



APPENDIX D Naval Spent Nuclear Fuel Management Part B

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and
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Environmental Restoration and
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Final Environmental Impact Statement
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Naval Spent Nuclear Fuel Management
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ATTACHMENT A - TRANSPORTATION OF NAVAL SPENT NUCLEAR FUEL

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ATTACHMENT A

TRANSPORTATION OF NAVAL SPENT NUCLEAR FUEL

A.1 PURPOSE AND SCOPE

This attachment provides an evaluation of the radiological and non-radiologic with the transportation of naval spent nuclear fuel and test specimens that origina commercial shipyards, prototypes, and related Department of Energy laboratories. T all past shipments through May 1995 and shipments planned in the 40-year period fro the end of 2035. This attachment evaluates the radiological risks associated with described in Section 3.

A.2 BACKGROUND

The transportation of naval spent nuclear fuel and test specimens covered in into the following four categories:

- Shipments of Naval Spent Nuclear Fuel from Shipyards and Prototypes
- Transfers of Naval Spent Nuclear Fuel to Storage Following Examination
- Transfers of Naval Test Specimen Assemblies Between the Examination Facili Reactor Area
- Shipments of Naval Test Specimens to Examination and Testing Facilities.

Each category is described in more detail below.

A.2.1 Shipments of Naval Spent Nuclear Fuel from Shipyards and

Prototypes

Since 1956, spent nuclear fuel has been removed from Navy nuclear-powered shi as a routine part of their operational cycle. The spent nuclear fuel has been tran Core Facility (ECF) for examination and evaluation. ECF is part of the Naval React within the Idaho National Engineering Laboratory (INEL). The examinations of the s irradiated test specimens have provided and will continue to provide engineering da designs used in technology development for naval nuclear reactors.

In the past, shipments have originated from two prototype sites, nine shipyar Shippingport Atomic Power Station (SAPS), located in Shippingport, Pennsylvania. T locations are the Kenneth A. Kesselring Site (KSO), located in West Milton, New Yor Site Operation (WSO), located in Windsor, Connecticut. The nine shipyard locations Shipbuilding (NNS), located in Newport News, Virginia; the Norfolk Naval Shipyard (Portsmouth, Virginia; the Pearl Harbor Naval Shipyard (PHNS), located in Pearl Harb Portsmouth Naval Shipyard (PNS), located in Kittery, Maine; the Puget Sound Naval S located in Bremerton, Washington; the Charleston Naval Shipyard (CNS), located in C Carolina; the Mare Island Naval Shipyard (MINS), located in Vallejo, California; th Division of General Dynamics (EB), located in Groton, Connecticut, and Ingalls Ship located in Pascagoula, Mississippi. Figure A-1 provides a map of the United States transportation origins for naval spent nuclear fuel. No future shipments from the Ingalls Shipbuilding, and Shippingport Atomic Power Station facilities are planned. Naval Shipyard, Charleston Naval Shipyard, and Windsor Site Operations facilities a

The naval spent nuclear fuel has been shipped in M-130, M-140, M-160, and S2W containers. Only the M-130, M-140, and M-160 shipping containers will be used in t description of the shipping containers to be used for naval spent nuclear fuel ship prototype sites is provided in Section A.4.1.

Figure A-1. Transportation origins for naval spent nuclear fuel. The naval spent spurs to the sites are not available. Therefore, the shipping containers are trans transporter to a nearby commercial rail line where the containers are then transpor Harbor Naval Shipyard, the containers are transported by ship to the Puget Sound Na the containers are then transported to ECF by rail. Since 1956, 599 containers of

have been shipped to ECF. An additional 16 containers of spent nuclear fuel were shipped from Shippingport Atomic Power Station to Hanford and 4 from ECF to Hanford); however, the transfers are covered by the DOE historic shipment calculations in Appendix I, Volume 1 of this EIS Statement. Table A-1 provides a list of these shipments made by year and origin.

A.2.2 Transfers of Naval Spent Nuclear Fuel to Storage Following

Examination

In the past, following examinations at ECF, the spent nuclear fuel has been processed and transferred to the Idaho Chemical Processing Plant (ICPP), also located on the INEL site. A description of the operations performed in the Expanded Core Facility is provided in Appendix I. Spent nuclear fuel is currently being held at ICPP until permanent disposition becomes available.

Since 1956, approximately 5400 transfers of naval spent nuclear fuel have been made to ICPP in shipping casks transported by truck dedicated to performing only such shipments. For alternatives involving continued transfers to storage, the transfers would be made in Peach Bottom, and Large Cell casks, in exclusive-use trucks. A detailed description of the transfers used for naval spent nuclear fuel transfers to storage is provided in Section A.4.2.

A.2.3 Transfers of Naval Test Specimen Assemblies Between the

Examination Facility and the Test Reactor Area

In addition to naval spent nuclear fuel from ships and prototypes, irradiated test specimen assemblies (fuel and non-fuel) have also been transported to ECF for examination. These assemblies are constructed of plant materials, reactor structural materials, and fuels used in the reactors. Table A-1. Number of past naval spent nuclear fuel containers shipped to ECF by year and origin.

| Year | Origin EB | SAPS | KSO | MINS | PHNS | PSNS | NNS | PNS | CNS | WS |
|------|--------------|------|-----|------|------|------|-----|-----|-----|----|
| 1957 | 1 | | | | | | | | | |
| 1958 | | | 1 | | | | | | | |
| 1959 | 1 | | | | | | | 1 | | |
| 1960 | | | | | | | | | | |
| 1961 | 1 | 2 | 2 | | | | | | | |
| 1962 | 5 | | | 1 | 1 | | | | | |
| 1963 | | 3 | | 1 | 1 | | | | | |
| 1964 | 2 | 1 | 2 | | | | | | | |
| 1965 | 2 | 1 | | 2 | | | 33 | 1 | 2 | |
| 1966 | 4 | 2 | | 1 | 1 | | | 1 | | 1 |
| 1967 | 2 | | 1 | | | 2 | 8 | 3 | 3 | |
| 1968 | 2 | | | 4 | | 4 | 2 | 3 | 2 | |
| 1969 | 8 | | 2 | 3 | 1 | 2 | 4 | | 2 | |
| 1970 | 4 | | | 7 | | 2 | 32 | 2 | 2 | |
| 1971 | 4 | | | 2 | | 8 | 4 | 2 | | |
| 1972 | 2 | | | 4 | | 2 | 2 | | 4 | |
| 1973 | 2 | 1 | 1 | 2 | 1 | 6 | 4 | 2 | 2 | |
| 1974 | 2 | 1 | | 6 | | 6 | 2 | 3 | | |
| 1975 | 2 | | 1 | 4 | 1 | 4 | 2 | | 2 | 1 |
| 1976 | 4 | | 3 | 7 | | | 2 | 4 | 2 | |
| 1977 | | | | 4 | 1 | 2 | 2 | 2 | 2 | |
| 1978 | | 2 | | 3 | 1 | 4 | 4 | | 2 | |
| 1979 | | | | 1 | | 2 | | | 2 | |
| 1980 | | | | 2 | | 6 | 4 | 1 | 1 | |

Table A-1 (Cont).

| Year | Origin EB | SAPS | KSO | MINS | PHNS | PSNS | NNS | PNS | CNS | WS |
|------|--------------|------|-----|------|------|------|-----|-----|-----|----|
| 1981 | | | | | 1 | | 4 | | 3 | |
| 1982 | | | | | 1 | | 6 | | 3 | |
| 1983 | | 3 | | 2 | | 6 | 4 | | 2 | 1 |
| 1984 | | 7 | | | 1 | 6 | 4 | 2 | | |
| 1985 | | | | | | 2 | 2 | 2 | 2 | |
| 1986 | | | | 2 | 1 | 4 | 4 | 2 | 2 | |
| 1987 | | | | 1 | | 4 | | 2 | 6 | |
| 1988 | | | | 4 | 1 | 5 | | 3 | 4 | |

| | | | | | | | | | | | |
|-------|---|--|----|----|----|----|-----|-----|----|----|---|
| 1989 | | | 4 | 1 | 7 | | 2 | 4 | | | |
| 1990 | | 3 | 4 | | 10 | 4 | 4 | 3 | | | |
| 1991 | | | 4 | 2 | 4 | | 1 | 7 | | | |
| 1992 | | 3 | 3 | 2 | 7 | | | 4 | | | |
| 1993 | | | | 2 | 8 | 12 | | | | | |
| 1994 | | 2 | 4 | | 1 | 5 | | 4 | | | |
| (1) | | | | | | | | | | | |
| 1995 | | | 2 | | 1 | | | | | | |
| (1) | | | | | | | | | | | |
| TOTAL | | 48 | 23 | 21 | 84 | 20 | 115 | 150 | 43 | 72 | 3 |
| EB | = | Electric Boat Division of General Dynamics | | | | | | | | | |
| SAPS | = | Shippingport Atomic Power Station | | | | | | | | | |
| KSO | = | Kenneth A. Kesselring Site Operations | | | | | | | | | |
| MINs | = | Mare Island Naval Shipyard | | | | | | | | | |
| PHNS | = | Pearl Harbor Naval Shipyard | | | | | | | | | |
| PSNS | = | Puget Sound Naval Shipyard | | | | | | | | | |
| NNS | = | Newport News Shipbuilding | | | | | | | | | |
| PNS | = | Portsmouth Naval Shipyard | | | | | | | | | |
| CNS | = | Charleston Naval Shipyard | | | | | | | | | |
| WSO | = | Windsor Site Operations | | | | | | | | | |
| NOR | = | Norfolk Naval Shipyard | | | | | | | | | |
| INGL | = | Ingalls Shipbuilding | | | | | | | | | |

(1) Shipments in these years cover those authorized by the court injunction. plants are tested and qualified to characterize their performance for the lifetime qualification program is to perform various irradiation tests of the materials for certification. Along with those tests are pre- and post-examinations that provide subsequent analysis of the material in question. This work is considered a fundame design and safe operation of naval reactor plants. Therefore, the transfers of tes the examination facility and shipments of the test specimens to the test facilities transportation evaluation. The test specimens have been assembled into test specim irradiated at the Test Reactor Area (TRA) on the INEL. The irradiated test specime returned to ECF for disassembly and examination.

Since 1956, approximately 3600 transfers of naval test specimen assemblies ha between ECF and TRA in shipping casks transported by exclusive-use truck. For alte future transfers of this type, the transfers would be made in the NR-1, ATR-2, NR-3 casks. A detailed description of the shipping casks used to transfer irradiated te provided in Section A.4.3.

A.2.4 Shipments of Naval Irradiated Test Specimens to Examination

and Testing Facilities

Following disassembly and examination of the test specimen assemblies at ECF, are shipped to off-site facilities for further testing or examination. These tests generally very specialized and ECF does not have the capability to perform them or in a timely manner due to other examination priorities. Specimens are also shipped examination or further irradiation at TRA.

Test specimen shipments have been shipped to or from several laboratories and They are the Bettis Atomic Power Laboratory (Bettis), located in West Mifflin, Penn Atomic Power Laboratory (KAPL), located in Niskayuna, New York; the Oak Ridge Natio (ORNL), located in Oak Ridge, Tennessee; the Argonne National Laboratory (ANL)-East Argonne, Illinois; the Battelle Memorial Institute, located in Columbus, Ohio; the Laboratories, located in Chalk River, Ontario, Canada (1 shipment only); the Hanfor Richland, Washington; and the ANL-West, Central Facilities Area (CFA), TRA, and ICP located on the INEL. Based on current schedules, Bettis and KAPL will be the only origins for future shipments. Figure A-2 provides a map of the United States transportation origins and destinations for the test specimen shipments.

Since 1956, approximately 850 shipments of naval test specimens have been mad and on- and off-site testing and examination facilities, in shipping containers tra truck. The shipments have been made in NRBK-41, -42, -43, and -44 shipping contain 39 and -40 shipping containers. For alternatives involving future shipments of thi would be made in the NRBK-41 and WAPD-40 shipping containers. A detailed descripti shipping containers used to ship irradiated test specimens between off-site facilit facility is provided in Section A.4.4.

A.3 ALTERNATIVES TO BE EVALUATED

A detailed description of the alternatives is provided in Section 3. The specific types of naval shipments (described in Section A.2) are described below for

A.3.1 Alternative 1 - No Action

Under this alternative, after implementation, there would be no further shipment of nuclear fuel from the shipyards and prototypes. The Expanded Core Facility would be used for spent nuclear fuel. Spent nuclear fuel would be stored at a facility at the site where it was removed, with the exception of naval spent nuclear fuel removed at Newport News Shipbuilding Shipyard, which would be transported to Norfolk Naval Shipyard for storage. All naval spent nuclear fuel currently at ECF would be transferred to ICPP prior to the start of the 40-year period for the fuel saved for future examinations, referred to as reference specimens. The reference specimens of the naval spent nuclear fuel which originated at the prototype sites at NRF would be transferred to ICPP sometime during the 40-year period. The TRA facility would perform any work associated with the assembly, disassembly, and routine examination of the test train assemblies; therefore, no transfers would be required. Specimens shipped off-site would remain at the destination following A-2 summarizes the shipments for the No Action alternative.

Figure A-2. Transportation origins and destinations for test specimen shipment

| | |
|--|--------------------|
| Type of Shipment | |
| Shipments of Naval Spent Nuclear Fuel from Shipyards and Prototypes | |
| - Shipyards and Prototypes to ECF | None |
| - Newport News to Norfolk | Yes |
| Transfers of Naval Spent Nuclear Fuel from ECF to ICPP | Reference Specimen |
| Transfers of Naval Test Specimen Assemblies Between ECF and TRA | Prototype Only |
| Shipments of Irradiated Test Specimens Between Off-Site Facilities and ECF | None |
| - Shipments from ECF | Yes |
| - Shipments back to ECF | None |

A.3.2 Alternative 2 - Decentralization

As described in Section 3.4, this alternative also involves storage of the naval spent nuclear fuel near the point of origin. An evaluation of each of the three subalternatives defined was performed. The impact of the transportation related to each subalternative is brief

A.3.2.1 Alternative 2a - Store Naval Spent Nuclear Fuel at or Close to Locations Where Removed Without Examination

From the standpoint of transportation, this subalternative is equivalent to the No Action alternative.

A.3.2.2 Alternative 2b - Examine a Limited Amount of Naval Fuel in the Puget Sound Naval Shipyard Water

Pit Facility and Store All Naval Fuel at Navy Facilities. For this alternative, the Pit Facility at NRF would be shut down and only high priority spent nuclear fuel would be shipped to Puget Sound Naval Shipyard for examination. For the naval spent nuclear fuel, approximately 10 percent of the total spent nuclear fuel for the 40-year period would be shipped. Following examination, the specimens would remain at Puget Sound Naval Shipyard. As in the No Action alternative, only reference specimens would remain at ECF after June 1995. Ten percent of the reference specimens would be transferred from ECF to Puget Sound Naval Shipyard. The remainder of the reference specimens of the naval spent nuclear fuel which originated at the prototype sites at NRF would be transferred to TRA. The TRA facility would perform any work associated with the assembly, disassembly, and examination of the test specimen assemblies; therefore, no transfers would be required. Shipments of naval spent nuclear fuel off-site facilities for specialized examinations would continue. Test specimens shipped

remain at the destination following examination. Table A-3 summarizes the shipment
 Table A-3. Summary of shipments for the Decentralization - Limited Inspection alte
 Type of Shipment

Shipments of Naval Spent Nuclear Fuel from Shipyards and
 Prototypes

- Shipyards and Prototypes to Puget Sound

Approximately
 of spent fuel

- Newport News to Norfolk

Yes

Transfers of Naval Spent Nuclear Fuel from ECF to ICPP

Reference Spec
 Prototype Only

Transfers of Naval Test Specimen Assemblies Between
 Puget Sound and TRA

None

Shipments of Irradiated Test Specimens to Off-Site Facilities

- Shipments from TRA

Yes

- Shipments back to TRA

None

A.3.2.3 Alternative 2c - Examine All Naval Spent Nuclear Fuel at the INEL and Return to Navy Facilities

for Storage. For this alternative, all naval spent nuclear fuel would be shipped t
 has been in the past. Only non-destructive examinations would be performed. The s
 would be returned in the same condition as originally shipped. Following examinati
 returned to the originating shipyard or prototype site for storage in the same type
 exception that naval spent nuclear fuel which originated at Newport News Shipbuildi
 to Norfolk Naval Shipyard for storage. New equipment would have to be designed and
 the spent nuclear fuel which returns to the shipyard. As in the No Action alternat
 specimens would remain at ECF after June 1995. The naval spent nuclear fuel which
 prototype sites at NRF (A1W and S5G) would be transferred to ICPP. Transfers of th
 specimen assemblies would continue, along with the shipments of test specimens from
 testing or examination facilities. Specimens shipped off-site would remain at the
 examination. Table A-4 summarizes the planned shipments for this alternative.

Table A-4. Summary of shipments for the Decentralization - Full Examination altern
 Type of Shipment

Shipments of Naval Spent Nuclear Fuel from Shipyards and
 Prototypes

- Shipyards and Prototypes to ECF

Yes

- Newport News to Norfolk

To Norfolk fro
 NRF Prototypes

Transfers of Naval Spent Nuclear Fuel from ECF to ICPP

Yes

Transfers of Naval Test Specimen Assemblies Between ECF and
 TRA

Shipments of Irradiated Test Specimens to Off-Site Facilities

- Shipments from ECF

Yes

- Shipments back to ECF

None

A.3.3 Alternative 3 - 1992/1993 Planning Basis

This alternative plans on making the same types of shipments described in Sec
 attachment. The only difference is that some of the historical origins of naval sp
 destinations for the test specimen shipments will not be used. Table A-5 summarize
 shipments for this alternative.

Table A-5. Summary of shipments for the 1992/1993 Planning Basis alternative.

Type of Shipment

Shipments of Naval Spent Nuclear Fuel from Shipyards and
 Prototypes

- Shipyards and Prototypes to ECF

Yes

- Newport News to Norfolk

No

Transfers of Naval Spent Nuclear Fuel from ECF to ICPP

Yes

Transfers of Naval Test Specimen Assemblies Between ECF and
 ATR

Yes

Shipments of Irradiated Test Specimens to Off-Site Facilities

- Shipments from ECF

Yes

- Shipments back to ECF

Yes

A.3.4 Alternative 4 - Regionalization

As described in Section 3.4, this alternative would distribute existing and new between various sites either on the basis of the fuel type or on the basis of dividing eastern and western parts of the United States. An evaluation of each of the options described in Section 3.4 was performed. The impact of the transportation related to this alternative is briefly described below.

A.3.4.1 Alternative 4a - Regionalization Using Storage at Three Sites. From the standpoint of

transportation of naval spent nuclear fuel and test specimens, this alternative is the Planning Basis alternative.

A.3.4.2 Alternative 4b - Regionalization Using Storage at Two Sites. This alternative would utilize an

existing DOE site in the eastern part of the United States and another existing DOE site of the country for storage of spent nuclear fuel. From the standpoint of transporting nuclear fuel and test specimens, this alternative is equivalent to the Centralization DOE sites because the Navy would operate a facility for examining naval spent nuclear fuel at the DOE sites and the naval spent nuclear fuel would be stored at the same site where

A.3.5 Alternative 5 - Centralization

This alternative considers consolidating all naval spent nuclear fuel and test specimens at INEL, Hanford Site, Savannah River Site, Oak Ridge Reservation, or Nevada Test Site. INEL is identical to the 1992/1993 Planning Basis alternative. For the other centralization alternatives, the number of shipments would be identical to the 1992/1993 Planning Basis alternative, the difference being the destination. The naval spent nuclear fuel will be shipped to the examination and subsequently transferred to a storage facility at the centralization site equivalent to ICPP. Naval spent nuclear fuel shipments from Newport News Shipbuilding Naval Shipyard would not be necessary. As in the No Action alternative, only reference specimens remain at ECF after June 1995. All reference specimens would be shipped to the centralization site. The test specimen assembly shipments would be shipped between the centralization site and the alternate site. The test specimen shipments would originate at the centralization site and would ultimately return to that site for storage.

A.4 GENERAL DESCRIPTIONS

The following general information is common to all of the alternatives evaluated.

A.4.1 Spent Nuclear Fuel Shipping Containers

For naval spent nuclear fuel, the M-130, M-140, and M-160 shipping containers are used for all alternatives. The shipping containers are primarily transported by railcars as part of general-use freight trains. Section A.2.1 describes the special circumstances under which shipping containers are transported by ship or heavy-lift transporter. A brief description follows.

A.4.1.1 M-130 Shipping Container. The M-130 shipping container is a large, lead-lined, steel-shelled

shipping container that is transported in the vertical position on a depressed center railcar. The major components of the M-130 shipping container include the shielded container body, a dust cover. Module holders are installed inside the container to hold the irradiated fuel modules and can be modified to accept different sized fuel modules. The container is shipped except for a small amount of residual water. Cooling fins on the outside of the

dissipate the heat generated by the spent nuclear fuel.

Figure A-3. M-130 shipping container mounted on railcar. The M-130 shipping container configuration. The container is approximately 13 feet tall and 7 feet in diameter. bottom cylindrical lead shell that is covered both on the inside and the outside with steel. The lead on the cylindrical sides is about 10 inches thick and is a minimum at the bottom. In the standard configuration, the closure head at the top of the container is constructed of 5.25 inches of lead and 7 inches of steel.

A.4.1.2 M-140 Shipping Container. The M-140 shipping container is a large, stainless steel shipping

container that is transported in the vertical position on a specially designed well. The major components of the M-140 shipping container include the shielded container protective dome. Module holders are installed inside the container to hold the irradiated fuel modules in place and can be modified to accept different sized fuel modules. The container is an exception of a small amount of residual water. Cooling fins on the outside of the container dissipate the heat generated by the fuel.

The M-140 shipping container weighs approximately 375,000 pounds in the loaded condition. The container is approximately 16 feet tall with a maximum diameter of 10.5 feet. It is made from stainless steel forgings with 14-inch thick walls and a 12-inch thick bottom and protective dome have a total thickness of 17.5 inches of stainless steel.

A.4.1.3 M-160 Shipping Container. The M-160 shipping container is a large, lead-lined, steel-shelled

shipping container that is transported in a horizontal position on a support structure on a flat bed railcar (Figure A-5). The major components of the M-160 shipping container include the container, closure head, and dust cover. Module holders are installed inside the container to hold the irradiated fuel modules in place. The container is shipped dry with the exception of a small amount of residual water. Cooling fins on the outside of the container are designed to dissipate the heat generated by the fuel.

