Reference 7.1). The radiological source terms were calculated, using ORIGEN2, for the range of initial enrichments and burnups given in Table 9.4-1. The source terms were calculated with cooling times for each assembly corresponding to a heat output of 0.66 kW. The fuel assembly with the largest source terms (both neutron and gamma) was found to be a 3.4 w/o initial enrichment, 42,000 MWD/MTU burnup, cooled for 8 years. These source strengths were used for shielding design throughout the ISFSI. The active fuel region neutron and gamma energy spectra for this reference fuel assembly is given in Table 7.2-1.

The source modeling methodology is similar to the NUHOMS-24P generic design methodology and is fully described in References 7.14 (active fuel region) and 7.16 (upper and lower end fittings and plenum region). This source term methodology has also been demonstrated to produce conservative dose rate results compared to actual Calvert Cliffs ISFSI benchmarks, when coupled with the shielding methodology discussed in Section 7.3.2. No other significant radiological sources (such as storage containers or tanks) are located in the vicinity of the ISFSI. Figure 7.2-1 is a bounding radiological limit curve for assemblies at or below a thermal power of 660 watts and have cooled at least 9 years. Assemblies meeting the requirements of both Table 9.4-1 and Figure 7.2-1 may be loaded and stored in the ISFSI.

### 7.2.2 AIRBORNE RADIOACTIVE MATERIAL SOURCES

The release of airborne radioactive material is postulated for three phases of system operation: fuel handling in the spent fuel pool, drying and sealing of the DSC, and DSC transfer and storage.

Potential airborne releases from irradiated fuel assemblies in the pool are discussed in Reference 7.3.

Dry shielded canister drying and sealing operations are performed using procedures which prohibit airborne leakage. During these operations, all vent lines are routed to the Auxiliary Building's existing radwaste systems. Once the DSC is dried and sealed, there are no design basis accidents which could result in a breach of the DSC and the airborne release of radioactivity.

During transfer of the sealed DSC and subsequent storage in the HSM, the only postulated mechanism for the release of airborne radioactive material is the dispersion of non-fixed surface contamination on the DSC exterior. By filling the cask/DSC annulus with demineralized water, placing a mechanical seal over the annulus, and utilizing procedures which require examination of the annulus surfaces for smearable contamination, the contamination limits on the DSC can be kept below the permissible level for off-site shipments of fuel. Therefore, there is no possibility of significant radionuclide release from the DSC exterior surface during transfer or storage.

## TABLE 7.2-1
## NEUTRON ENERGY SPECTRUM

| GROUP NUMBER | UPPER ENERGY (MeV) | GROUP FRACTION |
|---|---|---|
| 1 | 1.50E+01 | 4.65E-04 |
| 2 | 1.22E+01 | 1.88E-03 |
| 3 | 1.00E+01 | 5.76E-03 |
| 4 | 8.18E+00 | 1.92E-02 |
| 5 | 6.36E+00 | 4.00E-02 |
| 6 | 4.96E+00 | 5.17E-02 |
| 7 | 4.06E+00 | 1.09E-01 |
| 8 | 3.01E+00 | 8.80E-02 |
| 9 | 2.46E+00 | 2.09E-02 |
| 10 | 2.35E+00 | 1.16E-01 |
| 11 | 1.83E+00 | 2.09E-01 |
| 12 | 1.11E+00 | 1.92E-01 |
| 13 | 5.50E-01 | 1.33E-01 |
| 14 | 1.11E-01 | 1.35E-02 |
| 15 | 3.35E-03 | 0.00E+00 |
| 16 | 5.33E-04 | 0.00E+00 |
| 17 | 1.01E-04 | 0.00E+00 |
| 18 | 2.90E-05 | 0.00E+00 |
| 19 | 1.07E-05 | 0.00E+00 |
| 20 | 3.06E-06 | 0.00E+00 |
| 21 | 1.12E-06 | 0.00E+00 |
| 22 | 4.14E-07 | 0.00E+00 |
| | | 1.00E+00 |

For more information see Reference 7.11.