Figure A-4. M-140 shipping container mounted on railcar. Figure A-5. M-160 shipping container mounted on railcar. The container is approximately 16.5 feet long and 6.5 feet in diameter. The container consists of two concentric bottom closed steel cylinders with a 9.4-inch annulus between the cylinders. The outer shell is made from 1.5-inch thick steel, and the inner shell is made from 0.5-inch thick steel. The bottom plate is approximately 7 inches thick, and the closure head is approximately 10 inches thick.

A.4.1.4 Government Escorts for Spent Nuclear Fuel. Commercial railroads, exclusive-use heavy-lift

transporters, or exclusive-use ships are used to transport the naval spent nuclear fuel. The specific routes used to transport the spent nuclear fuel are set by the shipping companies. All naval spent nuclear fuel shipments are accompanied by government escorts perform the duties necessary to ensure the safe, expeditious transportation of spent nuclear fuel.

The government escorts receive specialized training in shipment safety procedures, controls, security, and emergency response. Routine shipment escort procedures involve authorization and shipping documentation, pre-shipment inspections, tracking shipment schedules, enroute inspections, shipment observation and surveillance, and periodic checks. The government escorts have been trained to use and are equipped with the monitoring equipment to verify the shipping container integrity.

A large amount of the government escorts' training involves emergency response procedures. government escorts are equipped to immediately notify emergency assistance personnel to assess the containment status of the shipping container, and communicate this information to support personnel. Depending on the situation, the technical and support personnel at emergency control centers that are prepared to provide the government escorts with the information to quickly and safely bring an emergency situation under control. All railroads, with heavy-lift shipments, also have specific emergency response procedures to safely expedite recovery that are involved in a rail line accident. Continually manned railroad operations have the capability to contact personnel from a combination of resources which provide appropriate manpower at the accident scene.

A.4.2 Spent Nuclear Fuel Shipping Casks for Transfers to Storage

Following Examination

For naval spent nuclear fuel being transferred from the examination facility ICPP), the Nuclear Fuel Services Model 100 cask (NFS-100), Peach Bottom cask, and it will be used for all alternatives. These shipping containers are transported by the description of each cask follows.

A.4.2.1 NFS-100 Cask. The NFS-100 cask is a large, lead-lined, steel-shelled shipping cask that is

transported in the horizontal position on a skid assembly attached to a tandem axle. The major components of the NFS-100 cask include the shielded cask and closure head insert is installed inside the cask to hold the irradiated fuel modules in place, with the exception of a small amount of residual water. The cask is enclosed on the during shipment.

The NFS-100 cask weighs approximately 110,000 pounds in the loaded configuration, approximately 10.5 feet tall and 7 feet in diameter. The cask is a closed bottom cask with a 0.375-inch thick steel inner shell and a 2-inch thick outer shell. The lead on the top is 8.75 inches thick and the lead on the bottom is 8.8 inches thick. The closure head is constructed of 9.75 inches of lead and 2 inches of steel.

A.4.2.2 Peach Bottom Cask. The Peach Bottom cask is a large, lead-lined, steel-shelled shipping cask that

is transported in the horizontal position on a skid assembly attached to a tandem axle. The major components of the Peach Bottom cask include the shielded cask and closure head holding insert is installed inside the cask to hold the irradiated fuel modules in place, dry with the exception of a small amount of residual water. The cask is enclosed on the cover during shipment.

Figure A-6. NFS-100 cask mounted on truck. Figure A-7. Peach Bottom cask mounted on truck. The Peach Bottom cask is approximately 16 feet tall and 3.5 feet in diameter. The cask is a stepped cylinder with a 0.25-inch thick steel inner shell and a 1.75-inch thick steel outer shell. The lead on the top ranges from 5.25 to 6.25 inches thick. The closure heads on each end of the cask are constructed of 8.5 inches of steel.

A.4.2.3 Large Cell Cask. The Large Cell cask, currently being designed for larger fuel types, will be a

large, stainless steel shipping cask that is transported in the vertical position on a skid assembly (Figure A-8). The major components of the Large Cell cask will include a shielded shipping cask, and external impact limiters. Fuel-holding inserts will be installed inside the cask to hold the irradiated fuel modules in place. The cask will be shipped dry with the exception of a small amount of residual water. Cooling fins on the outside of the shipping cask are designed to dissipate the heat generated by the fuel.

The Large Cell cask will weigh approximately 220,000 pounds in the loaded configuration. The shielded cask will be approximately 14 feet tall and 7 feet in diameter. The shielded cask is a closed bottom cylinder made from stainless steel forgings with 13.5-inch thick walls. The closure head will be a 14-inch thick stainless steel forging. The shielded cask is assembled to the shipping cask during transport. The shipping cask will be a 2-inch thick bottom cylinder with fins extending to a total diameter of 93.6 inches. The external cooling assemblies, located on both ends of the cask, will be constructed of encased bi-directional honeycomb and are approximately 10 feet in diameter. The total Large Cell cask height is approximately 17 feet.

A.4.2.4 Shipment Controls. All spent nuclear fuel transfers to a storage facility at the same site as the

examination facility will be accompanied by escorts. The escorts are personnel who perform the duties necessary to ensure the safe transportation of the spent nuclear fuel vehicles located in front of and behind the truck carrying the shipping cask.

The escorts receive specialized training in shipment safety procedures, radio communication, security, and emergency response. The escort vehicles are equipped with distinctive markings. The escorts are capable of radio contact with each other, the driver of the transport vehicle, and emergency coordinating personnel.

Figure A-8. Large Cell cask. A large amount of the escorts' training involves emergency response procedures.

emergency procedures for notification of site technical and safeguards support personnel equipped to immediately notify emergency assistance personnel, immediately assess the situation, the technical and support personnel may activate various emergency equipment with the equipment and manpower to provide the escorts with the necessary and safely bring an emergency situation under control.

Additional administrative controls are imposed on the transfers to further minimize, for example, the transfers are not allowed to travel during heavy traffic periods such as convoy travels at reduced speeds. The route itself also enhances safety, since the route and the highest possible drop distance in the event of an accident is approximately the location where the highway crosses a river bed.

A.4.3 Naval Test Specimen Assembly Casks for Transfers Between

TRA and the Examination Facility

For naval test specimen assemblies being transferred on-site between TRA and the examination facility, the NR-1, ATR-2, NR-3, NR-4, and Test Train casks will be used. These casks are exclusive-use trucks. For off-site shipments to the examination facility at the central Test Train cask will be used. A brief description of each cask follows.

A.4.3.1 NR and ATR Casks. The NR and ATR casks are large, lead-lined, steel-shelled casks that are

transported approximately 10y off horizontal in a cradle assembly attached to a tank (A-9). The major components of the casks include the shielded body, mast, and bottom.

The shielded bodies of the casks are all approximately 32 inches in diameter. The thickness ranges from 0.5 inch to 1.0 inch. The thickness of the inner steel shell for each cask. The lead ranges from approximately 10 inches to 11 inches for the vertical height of the shielded body ranges from approximately 6 feet to 12 feet. The mast is formed of reinforced aluminum and serves to support the structural end of the

Figure A-9. NR/ATR cask mounted on truck. specimen assemblies which require very heavy cask. The bottom closure/shield is constructed of 1.0 to 1.75 inches of steel and

The NR and ATR casks range in weight from approximately 19,000 to 48,000 pounds. The overall cask height ranges from approximately 20 to 30 feet.

A.4.3.2 Test Train Casks. A new test specimen container would be required to transport irradiated test

specimen assemblies between TRA and the examination facility located at the sites of the Centralization alternative. A new cask is currently being designed to replace the cask used to transport the test specimen assemblies between ECF and TRA, which are approaching their design lifetime. The basic concept for this new cask is a thick-walled, stainless steel cask with closures on each end. Energy absorbers will be attached to the cask to prevent damage to specimens. The current estimated size of this cask is 34 feet long by 5 feet in diameter and weighs approximately 40 tons. This cask would be shipped by exclusive-use truck.

A.4.3.3 Shipment Controls. All spent nuclear fuel transfers to an examination facility at the same site as

the irradiation facility will be accompanied by two escorts. The escorts are personnel trained to perform the duties necessary to ensure the safe transportation of the spent nuclear fuel. The escorts are in vehicles located in front of and behind the truck carrying the shipment.

The escorts receive specialized training in shipment safety procedures, radio communication, security, and emergency response. A large amount of the escorts' training involves this training involves emergency procedures for notification of site technical and support personnel. The escorts are equipped to immediately notify emergency assistance personnel to assess the containment status of the shipping cask, and communicate this information to support personnel. Depending on the situation, the technical and support personnel may activate various emergency control centers that are equipped with the equipment and manpower to provide the necessary support to quickly and safely bring an emergency situation under control. The vehicles are equipped with distinctive warning flashers, and the escorts are capable of communicating with each other, the driver of the transport vehicle, and emergency coordinating personnel.

Additional administrative controls are imposed on the shipments to further minimize, for example, the transfers are not allowed to travel during heavy traffic periods such

convoy travels at reduced speeds. The route itself also enhances safety, since the and the maximum possible drop in the event of an accident is from the bed of the tr

For the Centralization alternative, the casks would be shipped off-site. In certified for over-the-road transportation in accordance with the Nuclear Regulator regulations would be used for shipments of the test trains. No escorts or addition would be used.

A.4.4 Test Specimen Shipping Containers

For test specimens, the WAPD-40 and NRBK-41 shipping containers would be used the specimens between ECF and the off-site laboratories and test facilities for all shipping containers are transported by an enclosed truck using a commercial carrier each container follows.

A.4.4.1 WAPD-40 Shipping Container. The WAPD-40 shipping container (Figure A-10) is a cylindrical,

lead-shielded, steel-clad container that is shipped in a horizontal position. The inch thick, and the outer steel shell is 0.5-inch thick with 9.875 inches of lead s container is approximately 13 feet long and 2 feet in diameter. Steel clad, lead-s onto each end, and 0.5-inch thick plates are bolted over the end plugs. The specim special sealed inner containers prior to placement into the WAPD-40 shipping contain the container and skid assembly is approximately 28,000 pounds. The container and mounted into a special holddown cradle on the truck. This holddown cradle weighs a pounds.

A.4.4.2 NRBK-41 Shipping Container. The NRBK-41 shipping container (Figure A-11) is a cylindrical,

lead-shielded, steel-clad container that is shipped in the vertical position. The thick, and the outer steel shell is 0.5-inch thick with 10 inches of lead shielding has a 1-inch thick steel plate welded to the bottom with a second 1-inch thick steel plate with a 0.125-inch deep recess to provide a thermal break for the

Figure A-10. WAPD-40 shipping container. Figure A-11. NRBK-41 shipping contain provides a 0.125-inch air gap between the outer shell and the thermal shield. The approximately 4 feet tall and 2.25 feet in diameter. The container is bolted to a I-beam skid that is used to distribute the container load. The specimens are place inner container prior to placement into the NRBK-41 shipping container. The weight container is approximately 9,000 pounds.

A.4.5 Shipping Container Design Requirements

The M-130, M-140, M-160, NRBK-41, and WAPD-40 shipping containers have been d and built to meet the regulations specified in Title 49, Code of Federal Regulation entitled " Shippers - General Requirements for Shipments and Packagings" (CFR 1991) naval spent nuclear fuel and test specimens are further regulated by Title 10, Code Part 71 (10CFR71), entitled "Packaging of Radioactive Material for Transportation a Radioactive Material Under Certain Conditions" (CFR 1993). These regulations requi container to meet specific criteria under normal transport and accident conditions. must be evaluated under free drop, puncture, heat, cold, pressure, water spray, and conditions and a series of severe hypothetical accident conditions with the results criteria provided in 10CFR71.

The M-130, M-140, M-160, WAPD-40, and NRBK-41 shipping containers have underg rigorous engineering evaluations to assure compliance with 49CFR173 and 10CFR71 req addition, actual scale model or mock-up tests have been performed to verify selecte evaluations. This compliance has been certified by the U. S. Department of Energy Regulatory Commission. The new Test Train and Large Cell casks will also be design with the requirements of 49CFR173 and 10CFR71 and will undergo the same rigorous en evaluations and testing.

The safety analyses for the NFS-100, Peach Bottom, NR, and ATR casks demonstr with the requirements specified by the Department of Energy (DOE) in DOE Order 5480 "Safety Requirements for the Packaging and Transportation of Hazardous Materials, H Substances, and Hazardous Wastes" (DOE 1985) and supplemented by DOE Idaho Operatio

Order ID 5480.3, entitled "Hazardous Materials Packaging and Transportation Safety (DOE 1991). These requirements are similar to the requirements of 10CFR71 with the being that a worst credible accident can be defined based on site-specific informat

The NFS-100, Peach Bottom, NR, and ATR casks have undergone rigorous engineer evaluations to assure compliance with the DOE requirements. In addition, actual sc tests have been performed to verify selected engineering evaluations. The shipping requirements of DOE 5480.3 and DOE ID 5480.3 and this compliance is demonstrated by the Idaho Operations Office of the Department of Energy.

A.5 TECHNICAL APPROACH - GENERAL

Several computer codes were used to assess the radiological risks associated transportation of naval spent nuclear fuel and test specimens. Specifically, the R model, developed by Sandia National Laboratories (Neuhauser and Kanipe 1992), was u the general population and transportation crew (occupational) radiological risks as transportation of radioactive materials. This computer code was used extensively i accident risk assessments. In some cases, other methods were more appropriate than computer code for naval spent nuclear fuel. In these cases, other calculational mo specifically identified.

The RISKIND computer code, developed by Argonne National Laboratory (Yuan et specifically analyzes radiological consequences and health risks to individuals fro with transportation. For incident-free evaluations, RISKIND uses a generic truck c adjustments for different sized casks which is not appropriate for naval spent nucl specimen casks; therefore, this code was not used. RISKIND (a version which accept isotopes) was found to be the best code for calculation of the maximum individual a consequences for the accident scenario and was used for that purpose.

Several other computer codes were used to provide input for the RADTRAN 4 and computer codes. The codes include INTERLINE, HIGHWAY, SPAN4, and ORIGEN2. A descr each computer code and how the code was used is provided below.

The INTERLINE computer code, developed by Oak Ridge National Laboratory (John 1993a), was used to evaluate the rail routes used for the spent nuclear fuel shipme

The HIGHWAY computer code, also developed by Oak Ridge National Laboratory (J 1993b), was used to evaluate the truck routes used for the test specimen shipments.

The SPAN4 computer code (Wallace 1972) was used to perform gamma exposure rat calculations for the various shipping containers to assess the effect of increased on exposure. SPAN4 is a point kernel code where appropriate exponential kernels ar source distribution. SPAN4 was developed by the Bettis Atomic Power Laboratory spe spent nuclear fuel.

The ORIGEN2 is a computer code, developed by Oak Ridge National Laboratory (C that is used to simulate radiation and decay of materials that are irradiated in a ORIGEN2 computer code is widely accepted in the public domain and was used to indep the fission product inventory for naval fuel developed using the standard Bettis At method. In addition, the standard Bettis Atomic Power Laboratory method has been u Analysis Reports for Packaging, reviewed and accepted by the Nuclear Regulatory Com

The radiological risks associated with the transportation of spent nuclear fu specimens have been assessed for the general population, transportation workers (oc hypothetical maximum exposed individuals under incident-free and accident condition presented in Section A.3. The maximum consequences for an accident are also provid alternative. The radiation exposure to the government escorts for shipments was co in nature and was included with the transportation worker results.

The radiological impacts are first expressed as the calculated total exposure population, occupational workers, and the maximum exposed individuals. The calcula are then used to estimate the hypothetical health effects, expressed in terms of es The health risk conversion factors used in this evaluation are taken from the Inter Radiological Protection (ICRP Publication 60) which specifies 0.0005 fatal cancer c bers of the public, 0.0004 fatal cancer cases per person-rem for workers (ICRP 1991 estimated health detriment, the calculated exposure would be multiplied by the conv health detriments per person-rem for members of the public, and 0.00056 health detr

The numerical estimates of cancer deaths and other health detriments presente the practice of linear extrapolation from the nominal risk estimate for lifetime to rad. Other methods of extrapolation to the low-dose region could yield higher or l estimates of cancer deaths. Studies of human populations exposed at low doses are demonstrate the actual level of risk. There is scientific uncertainty about cancer

below the range of epidemiologic observation, and the possibility of no risk cannot 1992). In this appendix, the doses have been provided in all cases to allow indepe any relation between exposure and health effects.

Non-radiological risks related to the transportation of naval spent nuclear f The non-radiological risks are associated with vehicle exhaust emission for inciden fatalities resulting from transportation accidents. The non-radiological risks ass return empty containers to the origin are also included. Risk factors for vehicle state-level accident fatality rates were obtained from "Non-Radiological Impacts of Radioactive Material" (Rao et al. 1982), "Transportation Impacts of the Commercial Management Program" (Cashwell et al. 1986), and "Longitudinal Review of State-Level Statistics for Carriers of Interstate Freight" (Saricks and Kvitek 1994), respectiv

The shipments of radioactive waste at shipyards are not addressed. The expos incident-free transportation would be small and would be the same for all alternati affect the decision-making process. The consequences of an accident would also be compared to the accidents analyzed for spent nuclear fuel.

For the ocean-going portion of the shipments of naval spent nuclear fuel from prototypes, there would be no exposure to the general population. The basis for th ship's hull provides a considerable amount of additional shielding and that there w the general population close enough to the ship to receive appreciable exposure dur The consequences of an accident during the ocean-going portion have also not been e forces on the container during an accident aboard the ship would not be large enoug the container or fuel inside it since the ship itself would sustain the direct impa the fact that the impact forces to the container would be less than the regulatory release would occur.

A.5.1 Technical Approach for the Assessment of

Incident-free Transportation

For incident-free transportation of naval spent nuclear fuel, the RADTRAN 4 c used to calculate the radiologi- cal exposure for the general population and a portion of the occupational exposure.

Included in the RADTRAN 4 computer code incident-free risk calculations for t models describing (1) exposures to persons (e.g., residents) adjacent to the transp exposures), (2) exposures to persons (e.g., passengers on passing trains or vehicle route (on-link doses), (3) exposures to persons at stops (e.g., residents or rail a involved with the shipment), and (4) exposures to transportation crew members (occu exposures calculated for the first three groups were added together to estimate the exposure estimates for rail and truck transport; the exposure calculated for the fo occupational exposure to the rail crew exposures during inspections and truck crew inspections. Table A-6 summarizes the calculational methods used for each group fo naval spent nuclear fuel and test specimens.

As shown in Table A-6, simple calculations were performed to account for situ RADTRAN 4 computer code was not the best calculational model with respect to the tr naval spent nuclear fuel. The information used in the simple calculations was base information. The results obtained using these simple calculations are expected to than any exposures which might actually occur.

The maximum possible radiological exposure to an individual for the routine t spent nuclear fuel and test specimens off-site was estimated for transportation wor of the general population. For rail shipments, the three general population scenar worker who might be working at a distance of 10 meters (32.8 feet) from the shippin hours, (2) a resident who might live 30 meters (98.4 feet) from the rail line Table A-6. Calculational methods used to obtain exposures for population groups of

| | | | | General Population | |
|----------|--------|----------------|------|----------------------------|---|
| Shipment | Origin | Destination(a) | Mode | Off-Link and On-Link | S |
| Type | | | | | |

| | | | | | |
|--|----------------------|-------------------|-------|-----|---|
| Spent Nuclear Fuel to ECF or Equivalent | Kesselring Site | Ballston Spa | Truck | (1) | (|
| | Shipyard/Rail Siding | Various | Rail | (1) | (|
| | Windsor Site | Griffen Siding | Truck | (1) | (|
| | Pearl Harbor | Puget Sound | Ship | N/A | N |
| Spent Nuclear Fuel to Storage Test Specimen Assemblies | ECF or Equivalent | Various | Truck | (1) | (|
| Test Specimens | ECF or Equivalent | Bettis/KAPL, etc. | Truck | (1) | (|
| Calculational Methods: | | | | | |

- (1) RADTRAN 4 calculations.
- (2) RADTRAN 4 rail calculations for inspection exposure and simple calculations based on transportation data supplied by the government escorts for rail transit exposures
- (3) Simple calculation model based on truck transportation data supplied by site
- (4) Simple calculation model based on ship transportation data supplied by Pearl
- (5) Exposures based on historical TLD readings.
- (6) Simple calculation model based on scenarios provided in RISKIND.
- (a) The methods provided in this table apply to the destination for all the alter where the shipping container was being transported, and (3) a resident who could be (656.2 feet) from a rail stop where the shipping container was sitting for 20 hours and crew members from the rail, heavy-lift transporter, and ship were evaluated for workers (occupational). Based on records of past escorted rail shipments, the gove the same individual for as many as two-thirds of the shipments in a 5-year period. postulated to be the same individuals for all shipments in the 40-year period.

For off-site truck shipments, the three scenarios for the general population might be caught in traffic and located 1 meter (3 feet) away from the surface of the one-half hour, (2) a resident who might be living 30 meters (98.4 feet) from the highway the shipping container, and (3) a service station worker who might be working at a (65.6 feet) from the shipping container for 2 hours. The hypothetical maximum exposure radiological exposures were accumulated over the 40-year period. However, for the individual who might be caught in traffic next to a truck transporting spent nuclear exposures were only calculated for one event since it was considered unlikely that would be caught in traffic next to all containers for all shipments. For truck shipments maximum exposed individual is the driver. For each of the categories of truck shipments Sections A.4.2 through A.4.4, the calculations used a single individual as the driver in the past. For shipments in the 40-year period being evaluated, a single person calculations as the driver for all shipments of each category.

The hypothetical maximum exposed individual scenarios for the general population above were not applicable for on-site shipments of naval spent nuclear fuel and test reasons. The first is that there are no members of the general population in the vicinity of the shipments. The second reason is that an obstruction, if encountered, would be safe direction of the escorts. Two alternate scenarios were developed. They were: (1) disabled vehicle along the transport route, located 10 meters (32.8 feet) from the employee trailing the slow-moving transport vehicle for the entire trip. These scenarios be single-event occurrences.

As noted in Table A-6, simple methods were also used to calculate radiological exposures to personnel at a fixed distance from the shipping container was used.

Exposures to personnel at a fixed distance from the container:

$$= N \times NBA \times T \times SF \times K \times TI / D^2$$

where:

N = number of people

NBA = factor to account for exposure decrease at increased distance from the (attenuation/buildup). (Refer to Neuhauser and Kanipe 1993.)
 T = time
 SF = shielding factor
 K = transport index to exposure rate conversion factor
 TI = transport index (see Section A.7.1.1.2)
 D = distance from the centerline.

For the radiological exposures associated with the ship transport of spent nuclear fuel from the Pearl Harbor Naval Shipyard to the Puget Sound Naval Shipyard, the following general equation is used:

Exposures to personnel aboard ship during transport:

$$= N \times NBA \times T \times SF \times K \times TI \times (1/(X1 + X2)^2 + 1/X2^2)$$

where:

X1 = distance between the centerlines of the two shipping containers
 X2 = distance between centerline of the nearest shipping container and the individual

Exposures to personnel aboard ship during inspections:

$$= (N \times T \times TI) + (N \times NBA \times T \times K \times SF \times TI / (X1 - R - 1)^2)$$

where:

R = effective radius to account for the exposure from the second shipping container

Table A-7 provides an estimate of the number of people included in the analysis. This number, the basic equation used was:

(Distance Traveled) x (Exposure Path Width) x (Density of People).

In each alternative, there are many shipments from several different origin/destination combinations. Since the route would be the same for each shipment from the same origin/destination combination, the people along the route would also not change, therefore, the distance traveled for each origin/destination combination. The exposure path width is 1.6 kilometers with the RADTRAN 4 computer code methodology for incident-free calculations. The population was calculated by summing the product of the fraction of travel times the density in each area (rural, suburban, and urban). The fraction of travel and density were obtained from INTERLINE. The total number of people was then calculated by summing the results of origin/destination combinations for each alternative.

| Alternative | Number of People |
|---|------------------|
| No Action | 890,000 |
| Decentralization - No Examination | 890,000 |
| Decentralization - Limited Examination | 9,240,000 |
| Decentralization - Full Examination | 6,820,000 |
| 1992/1993 Planning Basis | 7,290,000 |
| Regionalization or Centralization at INEL | 7,290,000 |
| Regionalization or Centralization at Hanford | 8,370,000 |
| Regionalization or Centralization at Savannah River | 6,950,000 |
| Regionalization or Centralization at Oak Ridge | 5,660,000 |
| Regionalization or Centralization at Nevada Test Site | 8,320,000 |

A.5.2 Technical Approach for Transportation Accidents

The RADTRAN 4 computer code was used to calculate the radiological risk to the population and transportation (occupational) crew under accident conditions. The RADTRAN 4 code evaluates six pathways for radiation exposures resulting from an accident. They are:

- Direct Radiation Exposure from the Damaged Container
- Inhalation Exposure from the Plume of Radioactive Material Released from the Damaged Container
- Direct Radiation Exposure from Immersion in the Plume of Radioactive Material from the Damaged Container
- Direct Radiation Exposure from Ground Deposition of the Radioactive Material from the Damaged Container

- Inhalation Exposure from Resuspension of the Radioactive Material Deposite Ground
- Ingestion Exposure from Food Products Grown on the Soil Contaminated by Gr Deposition of Radioactive Material Released from the Damaged Container.

For each pathway, a specific formula is used to determine an estimate of the expressed in exposure, from that particular pathway with the total radiation exposu the exposure for each pathway. The total accident radiation exposure accounts for accident occurring and the probability of an accident of a particular severity. It consequences are included in the risk assessment, regardless of the probability. T the population exposure from all pathways is:

$$DR = -c, r (Nc \times Lr, c \times Pr \times -i, j, k (Pj \times RFj \times Di, j, k))$$

where: DR = population exposure from the accident

Nc = number of naval spent nuclear fuel modules shipped of fuel type c

Lr, c = shipment distance for fuel type c shipped through state r

Pr = frequency of traffic accidents

Pj = probability of occurrence of accident severity category j

RFj = fraction of curies released from shipping container by severity categ

Di, j, k = radiation exposure resulting from accident severity category j throu population density zone k.

The accident risk evaluation was performed using neutral and stable atmospher (Pasquill Stability Classes D and F, respectively). The neutral atmospheric condit estimate of the risk. Stable atmospheric conditions resulted in values approximate conditions, ignoring the lower probability of occurrence.

In addition to the estimation of the radiological risk of an accident describ of the consequences of an accident of the highest severity was performed. The cons radiological exposure, are calculated for the maximum exposed individual and the ge Exposures to the general population were calculated for each of the three populatio suburban, and urban). The maximum exposed individual was placed in the population in the highest exposure.

The RISKIND computer code, modified by its authors to accept the fission prod unique to naval spent nuclear fuel, was used to calculate the maximum consequences. evaluated by RISKIND are identical to those used in the RADTRAN 4 computer code for evaluation.

The maximum consequence evaluation presents the consequences for design basis defined as those accidents which have a probability of greater than 1×10^{-6} per y basis accidents, defined as those which have a probability of 1×10^{-6} to 1×10^{-7} probability of less than 1×10^{-7} were not analyzed in the maximum consequence eval

To determine the overall probabilities, the probability of an accident, the p quences, fraction of travel in each population area, and probability of the meteoro be determined.

The probability of the accident was calculated by multiplying the accident ra times the distance traveled in each state times the number of shipments. The resul combination of origin and destination for the alternative.

As described later in Section A.7, a study performed by Lawrence Livermore Na ry

entitled "Shipping Container Response to Severe Highway and Railway Accident Condit 1987) grouped accidents into categories by strain and container mid-wall temperatur probabilities of accidents of each category. Section A.7 also describes the conseq each accident category for the naval spent nuclear fuel and test specimen shipments were summed for the categories which have the same consequences.

The fraction of travel in each population area (rural, suburban, and urban) w INTERLINE and HIGHWAY for each origin/destination combination. Each alternative co shipments from various origin/destination combinations; therefore, an overall fract overall fraction, by alternative, was calculated by multiplying each origin/destina INTERLINE and HIGHWAY) by the number of shipments from that particular origin/desti combination, summing the results and dividing by the total number of shipments.

To calculate the probability of the meteorological conditions, Pasquill Class be equivalent to 50% meteorology; that is, 50% of the time, conditions are expected 50% of the time, conditions are expected to be less severe. Pasquill Class F was c equivalent to 95% meteorology; that is, 5% of the time, it is more severe, and 95%

severe. Since the difference in 50% (1 chance in 2) and 95% (1 chance in 20) is a probability of encountering Pasquill Class F was concluded to be a factor of 10 less. Analyses performed by the National Oceanic and Atmospheric Administration (Doty et al.) that this assumption is reasonable.

The overall probability of the consequence of an accident for each population calculated by multiplying the accident probability times the consequence probability distance traveled. Starting with the highest consequences, the probabilities were 1×10^{-6} per year criterion for the design basis accidents and 1×10^{-7} per year criterion for the design basis accidents. If the probability was greater than 10 times the criterion most severe Pasquill Class F results were presented. If not, and the probability was less than the criterion (1×10^{-6} or 1×10^{-7}), Pasquill Class D was presented. If the probability of the probabilities having the next most severe consequences were compared to the same step was repeated until all consequences were evaluated. As a minimum, the consequence release of 1% of the corrosion products (Pasquill Class D) were presented.

Careful attention was paid to ensure that the probabilities were not calculated for categories that the resulting probabilities were less than the criterion and result in less severe consequences. When the highest consequence accident did not meet the probability of the next highest accident was determined by summing both the accident evaluated and the probability of the higher consequence accidents previously shown less than the criterion. This same technique was applied to the fraction of travel equivalent to highest consequence, suburban fraction is next highest, etc.) as demonstrated in the example.

| | | |
|--|---|-----------------------|
| Probability of the accident of Consequence A | - | 1.17×10^{-7} |
| Fraction of distance traveled in rural area | - | 0.85 |
| Fraction of distance traveled in suburban area | - | 0.11 |
| Fraction of distance traveled in urban area | - | 0.04 |

The urban fraction was multiplied by the probability, and the resultant probability for Consequence A in an urban area was 4.68×10^{-9} . The consequences of this accident were evaluated. For the suburban area, the suburban and urban fractions were added and multiplied by the probability (1.75×10^{-8}). Again, the consequences of this accident would not be evaluated if the probability is less than 1×10^{-7} . Likewise, for the rural area, the rural, suburban, and urban fractions were added and multiplied by the probability. Using this technique, the probabilities were evaluated. The probability was 1.17×10^{-7} , which is greater than the 1×10^{-7} criterion and the consequence would be presented. If the fractions were used at face value, however, the probability for Consequence A would have been 4.68×10^{-9} in an urban area, 1.29×10^{-8} in a suburban area, and 1.29×10^{-8} in a rural area. When individually compared to the 1×10^{-7} criterion, this accident would not be presented for any area.

Accident results are presented for both the maximum exposed individual and the total population. These results include members of the transportation crew.

A.6 ROUTING ANALYSIS

In order to assess the radiological risks associated with transportation, it is necessary to determine route characteristics based on the origin and destination of each shipment.

For naval spent nuclear fuel shipments, the origin is the prototype or shipyard. Naval spent nuclear fuel is removed from a prototype or shipboard reactor. The des Savannah River Site, Hanford Site, Oak Ridge Reservation, Nevada Test Site, or Puget Sound Naval Shipyard, depending on the alternative. For each origin and destination pair, the routes have been generated and analyzed using the INTERLINE computer code (Johnson 1993a). For shipments originating from Pearl Harbor Naval Shipyard, the containers travel by ship to Puget Sound Naval Shipyard, where they are transferred to rail for shipment to the destination. For shipments originating from Puget Sound Naval Shipyard, the containers travel by ship to the destination. The time by ocean was based on historical data on the time in transit, independent of the type of heavy-lift transporter shipments from the Kesselring and Windsor prototype sites to the actual street routes and shipment duration times based on previous shipments were used.

INTERLINE is an interactive computer program designed to simulate routing using the INTERLINE system. The INTERLINE code used is the latest available from Oak Ridge National Laboratory and contains the 1990 census data. The INTERLINE data base consists of networks representing competing rail companies in the U.S. The routes used for the transportation evaluation are from the INTERLINE model which simulates the selection procedure that railroad companies would use for shipments of spent nuclear fuel. The code is updated periodically to reflect current conditions. It has been benchmarked against reported mileages and observations. INTERLINE also provides weighted population densities for rural, suburban, and urban populations for each shipment.

all states along the shipment route and the percentage of mileage traveled in each distance traveled, weighted population density, and percentage of distance in each input variables in the RADTRAN 4 code.

For the off-site transportation of the test specimen assemblies and test spec made by exclusive-use truck which includes no other freight. The destinations are Site, Hanford Site, Oak Ridge Reservation, Nevada Test Site, Puget Sound Naval Ship Power Laboratory, and Knolls Atomic Power Laboratory for the various alternatives. destination pair, the potential truck routes have been generated and analyzed using HIGHWAY.

HIGHWAY is an interactive computer code designed to simulate routing using th system. The HIGHWAY code used for this report is the latest available from Oak Rid Laboratory. The code is updated periodically as new roads are added. HIGHWAY prov between the origin and destination, the weighted population densities along the rou of distance traveled in each population density, all input variables for the RADTRA

For the on-site transportation, HIGHWAY only has two of the sites on the INEL origin/destination pair was run using HIGHWAY to determine the population densities travel in each population density. The actual distance between sites on the INEL w

A.7 INPUT PARAMETERS

The major input parameters and models used to evaluate the radiological risks five alternatives described in Section A.3 are provided in this section. Standard code values, as well as actual data gathered from historical naval spent nuclear fu shipments, were used as the basis for the input parameters. For those situations w available, the actual data were used in place of the standard RADTRAN 4 computer co the best estimate of the radiological risks associated with each alternative.

A.7.1 Shipments of Naval Spent Nuclear Fuel from Shipyards and

Prototypes

A.7.1.1 Incident-free Transportation of Spent Nuclear Fuel from Shipyards and Prototypes. This

section provides the input parameters used to determine the radiological impacts as routine, incident-free (i.e., no accident) transportation of spent nuclear fuel for

A.7.1.1.1 Planned Shipments. The list of planned shipments of naval spent nuclear fuel by origin

is provided in Table A-8.

Table A-8. Planned shipments of naval spent nuclear fuel from shipyards and protot

| Alternative | Generating Site | | NRF | TOTAL |
|---------------------------------|-----------------|------------|-----|-------|
| | East Coast | West Coast | | |
| No Action, Decentralization | -204 | 0 | 0 | 204 |
| No Exam | | | | |
| Decentralization - | 53 | 0 | 1 | 54 |
| Limited Exam | 181 | 0 | 0 | 181 |
| | 234 | 0 | 1 | 235 |
| Decentralization - | 314 | 261 | 0 | 575 |
| Full Exam | 314 | 261 | 0 | 575 |
| | 628 | 522 | 0 | 1150 |
| 1992/1993 Planning Basis, | 314 | 261 | 0 | 575 |
| Regionalization at INEL and | | | | |
| Centralization at INEL | | | | |
| All other Regionalization | 314 | 261 | 3 | 578 |
| and Centralization Alternatives | | | | |

A.7.1.1.2 Transport Index. Historical information from prior shipments was used to estimate the

expected external radiation exposure rates for future shipments. This information measured radiation levels and the recorded Transport Indexes (TIs) from past shipments. This analysis is the sum of the maximum neutron and gamma radiation measured at 1 m from the surface of the cask. The TIs that were used ranged from 0.1 to 1.8.

A.7.1.1.3 Transportation Distances and Population Densities. Section A.6 provided a

description of the general methodology used for determining transportation distance densities along the transportation routes. Historical data were obtained on the distances traveled for shipments from the shipyards and prototype sites to ECF. These data were averaged and compared to the value calculated by INTERLINE. The actual data were approximately the distance predicted by INTERLINE on average. In order to provide the best estimate based on the distance traveled, the INTERLINE distances were increased by 11% for the Planning Basis alternative. One of the primary reasons the actual distances traveled longer than the INTERLINE prediction was the escort responsibility to avoid potential or security problems. The shipments to the alternative sites will also be escorted and increased travel distance is expected. The 11% increase in distance traveled was a conservative alternative. This technique allowed for comparison of the alternatives on an equal basis of distance traveled in each population density calculated by INTERLINE were applied and increased by 11%.

A.7.1.1.4 Train Speed. The RADTRAN 4 computer code provides standard values for train

speeds that are dependent on the population density. For rural areas, the standard value is 40 miles per hour (40 mph). For suburban areas, the standard value is 40.2 miles per hour (40.2 mph), and for urban areas, the standard value is 24.1 kilometers per hour (15 mph). Nuclear fuel shipments are required to be transported at speeds not to exceed 56.3 miles per hour (45 mph). Government escort logs from historical spent nuclear fuel shipments support the 15 mph standard. This 24.1 kilometers per hour (15 mph) train speed estimate was used for all five alternatives.

A.7.1.1.5 Train Stop Time. The RADTRAN 4 computer code provides standard values for train

stop times that are either dependent or independent of the distances traveled. For shipments transported by rail, the government escorts are responsible for ensuring that the shipments are transported in the most efficient and safe manner. The government escort logs for historical spent nuclear fuel shipments were reviewed, and actual stop times were determined to be much shorter than the standard computer code values. The recorded stop times were divided by the actual distance traveled to obtain historical data over the last 3 years and an average of 0.02 hour per kilometer (0.02 hours/km) was calculated. This value was used to evaluate all five alternatives since the rail shipments of nuclear fuel will always be accompanied by government escorts and all alternatives use the same locations.

A.7.1.1.6 Number of Train Crew Members. The standard RADTRAN 4 computer code value for the

number of train crew members is five. For all shipments to NRF, all rail companies including Burlington Northern have two crew members during shipments, located in the locomotive and the caboose. Burlington Northern adds a third crew member in a caboose immediately behind the government escort. The RADTRAN 4 computer code, exposure to the crew members is not calculated since the number of crew members is large. In actuality, the distance to the Burlington Northern crew caboose is less than that used in the RADTRAN 4 computer code and therefore simple calculations were performed to determine the radiological exposure. In addition, naval spent nuclear fuel is shipped periodically by "special train." In the special train configuration, the locomotive and the railcar are one car from the railcar with the shipping container. Historically, approximately 42 percent of the time. The majority of shipments by "special" train are made by railroad companies to meet railroad schedules. On occasion, the Navy requests "special" shipments with high-priority examination material. Simple calculations were also performed for the radiological exposure during these special shipments. For shipments to the site, there was no experience with all railroad companies which would have to be used; however, it is expected the rail companies to change their standard practices. In these cases, there

crewmembers, both located in the engine area. Forty-two percent of the shipments would train to the alternate sites. When applicable, the third Burlington Northern crew accounted for.

A.7.1.1.7 Transport Index to Exposure Rate Conversion Factors. Container transport index to

exposure rate conversion factors for the M-130 and M-140 shipping containers were a standard equation in the RADTRAN 4 computer code. The results were compared to the analyses performed using SPAN4, and the RADTRAN 4 results were found to overestimate by a factor of two to three. Using the SPAN4 computer code results, the effective the containers used in the RADTRAN 4 calculations were adjusted to provide a conservative realistic value of the transport index to exposure rate conversion factor. Due to construction and fuel shipped, the M-130 conversion factor was applicable to the M-140 are provided in Table A-9.

Table A-9. Transport index to exposure rate conversion factors for the M-130, M-140 shipping containers.

| Container | Effective Package Dimension (meters) | Transport Index to Exposure Rate Conversion Factor |
|-------------|---|---|
| M-130/M-160 | 2.50 (8.2 feet) | 5.06 |
| M-140 | 3.20 (10.5 feet) | 6.76 |

A.7.1.1.8 Train Stop Shield Factors. For train stops, the standard RADTRAN 4 computer code

gamma and neutron radiation shield factors are both assigned as 0.1. This value in substantial railyard steel structures equivalent to approximately 4 inches of steel reduces gamma radiation by more than a factor of 10; however, the steel only reduce a factor of approximately 2. Therefore, a shield factor of 0.5 was conservatively radiation. In order to incorporate this shielding into the RADTRAN 4 computer code neutron radiation exposure calculations were performed. However, since RADTRAN 4 does separate shielding factors to be used for different types of radiation, the stop time evaluations were increased by a factor of 5 to provide an equivalent increase in more realistic changes to the standard RADTRAN 4 computer code values were incorporated alternatives.

A.7.1.1.9 Radiation Exposure Decrease Due to Distance. The RADTRAN 4 computer code

provides standard values for determining the gamma and neutron radiation exposure at distance from the source. For gamma radiation, the RADTRAN 4 computer code uses the due to distance. The RADTRAN 4 computer code also specifically calculates the decrease exposure at increased distances. The adequacy of the RADTRAN 4 radiation exposure evaluated. The gamma radiation decrease factor used by RADTRAN 4 was consistent with predicted for naval fuel. The RADTRAN 4 prediction for neutron radiation slightly decrease in exposure at far distances for the shipping containers used for naval ship basic equation used by RADTRAN, a value of 2.0×10^{-10} was used for the RADTRAN 4 coefficient of 0. The value of 2×10^{-10} produces results which are slightly higher than agree with measurements of neutron exposure rates from naval spent nuclear fuel shipments.

A.7.1.1.10 Shipment Storage Time. As noted previously, the government escorts accompanying

the rail shipments of spent nuclear fuel are responsible for ensuring that the naval shipments are made in the most efficient and safe manner. Naval spent nuclear fuel being shipped; therefore, there was no intermediate shipment storage time associated alternatives. There is also no intermediate storage time during the heavy-lift transport prototype sites and the ocean shipments from Pearl Harbor Naval Shipyard.

A.7.1.1.11 Heavy-lift Transporter Transportation Crew. Information from records of naval

spent nuclear fuel shipments was reviewed to determine a realistic estimate of the involved, the amount of time required, and the distances between individuals and the The number of hours worked ranged from 1 to 10 and the distance from the container

91 meters (5 to 300 feet). For simplicity, weighted averages of the number of hour shipping container were calculated and are provided in Table A-10.
 Table A-10. Summary of the number of people involved and distance from the contain heavy-lift transporter shipments to the rail siding at the prototype sites.

| Prototype | Number of People | Number of Hours per Worker | Distance the S (meters) |
|-----------------|------------------|-------------------------------|-------------------------------|
| Windsor Site | 37 | 5.08 | 25.0 |
| Kesselring Site | 36 | 5.11 | 32.3 |

This information was used to evaluate all five alternatives.

A.7.1.1.12 Time to Ship by Heavy-lift Transporter. Based on discussions with personnel at the

prototype facilities who have made shipments and a review of records, the average d lift transporter shipment from the prototype sites to the local rail siding is 2 ho

A.7.1.1.13 Number of Heavy-lift Transporter Inspections. The shipments are inspected prior to

leaving the prototype's site boundaries, and no additional inspections are performe lift transporter shipment. As a result, there are no inspections during the heavy- the evaluation of the five alternatives.

A.7.1.1.14 Heavy-lift Transporter Stop Time. Shipments of spent nuclear fuel from the two

prototype locations are first transported by heavy-lift transporter to the nearest from records of naval spent nuclear fuel shipments was reviewed to determine a real heavy-lift transporter stop times. For naval spent nuclear fuel heavy-lift transpo Windsor Site, a heavy-lift transporter stop time of 24 hours was used. For heavy- from the Kenneth A. Kesselring Site, a stop time of 10 hours was used. The heavy-l shipments from the prototypes to the rail sidings occur through suburban population lift transporter stop times were used to evaluate all five alternatives.

A.7.1.1.15 Standard RADTRAN 4 Computer Code Values Used. The following standard RADTRAN 4

computer code value was reviewed and determined to reflect the best estimate of cur practice:

- Number of Inspections of the Shipping Container and Railcar.

The following standard RADTRAN 4 computer code estimates of the populations t affected by the shipment of spent nuclear fuel were also used for the five alterna- tives:

- Number of People per Vehicle Sharing the Transport Route (On Link)
- Traffic Count Passing a Specific Point - Rural, Suburban, and Urban Zones
- Average Exposure Distance When Stopped
- Persons Exposed While Stopped
- Fraction of Travel During Rush Hour, on City Streets, and on Freeways.