**TABLE 7.2-2**
**GAMMA ENERGY SPECTRUM**

| **GROUP NUMBER** | **UPPER ENERGY (MeV)** | **GROUP FRACTION** |
|---|---|---|
| 1 | 1.00E+01 | 2.99E-11 |
| 2 | 8.00E+00 | 2.60E-10 |
| 3 | 6.50E+00 | 2.26E-09 |
| 4 | 5.00E+00 | 0.00E+00 |
| 5 | 4.00E+00 | 2.41E-07 |
| 6 | 3.00E+00 | 1.88E-06 |
| 7 | 2.50E+00 | 3.28E-05 |
| 8 | 2.00E+00 | 5.16E-04 |
| 9 | 1.66E+00 | 0.00E+00 |
| 10 | 1.33E+00 | 2.93E-02 |
| 11 | 1.00E+00 | 5.94E-02 |
| 12 | 8.00E-01 | 0.00E+00 |
| 13 | 6.00E-01 | 4.29E-01 |
| 14 | 4.00E-01 | 1.20E-02 |
| 15 | 3.00E-01 | 2.29E-02 |
| 16 | 2.00E-01 | 2.91E-02 |
| 17 | 1.00E-01 | 7.39E-02 |
| 18 | 5.00E-02 | 3.44E-01 |
| | | 1.00E+00 |

For more information see Reference 7.11.

## 7.3 RADIATION PROTECTION DESIGN FEATURES – NUHOMS-24P

### 7.3.1 INSTALLATION DESIGN FEATURES

The Calvert Cliffs ISFSI design features are discussed in detail in the NUHOMS-24P Topical Report (Reference 7.1).

### 7.3.2 SHIELDING

Reference 7.1 contains a complete description and illustration of the shielding design for the Calvert Cliffs ISFSI, with the exception of the shielding on the automatic welding machine, which is described in Reference 7.19. The operational shielding analyses performed in support of the Calvert Cliffs ISFSI design utilized the MCNP code. The storage term shielding analyses are identical in form to the calculations performed in support of the generic NUHOMS-24P design. References 7.1 and 7.15 through 7.17 contain a complete description of those shielding methodologies and models. References 7.4 through 7.9 and 7.18 are the shielding computer program packages and cross-section data used in the analyses.

As indicated above, the MCNP code was utilized to calculate operational dose rates at the locations of interest. The MCNP code is a general-purpose Monte-Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/ photon/electron transport, including the capability to calculate $k_{eff}$ for critical systems. A benchmark of the MCNP models for the CCNPP ISFSI produced results that matched the actual ISFSI surveys or exceeded them by an average factor of 1.8.

The results of the shielding analyses are presented in Table 7.3-1. The CCNPP HSM doorway dose rate (3240 mrem/hr) is higher than the corresponding generic NUHOMS-24P dose rate (760 mrem/hr) due to design differences in the fuel to be stored, the DSC shield geometries, and the analysis methodology. The former value was calculated using the benchmarked MCNP model, while the latter value was calculated using the methodology presented in the NUHOMS-24P topical report. A number of different combinations of DSC, transfer cask, and HSM door shield thickness combinations were evaluated and the present design was determined to be acceptable in terms of the operational, regulatory, and ALARA objectives. Note that the only operation which takes place in the HSM doorway radiation field is the seismic restraint installation. The CCNPP NUHOMS seismic restraint design includes refinements which allow the restraint to be more easily handled and quickly placed than the generic design. Since the remainder of the operational and storage term radiological exposure rates are heavily influenced by the HSM door exterior dose rates, an improved HSM door design has been developed for the CCNPP ISFSI which reduces the HSM door centerline dose rate from 77 to 10 mrem/hr. The resulting dose rate is lower than that calculated for the topical report.

### 7.3.3 VENTILATION

The Calvert Cliffs ISFSI ventilation design is described in Section 4.3.1 and in the NUHOMS-24P Topical Report (Reference 7.1). Ventilation during DSC drying operations is described in Section 7.2.2.

### 7.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

The Calvert Cliffs ISFSI is expected to result in very low direct dose rates and zero radionuclide release during all credible phases of operation. In order to assure public and employee safety, the CCNPP environmental monitoring program has been expanded to include Dosimeters of Legal Record, air samplers, and vegetation and soil samples at the ISFSI site. Equipment at existing nearby monitoring sites are also upgraded as necessary to monitor the ISFSI.