A.7.1.1.16 Number of Ship Inspections. Shipments of spent nuclear fuel from Pearl Harbor

Naval Shipyard must first be transported by ship to the Puget Sound Naval Shipyard. values in the RADTRAN 4 computer code, the radiological exposures to the crew and g are negligible since the distances from these individuals to the shipping container the radiological exposure estimates are only expected to occur during inspections. monitoring results for past naval spent nuclear fuel shipments, this is not realist fuel, and a separate calculational model was developed to account for this potentia

The model uses the standard point source formula (see Section A.5.1) to calculate t ment escort exposures during transport by ship. The model took into account the sh index, transport time, distance between shipping containers, distance from the ship living quarters, distance from the shipping containers and the engine room, the num and government escorts, and the time required for inspections based on records from of spent nuclear fuel. After reviewing historical shipment records, it was determi sized ships have recently been used. The smallest one, Ship 1, was used once and i used in the future. Only the other two, Ships 2 and 3, would be used in the future Table A-11 below provides the information used to calculate the radiological exposu transporting naval spent nuclear fuel by ship. This model was used to evaluate all Table A-11. Parameters used to calculate crew and escort exposure during ocean tra Pearl Harbor Naval Shipyard to Puget Sound Naval Shipyard.

| Parameter | Ship 1 | Ship |
|--|---|------|
| Transport Time, T, in days | 11 | 8 |
| Separation Between M-130s, X1, in feet | 92 | 43 |
| Nearest Distance to Living Quarters, X2, in feet | 40 | 80 |
| Nearest Distance to Engine Room, X3, in feet | 20 | 80 |
| Number of Crew Members, Nc | 11 | 22 |
| Number of Government Escorts (not part of crew size), Ne | 2 | 2 |
| Escort Inspection Time (per Escort), in hr/day | 0.50 for histo 0.25 for futur (1/3) for gamm every 40-foot centerline | |
| Shielding Factor | | |

A.7.1.2 Accident During Transportation of Spent Nuclear Fuel. This section provides the input

parameters used to calculate the radiological impacts for accidents during transpor fuel for evaluation of the five alternatives. The planned shipments, transportatio densities, and the percentages of travel in each population density described in Se used for the accident analyses. Unless otherwise described in this section, the st the RADTRAN 4 and RISKIND computer codes were used.

A.7.1.2.1 Accident Probability. The probability of a rail accident used for evaluation of all

alternatives was obtained from "Longitudinal Review of State-Level Accident Statist Interstate Freight" (Saricks and Kvitek 1994). The probabilities are provided both average. The state dependent probabilities were used for the accident risk assessm nuclear fuel shipments have traveled approximately 2 million kilometers (1.24 milli without an accident, which is consistent with the national average of 5.57×10^8 a

A.7.1.2.2 Accident Severity Categories and Probabilities. In the "Shipping Container

Response to Severe Highway and Railway Accident Conditions" (NUREG 1987), referred "Modal Study," Lawrence Livermore National Laboratory categorized the potential dam containers according to the magnitude of the thermal and mechanical forces that cou accident. The structural and thermal forces were categorized into 20 regions. Giv occurs, the probability that the accident would be in each region was calculated fo shipments. Table A-12 provides the probabilities for rail accidents by region.

Table A-12. Accident severity probabilities for rail shipments.

| | | | | |
|-------|-------------------------|------------------------|------------------------|------------------------|
| | R(4,1) | R(4,2) | R(4,3) | R(4,4) |
| | 1.786×10^9 | 3.290×10^{13} | 2.137×10^{13} | 1.644×10^{11} |
| S3 | R(3,1) | R(3,2) | R(3,3) | R(3,4) |
| (30) | 5.545×10^4 | 1.0217×10^7 | 0.634×10^8 | 5.162×10^8 |
| S2 | R(2,1) | R(2,2) | R(2,3) | R(2,4) |
| (2) | 2.7204×10^{35} | 0.11×10^7 | 3.255×10^7 | 2.531×10^7 |
| S1 | R(1,1) | R(1,2) | R(1,3) | R(1,4) |
| (0.2) | 0.993962 | 1.2275×10^3 | 7.9511×10^4 | 6.140×10^4 |
| | | | | T4 (1050) |

Thermal Response (lead mid-thickness temperature, yF)

A.7.1.2.3 Naval Spent Nuclear Fuel Integrity Following an Accident. Detailed structural and

thermal analyses were performed for the shipping containers used for naval spent nuclear fuel up to an equivalent strain of 30% and mid-wall temperature of 1050yF. For these conditions, the naval spent nuclear fuel was not damaged. For the thermal and structural regions above 1050yF, the modal study defines the upper limits as unbounded. The naval spent nuclear fuel was not damaged and the fission products and corrosion products would be released in the quantities specified in Table A-13 for the risk analyses.

A.7.1.2.4 Release Fractions. The release fractions were derived based on the results presented

in the NRC modal study (NUREG 1987) and the results of the structural and thermal analyses presented above. Although the naval spent nuclear fuel is stronger, the release fractions for the modal study, the release fraction in lower left region R(1,1) is zero for the maximum consequence evaluation, 1% of the corrosion products might be released for region, R(1,1). Based on the results of the structural and thermal analyses up to mid-wall temperature, the naval spent nuclear fuel is not damaged; therefore, regions R(2,1), R(2,2), R(2,3), R(1,4), R(2,4), R(3,4), R(3,1), R(3,2) and R(3,3) do not release ten percent of the corrosion products might be released. In the remaining regions, products might be available for release and released at the fractions specified below 10% of the corrosion products. Table A-13 provides the release fractions used. Table A-13. Cask release fractions used for the RADTRAN 4 risk analyses.

Release Fractions^a

| Cask Response Region | Inert Gas | Iodine | Cesium | Ruthenium |
|------------------------|------------|------------|------------|------------|
| R(1,1) | 0.0 | 0.0 | 0.0 | 0.0 |
| R(1,2), R(1,3) | 0.0 | 0.0 | 0.0 | 0.0 |
| R(2,1), R(2,2), R(2,3) | 0.0 | 0.0 | 0.0 | 0.0 |
| R(1,4), R(2,4), R(3,4) | 0.0 | 0.0 | 0.0 | 0.0 |
| R(3,1), R(3,2), R(3,3) | 0.0 | 0.0 | 0.0 | 0.0 |
| R(1,5), R(2,5), R(3,5) | 6.3 x 10y1 | 4.3 x 10y2 | 2.0 x 10y3 | 4.8 x 10y4 |
| R(4,5), R(4,1), R(4,2) | | | | |
| R(4,3), R(4,4) | | | | |

^a The release fraction represents the fraction of the fuel inventory available for release that would be released into the atmosphere following an accident of the given severity.

A.7.1.2.5 Plume Release Height. For the accident risk assessment, a ground level release was

used. For the maximum consequence assessment, a plume release height of 10 meters was used.

A.7.1.2.6 Direct Exposure from a Damaged Shipping Container. A radiation level following the

accident at the 10CFR71 regulatory limit of 1 rem at 1 meter (3.3 feet) from the container was used.

A.7.1.2.7 Food Transfer Factors. Food transfer factors were derived for the isotopes related to

naval spent nuclear fuel in accordance with the methods described in Nuclear Regulatory Guide 1.109 (NUREG 1977).

A.7.1.2.8 Distance from the Accident Scene to the Maximum Exposed Individual. No shielding

was accounted for as the plume passes for the calculation of the exposure to the maximum exposed individual. The location was determined using RISKIND based on the atmospheric stability and plume characteristics. The maximum exposed individual could be a member of the rail crew or the general public.

A.7.1.2.9 RISKIND Population Density. The standard national average for each population

density from the RADTRAN 4 computer code was used for the RISKIND maximum consequence assessment (6 people per square kilometer for rural, 719 for suburban, and 3861 for

A.7.1.2.10 Radionuclide Inventory. The amount of radionuclides which would be released from

an average shipment are provided in Table A-14. The values factor in the damage for Section A.7.1.2.3 and release fractions described in Section A.7.1.2.4. The radionuclides 99 percent of the exposure in all pathways. Table A-14. Radionuclides which would be released from an average shipment of naval nuclear fuel from a shipyard or prototype.

| For Accidents which Release Both Fission and Corrosion Products | | For Accidents which Release Corrosion Products |
|--|------------------------|---|
| Nuclide | Activity (Ci) | Nuclide |
| Kr-85 | 9.85 x 10 ² | Co-58 |
| Cs-134 | 3.72 x 10 ¹ | Mn-54 |
| Cs-137 | 3.44 x 10 ¹ | Fe-55 |
| H-3 | 1.39 x 10 ¹ | Co-60 |
| Ru-106 | 9.02 x 10 ¹ | Sr-90 |
| Ce-144 | 4.89 x 10 ¹ | Ni-63 |
| Co-60 | 3.63 x 10 ¹ | |
| Sr-90 | 3.41 x 10 ¹ | |
| Pu-238 | 1.02 x 10 ² | |
| Pu-241 | 3.43 x 10 ³ | |
| Cm-244 | 1.36 x 10 ⁴ | |

A.7.2 Transfers of Naval Spent Nuclear Fuel to Storage Following

Examination

A.7.2.1 Incident-free Transportation of Naval Spent Nuclear Fuel to Storage. This section provides the

input parameters used to determine the radiological impacts associated with the routine (no accident) transportation of naval spent nuclear fuel to storage for each of the

A.7.2.1.1 Planned Shipments. Table A-15 provides the number of planned transfers in each cask.

Table A-15. Planned transfers of naval spent nuclear fuel to storage.

| | NFS-100 | Peach Bottom | Large Cell |
|-----------------------------------|---------|--------------|------------|
| No Action, | 0 | 0 | 15 |
| Decentralization - No Exam, | | | |
| Decentralization - Limited Exam | | | |
| Decentralization - Full Exam | | 0 | 14 |
| 1992/1993 Planning Basis, 196 | | 64 | 468 |
| All Regionalization Alternatives, | | | |
| All Centralization Alternatives | | | |

A.7.2.1.2 Transport Index (TI). A TI of 0.3 was used for all NFS-100 cask transfers. This value

was determined from recorded measurements over the last 3 years for the same fuel transferred in the future. The Peach Bottom and Large Cell casks have not previous planned transfers and therefore historic data were not available. Based on a comparison of values from conservative safety analyses to the actual measured TI's for similar casks of 1.0 was calculated for both the Peach Bottom and Large Cell casks.

A.7.2.1.3 Transportation Distances and Population Densities. Section A.6 provided a

description of the general methodology used for determining transportation distance densities along the transportation routes. The distance between ECF and ICPP is 9. From the HIGHWAY computer code, the transfer of naval spent nuclear fuel to storage area. As stated in Section A.3.5, the storage facility at the alternative sites was. Therefore, for the evaluation of the alternatives, the distance traveled and population ICPP transfer were also used for the evaluation of the other alternatives.

A.7.2.1.4 Truck Speed. The standard RADTRAN 4 computer code speed for truck shipments in a

rural population is 88.5 kilometers per hour (55 miles per hour). One of the reasonable accident is less severe than the 10CFR71 hypothetical accident is that the by the on-site transportation procedures. An average speed of 24.1 kilometers per hour was used.

A.7.2.1.5 Truck Stop Time. The standard RADTRAN 4 computer code provides values for truck

stop times that are either dependent or independent of the distances traveled. The transfers of naval spent nuclear fuel to storage were reviewed, and it was determined times (10 minutes) were much shorter than the standard RADTRAN 4 computer code value of 10 minutes was used to evaluate all five alternatives.

A.7.2.1.6 Radiation Exposure Decrease Due to Distance. The radiation exposure decrease due to

distance described in Section A.7.1.1.9 was also applied to the truck transfers of storage.

A.7.2.1.7 Distance from Source to Crew. A distance of 6.1 meters (20 feet) was measured

between the shipping cask and the driver for the exclusive-use truck transfers of nuclear shipments to storage. Two escorts, one located approximately 46 meters (150 feet) same distance behind the transport vehicle, are also present. These data were used in analyses for all alternatives.

A.7.2.1.8 Transport Index to Exposure Rate Conversion Factors. Transport index to exposure

rate conversion factors for the casks used for transfers of naval spent nuclear fuel calculated using the standard equation in RADTRAN 4. The results were compared to analyses performed using SPAN4, and RADTRAN 4 results were found to overestimate them. Using the SPAN4 computer code results, the effective package dimensions of the cask RADTRAN 4 calculations were adjusted to provide a conservative yet more realistic value index to exposure rate conversion factor. The values used are provided in Table A-Table A-16. Transport index to exposure rate conversion factors for the NFS-100, P and Large Cell casks.

| Cask | Effective Package Dimensions (meters) | Transport Index to Exposure Rate Conversion Factor |
|--------------|--|---|
| NFS-100 | 3.8 (12.5 feet) | 8.41 |
| Peach Bottom | 2.8 (9.2 feet) | 5.76 |
| Large Cell | 3.2 (10.5 feet) | 6.76 |

A.7.2.1.9 Storage. There is no intermediate storage time during transfers of naval spent nuclear

fuel to its destination.

A.7.2.1.10 Persons Exposed While Stopped. The only stop time for the transfer of naval spent

nuclear fuel to storage occurs during routine surveys at the destination entrance. from highway and general population and therefore no people were considered to be in short 10-minute stop. The escorts are not present during the surveys and the drive

the truck, 6.1 meters (20 feet) from the cask during the surveys. The people performing the surveys are badged and all exposure received during the surveys is included in the normal occupational exposure which is regularly monitored.

A.7.2.1.11 Traffic Count Passing a Specific Point. The RADTRAN 4 computer code uses 470

vehicles per hour passing the transport vehicle. Travel on the transport path is restricted by a security checkpoint, the majority of INEL employees ride the INEL shuttle. If the transfers are not made during high traffic times (i.e., shift changes when buses are used), therefore, using the standard 470 vehicles per hour value would be extremely conservative. A realistic estimate of 25 vehicles per hour was used.

A.7.2.1.12 Standard RADTRAN 4 Computer Code Values Used. The following standard RADTRAN 4

computer code value was reviewed and determined to reflect the best estimate of current conditions and was consistent with historical data from transfers of naval spent nuclear fuel to storage.

- Minimum Number of Inspections.

The following standard RADTRAN 4 estimate of the population that could be affected by the transfer of naval spent nuclear fuel to storage was used to evaluate the five alternatives:

- Number of People per Vehicle Sharing the Transport Route (On Link).

A.7.2.2 Accident During Transportation of Spent Nuclear Fuel to Storage. This section provides the

input parameters used to calculate the radiological impacts for accidents during transport of naval spent nuclear fuel to storage for evaluation of the five alternatives. The planned transport distances, population densities, and the percentages of travel in each population density category from Section A.7.2.1 were also used for the accident analyses. Unless otherwise described, the standard values provided by the RADTRAN 4 and RISKIND computer codes were used.

A.7.2.2.1 Accident Probability. The probability of a truck accident used for evaluation of all

alternatives was obtained from "Longitudinal Review of State-Level Accident Statistics for Interstate Freight" (Saricks and Kvitek 1994). The truck accident rates are state specific. The probability of a truck accident used for evaluation of the alternatives which involve transfer of naval spent nuclear fuel to storage for the alternatives are Idaho, Washington, South Carolina, Tennessee, and Nevada. The corresponding accident rates on rural interstates in accidents per kilometer are 2.30×10^{-7} for Idaho, 1.83×10^{-7} for South Carolina, 1.48×10^{-7} for Tennessee, and 1.57×10^{-7} for Nevada. These correspond to 3.70×10^{-7} (Idaho), 4.02×10^{-7} (Washington), 2.94×10^{-7} (South Carolina), and 2.53×10^{-7} (Nevada) accidents per mile.

A.7.2.2.2 Accident Severity Categories and Probabilities. In the modal study, Lawrence

Livermore National Laboratory categorized the potential damage to shipping containers by the magnitude of the thermal and mechanical forces that could result from an accident. The thermal forces were categorized into 20 regions. Given that an accident occurs, the probability of an accident occurring in each region was calculated for both rail and truck shipments. The probabilities for truck accidents by region.

Table A-17. Accident severity probabilities for truck shipments.

| | R(4,1) | R(4,2) | R(4,3) | R |
|-------|-------------------------|-------------------------|------------------------|--------|
| | 1.532×10^{-7} | 3.926×10^{-14} | 1.495×10^{-1} | 7 |
| S3 | R(3,1) | R(3,2) | R(3,3) | |
| (30) | 1.7984×10^{-3} | 1.574×10^{-7} | 2.034×10^{-7} | 1.076 |
| S2 | R(2,1) | R(2,2) | R(2,3) | R(2,4) |
| (2) | 3.8192×10^{-3} | 2.330×10^{-7} | 3.008×10^{-7} | 1.592 |
| S1 | R(1,1) | R(1,2) | R(1,3) | R(1,4) |
| (0.2) | 0.994316 | 1.687×10^{-5} | 2.362×10^{-5} | 1.525 |
| | T1 | T2 | T3 | |
| | (500) | (600) | (650) | |

Thermal Response (lead mid-thickness temperature, yF)

A.7.2.2.3 Naval Spent Nuclear Fuel Integrity Following an Accident. Detailed structural and

thermal analyses have been performed for the casks used for shipments of naval spent storage. As described in Section A.4.5, these analyses are performed using a worst is defined based on the site specific terrain and administrative controls during th The probability of the worst credible accident is equal to that listed in region R(conditions in excess of the worst credible accident, the fission product and corros fractions described in the next section were used.

A.7.2.2.4 Cask Release Fractions. The cask release fractions were derived based on the results

presented in the NRC modal study (NUREG 1987). Although the naval spent nuclear fu release fractions for the BWR, PWR, and aluminum-clad fuel from the modal study wer modal study, the release fraction for lower left region R(1,1) is zero for the risk maximum consequence evaluation, 1% of the corrosion products were released for the R(1,1). The remaining regions used 10% of the fission products available for relea fractions specified below, and release of 10% of the corrosion products. Table A-1 fractions used. The release fractions in Table A-18 for the less severe conditions A-13 because supplementary structural and thermal analyses have not been performed discussed in this section.

Table A-18. Cask release fractions used for the RADTRAN 4 risk analyses.
Release Fraction^a

| Cask Response Region | Inert Gas | Iodine | Cesium | |
|------------------------|-------------------|-------------------|-------------------|--------------|
| R(1,1) | 0.0 | 0.0 | 0.0 | 0.0 |
| R(1,2), R(1,3) | $9.9 \times 10y3$ | $7.5 \times 10y5$ | $6.0 \times 10y6$ | $8.1 \times$ |
| R(2,1), R(2,2), R(2,3) | $3.3 \times 10y2$ | $2.5 \times 10y$ | $4.2 \times 10y5$ | $2.7 \times$ |
| R(1,4), R(2,4), R(3,4) | $3.9 \times 10y1$ | $4.3 \times 10y3$ | $2.0 \times 10y4$ | $4.8 \times$ |
| R(3,1), R(3,2), R(3,3) | $3.3 \times 10y1$ | $2.5 \times 10y3$ | $2.0 \times 10y4$ | $2.7 \times$ |
| R(1,5), R(2,5), R(3,5) | $6.3 \times 10y1$ | $4.3 \times 10y2$ | $2.0 \times 10y3$ | $4.8 \times$ |
| R(4,5), R(4,1), R(4,2) | | | | |
| R(4,3), R(4,4) | | | | |

^a The release fraction represents the fraction of the fuel inventory available for released into the atmosphere following an accident of the given severity.

A.7.2.2.5 Plume Release Height. For the accident risk assessment, a ground level release was

used. For the maximum consequence assessment, a plume release height of 10 meters used.

A.7.2.2.6 Direct Exposure from a Damaged Shipping Container. A radiation level following the

accident at the 10CFR71 regulatory limit of 1 rem at 1 meter (3.3 feet) from the ca

A.7.2.2.7 Food Transfer Factors. Food transfer factors were derived for the isotopes related to

naval spent nuclear fuel in accordance with the methods described in Nuclear Regula Guide 1.109 (NUREG 1977).

A.7.2.2.8 Distance from the Accident Scene to the Maximum Exposed Individual. No shielding

was accounted for as the plume passes for the calculation of the exposure to the ma location was determined using RISKIND based on the selected atmospheric stability a height. The maximum exposed individual could be a member of the track crew or the

A.7.2.2.9 RISKIND Population Density. From the HIGHWAY computer code, the population

density for the on-site shipment was determined to be one person per square kilometer (one person per square mile) in a rural area. For on-site transportation at INEL, the population density of the populated sector, from 1990 census data, is 55 people per square kilometer, with the population density in the area 64.4 to 80 kilometers (40 to 50 miles) from the site. This population density is the lower region of the suburban density range of 53.7 to 1284.7 people per square kilometer (one person per square mile) used in HIGHWAY and INTERLINE. The standard value of 6 (rural) people per square kilometer (15.5 and 1861 people per square mile, respectively) was used in the evaluation of all alternatives.

A.7.2.2.10 Radionuclide Inventory. The transfers of naval spent nuclear fuel to storage contain

the same radionuclides as listed in Table A-14. On average, there is approximately equal activity of each radionuclide.

A.7.3 Transfers of Naval Test Specimen Assemblies Between the

Examination Facility and the Test Reactor Area

A.7.3.1 Incident-free Transportation of Naval Test Specimen Assemblies. This section provides the

input parameters used to determine the radiological impacts associated with the routine (no accident) transportation of naval test specimen assemblies for each of the five

A.7.3.1.1 Planned Shipments. Table A-19 provides the number of planned transfers in each cask.

Table A-19. Planned transfers of naval test specimen assemblies.

| | NR/ATR | Test Train |
|---------------------------------|--------|------------|
| No Action, | 0 | 0 |
| Decentralization - No Exam, | | |
| Decentralization - Limited Exam | | |
| Decentralization - Full Exam | 38 | 922 |
| 1992/1993 Planning Basis, | | |
| Regionalization at INEL, and | | |
| Centralization at INEL | | |
| All other Regionalization and | 0 | 960 |
| Centralization Alternatives | | |

A.7.3.1.2 Transport Index. A TI of 130.0 was used for all NR and ATR cask transfers. This

value was derived from historic measurements over the last several years. The new facilities which are currently being designed, would have a TI of 1.0.

A.7.3.1.3 Transportation Distances and Population Densities. Section A.6 provided a

description of the general methodology used for determining transportation distance and population densities along the transportation routes. The distance between ECF and TRA is 8.0 kilometers. From the HIGHWAY computer code, this on-site transfer of naval test specimen assemblies is in a rural area. For shipments from TRA to the centralization sites, the HIGHWAY computer code calculates the distance traveled, the population densities, and the percent distance traveled. As described in Section A.7.4.1.3, the HIGHWAY predicted distances for shipments were increased by 3%.

A.7.3.1.4 Truck Speed. The standard RADTRAN 4 computer code speed for truck shipments in a

rural population is 88.5 kilometers per hour (55 miles per hour). One of the reasonable accident scenarios is less severe than the 10CFR71 hypothetical accident is that the speed is limited. An average speed of 16.1 kilometers per hour (10 miles per hour) was used

shipments. For off-site shipments to the centralization sites, the standard RADTRA values were used.

A.7.3.1.5 Truck Stop Time. The standard RADTRAN 4 computer code provides values for truck

stop times that are either dependent or independent of the distances traveled. The transfers of naval test specimen assemblies were reviewed, and it was determined that (one and one-half hours) was less than the standard RADTRAN 4 computer code values. alternative in which on-site transfers would continue, the one and one-half hour stop times for the off-site shipments of test specimen assemblies to the centralization sites, a stop time of 0.01 hour per mile was used, consistent with the value used for other projects outside the boundaries of DOE facilities (see Section A.7.4.1.4).

A.7.3.1.6 Radiation Exposure Decrease Due to Distance. The radiation exposure decrease due to

distance described in Section A.7.1.1.9 was also applied to the truck transfers of

A.7.3.1.7 Distance from Source to Crew. A distance of 3.6 meters (12 feet) was measured

between the NR/ATR shipping cask and the driver for the exclusive-use truck transfers of test specimen assemblies on-site. Two escorts, one located approximately 46 meters (150 feet) in distance behind the transport vehicle, are also present for on-site shipments.

For off-site shipments to the centralization sites, the standard RADTRAN 4 code for the number of crew members was used (2). The value used for the distance from centerline of the cask for off-site shipments was 5.85 meters (20 feet), based on the new Test Train cask.

A.7.3.1.8 Transport Index to Exposure Rate Conversion Factors. Transport index to exposure

rate conversion factors for the casks used for test specimen assembly transfers were standard equation used by RADTRAN 4. The results were compared to detailed computations performed using SPAN4, and RADTRAN 4 results were found to overestimate the exposure rate. SPAN4 computer code results, the effective package dimensions of the casks used in calculations were adjusted to provide a conservative yet more realistic value of the exposure rate conversion factor. The values used are provided in Table A-20.

Table A-20. Transport index to exposure rate conversion factors for the NR/ATR and casks.

| Cask | Effective Package Dimension (meters) | Transport Index to Exposure Rate Conversion Factor |
|------------|---|---|
| NR/ATR | 0.61 (2 feet) | 1.70 |
| Test Train | 1.70 (5.6 feet) | 3.42 |

A.7.3.1.9 Storage. There is no intermediate storage time during transfers of naval test specimen

assemblies.

A.7.3.1.10 Persons Exposed While Stopped. The only stop time for the transfer of naval test

specimen assemblies on-site occurs during routine surveys at the destination entrance. The assemblies are removed from highway and population and therefore no people were considered to be exposed during the one and one-half hour stop. The escorts are not present during the surveys and the approximately 46 meters (150 feet) from the source during the surveys. The people are badged and all exposure received during the survey is included in the normal occupancy which is regularly monitored. For off-site shipments, the standard RADTRAN 4 computer code was used.

A.7.3.1.11 Traffic Count Passing a Specific Point. The RADTRAN 4 computer code uses 470

vehicles per hour passing the transport vehicle. Travel on the on-site transport path

employees, the majority of INEL employees ride the INEL site buses to work, and the made during high traffic times (i.e., shift changes); therefore, using the standard value would excessively overestimate the number of persons involved. A more realistic vehicles per hour was used for on-site shipments. For off-site shipments, the standard computer code values were used.

A.7.3.1.12 Standard RADTRAN 4 Computer Code Values Used. The following standard RADTRAN 4

computer code value was reviewed and determined to reflect the best estimate of current and was consistent with recorded data from transfers of naval test specimen assemblies.

- Minimum Number of Inspections.

The following standard RADTRAN 4 estimate of the population that could be affected by the transfer of test specimen assemblies was used for evaluation of the five alternatives:

- Number of People per Vehicle Sharing the Transport Route (On Link).

A.7.3.2 Accident During Transportation of Naval Test Specimen Assemblies. This section provides the

input parameters used to calculate the radiological impacts for accidents during test specimen assemblies for evaluation of the five alternatives. The planned transfer distances, population densities, and the percentages of travel in each population density. Section A.7.3.1 were also used for the accident analyses. Unless otherwise described, standard values provided by the RADTRAN 4 and RISKIND computer codes were used. All described in Section A.7.2.2 are applicable to these transfers with the exception of population density.

A.7.3.2.1 RISKIND Population Densities. For the Decentralization, 1992/1993 Planning Basis,

Regionalization at INEL, and Centralization at INEL alternatives, the test specimen would occur on the INEL site. For these transfers, the same conditions described in Section A.7.2.2 were used. For the other Regionalization and Centralization alternative risk assessment, standard population densities from RADTRAN 4 were used.

A.7.3.2.2 Release Fractions. For the Decentralization, 1992/1993 Planning Basis, and

Regionalization at INEL, and Centralization at INEL alternatives, the test specimen would occur on the INEL site. For these transfers, the same conditions described in Section A.7.2.2 were used.

A.7.2.2.4 were used. For the other Regionalization and Centralization alternatives, the conditions

described in Sections A.7.1.2.3 and A.7.1.2.4 were used.

A.7.3.2.3 Radionuclide Inventory. The radionuclides which would be released from an average

transfer are listed in Table A-21, along with the activity. The values factor in the release fractions described in Section A.7.3.2.2. The radionuclides listed result in exposure in each pathway.

Table A-21. Radionuclides which would be released from an average transfer of test assemblies.

| For Accidents which Release Both Fission and Corrosion Products | | For Accidents which Release Corrosion Products | |
|--|------------------------|---|--|
| Nuclide | Activity (Ci) | Nuclide | |
| I-131 | 1.30 x 10 ³ | Eu-156 | |
| H-3 | 3.51 x 10 ² | Lu-177 | |
| I-132 | 3.10 x 10 ² | Eu-152 | |
| Eu-156 | 3.75 x 10 ¹ | Zr-95 | |
| Eu-152 | 1.41 x 10 ¹ | Zn-65 | |
| Zr-95 | 1.09 x 10 ¹ | Co-60 | |
| Zn-65 | 9.80 x 10 ⁰ | Ce-141 | |
| Co-60 | 7.68 x 10 ⁰ | Eu-154 | |

| | | |
|--------|-------------|--------|
| Eu-154 | 6.15 x 100 | Cs-136 |
| Sc-46 | 3.25 x 100 | Sc-46 |
| Cs-137 | 1.78 x 100 | I-131 |
| Ru-106 | 3.36 x 10y1 | Hf-181 |
| Nb-95 | 2.64 x 10y1 | |
| Pr-144 | 2.19 x 10y1 | |
| Ce-144 | 2.19 x 10y1 | |

A.7.4 Shipments of Naval Irradiated Test Specimens to Examination

and Testing Facilities

A.7.4.1 Incident-free Transportation of Test Specimens. This section provides the input parameters

used to determine the radiological impacts associated with the routine, incident-free transportation of test specimens for evaluation of the five alternatives.

A.7.4.1.1 Planned Shipments. Table A-22 provides the estimated number of shipments used in

the analysis.

Table A-22. Planned shipments of naval test specimens.
NRBK-41/WAPD-40

| Alternative | Centralization | | |
|--|----------------|------|------|
| | ICPP | PSNS | Site |
| No Action | 29 | 0 | 0 |
| Decentralization - No Exam | | | |
| Decentralization - Limited Exam | 26 | 3 | 0 |
| Decentralization - Full Exam | 0 | 0 | 0 |
| 1992/1993 Planning Basis, Regionalization at INEL, and Centralization at INEL Alternatives | 0 | 0 | 0 |
| All other Regionalization and Centralization Alternatives | 0 | 0 | 29 |

A.7.4.1.2 Transport Index. A TI of 0.1 was used for all NRBK-41 and WAPD-40 shipping

container shipments. These values were derived from recorded measurements over the

A.7.4.1.3 Transportation Distances and Population Densities. Section A.6 provided a

description of the general methodology used for determining transportation distance densities along the transportation routes. Historical data were obtained for shipment. The distance traveled was averaged based on the point of origin and compared to the HIGHWAY. The actual distance traveled was approximately 3% higher on the average. provide the best estimate exposure, which is based on the distance traveled, the HI increased by 3% for all alternatives. This technique allowed for comparison of the basis. The percentages of distance traveled in each population density calculated to the distances which were increased by the 3%.

A.7.4.1.4 Truck Stop Time. The RADTRAN 4 computer code provides standard values for truck

stop times that are either dependent or independent of the distances traveled. The historical test specimen shipments were reviewed, and it was determined that the actual stop times were much shorter than the standard RADTRAN 4 computer code values. The recorded stop time divided by the actual distance traveled from historical data over the last three years per hour per kilometer (0.01 hour per mile) was calculated. This value was used to evaluate alternatives.

A.7.4.1.5 Radiation Exposure Decrease Due to Distance. The radiation exposure decrease due to

distance described in Section A.7.1.1.9 was also applied to the truck shipments of

A.7.4.1.6 Transport Index to Exposure Rate Conversion Factors. Container transport index to

exposure rate conversion factors for the casks used for test specimen shipments were standard equation used by RADTRAN 4. The results were compared to detailed computations performed using SPAN4, and RADTRAN 4 results were found to overestimate the exposure rate conversion factors. The effective package dimensions of the containers used in the SPAN4 computer code results, the effective package dimensions of the containers used in the RADTRAN 4 calculations were adjusted to provide a conservative yet more realistic value of exposure rate conversion factor. The values used are provided in Table A-23. Table A-23. Transport index to exposure rate conversion factors for the NRBK-41 and WAPD-40 shipping containers.

| Container | Effective Package Dimensions (meters) | Transport Index to Exposure Rate Conversion Factor |
|-----------|--|---|
| NRBK-41 | 0.74 (2.4 feet) | 1.88 |
| WAPD-40 | 3.2 (10.5 feet) | 6.76 |

A.7.4.1.7 Storage. The test specimen shipping containers are not stored during shipment.

A.7.4.1.8 Standard RADTRAN 4 Computer Code Values Used. The following standard RADTRAN 4

computer code values were reviewed and were determined to reflect the best estimate practice and were consistent with historical data from shipments of naval test specimens.

- Truck Speed
- Distance from Source to Crew
- Number of Crewmen
- Minimum Number of Inspections.

The following standard RADTRAN 4 estimates of the populations that could be a shipment of test specimens were also used to evaluate the five alternatives:

- Persons Exposed While Stopped
- Average Exposure Distance While Stopped
- Number of People per Vehicle Sharing the Transport Route (On Link)
- Traffic Count Passing a Specific Point - Rural, Suburban, and Urban Zones
- Fraction of Travel During Rush Hour, on City Streets, and on Freeways.

A.7.4.2 Accident During Transportation of Test Specimens. This section provides the input parameters

used to calculate the radiological impacts for accidents during transportation of the five alternatives. The planned shipments, transportation distances, population percentages of travel in each population density described in Section A.7.4.1 were used in the accident analyses. Unless otherwise described in this section, the standard values for RADTRAN 4 and RISKIND computer codes were used. All the conditions and variables described in Section A.7.1.2 are applicable to these shipments with the exception of the Accident

A.7.4.2.1 Accident Probability. The probability of a truck accident used for evaluation of all

alternatives was obtained from "Longitudinal Review of State-Level Accident Statistics for Interstate Freight" (Saricks and Kvitek 1994). The truck accident rates are state specific and were obtained from HIGHWAY. The accident rate values are consistent with past test results which have traveled approximately 2.4 million kilometers (1.5 million miles) without

A.7.4.2.2 Test Specimen Integrity Following an Accident. Detailed structural and thermal

analyses were performed for the shipping containers used for naval test specimen shipments. The equivalent strain of 30% and mid-wall temperature of 1050°F. For these cases, the container was not damaged; therefore, only the activity on the outside of the inner container corrosion products, was released. For the thermal and structural regions above 105

modal study defines the upper limits as unbounded. For these cases, the sealed inn test specimens was postulated to be damaged and the fission products and corrosion released in the quantities described in Section A.7.1.2.4.

A.7.4.2.3 Radionuclide Inventory. The test specimen shipments contain the same radionuclides

as listed in Table A-21. On average, there is approximately 1.5 percent of the act

A.8 SUMMARY OF RESULTS

A.8.1 Historical - Incident Free

This section summarizes the results of the calculations for the radiological impacts of the incident-free transportation of naval spent nuclear fuel and test sp shows the radiological impact on the general population, transportation workers (oc maximum exposed individual, and the non-radiological impact on all persons. The ra the general population for all historical shipments is 1.95 person-rem, which stati 0.00098 cancer fatalities in the entire population over the 40-year period consider impact on transportation workers for all historical shipments is 16.6 person-rem, w corresponds to 0.0066 cancer fatalities. As can be seen from Table A-24, the radio general population is greatest for the highway transportation of test specimens. I impacts tend to be greater for highway transportation than for rail transportation population and transportation workers are closer to the shipping container in trans maximum exposed individual is a transportation worker, since the workers are closer a longer time than any member of the general population. The maximum exposed indiv is a driver for the trucks transferring test specimen assemblies between ECF and TR modeling approach that the same person drove every shipment for the entire period, total exposure of 7.5 rem over the approximate 40-year period, or about 0.19 rem pe occupationally exposed individuals. By comparison, the maximum exposed individual population received only 0.062 rem over the entire historical period, which is much to the maximum exposed individual transportation worker and corresponds to 0.0016 m year. It should be noted that the majority of the exposure to the transportation w exposed worker is already accounted for since most transportation workers are badge exposure is included with all other exposure they would receive on the job. The ra off-site truck drivers are the only transportation workers who are not badged. The calculated to be only approximately 30% of the total.

The estimated non-radiological fatalities due to vehicle emissions is 0.028 f period.

A.8.2 Incident Free

Table A-25 provides a summary of the annual exposures and risks from incident transportation of naval spent nuclear fuel and test specimens for all alternatives. by dividing the values in Table A-26 by the 40 years evaluated to obtain the averag

The annual radiological impact on the general population ranges from 0.0085 t The general population annual radiological risk ranges from 0.0000043 to 0.00015 fo

The radiological impact on the transportation crew (occupational) ranges from person-rem. The transportation crew annual radiological risk ranges from 0.000015 fatalities.

Table A-25. Summary of annual incident-free impacts during transportation of naval test specimens.

| | General Population | | Occupational | | MEI-General Po | |
|-----------|----------------------------------|--|----------------------------------|--|----------------|--|
| | Collective Dose (person-r em/yr) | Estimated Cancer Fatalities (per year) | Collective Dose (person-rem /yr) | Estimated Cancer Fatalities (per year) | Dose (rem/yr) | |
| No Action | 0.0085 | 4.3 x | 0.038 | 1.5 x | 0.00 | |

| | | | | | |
|--|--------|-----------------------|-------|-----------------------|---------------|
| Decentralization - No Exam | 0.0085 | 10y6 4.3 x 10y6 | 0.038 | 10y5 1.5 x 10y5 | 8 0.0 8 |
| Decentralization - Limited Exam | 0.021 | 1.1 x 10y5 | 0.068 | 2.7 x 10y5 | 0.0 |
| Decentralization - Full Exam | 0.083 | 4.2 x 10y5 | 0.30 | 1.2 x 10y4 | 0.0 |
| 1992-1993 Planning Basis | 0.053 | 2.7 x 10y5 | 0.18 | 7.2 x 10y5 | 0.0 |
| Regionalization or Centralization at INEL | 0.053 | 2.7 x 10y5 | 0.18 | 7.2 x 10y5 | 0.0 |
| Regionalization or Centralization at Hanford | 0.12 | 6.0 x 10y5 | 0.25 | 1.0 x 10y4 | 0.0 |
| Regionalization or Centralization at Savannah River | 0.30 | 1.5 x 10y4 | 0.38 | 1.5 x 10y4 | 0.0 |
| Regionalization or Centralization at Oak Ridge | 0.28 | 1.4 x 10y4 | 0.35 | 1.4 x 10y4 | 0.0 |
| Regionalization or Centralization at Nevada Test Site | 0.15 | 7.5 x 10y5 | 0.28 | 1.1 x 10y4 | 0.0 |

Table A-26. Summary of 40-year cumulative incident-free impacts during transport of nuclear fuel and test specimens.

| | General Population | Occupational | MEI-General Population | | |
|------------------------------------|---------------------------------|--------------------------------|---------------------------------|--------------------------------|---------------|
| | Collective Dose (person-rem) | Estimated Cancer Fatalities | Collective Dose (person-rem) | Estimated Cancer Fatalities | Dose (rem) |
| No Action | 0.34 | 1.7 x 10y4 | 1.5 | 6.0 x 10y4 | 0.039 |
| Decentralization - No Exam | 0.34 | 1.7 x 10y4 | 1.5 | 6.0 x 10y4 | 0.039 |
| Decentralization - Limited Exam | 0.83 | 4.2 x 10y4 | 2.7 | 1.1 x 10y3 | 0.045 |
| Decentralization - Full Exam | 3.3 | 1.7 x 10y3 | 12 | 4.8 x 10y3 | 0.17 |
| 1992-1993 Planning Basis | 2.1 | 1.1 x 10y3 | 7.3 | 2.9 x 10y3 | 0.086 |

| | | | | | | |
|--|-----|---------------|-----|---------------|-------|--------|
| Regionalization or Centralization at INEL | 2.1 | 1.1 x 10y3 | 7.3 | 2.9 x 10y3 | 0.086 | 4 1 |
| Regionalization or Centralization at Hanford | 4.7 | 2.4 x 10y3 | 9.8 | 3.9 x 10y3 | 0.16 | 8 1 |
| Regionalization or Centralization at Savannah River | 12 | 6.0 x 10y3 | 15 | 6.0 x 10y3 | 0.16 | 8 1 |
| Regionalization or Centralization at Oak Ridge | 11 | 5.5 x 10y3 | 14 | 5.6 x 10y3 | 0.16 | 8 1 |
| Regionalization or Centralization at Nevada Test Site | 6.0 | 3.0 x 10y3 | 11 | 4.4 x 10y3 | 0.16 | 8 1 |

For all alternatives, the maximum exposed individual is a transportation work truck shipments. The annual radiological impact on the maximum exposed individual to 0.12 rem. These values were calculated based on the modeling approach that for of shipments described in Sections A.4.2 through A.4.4, the same person would drive maximum exposed individual annual radiological risk ranges from 0.0000035 to 0.0000 fatalities. The annual exposure to the maximum exposed individual of the general p 0.00098 to 0.0043 rem for the various alternatives. The estimated exposure and hea maximum exposed individual for the general population correspond to approximately a than those estimated for the transportation worker.

The annual non-radiological risk ranges from 0.00015 to 0.00093 fatalities.

The summary of exposures and risks from incident-free transportation of naval and test specimens for all alternatives are included in Table A-26 for the 40-year. The radiological impact on the general population ranges from 0.34 to 12 pers population radiological risk for the entire 40-year period ranges from 0.00017 to 0

The radiological impact on the transportation crew (occupational) ranges from rem. The transportation crew radiological risk for the entire 40-year period range for cancer fatalities.

For all alternatives, the maximum exposed individual is a transportation work truck shipments. The radiological impact on the maximum exposed individual ranges rem. These values were calculated based on using the same driver for all shipments categories of shipments described in Sections A.4.2 through A.4.4. The maximum exp radiological risk for the entire 40-year period, 1995 through 2035, ranges from 0.0 cancer fatalities. The exposure to the maximum exposed individual of the general p 0.039 to 0.17 rem for the various alternatives. The estimated exposure and health exposed individual for the general population correspond to approximately a factor estimated for the transportation worker.

The non-radiological risk ranges from 0.0059 to 0.037 fatalities for the enti

There are appreciable differences in exposure to the general population, tran the maximum exposed individual among the various alternatives. Part of these diffe varying number of shipments. For example, for the Decentralization - Full Examinat shipments of naval spent nuclear fuel are shipped to the INEL and then returned to prototypes, thereby doubling the number of shipments. However, the single most imp the differences among the alternatives is the shipment of test specimen assemblies. Decentralization - No Examination, and Decentralization - Limited Examination alter shipments; for the Decentralization - Full Examination, 1992/1993 Planning Basis, R INEL, and Centralization at INEL alternatives, the exposure is minimal since the sh INEL site. However, for the other Regionalization and Centralization alternatives, assemblies would be shipped off-site between the INEL and the alternative sites. W on the casks are low, the number of shipments and the distances involved increase t

on the transportation crew and the general population.

Tables A-27 and A-28 provide the 40-year cumulative incident-free results sep and off-site shipments. For all alternatives, the shipments of naval spent nuclear prototypes and shipments of naval irradiated test specimens are off-site. Likewise spent nuclear fuel to storage following examination are on-site for all alternative test specimen assemblies are off-site for the Regionalization and Centralization at Savannah River, Oak Ridge, and the Nevada Test Site, otherwise they would be on-site.