**TABLE 7.3-1**
**NUHOMS-24P SHIELDING ANALYSIS RESULTS NOMINAL DOES RATES (MREM/HR)**

| | LOCATION | NEUTRON | GAMMA (PRI + SEC) | TOTAL |
|---|---|---|---|---|
| | NUHOMS-24P DSC in HSM | | | |
| 1. | HSM Wall or Roof | 0.5 | 12 | 12.5 |
| 2. | HSM Air Outlet | 1 | 81 | 82 |
| 3. | Center of Door | 5 | 5 | 10 |
| 4. | Center of Doorway | 621 | 2619 | 3240 |
| 5. | Air Inlet Vent | 1 | 72 | 73 |
| 6. | 1m from HSM Door | 2 | 4 | 6 |
| | NUHOMS-24P DSC in Cask | | | |
| 1. | Centerline [a] DSC Shield Plug (Flooded DSC) | 4 | 76 | 80 |
| 2. | DSC Cover Plate (Dry DSC) | | | |
| | 2.1 Center | 45 | 96 | 141 |
| | 2.2A Edge[b] (Wet Gap) | 80 | 62 | 142 |
| | 2.2B Edge[b] (Dry Gap) | 124 | 136 | 260 |
| 3. | Transfer Cask | | | |
| | 3.1 Side | 69 | 72 | 141 |
| | 3.2 Top | 6.0 | 1.0 | 7.0 |
| | 3.3 Bottom | 56 | 63 | 119 |

For more information see References 7.15 and 7.16.
Dose rates associated with the use of NUHOMS-32P DSCs are discussed in Section 12.7.

[a] The DSC/cask annular gap is filled with water. All but the top 6” of the DSC inner cavity is filled with water.
[b] Nominal at edge of cover plate. The total dose rate is approximately a factor of 3 lower at the top edge of the transfer cask, and several times higher inside the dry annulus.

## 7.4 ESTIMATED ON-SITE COLLECTIVE DOSE ASSESSMENT

### 7.4.1 OPERATIONAL EXPOSURE

This section establishes the expected cumulative dose delivered to operational personnel during the DSC fuel loading, closure, and transfer activities associated with placing one DSC into dry storage in a NUHOMS module. Chapter 5 describes the ISFSI operational procedures, a number of which involve radiation exposure to personnel.

The occupational dose received during fuel loading, closure, and transfer of the DSC at the HSM is estimated utilizing MCNP (Reference 7.18). Details on the modeling techniques used and results are in Reference 7.17.

The estimated occupational exposures for one HSM load are documented in Reference 7.17 and are based on the experience gained during the fuel loads at the Carolina Power and Light Company's H. B. Robinson NUHOMS-07P ISFSI and CCNPP. The estimated occupational exposures have been benchmarked (References 7.17 and 7.20) and demonstrated conservative to actual accumulated total occupational dose measured at fuel loads performed at CCNPP. The results are also consistent with subsequent experience gained during the fuel loads at the Duke Power Company's Oconee NUHOMS-24P ISFSI. The apparent differences between the NUTECH topical report and Calvert Cliffs ISFSI are due largely to the regrouping of the various fuel loading tasks and utilization of different calculational methodology.

The expected dose rates from transfer components are determined by the design of the permanent shielding. The design requirements for the shielding were based on a desire to attain the maximum permanent shielding that was economically practical. The system has been designed to meet the requirements of the ALARA program at Calvert Cliffs (Section 7.1). Since the shielding is fixed, the only credible cause for the operational dose rates to be significantly higher than expected is the misloading of fuel assemblies in the canister. Other operational controls are sufficient to preclude this condition such that additional operational controls on surface dose rates at specific locations during transfer operations are not necessary. Small variations above the expected operational dose rates at specific locations will be alleviated in accordance with normal plant ALARA procedures.