As described in Section 3.8 of the main body of this Appendix, all alternative use of the existing Expanded Core Facility at INEL would require a transition period for examination and storage of naval spent nuclear fuel were developed. During the approximately 80 shipments from Navy sites to ECF would be needed. These shipments explicitly in the detailed analyses; however, the appropriate number of shipments in alternative during this period is explicitly included, so the range of environmental shipments is bounded. For example, the estimated fatalities for the No Action, Dec Examination, and Decentralization - Limited Examination alternatives would actually the transition shipments were included. The estimated fatalities for the alternative continues to receive shipments would remain the same. For the Regionalization and alternatives at sites other than INEL, the estimated fatalities would

Table A-27. Summary of 40-year cumulative incident-free impacts of on-site transpo

| | General Population | | Occupational | | MEI-General Population | |
|---|------------------------------|-----------------------------|------------------------------|-----------------------------|------------------------|-----------------------------|
| | Collective Dose (person-rem) | Estimated Cancer Fatalities | Collective Dose (person-rem) | Estimated Cancer Fatalities | Dose (rem) | Estimated Cancer Fatalities |
| No Action | 0.00010 | 5.0 x 10 ^{y8} | 0.0018 | 7.2 x 10 ^{y7} | 0.000017 | 8.10 |
| Decentralization - No Exam | 0.00010 | 5.0 x 10 ^{y8} | 0.0018 | 7.2 x 10 ^{y7} | 0.000017 | 8.10 |
| Decentralization - Limited Exam | 0.00010 | 5.0 x 10 ^{y8} | 0.0018 | 7.2 x 10 ^{y7} | 0.000017 | 8.10 |
| Decentralization - Full Exam | 0.013 | 6.5 x 10 ^{y6} | 0.44 | 1.8 x 10 ^{y4} | 0.062 | 3.10 |
| 1992-1993 Planning Basis | 0.015 | 7.5 x 10 ^{y6} | 0.50 | 2.0 x 10 ^{y4} | 0.062 | 3.10 |
| Regionalization or Centralization at INEL | 0.015 | 7.5 x 10 ^{y6} | 0.50 | 2.0 x 10 ^{y4} | 0.062 | 3.10 |
| Regionalization or Centralization at Hanford | 0.0024 | 1.2 x 10 ^{y6} | 0.067 | 2.7 x 10 ^{y5} | 0.000017 | 8.10 |
| Regionalization or Centralization at Savannah River | 0.0024 | 1.2 x 10 ^{y6} | 0.067 | 2.7 x 10 ^{y5} | 0.000017 | 8.10 |
| Regionalization or Centralization at Oak Ridge | 0.0024 | 1.2 x 10 ^{y6} | 0.067 | 2.7 x 10 ^{y5} | 0.000017 | 8.10 |
| Regionalization or Centralization at Nevada | 0.0024 | 1.2 x 10 ^{y6} | 0.067 | 2.7 x 10 ^{y5} | 0.000017 | 8.10 |

Test Site

Table A-28. Summary of 40-year cumulative incident-free impacts of off-site transp
General Population Occupational MEI-General
Population

| | Collectiv e Dose (person-r em) | Estimate d Cancer Fataliti es | Collectiv e Dose (person-r em) | Estimate d Cancer Fataliti es | Dose (rem) | Es Ca Fa s |
|---|---|--|---|--|---------------|---------------------|
| No Action | 0.34 | 1.7 x 10y4 | 1.5 | 6.0 x 10y4 | 0.039 | 2. 10 |
| Decentralizat ion - No Exam | 0.34 | 1.7 x 10y4 | 1.5 | 6.0 x 10y4 | 0.039 | 2. 10 |
| Decentralizat ion - Limited Exam | 0.83 | 4.2 x 10y4 | 2.7 | 1.1 x 10y3 | 0.045 | 2. 10 |
| Decentralizat ion - Full Exam | 3.3 | 1.7 x 10y3 | 11 | 4.4 x 10y3 | 0.17 | 8. 10 |
| 1992-1993 Planning Basis | 2.1 | 1.1 x 10y3 | 6.8 | 2.7 x 10y3 | 0.086 | 4. 10 |
| Regionalizati on or Centralizatio n at INEL | 2.1 | 1.1 x 10y3 | 6.8 | 2.7 x 10y3 | 0.086 | 4. 10 |
| Regionalizati on or Centralizatio n at Hanford | 4.7 | 2.4 x 10y3 | 9.7 | 3.9 x 10y3 | 0.16 | 8. 10 |
| Regionalizati on or Centralizatio n at Savannah River | 12 | 6.0 x 10y3 | 15 | 6.0 x 10y3 | 0.16 | 8. 10 |
| Regionalizati on or Centralizatio n at Oak Ridge | 11 | 5.5 x 10y3 | 14 | 5.6 x 10y3 | 0.16 | 8. 10 |
| Regionalizati on or Centralizatio n at Nevada Test Site | 6.0 | 3.0 x 10y3 | 11 | 4.4 x 10y3 | 0.16 | 8. 10 |

also remain approximately the same since the number of shipments is approximately e
between the east and west coast origins and therefore the total distance traveled i

A.8.3 Accident Risk

This section summarizes the results of the calculations for radiological and
from accidents which could occur during shipments of naval spent nuclear fuel and t
A-29 and A-30 provide the results of the accident risk assessment for each alternat
provided for the general population in terms of exposure and estimated cancer fatal
presented for 50% meteorological conditions, Pasquill Stability Class D. Table A-2
an annual basis and Table A-30 provides the total risks over the entire 40-year per

The annual radiological impact, from Table A-29, on the general population ra
to 0.021 person-rem. These exposures equate to 0.00000011 to 0.000011 estimated ca
non-radiological impacts, the estimated annual fatalities from traffic accidents ra

The cumulative radiological impact, from Table A-30, on the general populatio

0.0082 to 0.84 person-rem. These exposures equate to 0.0000041 to 0.00042 estimate For non-radiological impacts, the estimated fatalities from traffic accidents range

There are appreciable differences in exposure to the general population, tran the maximum exposed individual among the various alternatives. Part of these diffe varying number of shipments. For example, for the Decentralization - Full Examinat shipments of naval spent nuclear fuel are shipped to the INEL and then returned to prototypes, thereby doubling the number of shipments. As in the incident-free asse test specimen assemblies is a large factor. For the No Action, Decentralization - Decentral-

ization - Limited Examination alternatives, there are no shipments; for the Decentr Examination, 1992/1993 Planning Basis, Regionalization at INEL, and Centralization the exposure is minimal since the shipments remain

Table A-29. Summary of annual accident risk for transportation of naval spent nucl test specimens.

| | General Population Collective Dose (person-rem/yr) Class D | Estimated Cancer Fatalities (per year) Class D | Estimated Traffic Fatalities (per year) |
|---|---|--|--|
| No Action | 0.00021 | 1.1×10^7 | 1.2×10^3 |
| Decentralization - No Exam | 0.00021 | 1.1×10^7 | 1.2×10^3 |
| Decentralization - Limit Exam | 0.00043 | 2.2×10^7 | 1.6×10^3 |
| Decentralization - Full 1992/1993 Planning Basis | 0.0028 0.0020 | 1.4×10^6 1.0×10^6 | 2.2×10^2 1.3×10^2 |
| Regionalization or Centralization at INEL | 0.0020 | 1.0×10^6 | 1.3×10^2 |
| Regionalization or Centralization at Hanford | 0.0033 | 1.7×10^6 | 1.3×10^2 |
| Regionalization or Centralization at Savannah River | 0.0210 | 1.1×10^5 | 1.5×10^2 |
| Regionalization or Centralization at Oak Ridge | 0.015 | 7.5×10^6 | 1.4×10^2 |
| Regionalization or Centralization at Nevada Test Site | 0.0070 | 3.5×10^6 | 1.5×10^2 |

Table A-30. Summary of cumulative accident risk over the 40-year period for transp spent nuclear fuel and test specimens.

| | General Populatio Collective Dose (person-rem) Class D | Estimated Cancer Fatalities Class D | Estimated Traffic Fatalities |
|---|---|---|------------------------------------|
| No Action | 0.0082 | 4.1×10^6 | 4.7×10^2 |
| Decentralization - No Exam | 0.0082 | 4.1×10^6 | 4.7×10^2 |
| Decentralization - Limited Exam | 0.017 | 8.5×10^6 | 6.5×10^2 |
| Decentralization - Full Exam | 0.11m | 5.5×10^5 | 8.6×10^1 |
| 1992/1993 Planning Basis | 0.079 | 4.0×10^5 | 5.1×10^1 |
| Regionalization or Centralization at INEL | 0.079 | 4.0×10^5 | 5.1×10^1 |
| Regionalization or Centralization at Hanford | 0.13 | 6.5×10^5 | 5.3×10^1 |
| Regionalization or Centralization at Savannah River | 0.84 | 4.2×10^4 | 6.0×10^1 |
| Regionalization or | 0.61 | 3.1×10^4 | 5.7×10^1 |

Centralization at Oak Ridge
 Regionalization or 0.28
 Centralization at
 Nevada Test Site

1.4 x 10y4

6.1 x 10y1

on the INEL site. However, for the other Regionalization and Centralization altern assemblies would be shipped off-site between the INEL and the alternate sites. Whi on the containers are low, the number of shipments and the distances involved incre impact on the transportation crew and the general population. In addition, the rou important factor. While differences in distance and population densities are impor the Regionalization at Savannah River and Centralization at Savannah River alternat due to the higher accident rates along the route taken and higher food transfer fac through farming states with much higher ingestion rates.

Table A-31 provides the 40-year cumulative risk, separated by on-site and off

As described in Section 3.8 of the main body of this Appendix, a transition p necessary which would require approximately 80 shipments from Navy sites to ECF. T not included explicitly in the detailed analyses; however, the appropriate number o by each alternative during this period is explicitly included, so the range of envi shipments is bounded. The addition of the transition shipments would increase the the No Action, Decentralization - No Examination, and Decentralization - Limited Ex alternatives. Since the accident risk is proportional to the distance traveled, th for these alternatives, which were the lowest of all alternatives. All other alter same. Therefore, incorporating the transition period would actually reduce the dif alternatives from the standpoint of transportation effects.

A.8.4 Accident Maximum Consequences

This section summarizes the results of the calculations of maximum consequenc which could occur during shipments of naval spent nuclear fuel and test specimens. provide the results of the maximum consequence assessment for each alternative. Th consequences are provided for the general population by population area (rural, sub the maximum exposed individual in terms of exposure. The members of the transporta the maximum exposed individual.

Table A-31. Summary of cumulative risk over the 40-year period for transportation fuel and test specimens (on-site/off-site).

| | ON-SITE | | |
|---|------------------------------------|-----------------------------------|------------------------------------|
| | General Population | | |
| | Collective Dose (person-rem) | Estimated Cancer Fatalities | Estimated Traffic Fatalities |
| No Action | 1.3 x 10y6 | 6.5 x 10y10 | 6.8 x 10y6 |
| Decentralization - No Exam | 1.3 x 10y6 | 6.5 x 10y10 | 6.8 x 10y6 |
| Decentralization - Limited Exam | 1.3 x 10y6 | 6.5 x 10y10 | 6.8 x 10y6 |
| Decentralization - Full Exam | 4.1 x 10y5 | 2.1 x 10y8 | 3.2 x 10y4 |
| 1992-1993 Planning Basis | 1.3 x 10y4 | 6.5 x 10y8 | 6.1 x 10y4 |
| Regionalization or Centralization at INEL | 1.3 x 10y4 | 6.5 x 10y8 | 6.1 x 10y4 |
| Regionalization or Centralization at Hanford | 8.7 x 10y5 | 4.4 x 10y8 | 2.1 x 10y4 |
| Regionalization or Centralization at Savannah River | 8.7 x 10y5 | 4.4 x 10y8 | 3.6 x 10y4 |
| Regionalization or Centralization at Oak Ridge | 8.7 x 10y5 | 4.4 x 10y8 | 2.3 x 10y4 |
| Regionalization or Centralization at Nevada Test Site | 8.7 x 10y5 | 4.4 x 10y8 | 1.6 x 10y4 |

Table A-32. Summary of maximum consequences (person-rem) of an accident (Design Basis)
 MAXIMUM CONSEQUENCES
 DESIGN BASIS
 (accident probability between 1 and 1 x 10⁶)

| | Maximum Exposed Individual (rem) | Rural (person-rem) | Suburban (person) |
|---|----------------------------------|--------------------|-------------------|
| No Action | 0.0034 | 0.51 | 4.3 |
| Decentralization - No Exam | 0.0034 | 0.51 | 4.3 |
| Decentralization - Limited Exam | 0.014 | 4.0 | 4.3 1 |
| Decentralization - Full Exam | 0.045 | 7.4 | 25 1 |
| 1992/1993 Planning Basis | 0.045 | 7.4 | 25 |
| Regionalization or Centralization at INEL | 0.045 | 7.4 | 25 |
| Regionalization or Centralization at Hanford | 0.25 | 38 | 100 |
| Regionalization or Centralization at Savannah River | 0.25 | 38 | 320 |
| Regionalization or Centralization at Oak Ridge | 0.25 | 38 | 320 |
| Regionalization or Centralization at Nevada Test Site | 0.25 | 38 | 320 |

Table A-33. Summary of maximum consequences (person-rem) of an accident (Beyond Design Basis).

| | MAXIMUM CONSEQUENCES BEYOND DESIGN BASIS (accident probability between 1 x 10 ⁶ and 1 x 10 ⁷) | | | | |
|---|--|-----------------------------|------------------------------|-----------------------------|---------------------------------------|
| | Maximum Exposed Individual | Rural | | | Su |
| | Estimated Dose (rem) | Estimated Cancer Fatalities | Collective Dose (person-rem) | Estimated Cancer Fatalities | Estimate Collective Dose (person-rem) |
| No Action | 0.014 | 7.0 x 10 ⁶ | 4.0 | 2.0 x 10 ³ | 25 |
| Decentralization No Exam | 0.014 | 7.0 x 10 ⁶ | 4.0 | 2.0 x 10 ³ | |
| Decentralization Limited Exam | 0.045 | 2.3 x 10 ⁵ | 7.4 | 3.7 x 10 ³ | 25 |
| Decentralization Full Exam | 1.8 | 9.0 x 10 ⁴ | 2700 | 1.4 | 3300 |
| 1992/1993 Planning Basis | 2.2 | 1.1 x 10 ³ | 3300 | 1.7 | 4100 |
| Regionalization or Centralization at INEL | 2.2 | 1.1 x 10 ³ | 3300 | 1.7 | 4100 |
| Regionalization or Centralization at Hanford | 2.2 | 1.1 x 10 ³ | 3300 | 1.7 | 41 |
| Regionalization or Centralization at Savannah River | 2.2 | 1.1 x 10 ³ | 3300 | 1.7 | 4100 |
| Regionalization or Centralization at Oak Ridge | 2.2 | 1.1 x 10 ³ | 3300 | 1.7 | 4100 |
| Regionalization or Centralization at Nevada Test Site | 2.2 | 1.1 x 10 ³ | 3300 | 1.7 | 4100 |

For design basis accidents, the calculated exposure to the general population person-rem in a rural area to 560 person-rem in an urban area. The risk associated ranges from 0.00026 to 0.28 cancer fatalities. The exposure to the maximum exposed from 0.0034 rem to 0.25 rem. The risk to the maximum individual ranges from 0.0000 cancer fatalities.

For beyond design basis accidents, the exposure to the general population ranges from 0.002 to 2.2 cancer fatalities. The risk associated with the maximum exposure ranges from 0.000007 to 0 fatalities.

The shipments of naval spent nuclear fuel from shipyards and prototypes, transfer of nuclear fuel to storage, transfers of test specimen assemblies to the examination of test specimens to test facilities were evaluated for the maximum consequences of an accident. The naval spent nuclear fuel shipments contain a higher amount of activity per shipment where the test specimen shipment consequences are larger. The consequences are larger for the higher number of shipments which increases the probabilities such that a more serious accident is evaluated.

Tables A-34 and A-35 provide the maximum consequences, separated by on-site and off-site accidents, respectively.

As described in Section 3.8 of the main body of this Appendix, a transition period is necessary which would require approximately 80 shipments from Navy sites to ECF. This period is not included explicitly in the detailed analyses; however, the appropriate number of shipments by each alternative during this period is explicitly included, so the range of environmental consequences is bounded. Since all alternatives ship the same basic fuel types, the maximum consequences are determined by the probability of the accident which is a function of the distance from the site. In Section A.8.3, only the No Action, Decentralization - No Examination, and Decentralization - Examination alternatives, which have the lowest estimated maximum consequences, would be evaluated if the distance traveled is less than 100 miles.

Table A-34. Summary of maximum consequences of an on-site accident (Beyond Design Basis)

| | MEI Collective Dose (person-rem) | Estimated Cancer Fatalities | Rural Collective Dose (person-rem) | Estimated Cancer Fatalities | Suburban Collective Dose (person-rem) |
|--|---|-----------------------------------|---|-----------------------------------|--|
| No Action | 0.0013 | 6.5 x 10 ⁻⁷ | 0.37 | 1.9 x 10 ⁻⁴ | 2.4 |
| Decentralization - No Exam | 0.0013 | 6.5 x 10 ⁻⁷ | 0.37 | 1.9 x 10 ⁻⁴ | 2.4 |
| Decentralization - Limited Exam | 0.0013 | 6.5 x 10 ⁻⁷ | 0.37 | 1.9 x 10 ⁻⁴ | 2.4 |
| Decentralization - Full Exam | 0.51 | 2.6 x 10 ⁻⁴ | 200 | 1.0 x 10 ⁻¹ | 100 |
| 1992-1993 Planning Basis | 2.2 | 1.1 x 10 ⁻³ | 3300 | 1.7 | 4100 |
| Regionalization or Centralization at INEL | 2.2 | 1.1 x 10 ⁻³ | 3300 | 1.7 | 4100 |
| Regionalization or Centralization at Hanford | 2.2 | 1.1 x 10 ⁻³ | 3300 | 1.7 | 4100 |
| Regionalization or Centralization at Savannah River | 2.2 | 1.1 x 10 ⁻³ | 3300 | 1.7 | 4100 |
| Regionalization or Centralization at Oak Ridge | 2.2 | 1.1 x 10 ⁻³ | 3300 | 1.7 | 4100 |

n or
Centralization
at
Nevada Test
Site

Table A-35. Summary of maximum consequences of an off-site accident

| | MEI Collectiv e Dose (person-r em) | Estimated Cancer Fatalitie s | Rural Collectiv e Dose (person-r em) | Estimated Cancer Fatalitie s | Suburban Collectiv e Dose (person-r em) |
|---|--|---------------------------------------|--|---------------------------------------|---|
| No Action | 0.014 | 7.0 x 10y6 | 4.0 | 2.0 x 10y3 | 25 |
| Decentralizati on - No Exam | 0.014 | 7.0 x 10y6 | 4.0 | 2.0 x 10y3 | 25 |
| Decentralizati on - Limited Exam | 0.045 | 2.3 x 10y5 | 7.4 | 3.7 x 10y3 | 25 |
| Decentralizati on - Full Exam | 1.8 | 9.0 x 10y4 | 2700 | 1.4 | 3300 |
| 1992-1993 Planning Basis | 1.8 | 9.0 x 10y4 | 2700 | 1.4 | 79 |
| Regionalizatio n or Centralization at INEL | 1.8 | 9.0 x 10y4 | 2700 | 1.4 | 79 |
| Regionalizatio n or Centralization at Hanford | 1.8 | 9.0 x 10y4 | 2700 | 1.4 | 320 |
| Regionalizatio n or Centralization at Savannah River | 1.8 | 9.0 x 10y4 | 2700 | 1.4 | 320 |
| Regionalizatio n or Centralization at Oak Ridge | 1.8 | 9.0 x 10y4 | 2700 | 1.4 | 320 |
| Regionalizatio n or Centralization at Nevada Test Site | | | | | |

transition shipments were included. Therefore, incorporating the transition period the difference between alternatives from the standpoint of transportation effects.

A.9 EFFECT ON ENVIRONMENTAL JUSTICE

The only method used to ship naval spent nuclear fuel to INEL in the past and proposed for future shipments is by rail. The only exceptions to this are that nav Pearl Harbor Naval Shipyard is transported by ship from Hawaii to Puget Sound Naval shipping containers are transferred to railcars for the journey to INEL, and a heav to move the shipping containers from the Kesselring Site a few miles to the nearest shipment used for naval spent nuclear fuel tends to limit the exposure to members o during transportation. The shipments pass through urban, suburban, and rural areas by the railroads in accordance with applicable regulations and the requirements of

of the distance traveled in urban, suburban, and rural areas range from about 2.5% and 85% rural to approximately 4% urban, 35% suburban, and 61% rural, depending on considered.

As shown in the analyses in this Attachment, the impacts on human health or t resulting from routine transport of naval spent nuclear fuel and hypothetical trans be small for all of the alternatives considered. For example, it is unlikely that would occur as a result of the transportation of naval spent nuclear fuel under any accidents could occur at any location along the routes used, so it is not possible low-income composition of the populations along the routes. However, the fact that due to an accident for any of the alternatives considered would present no signific constitute a credible adverse impact on the population along the shipping routes ma that no adverse effects from accidents associated with the management of naval spen be expected for any specific segment of the population, minorities and low-income g

To place the impacts on environmental justice in perspective, the risk from r activities or hypothetical accidents associated with transportation of naval spent the alternatives considered would amount to less than one additional fatality per y population. For comparison, in 1990 there were approximately 40,000 traffic fatali States population and there were about 7,400 deaths caused by traffic accidents amo the U. S. Even if all of the additional cancer deaths associated with an accident considered for naval spent nuclear fuel management were assumed to occur only among that group would experience far less than one additional fatality per year. The sa drawn for low-income groups.

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ATTACHMENT B - DESCRIPTION OF NAVAL SPENT NUCLEAR FUEL RECEIPT AND

HANDLING AT THE EXPENDED CORE FACILITY AT THE
IDAHO NATIONAL ENGINEERING LABORATORY

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ATTACHMENT B

DESCRIPTION OF NAVAL SPENT NUCLEAR FUEL RECEIPT AND HANDLING AT THE EXPENDED CORE FACILITY AT THE

IDAHO NATIONAL ENGINEERING LABORATORY

B.1 GENERAL DESCRIPTION AND OPERATION OF FACILITIES

The Expended Core Facility (ECF) is located within the confines of the Naval Facility (NRF) at the Idaho National Engineering Laboratory (INEL). It is a large used to receive, examine, prepare for storage, and ship naval spent nuclear fuel an specimen assemblies. The information derived from the examinations performed at EC engineering data on nuclear reactor environments, material behavior, and design per data are used to develop new technology and to improve the cost-effectiveness of ex Naval spent nuclear fuel is prepared at ECF for storage and shipment to the Idaho C Processing Plant (ICPP). Some naval equipment contaminated by radioactive material the fleet is refurbished for reuse.

The building which houses ECF is a concrete block structure approximately 100 feet. This space provides offices and enclosed work areas, including an array of i reinforced concrete water pools which permit visual observation of naval spent nucl handling and inspection while shielding workers from radiation. Adjacent to the wa shielded cells used for operations which must be performed dry. Access to ECF for shipping of large containers is provided by large roll-up doors that allow railcar schematic view of ECF is shown in Figure B-1 and a photograph of the water pool are Figure B-2.

ECF has been specifically designed to provide the unique physical and adminis required by the Naval Nuclear Propulsion Program to ensure safe handling of irradiated contaminated nuclear fuels and components with a high degree of worker safety and p
 Figure B-1. Schematic view of Expanded Core Facility. Figure B-2. Expanded Core a shielded cell with a connecting transfer canal. The facility has been modified to accomplish the expanding mission of the facility since then, including the addition of water pools, several shielded cells, and other capabilities dictated by the nature

B.1.1 Water Pools

The purpose of the four interconnected water pools is to permit viewing and e radioactive reactor components and specimens while providing radiation shielding for

Walls and stainless steel gates divide the water pools into smaller work areas. This partitioning makes it possible to drain a small portion of the total water pool facility equipment maintenance or repair is required. It also would permit isolation of a zone if a leak were to develop which, combined with transfer of the water from that facilities, would minimize the loss of water.

B.1.1.1 Water Pit No. 1. This pool is used for the removal of spent fuel from shipping contain-

ers, and for preparation of fuel and low-level waste for shipment to ICPP. It also non-fuel storage areas.

B.1.1.2 Water Pit No. 2. This water pool is used for handling irradiation test assemblies.

Various components are tested for their reaction to radiation. Test assemblies returned from the Advanced Test Reactor (ATR) at INEL are unloaded from the shipping cask and disassembled. Verification of test integrity and connection of electrical and mechanical monitoring is performed.

B.1.1.3 Water Pit No. 3. Radioactive components are separated by milling machines into smaller

units for examination in this water pool. Dimensional measuring equipment is used to examine selected components. Fuel storage racks are also located in Water Pit No. 3.

Observation rooms are located along the northern wall of this water pool. They are below the level of the water surface and have viewing windows into the water pool. Components may be visually examined and remotely handled underwater for shielding purposes from

B.1.1.4 Water Pit No. 4. Operations performed in this water pool include spent fuel removal

from transfer containers, temporary fuel storage in racks, fuel examination, and preparation for fuel shipments. Observation rooms are located along the northern wall of the water pool. The pool also contains the transfer canals that would link the water pools with the process Project, which would prepare spent fuel for shipment in a dry, enclosed environment

B.1.1.5 Construction. All of the water pools are constructed of reinforced concrete in such a

manner that they are watertight. The water pool floors are designed to support inside and shielded shipping containers weighing up to 100 tons with a minimum base area of 100 sq ft. Water pool zone depths range from 20 feet to 45 feet. Water pool walls and floors are coated with a thermo-setting plastic coating which is highly resistant to radiation damage, is sealed, and serves as an extra barrier to water leakage.

B.1.1.6 Water Treatment and Minimizing Radioactive Contamination. Radioactive contaminants which

have accumulated in the ECF water pools through the introduction of corrosion products from irradiation test assemblies and the unloading of spent fuel are removed by various techniques. The design basis for the ECF water treatment system is to allow no discharge of radioactive material to the environment, maintain water clarity, and minimize the a

radioactive contaminants in the water.

The design goals are accomplished through the use of water purification modules, surface skimming to remove film and floating material, and water recycling systems. Purification modules prefilter the water to remove particles larger than 60 microns, remove any dissolved solids in ion-exchange resin beds, and remove any organic or solid material by absorption in an activated carbon bed. Spent resin, carbon, and filter are disposed of as solid radioactive waste.

B.1.1.7 Water Management. The total volume of the ECF water pools (excluding the two new

transfer canals that are empty) is 3,000,000 gallons. A 1-inch difference in the water level is equivalent to approximately 9,300 gallons.

The water pools are maintained at a nearly constant level. Alarms are installed both high and low level conditions. The total water volume is accounted for monthly. The amount of water to the system is reported to a separate NRF site organization for an independent verification of water volume.

Water leaves the water pools via evaporation, temporary filling of shipping containers for decontamination of equipment, and transfers to retention basins. The water pool evaporation has been calculated theoretically and confirmed by experiment. Water returns to the water pools from the retention basins and by draining shipping containers. Water removed from the system due to evaporation and equipment decontamination is replaced by adding demineralized water.

ECF has the capability of storing 235,000 gallons of water pool water in three steel-reinforced, concrete storage basins. Two of the vaults each have a 40,000-gallon capacity, and the third has a 155,000-gallon capacity. These basins provide the capability to receive water from the pools and receive water pool water if draining a water pool zone is necessary.

B.1.2 Shielded Cells

There are 14 concrete shielded cells in the facility. These shielded cells are used for the examination of smaller components, such as specimens which have been removed from irradiation tests that have been exposed to a neutron flux in the ATR, and fuel and non-fuel components from the water pools.

The shielded cells are constructed of concrete, with walls 3 feet thick to protect from radiation. Ventilation in the cell bank maintains negative pressure inside the cells relative to the rest of the facility. This ensures that radiological contamination is contained within the cells.

All work in the shielded cells is performed remotely by equipment controlled from a remote gallery, and is viewed through shielded lead glass windows. The windows are 3 feet thick and provide the same shielding value as the concrete walls. The interior of the cells is viewed through wall periscopes that permit undistorted viewing of equipment and components.

B.2 RECEIPT AND HANDLING OF NAVAL SPENT NUCLEAR FUEL

B.2.1 Receipt of Spent Fuel

Nuclear-powered ship assignments for refueling, defueling, and overhaul are completed by the six nuclear-capable public shipyards (Mare Island, Puget Sound, Portsmouth, Norfolk, and Charleston) and one nuclear-capable private shipyard (Newport News). In 1993, the federal base closing commission included Mare Island and Charleston Naval Shipyard among the bases to be closed in the near future. The spent fuel is removed from nuclear ships and loaded into shipping containers designed specifically for naval spent nuclear fuel. Containers are loaded and sealed at the shipyard and shipped to ECF via railcar. Attachment A. A maximum of 48 containers can be staged on the rail siding at NRF while awaiting transfer of the spent fuel to the water pools. ECF also receives spent nuclear fuel from prototype plants in a similar manner.

B.2.2 Handling of Spent Fuel

The shipping containers are brought into the ECF building at one of the two d and are prepared for defueling by removing the dust cover, leveling, and filling wi Appropriate containments to prevent release of radioactive material are installed a access plug is removed to allow access to the fuel modules.

The containers are unloaded at either the west end defueling station or the e station. Regardless of the defueling station used, the fuel modules are removed fr container one at a time using a fuel handling machine which draws the module out of into a shielded volume, and the entire machine is transferred to the water pools. then discharged into a receiving receptacle in the water pools. Photographs of the machines used are provided in Figures B-3 and B-4.

Every item containing nuclear fuel received at ECF has a unique serial number fuel is removed from its shipping container, two ECF fuel handlers independently re number and compare it to the shipping paperwork. After the serial number is confir

Figure B-3. M-140 container fuel handling machine. Figure B-4. M-130 containe that the fuel is stored in the correct storage location. ECF has a computer-based system which maintains a record of the location and type of every piece of nuclear many grams of uranium are contained within the fuel. This system tracks every fuel during the time that the fuel is at ECF.

All naval fuel modules have metal structures which contain no fuel above and region to facilitate coolant flow and maintain proper support and spacing within th upper and lower non-fuel bearing structures must be removed to provide access to th sections to permit inspection of the module. Removal also reduces the storage spac required for the fuel by approximately 50 percent. The upper and lower non-fuel be removed during the preparation of fuel modules are evaluated using the waste classi established by federal regulations in 10CFR61 and DOE Order 5820.2. These non-fuel structures do not contain any fuel, or fission products from fuel, and therefore ca "spent nuclear fuel." They also do not contain transuranic elements or fission pro cannot be considered high-level waste or transuranic waste. Therefore, the amounts in the end boxes cause them to be classified as low-level waste. As indicated in S amount of low-level waste generated each year at the Expended Core Facility is 425 The radioactive isotopes which represent 99 percent of the activity in this materia follows:

| ISOTOPE | HALF-LIFE (Years) | PRIMARY MODE OF DECAY |
|---------|----------------------|--------------------------|
| Fe-55 | 2.73 | Electron Capture (x-ray) |
| Co-60 | 5.271 | Beta and Gamma |
| Ni-59 | 76,000 | Electron Capture |
| Ni-63 | 100 | Beta |

U.S. Nuclear Regulatory Commission 10CFR61 identifies three classes of low-le which are generally suitable for near-surface disposal, namely, Classes A, B, and C the requirements for near-surface disposal are shipped to the INEL Radioactive Waste Complex using a shielded cask. Wastes with concentrations greater than those speci for certain short- and long-lived isotopes were found to be not generally suitable disposal. These wastes are classified as Greater Than Class C Low-Level Radioactiv May 1989, the Nuclear Regulatory Commission promulgated a rule that requires dispos commercially generated low-level waste with concentrations of radioactivity greater deep geologic repository, unless disposal elsewhere is approved by the Nuclear Regu

Currently, a small amount (about 25 cubic meters) of greater than Class C low material removed from the ends of naval spent nuclear fuel modules over the years i the Naval Reactors Facility pending availability of a disposal facility licensed by Regulatory Commission. This material has been collected and held at the Expended C many years. This practice is expected to continue over the period of time covered mental Impact Statement.

After these upper and lower metal structures have been removed from a fuel mo fixture is installed to facilitate handling. Prepared fuel may then be inspected i be held for a time prior to inspection in storage racks in the water pool. In the temporarily stored while awaiting inspection, spacers are placed at the bottom of t the storage rack to maintain the position of the fuel module close to the top of th movement of the module easier.

Visual examinations of all modules are performed to verify that the fuel has expected. As discussed in Section 2.4.1, about 10 to 20 percent of the spent react selected for more detailed examination or destructive analysis in accordance with t

Naval Reactors fuel development program. The more extensive examinations performed pools include measurements of key dimensions of the modules and collection of specimens examined in the shielded cells. The specialized equipment used to perform examinations of spent nuclear fuel are described in more detail in the section of this attachment devoted to destructive analyses are performed at the Expanded Core Facility or at other laboratories. Material subjected to such analysis must be removed from the spent fuel modules at the Core Facility.

The last steps of spent fuel handling performed at ECF are staging the module and loading the module into the shipping cask used to transport spent fuel from ECF. Spent fuel may be temporarily stored in the racks in the ECF water pools until a cask is available to transfer the material to ICPP.

B.2.3 Shipment of Fuel to the Idaho Chemical Processing Plant

A lead-filled, stainless steel shipping cask is used to transport naval and power modules from ECF to ICPP. The cask is removed from its transport truck and lowered into the water pool until it rests on the floor of the pool. The closure head is removed, and spacers are placed in the cask to provide proper spacing of fuel and to maintain proper positioning of the modules. The modules are inserted into the cask, the closure head is reinstalled, and the cask is lifted from the water. The cask is drained, the exterior is decontaminated, and then it is loaded onto the truck for shipment. The transport of the cask to ICPP is described in Attachment B.2.4.

B.2.4 Library of Naval Reactor Components

As the first modules of a given fuel design are received at the Expanded Core Facility, selected key operating components are retained in "library" storage to provide a source of reference. These older components are kept to ensure that they are representative items available to assist in diagnosis of problems which may occur in the fleet. The items chosen for this library are usually those that have the longest life so that they display the most pronounced effects of use. As the components are replaced in fleet service by newer designs, fuel components related to the fuel are removed from library storage and shipped to ICPP.

B.3 HANDLING OF IRRADIATED TEST SPECIMENS

The irradiated materials program evaluates small specimens of materials for use in reactor systems. The specimens are loaded in sample holders, and the holders are placed in assemblies at ECF. The assemblies are irradiated at ATR, and returned to ECF for disassembly. The specimens are cleaned, examined, reloaded in a test assembly, and returned to the ATR for continued irradiation. A typical specimen undergoes several cycles of irradiation over several months or years. Examinations include nondestructive and destructive tests. Historically, sectioning of specimens for mechanical testing and metallographic work was performed in the ECF hot cells in the past and is planned to continue on specimens in the future.

After completion of the final examination, specimens are shipped to ICPP for the INEL Radioactive Waste Management Complex for disposal. Other specimens are shipped either to the Bettis Atomic Power Laboratory near Pittsburgh, Pennsylvania, or the Knolls Atomic Power Laboratory near Schenectady, New York for more detailed examinations.

B.4 DESCRIPTION OF MAJOR ITEMS OF EQUIPMENT

The normal method for moving the fuel in the water pools to designated examination equipment areas is by use of one of five bridge cranes which move on rails located on the walls of the water pools. The fuel is handled remotely. All fuel movements are coordinated by personnel, and accountability is maintained both by computer and by personnel using log forms.

B.4.1 Water Pool Equipment

ECF has unique equipment in the water pools that has been designed for remote underwater to perform specific examinations on naval spent nuclear fuel and irradiated Special consideration was given during equipment design to provide for remote repairment of components. A description of the water pool spent nuclear fuel and irradiation equipment is presented below.

B.4.1.1 Water Pool Band Saws. There are two underwater band saws in the ECF water pools.

These band saws are used to remove the non-fuel bearing structural material from the fuel cells in preparation for inspection. The fuel region of the fuel cell remains cutting procedure.

B.4.1.2 Water Pool Milling Machines. Three milling machines in the water pools are used to

separate spent nuclear fuel components into smaller sections for examination in the fuel region of the fuel cell remains intact during the machining. The mills are used to spent fuel into pieces which can be handled in the shielded cells for examinations, radiation measurement, or for obtaining smaller specimens for metallurgical analysis measurement. The mill head of the largest milling machine can be remotely interchanged with a saw attachment to convert the machine into a cutoff saw.

B.4.1.3 Universal Inspection Station. This equipment is used to obtain dimensional measurements

using specially designed probes that are inserted in the fuel module. This equipment can position and rotate the probe in any orientation by a dedicated computer. This information is used to assess dimensional changes in the fuel module.

B.4.1.4 Vertical Inspection Gage. The vertical inspection gage is used for obtaining dimensional

measurements or to trace the contour of the external surfaces of fuel cell assemblies. This information can be used to provide a three-dimensional image of the fuel cell at the end of fuel life to determine the effects of fuel element changes on the overall dimensions over fuel life and the effects of radiation on control rod dimensions over

B.4.1.5 Video Visual Equipment. Underwater television cameras and lighting can be set up in

any zone in the water pools to obtain images of the external surfaces of the fuel cell control rods. These visual inspections are used to search for anomalies such as external wear on external surfaces. The bottom end of the fuel cell assemblies can also be inspected for blockage, corrosion, and wear.

B.4.1.6 Assembly and Disassembly Tables. These tables are used to assemble and disassemble

irradiated test assemblies that are inserted in the ATR. There are two identical assembly and disassembly tables installed side by side in the water pools. Each is mounted on a base used to rotate the table from a horizontal position for test assembly and disassembly position for loading and unloading the test assembly.

B.4.1.7 Headwork Station. The Headwork Station provides containment and shielding for the

mechanical connection and disconnection of components to and from the unirradiated assembly and disassembly of irradiated test assemblies for the ATR. There are two independent stations; each consists of an elevator platform which raises the top of the unirradiated assembly above the water surface. A containment is positioned above the water surface to prevent release of contamination while the examination is performed above the water.

B.4.1.8 Fuel Storage Racks. Storage racks are required at ECF since, at times, fuel is received

into the facility faster than fuel can be prepared and shipped out of the facility. to store the small amount of naval spent nuclear fuel selected for retention as lib future reference and study. Ensuring that the racks are conservatively designed to credible accident and continue to provide adequate nuclear separation are the major storage racks.

The basic configuration of a fuel storage rack is a rectangular structural ar ports. Each port has a square opening, but depth is variable. All storage ports i stainless steel. Stainless steel is used exclusively to resist corrosion during th racks. The storage ports are designed to withstand the weight of the heaviest fuel be placed in the port, and the frame assembly is designed to support the entire wei ports fully loaded with the heaviest fuel type.

All the fuel racks are designed to maintain their structural integrity during earthquake and to withstand the impact of a fuel module dropped onto the fuel racks fuel racks in the event of seismic activity has demonstrated that they will not col postulated earthquake. ECF also performed a full analysis of the strength of the p module were dropped over the fuel racks, including the kinetic energy which the dro module would impart to the rack. It was determined that all fuel racks at ECF were designed to withstand the energy of dropped fuel. The analysis also identified tha handled at ECF was heavy enough that the racks might be deformed if the equipment w Thus, operating rules and procedures prohibit the movement of large loads over the ensure that no accidental damage to the racks can occur.

Fuel storage racks were also designed to prevent arrangement of the modules i ly critical configuration. The fuel racks are designed so that each port separates contains from every other module by a distance great enough to prevent criticality limiting conditions possible. To assure that only one piece of fuel is placed in a ports are equipped with lids which can be locked and sealed. Finally, the frame as storage racks are covered with stainless steel sheeting to prevent fuel from inadve between fuel storage ports.

B.4.2 Water Pool to Shielded Cell Transfer Systems

Components that have been removed from spent nuclear fuel cells or test assem transferred into the shielded cells using one of the three available water pool to systems. The transfer systems use carts that are driven through underwater tunnels

B.4.3 Shielded Cell Examination Equipment

ECF has specialized equipment installed in the shielded cells which is design examinations on fuel elements and components removed from spent fuel cell assemblie specimens that have been irradiated in the ATR. A description of the major shielde follows.

B.4.3.1 Electronic Balances. These are commercially available electronic balances that have been

modified to operate remotely in the shielded cells. Components on these balances t deteriorate from exposure to radiation have been replaced using materials that are radiation damage. The equipment is interfaced with computer data acquisition syste operators in tracking and reducing the data. These balances are used primarily to changes that result from corrosion testing of materials in the ATR.

B.4.3.2 Descale Tanks. Corrosion removal is performed for test specimens that have been

irradiated in the ATR and structural components and fuel elements removed from spen modules. These tanks use heat, chemicals, and ultrasound to dislodge corrosion tha on the specimens or components. The corrosion removal aids in visual examination o specimens.

B.4.3.3 Bridgeport Milling Machine. This is a high-precision milling machine that has been

modified for remote operation in the ECF shielded cells. The mill is controlled by controller located in the shielded cell gallery. The Bridgeport mill is used for non-fuel components removed from spent nuclear fuel cell assemblies.

B.4.3.4 Specimen Coordinate Automated Measuring Machine. The specimen coordinate

automated measuring machine is a fully automated unit specifically designed to perform dimensional measurements on irradiated test specimens and structural components removed from nuclear fuel cells. The equipment is completely computer controlled and has an accuracy of 50 microinches. The information obtained from this equipment is used to assess the effect of radiation on material growth and fuel burnup on swelling of specimens.

B.4.3.5 Fiducial Automated Measuring Machine. This machine is used to measure the distance

between scribe marks that are put on some types of specimens during fabrication. The machine measures the position of the scribe marks in relation to other fiducial marks. These data are used to assess the effects of radiation on specimen growth and distortion. The effect of fuel depletion on fuel element swelling.

B.4.3.6 Gamma Scan System. This system measures gamma radiation emitted by fission

products to identify isotopes present in the fuel as a result of fuel depletion. The system is controlled by a dedicated computer which positions the specimen, provides for data acquisition and provides an output of the isotopes detected by the system at each location along the specimen.

B.4.3.7 Alpha Box. The Alpha Box is a carbon steel containment inside the shielded cells. It

provides isolation within the shielded cells for fuel cutting to prevent the spread of contamination. This is the only location in the facility where cutting through the fuel region is allowed.

B.5 FACILITY DESIGN AND INTEGRITY REQUIREMENTS

B.5.1 Flood

A flood at ECF due to overflow of any source of surface water within the INEL is a low probability event. With the construction of the INEL flood control diversion, the threat of a flood from overflowing of the Big Lost River, the primary source of water for the INEL, has become very small.

The maximum water elevation postulated at ECF would be caused by a hypothetical Maximum Flood resulting from failure of the Mackay Dam, located approximately 35 miles from the INEL. The hypothetical flood could result in a maximum water level approximately 10 feet above the floor elevation of the ECF building. This flood is postulated to result from overtopping of the Mackay Dam and causing it to fail due to high water levels. This is unlikely. (Koslow and Van Haaften 1986)

Dam failure due to other causes, such as seismic activity, is more likely. A Mackay Dam survived the 1983 Borah Peak earthquake without damage, it was built with design criteria. Additionally, it is not clear how resistant the dam structure is. A fault segment runs within 6 kilometers of the Mackay Dam.

Flooding of the ECF building is possible should the Mackay Dam fail. Flooding of the building would not create a nuclear criticality hazard. Flooding of the building could result in the release of water containing low levels of radioactive contamination to the environment.

equipment in flooded areas. Following the dam break, it would take over 16 hours for water to reach NRF. This is adequate time to complete emergency procedure preparat filling and placing sandbags, for the expected flood conditions.

B.5.2 Earthquake

The ECF building structure was built in accordance with the Uniform Building particular phase of construction. Water Pit No. 1, Water Pit No. 2, and Water Pit to "Zone 2" earthquake requirements which were judged to be appropriate under the U Survey classification of the area at the time of their construction. Water Pit No. transfer canals were built to the more restrictive "Zone 3" earthquake requirements time they were built.

A seismic assessment has been performed for the ECF using the actual characte existing facility. Based on this assessment, a design basis seismic event at ECF c ground acceleration of 0.24 g (Rizzo 1994). This peak ground acceleration is deriv that a moment magnitude 6.9 seismic event centered near Howe on the Lemhi fault wou rupture of approximately 34 kilometers along the Lemhi fault. The Howe epicenter i located closest to ECF, and 6.9 was the moment magnitude of the Borah Peak earthqua This approach for postulating the location of the seismic event is consistent with Regulatory Commission methodology used for commercial power plants. The beyond des seismic event was based on the entire 150 kilometers of the Lemhi fault rupturing. design basis earthquake might have a peak ground acceleration of 0.4 g at ECF.

B.5.3 Tornado

A tornado at ECF is a low probability event. The document "Technical Basis f Regional Tornado Criteria," WASH-1300, provides the technical basis for Nuclear Rea sion Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants." The W document identifies the probability of occurrence of a tornado at ECF to be 7.8×10^{-5} on historical records. Regulatory Guide 1.76 identifies the maximum wind speed app to be 240 mph. Data collected by Dr. T. Fugita of the University of Chicago perfor request of the DOE for the period between 1950 and 1976 indicate the probability of winds of that speed occurring at the INEL is about 1.3×10^{-9} per year. Based on a speed for tornado damage of 75 mph (refer to P. L. Doan, "Tornado Considerations fo Power Plant Structures," Nuclear Safety, Volume 11, No. 4) and a probability of 0.8 occurrence of tornado-induced wind speeds greater than or equal to 75 mph (WASH-130 the probability of a damaging tornado occurring at ECF is 7.8×10^{-5} per year $\times 0.8$ year.

A tornado could not affect the fuel storage area in ECF in such a way that th rearranged into a critical configuration. The article by Doan cited above analyzes tornados for the general case of spent fuel in water pools and concludes "... massi to either tornado-induced wind forces or tornado-generated missiles cannot happen. however, that a couple of feet of water could be lost owing to the combination of w water entrainment, and pressure differentials. The spent fuel at the bottom of the however, remain completely covered.... By the same token, the radiation dose level surface would not increase by any meaningful amount."

B.5.4 Fires

The entire ECF facility is protected against fires by one of several types of A large, intense fire in fuel handling areas is a low probability event because of materials of construction in these areas, the amounts and kinds of material present protection system. Most of the spent fuel is under many feet of water, providing a against a fire which might involve fuel. Fires at other locations in the facility by the sprinkler system and by manual fire protection equipment (e.g., fire extingu hoses). An extensive fire involving the ECF building structure is highly unlikely constructed of non-combustible or fire-resistive material to the greatest extent po with applicable Atomic Energy Commission, Energy Resource and Development Administr DOE design criteria.

B.5.5 Loss of Water Pool Water

Loss of all water in a section of the water pool is extremely unlikely. How a heavy object be dropped onto a water pool floor, a crack could develop. If this were cracked water pool area would be isolated and drained in a controlled manner to one basin before a substantial loss of water to the environment would occur. Even in severe damage to a water pool floor were to result in the loss of substantial amount of water, no nuclear criticality hazard would result and no melting of fuel would occur.

B.6 CRITICALITY CONTROL

There has never been an inadvertent criticality at the Expanded Core Facility as a result of strict application of the following principles.

A fundamental principle of nuclear safety is Criticality Control. When a mass reaches a condition at which its atoms are capable of undergoing a self-sustaining fissioning (fissioning) into new elements, the result is called a criticality. Nuclear energy in the form of radiation and heat. Controlled criticality within a shielded container produces energy within a confined space without harm to personnel or the environment. Water pools, the shielded cells, and the ECF building are designed to shield and control radioactive contamination, an uncontrolled criticality (or nuclear excursion) within the facility, and comprehensive measures are taken to prevent such an occurrence. Criticality at ECF could be described more accurately as "absolute criticality prevention." Conditions identified, equipment or processes are designed, rules and procedures are formulated and are trained to prevent occurrence of an accidental criticality.