### 7.4.2 STORAGE TERM EXPOSURE

Figures 7.4-1 and 7.4-2 are graphs of the dose rate versus distance from the end and face, respectively, of an array of 120 NUHOMS HSMs (bounds NUHOMS-24P and NUHOMS-32P DSC designs) configured as shown in Figure 1.2-1 and loaded with design basis fuel. The curve was constructed from the shielding analysis described in the previous sections. Direct neutron and gamma flux, as well as the air-scattered radiation from the module surfaces are included. No credit is taken for shielding by existing structures or terrain.

The radiation sources used for the direct and air-scattered dose calculations are in References 7.16 (NUHOMS-24P DSC) and 7.23 (NUHOMS-32P DSC). Direct and air-scattered dose rates are calculated using the computer code MCNP (Reference 7.18). The results depicted in Figures 7.4-1 and 7.4-2 were generated using the bounding dose results at each location from the MCNP models discussed in Sections 7.3 (Reference 7.21) and 12.7.3 (Reference 7.23).

The generic design depicted in Figure 7.4-1 of the NUHOMS-24P topical report cannot be compared directly to the Calvert Cliffs design since it only shows dose versus distance from a single 2x10 array of HSMs. Figures 7.4-1 and 7.4-2 also incorporate the effects of an improved HSM door design which substantially lowers area dose rates.

The ISFSI is surrounded by a large open area for operational and security purposes. Access to the storage modules is restricted such that during storage, no access closer than 50' is allowed except for security and surveillance inspection purposes. There are no work areas close to the ISFSI. Dose to workers at the power plant and other individuals in the unrestricted area due to exposure from the ISFSI is minimal and below regulatory limits (Reference 7.11).

If DSC transfers are performed during Camp Conoy hours, visitor access is restricted and controlled by Calvert Cliffs Nuclear Power Plant, Inc. Security. During the DSC seismic restraint installation, radiation dose rate measurements will be made by Radiation Safety Technicians at selected locations, to supplement environmental surveillance stations and Dosimeters of Legal Record and to validate calculated dose rates.

During the seismic restraint installation, a person standing outside the ISFSI security fence would be exposed to radiation from all closed/loaded HSMs as well as from the HSM where the seismic restraint is being installed. The dose contribution from each of these sources is as follows:

- Site: Figure 2.4-1 shows that the outer fence is 94 feet from the HSMs in the N/S direction, and 53 feet from the HSMs in the E/W direction. From Figures 7.4-1 and 7.4-2, the maximum dose rates at these locations from a site consisting of 120 HSMs loaded with NUHOMS-32P or NUHOMS-24P DSCs is 0.67 mrem/hr in the N/S direction, and 0.73 mrem/hr in the E/W direction.

  These site dose rates also bound storage of a NUHOMS-32P in an high burnup horizontal storage module (HSM-HB) since dose rates at the module surface are generally lower than the HSM (see Table 12.7-1).

- Seismic Restraint Installation: The minimum distance between the DSC surface and an individual at the closest approach to the outer ISFSI security fence (east/west portion of the fence) is estimated to be 66.7'. Note that at 66.7' from the DSC surface an individual at the outer security fence will not be directly in front of the HSM and can only see a fraction of the DSC end surface because it is partially shielded by the cask, HSM front wall, and the HSM door which is raised 2 feet from the closed position during seismic restraint installation. The dose rate at a distance of 66.7' directly in front of a loaded NUHOMS-24P HSM with the door fully open and unobstructed by the transfer cask is conservatively reported as 41 mrem/hr (Reference 7.16). At the same location, unobstructed by the transfer cask, the dose rate for the NUHOMS-32P HSM with the door open 2' for seismic restraint installation is 10.2 mrem/hr (Reference 7.23). The latter is used for this evaluation since it remains conservative relative to the actual configuration. NUHOMS-32P stored in the HSM is bounding for

HSM-HB since dose rates in the open doorway are higher for the HSM than the HSM-HB (see Table 12.7-1).

Experience at Calvert Cliffs and Oconee indicates that seismic restraint installation requires a maximum of 5 minutes (0.08 hours) and can nominally be done in 1 minute (0.017 hours). Thus, a person just outside the fence would conservatively receive a maximum of 0.82 mrem (0.08 hr x 10.2 mrem/hr) during the seismic restraint installation process. If that person remained outside the fence for an entire hour during the seismic restraint installation, their dose would conservatively not exceed 1.55 mrem (0.82 mrem from seismic restraint installation + 0.73 mrem from the fully loaded ISFSI site). This is within the 10 CFR 20.1301 requirement that no individual (member of the public) may receive greater than 2 mrem in any 1 hour.

The air inlet and outlets of the HSM are localized areas compared to the overall HSM surfaces. Therefore, the surface dose rates at these locations are less representative than dose rates at the HSM walls and door for assessing the effect on the direct radiation levels associated with ISFSI operations to individuals located beyond the controlled area while the DSC is in storage. The HSM air inlet vent inspection dose is estimated to be less than 108 mrem/yr for both NUHOMS-24P and NUHOMS-32P DSC designs (Reference 7.22). Normally, remote cameras will be used for air inlet and outlet surveillance. Therefore, no exposure will result from their inspection. If the outlets were manually inspected, an additional exposure of 408 mrem/yr (bounds both NUHOMS-24P and NUHOMS-32P DSC designs) would be incurred (Reference 7.22). These values are derived by assuming that one inspector performs a daily inspection, walking at an average speed of 3 mph on a path passing directly between the north and south outlets on the HSM roof and far away (visual distance) from the front of the air inlet vents of the ISFSI. The distance between the inspector and each HSM front wall is assumed to be half-distance between each row of modules. It is further assumed that all five phases (120 HSMs) are filled with design basis fuel. No credit is taken for radioactive decay of the fuel during storage. Dose rates are based on those given in Figure 7.4-2 (front) and Tables 7.3-1 and 12.7-1 (roof).

## 7.5 HEALTH PHYSICS PROGRAM

### 7.5.1 ORGANIZATION

Since the Calvert Cliffs ISFSI is located adjacent to the CCNPP and within the owner-controlled area, the GS-RS has responsibility for administration of the radiation safety activities at the ISFSI. The administrative organization of the Radiation Safety Program is described in the CCNPP Updated Final Safety Analysis Report (Reference 7.3), Section 11.3, RADIATION SAFETY.

### 7.5.2 EQUIPMENT, INSTRUMENTATION, AND FACILITIES

Radiation Safety Program equipment, instrumentation, and facilities are discussed in the CCNPP Updated Final Safety Analysis Report (Reference 7.3). Procedures and equipment for personnel and decontamination are in place at Calvert Cliffs and are utilized as needed for ISFSI operations.

### 7.5.3 PROCEDURES

Calvert Cliffs Radiation Safety Program Procedures, as described in the Radiation Safety Manual and ALARA Program, are utilized for the Calvert Cliffs ISFSI.

## 7.6 ESTIMATED OFF-SITE COLLECTIVE DOSE ASSESSMENT

### 7.6.1 EFFLUENT AND ENVIRONMENTAL MONITORING PROGRAM

No effluents are released from the ISFSI during operation. Effluents released during DSC loading are treated using existing power plant systems as described in Chapter 6. Since no effluents are released from the Calvert Cliffs ISFSI site, no monitoring program is required.

### 7.6.2 ANALYSIS OF MULTIPLE CONTRIBUTION

An analysis of multiple contributions was performed to determine the additional radiological impact that the ISFSI will impose on the population surrounding the CCNPP site. This impact, added to the contributions made by other uranium fuel cycle operations in the vicinity, was compared to the natural background radiation and the regulatory requirements of 10 CFR 72.104 and 40 CFR Part 190.

The maximally exposed member of the public is assumed to have continuous occupancy in the nearest residence to the ISFSI which is located 4705' from the facility. At that location, the dose rate from 120 HSMs filled to capacity with design basis fuel would be less than 2 mrem/yr from the ISFSI, and less than 13.5 mrem/year from the remaining fuel cycle operations in the vicinity (Reference 7.10). The collective dose due to the ISFSI for persons located within 0-2 miles is conservatively estimated as 28.25 person-mrem spread over 215 people. This is less than 1% of the collective dose from the remaining fuel cycle operations. It can be concluded that the radiation exposure due to the Calvert Cliffs ISFSI, combined with all other fuel cycle operations, will not exceed the regulatory requirements of 25 mrem/year in 10 CFR 72.104 and 40 CFR Part 190.

### 7.6.3 ESTIMATED DOSE EQUIVALENTS

Since no liquid or airborne effluents are postulated to emanate from the ISFSI, the direct and air-scattered radiation exposure discussed in previous chapters comprises the total radiation exposure to the public. No estimation of effluent dose equivalents is necessary.

### 7.6.4 LIQUID RELEASE

No liquids are released from the Calvert Cliffs ISFSI.

## 7.7 REFERENCES


7.1 Topical Report for the Nutech Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS-24P, Nutech Engineers, Inc., NUH-002, Revision 1A, July 1989

7.2 Topical Report on the Nutech Horizontal Modular Storage System for Irradiated Fuel, NUHOMS-07P, Nutech Engineers, Inc., NUH-001, Revision 1A, June 1986

7.3 Calvert Cliffs Nuclear Power Plant, Updated Final Safety Analysis Report, Docket Nos. 50-317 and 50-318, Baltimore Gas and Electric Company

7.4 ANISN/PC, Multigroup One-Dimensional Discrete Ordinates Transport Code System with Anisotropic Scattering, CCC-514 Micro, Oak Ridge National Laboratory, January 1988

7.5 CASK, 40 Group Coupled Neutron and Gamma-Ray Cross-Section Data, DLC-23, Oak Ridge National Laboratory, June 1987

7.6 QAD-CGGP, A Combinatorial Geometry Version of QAD-P5A, A Point Kernel Code System for Neutron and Gamma-Ray Shielding Calculations Using the GP Buildup Factor, CCC-493, Oak Ridge National Laboratory, July 1986

7.7 MicroSkyshine Manual, Version 2, Grove Engineering, Washington Grove, MD, July 1987

7.8 Microshield User's Manual, A Program for Analyzing Gamma Radiation and Shielding, Version 3, Grove Engineering, Inc., Washington Grove, MD

7.9 Croff, A. G., ORIGEN2- A Revised and Updated Version of the Oakridge Generation and Depletion Code, ORNL-5621, Oak Ridge National Laboratory, 1980

7.10 Calvert Cliffs Nuclear Power Plant Environmental Report, Docket Nos. 50-317 and 50-318, Baltimore Gas and Electric Company, November 16, 1970

7.11 Letter from Mr. G. C. Creel (BGE) to Director, Office of Nuclear Material Safety and Safeguards (NRC), dated November 1, 1990, Response to NRC's Comments on Environmental Issues Regarding BGE's License Application for Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI)

7.12 J. F. Hall, "Corrosion of Low Alloy Steel Fastener Materials Exposed to Borated Water", Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors, Ed. by G. J. Theus and J. R. Weeks, The Metallurgical Society, 1988

7.13 Deleted

7.14 CCNPP Calculation CA05803, ISFSI 24P Assembly Insertion Requirements, December 31, 2001

7.15 CCNPP Calculation CA05924, Calvert Cliffs ISFSI/NUHOMS-24P Radiation Dose Rates for Cask Loading and Transfer, September 18, 2002

7.16 CCNPP Calculation CA05925, Calvert Cliffs ISFSI/NUHOMS-24P HSM Dose Rates, September 18, 2002

7.17 CCNPP Calculation CA05926, Calvert Cliffs ISFSI/NUHOMS-24P Occupational Doses, September 18, 2002

7.18 Los Alamos National Laboratory, LA-13709-M, MCNP-A General Monte Carlo N-Particle Transport Code, Version 4C, March 2000

7.19 CCNPP Drawing 84011SH0001, Revision 00, ISFSI Weld Machine Radiation Shield

7.20 CCNPP Engineering Package ES200101042, ISFSI 24P Design Basis Dose Calculations with MCNP, September 25, 2002

7.21 CCNPP Calculation CA06058, ISFSI 24P 5 Phase Site Dose Rates, September 25, 2002

7.22 CCNPP Calculation CA03904-002, HSM Inlet Vent Surveillance Dose

7.23 CCNPP Calculation CA06751, Horizontal Storage Module Dose Rates for ISFSI 32P Burnup Extension