Safety analyses are performed on all fuel types and system designs where all likely and unlikely accidents are considered. Conservatism is employed in establishing limits and spent fuel is handled to the more restrictive as-built values. Then a "double check" is applied to all fuel handling equipment and procedures. The double check criteria must be handled and equipment designed so that acceptable margins to criticality exist, limiting, unlikely, independent, and concurrent accidents. In this context, two separate administrative procedures are considered to be a single accident, not two." As a result of this criterion to equipment and procedures at ECF, the amount of fuel which may be in any operation is typically restricted to one quarter of the minimum amount which could cause criticality minus a safe margin to criticality.

All nuclear fuel operations must be performed in accordance with approved criticality procedures. Nuclear safety analyses are carefully reviewed by the responsible management and independent nuclear safety committees. Naval Reactors must approve each analysis before it is implemented. Strict reviews and approvals are also applied to implementation of safety analyses in fuel handling procedures.

The successful criticality control program at ECF is also due to thorough training of fuel handling personnel. Employees are educated concerning the principles associated hazards, and prevention. A system of checks to ensure that the rules are observed is employed. It includes detailed training documentation, qualification and a self-assessment (audit) program, and an array of accountability and nuclear safety.

B.7 PROPOSED DRY CELL FACILITY

The Dry Cell Facility consists of a shielded, radiologically controlled area containing operated equipment. The facility is designed for a 40-year life, built of structural steel and would be integral with the existing ECF building.

The major element of the Dry Cell Facility is a large reinforced concrete shielded interior dimensions of 22 feet wide by 84 feet long by 21 feet high, containing all necessary equipment to inspect and disassemble fuel modules. The facility will have the capacity to load one fuel module per shift in a shipping cask. Based on a two shift per day (two shifts per year), and a 25-percent maintenance downtime, the Dry Cell Facility year expected to be 375 modules. Shielded decontamination and repair cells will be attached to the Dry Cell Facility to allow remote decontamination and repair of equipment used throughout the Dry Cell Facility and the associated Cask Loading System are shown in Figure B-6.

The dry cell design incorporates 4-foot thick, radiation shielding walls consisting of high-density and normal-density concrete. The shielding is designed to limit radiation levels in occupied areas around the cell to 0.1 millirem per hour or less. At the INEL Site there would be no measurable elevation above the naturally occurring background radiation level. The cell design meets the latest seismic requirements and includes negative pressure air handling for radiological contamination control. Shielded lead glass windows and viewing aids are required at the workstations. Power, lighting, and a fire suppression system are also required.

The Dry Cell Facility is also designed to facilitate decontamination and decommissioning of the facility at some future date. This is achieved by including cell liner contamination control.

fixed embedded piping, a minimum of cracks and crevices, smooth surfaces, and wall large enough to be radiologically surveyed to verify decontamination effectiveness.

B.8 REFERENCES

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 Rizzo (Paul C. Rizzo Associates), 1994, Natural Phenomena Hazards - Expended Core F Idaho National Engineering Laboratory, Monroeville, Pennsylvania, June.

ATTACHMENT C - COMPARISON OF STORAGE IN NEW WATER POOLS VERSUS DRY CONTAINER STORAGE

Figure B-5. Proposed ECF Dry Cell Facility. Figure B-6. ECF Dry Cell Facility Cask Loading System.

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ATTACHMENT C

COMPARISON OF STORAGE IN NEW WATER POOLS VERSUS DRY CONTAINER STORAGE

C.1 INTRODUCTION

This attachment discusses the advantages and disadvantages of water pools versus container storage should construction of additional interim storage be required. It considers the generic safety aspects of water pools and dry container storage based on performance by the Nuclear Regulatory Commission (NRC) and the Department of Energy (DOE) as well as experience with naval spent nuclear fuel.

C.2 WATER POOLS

During the last four decades, the Expended Core Facility (ECF) at the Idaho National Engineering Laboratory (INEL) has demonstrated the safety and reliability of water control of the Naval Reactors Program. Water pools have historically been the method of interim storage and fuel handling because: (1) water has a high thermal capacity to absorb heat from the fuel, (2) the transparency of water facilitates the inspection and monitoring of the fuel, (3) water is an excellent gamma and neutron shield, (4) water is easy to purify and water provides a means to prevent release of radioactive material into the air.

The safety of spent fuel storage in a water pool can be considered in terms of three criteria. They are: (1) the integrity of spent fuel under water pool storage conditions, (2) the structural and component safety of the facility, and (3) the potential risks of accident or sabotage at the spent fuel facility.

The NRC conducted an extensive investigation into the storage of spent fuel and the findings in the Waste Confidence Decision (NUREG 1575-01, 1984). Based on the technical data cited in that document, the NRC found that the Zircaloy cladding which encases spent fuel is resistant to failure under pool storage conditions and concluded that Zircaloy-clad fuel satisfied the first generic criterion. This conclusion is consistent with the extensive experience with naval spent nuclear fuel. Naval fuel is Zircaloy clad and thus is highly resistant to corrosion.

water. In addition, a Navy fuel assembly has much higher mechanical integrity than since it is designed for military application and is capable of withstanding shock be encountered in battle conditions.

The NRC also conducted an extensive evaluation of the structural and component water pools. The NRC found no reason why spent fuel storage pools would not be capable of performing their cooling and storage functions for a number of years past the design if the water pools are properly maintained; therefore, the second generic criterion. This conclusion is consistent with the naval fuel experience of over 35 years of operation.

The risk of major accidents at spent fuel storage pools resulting in off-site release because of the secure and stable character of the spent fuel in the storage and the absence of driving forces (i.e., high pressure or temperature) which might release radioactive material (NUREG 1984). The consequences of terrorist attacks on a storage pool would be limited by the realities that the radioactive content of spent fuel is material encapsulated in high-integrity metal cladding and stored underwater in a storage structure. Under these conditions, the radioactive content of spent fuel is relatively dispersed to the environment (NUREG 1984).

These considerations led the NRC to conclude that storage pools can be designed to withstand accidents caused either by natural or man-made phenomena such that there would be no impact to the environment. Therefore, the third generic criterion would be satisfied.

The NRC concluded that all areas of safety and environmental concern (e.g., systems and components, prevention of material degradation, protection against sabotage) have been addressed for water pools, and that spent fuel can be stored without impact. This conclusion is supported by the Organization for Economic Co-Operation and Development of the Nuclear Energy Agency (NEA 1993).

C.3 DRY CONTAINER STORAGE

Dry container storage technologies have been in use in the United Kingdom since 1963 (MOCSG 1993). In the United States, demonstration projects have been underway since 1970. Dry container storage, multiple barriers prevent gaseous as well as particulate fission products from being released: (1) the fuel cladding, (2) the inner secondary seal, (3) the outer secondary seal. In addition, dry storage systems provide shielding to reduce the external radiation to acceptable limits.

The NRC concluded that dry container storage involves a simpler technology than represented by water storage systems. Water storage relies to a certain extent upon such as pumps, renewable filters, and cooling systems to maintain safe storage. Fuel chemistry must also be maintained to retard corrosion. Dry container storage uses circulation of an inert atmosphere in a sealed dry system so there is little opportunity for leakage (NUREG 1984).

The NRC also found that dry container storage of spent fuel in dry wells, vacuum metal casks is relatively invulnerable to sabotage and the forces of nature, because of the size of the sealed, protective enclosures, which may include 100-ton steel casks, 1 cask, and surface concrete silos (NUREG 1980).

The NRC concluded that for dry interim storage, all areas of safety and environmental concern (e.g., maintenance of systems and components, prevention of material degradation against accidents and sabotage) have been addressed and shown to present no more potential adverse impact on the environment and the public health and safety than storage of water pools. This conclusion is supported by the Organization for Economic Co-Operation and Development of the Nuclear Energy Agency (NEA 1993).

As stated earlier, naval fuel uses Zircaloy cladding and has a much higher mechanical integrity than commercial fuel since naval fuel is designed for military application. The generic conclusions reached for commercial spent fuel are directly applicable to naval fuel.

C.4 NON-RADIOLOGICAL CONSEQUENCES OF SPENT FUEL

STORAGE

The NRC concluded (NUREG 1984) that "there are no significant non-radiological consequences due to the extended storage of spent fuel which could adversely affect the construction of an interim spent fuel storage facility (i.e., the construction of a

pad, a building, rail spur, etc.) would have little impact on the environment. The given off by spent fuel decreases with time as the fuel ages and decays radioactive of additional energy and water needed to maintain spent fuel storage is also small.

C.5 LAND UTILIZATION

With the use of water pool storage or dry container storage at an existing sh already devoted to industrial use is planned to be used for the spent fuel storage of land required for storage at specific shipyards is addressed in Attachment D.

C.6 COST

The use of alternate sites other than INEL would involve the construction of facilities. Both water pools and dry container storage could be used, with little therefore, the relative cost between these two options could be relevant. Conceptu have been prepared for each storage option at each location that is being evaluated comparisons are found in Attachments D and E.

C.7 SUMMARY

Based on the above discussion, both a new water pool and dry container storag suitable for the interim storage of spent naval fuel with no important radiological environmental impact. If a facility would be required to be used for the inspectio well as storage, then a water pool offers an advantage since water is an inexpensiv form of transparent shielding. If it were not necessary for a new facility to be u fuel, then the cost of the facility and the amount of land required could be factor option.

C.8 REFERENCES

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 NUREG (U.S. Nuclear Regulatory Commission), 1980, Dry Storage of Spent Nuclear Fuel NUREG/CR-1223, April.
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ATTACHMENT D - DESCRIPTION OF STORAGE OF NAVAL SPENT NUCLEAR FUEL

AT SERVICING LOCATIONS (SHIPYARDS AND PROTOTYPES)

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D-

ATTACHMENT D

DESCRIPTION OF STORAGE OF
NAVAL SPENT NUCLEAR FUEL AT SERVICING
LOCATIONS

(SHIPYARDS AND PROTOTYPES)

D.1 STORAGE OF NAVAL SPENT NUCLEAR FUEL IN CONTAINERS
AT SHIPYARDS AND PROTOTYPES

D.1.1 Introduction

This attachment examines the alternative of storing naval spent nuclear fuel prototype sites where the fuel is removed from the reactor plant. Water pool storage containers, and dry storage in shipping containers are evaluated for each prototype location. Under the No Action alternative, naval spent nuclear fuel would be stored in shipping containers. For the other alternatives where naval spent nuclear fuel would be stored in prototype sites, the storage mode would be selected by the Record of Decision. Attachment C has addressed the generic safety of water pool and dry storage and construction methods would be suitable for the interim storage of naval spent nuclear fuel with minimal environmental impact. This attachment addresses the design requirements, operation costs, and land requirements for the Puget Sound Naval Shipyard, Pearl Harbor Naval Shipyard, Portsmouth Naval Shipyard, and the Kesselring Site.

The interim storage facilities for naval spent nuclear fuel at shipyards and prototype sites would be designed to comply with applicable requirements. The storage facilities would be maintained in compliance with Naval Reactors Program requirements for radiation protection of workers and the public and the environment. Specifically, exposure to workers at the site would be maintained as low as reasonably achievable and would be controlled to Navy Program radiation exposure standards. As with current naval practices, no measurable radiation levels at the site boundary would result from the storage of naval spent nuclear fuel at the alternate site.

D.1.2 Shipping Containers

D.1.2.1 Container Design Features. Shipping containers and immobile dry storage containers

position the spent naval fuel modules within sealed structures designed to provide physical support, and remove residual heat from the fuel in an environment that prevents corrosion. The massive size of the containers provides not only strength, but also shielding against radiation from the spent fuel within.

The shipping containers might be M-140 shipping containers with long-lived seals for storage of spent nuclear fuel for the duration of the period covered by this Environmental Statement (EIS). A description of the M-140 shipping container is provided in Attachment C. The container is already certified to meet the requirements of the U.S. Nuclear Regulatory Commission in 10CFR71, for the transportation of naval spent nuclear fuel. With an intact seal, the M-140 container could be qualified for storage for 40 years. The seal could either be positioned on railcars at the storage site or on concrete pads. The design of the shipping container long-lived seal would commence with the Record of Decision option were selected. The cost associated with the design and recertification of the seal would range from approximately \$1 million to \$5 million. The cost to manufacture the container would be about \$5 million. Some uncertainties in estimated costs exist for a detailed design for the shipping container long-lived seal is not yet available.

If the Record of Decision were to choose shipping containers, a more detailed design would need to be performed to determine whether it is more appropriate to modify the existing shipping container design or whether a new container design should be used. Since the container is designed as a shipping container, the modifications that would need to be made to convert it to accommodate interim storage might involve substantial new design work and recertification.

About 500 additional containers with holding capacity equivalent to the M-140 would need to be fabricated to cover the projected reactor servicing from 1995 through 2015. If the alternative using the shipping containers were to be chosen, an expanded manufacturing facility would need to be developed to meet the projected container requirements. With the current manufacturing capabilities, 3 years are required to build an M-140 container and the rate is about 6 containers per year.

The shipping containers loaded during the period preceding the Record of Decision also need to be modified to meet the storage container design criteria. An evaluation would need to be performed to determine whether these modifications could be safely made with spent nuclear fuel present in the containers. In the event that the spent nuclear fuel must be removed from the containers, the containers would be unloaded and the spent nuclear fuel would be transferred to modified shipping containers at a suitable facility under controls which would protect the public, and the environment. The unloading of spent nuclear fuel from the original

containers and reloading into modified shipping containers would introduce additional fuel handling, transportation, and risks.

D.1.2.2 Operations. The process of loading spent nuclear fuel into shipping containers for storage

would be similar to that used for loading M-140 shipping containers. During reactor operations, spent nuclear fuel is normally loaded into M-140 shipping containers with water. The spent nuclear fuel is staged in this configuration for sufficient time produced by radioactive decay of fission products is adequately dissipated. When removed from the M-140 container, the loaded M-140 can be shipped. After water is removed from the shipping container, it would be transported to the storage site. The water is The transportation procedures would be essentially unchanged from current procedure containers would be moved to the interim storage site instead of being shipped to the Facility (ECF) at the Idaho National Engineering Laboratory (INEL) for inspection. storage, the railcar would be positioned in the storage area. For cases where the stored on a concrete pad, the container would be off-loaded from the railcar or truck then secured to the pad (if securing would be required). In order to accomplish this capacity crane would be needed at each site, and the site would need to be prepared to accommodate the mode of storage.

D.1.3 Immobile Dry Storage Containers

D.1.3.1 Container Design Features. There are currently no immobile dry storage containers

designed for interim storage of naval spent nuclear fuel. The container design would be that of containers which are presently certified by the Nuclear Regulatory Commission for spent nuclear fuel from commercial reactors. The design, approval, and construction of a dry storage container would commence with the Record of Decision if this option were selected. The effort could require up to 5 years to complete. The cost associated with the design of the immobile storage container would be about \$2 million. The cost to construct each storage container would be about \$2 million. These estimates are based on costs of available containers with contingencies added to account for additional design features required.

Two concepts for storing naval spent nuclear fuel in immobile dry storage containers have been developed in order to provide a baseline for assessing the impacts. Other dry storage approaches (such as dry storage vaults) exist and would be considered in more detail in the Record of Decision were to choose the immobile dry container storage alternative. The first approach (referred to as the minimum fuel loading concept) is based on the number of spent fuel assemblies that could be stored in the immobile dry storage container being about the same as that which is loaded into M-140 shipping containers. This approach results in the need for about 500 immobile dry storage containers. The second approach (referred to as the maximum fuel loading concept) maximizes the number of assemblies that would be stored in the immobile dry storage containers. The number required for the second approach is about 300.

The minimum fuel loading concept results in a container with a comparatively less maintenance, and lower unit costs (~\$1.9 million/container). Under the maximum fuel loading concept, the container would need to be equipped with additional active cooling features to ensure that the heat produced by radioactive decay of fission products is removed. These additional cooling features would be needed for a period of several years after spent nuclear fuel is removed from the reactor vessel. For the minimum fuel loading concept, additional active cooling features such as recirculating water would not be required. As with the shipping containers, an expanded vendor base would be necessary in order to build the immobile dry storage containers at the rate they would be needed.

Figures D-1 and D-2 provide conceptual layouts of candidate immobile dry storage containers for naval spent nuclear fuel.

Figure D-1. Conceptual concrete immobile dry storage container for naval spent nuclear fuel would be approximately the same as the M-140 shipping container (i.e., 10 to 16 feet high and 8 to 10 feet wide). The fuel spacing within the container itself would be designed to prevent any nuclear chain reaction, to ensure that decay heat is dissipated, and to ensure that the spent fuel would be protected from hazards associated with natural phenomena or human activities for each storage site.

D.1.3.2 Operations. Operations commence following the defueling of the reactor, after fuel

modules are in a suitable holding container such as an M-130 or M-140 shipping container. The immobile dry storage container would be positioned at the storage location. Transfer of a module from the holding container to the dry storage container would be accomplished by using a shielded transfer container. All fuel transfers would be in accordance with procedures which would have been written, reviewed, and approved by trained, qualified, and specifically authorized personnel to perform such work. The transfer container would be landed on the holding container, and a module would be withdrawn from the holding container. The module would be secured and the loaded transfer container closed, moved into the dry storage container, and landed. The transfer container would be reopened and the module would be seated in the immobile storage container. The transfer container would then be removed and the process would be repeated until the container is filled with spent fuel modules. The transfer container would then be sealed.

Transfers of spent nuclear fuel to the immobile dry storage container would be in accordance with Naval Reactors Program requirements for radiation protection. Radiation containment devices would be used where necessary to prevent radioactivity from the workplace and from becoming airborne. The transfer and storage containers would contain shielding that minimizes radiation exposure to the workers during transfer and storage, ensuring that radiation levels at the site perimeter are indistinguishable from natural background levels.

D.1.4 Water Pool Storage

D.1.4.1 Water Pool Design Features. If the Record of Decision were to choose the alternative of

storing naval spent nuclear fuel in water pools, five water pools could be constructed at the designated storage site. Each water pool facility would be designed, built, and operated in accordance with DOE Order 6430.1A and consistent with the intent of Nuclear Regulatory Commission requirements in 10CFR72 and associated Regulatory Guides. The siting, design, construction, and approval of a water pool storage facility would commence with the Record of Decision and would take 6 to 9 years to complete. The design and construction of each water pool facility would be in accordance with local construction standards for each site.

Water pools operate by holding spent fuel modules in a deep pool of water. The water provides cooling for the spent fuel, a transparent medium for work activities, and shielding from radiation (see Attachment C). The structural materials of the fuel modules and naval vessels, as well as temperature and chemistry control of the water, would result in the spent fuel being resistant to corrosion. Corrosion-resistant racks below the water surface would be used to hold and position the fuel modules in place for handling and to prevent a critical mass. The water depth would be sufficient to provide shielding to protect workers and the environment during module movement and storage.

D.1.4.2 Operations. The naval spent nuclear fuel would be transferred to the water pool in a

suitable container, such as an M-130 or M-140 shipping container. The fuel modules would be transferred into the water pool using equipment and procedures that are similar to those used at ECF for unloading spent nuclear fuel from shipping containers. The fuel modules would be individually lowered and secured in the storage racks located on the pool floor. The use of a water pool for storage of naval spent nuclear fuel would provide an opportunity for limited visual inspection of the exterior of the fuel modules after removal from the naval vessels. This opportunity would not exist to the same extent for the dry storage alternatives.

D.1.5 Design Basis Considerations for Storage Containers and Water

Pools

The design of both the shipping and immobile dry storage containers would be in accordance with DOE Order 6430.1A and consistent with the intent of Nuclear Regulatory Commission requirements for independent spent fuel storage installations found in 10CFR72 and associated Regulatory Guides. Attachment F describes the exposures which would be expected during operational exposures and the exposures calculated for hypothetical accidents that could occur during interim storage of spent fuel at each shipyard and prototype location. The accident scenarios used to establish the requirements for the design of the interim storage facilities

D.1.5.1 Design Basis Considerations for Storage Containers.

- (1) Natural Phenomena. The fuel spacing within the container and the container would be designed to prevent a nuclear criticality, to ensure that heat produced by radioactive decay of fission products is adequately dissipated, and to ensure that the container would safely survive hazards associated with natural phenomena or flooding for each storage site. The shipping containers and the immobilized containers would be designed to withstand the most severe design basis seismic event expected for the storage sites. The seismic analysis would evaluate the external structures of the containers and the components associated with the containers. The containers and associated components would be designed to withstand the environment during other natural phenomena such as tornado winds, tornadoes, hurricanes, volcanic activity, design basis floods, and very large waves. The Record of Decision involves the need for new facilities for the interim storage of naval spent nuclear fuel, detailed site-specific seismic evaluations would be conducted and the results would be incorporated into the design of new facilities. The construction of these facilities to seismic standards which take into account the seismic character of the area would ensure that structures could withstand a seismic event. The adequacy of the storage facility would be documented in an assessment report for each location.

- (2) Man-made Hazards. The containers would be arranged to allow access for inspections, maintenance, and emergencies. This includes sufficient access for pressure, temperature, and radiological monitoring as well as for fire fighting and ambulances.

The containers would be designed to withstand a fire without losing fission product containment. Flammable liquids and gases as well as explosive materials would be prohibited in the storage area with the exception of fuel in motor vehicle support operations. Combustible materials such as wood, paper, and plastic would be kept to a minimum in the spent nuclear fuel storage areas.

The fuel spacing within the container and the container itself would be designed to prevent nuclear criticality, to ensure that the heat produced by radioactive decay is adequately dissipated, and to ensure that it would safely survive credible accidents for each storage site. Other man-made hazards such as truck accidents, airplane crashes, and objects dropped by cranes would also be addressed in an assessment report.

D.1.5.2 Design Basis Considerations for Water Pools.

- (1) Natural Phenomena. The spent nuclear fuel spacing within the water pool and the building support structures would be designed to prevent nuclear criticality, to ensure that heat produced by radioactive decay is adequately dissipated, and to ensure that it would safely survive hazards associated with natural phenomena for each storage site (i.e., seismic, tornadoes, hurricanes, volcanic activity, maximum expected floods, and very large waves). The water pools would be equipped with spent fuel storage racks for reactor modules. The racks would be designed to safely survive the above hazards. The Record of Decision involves the need for new facilities for the interim storage of naval spent nuclear fuel, detailed site-specific seismic evaluations would be conducted and the results would be incorporated into the design of new facilities. The construction of these facilities to seismic standards which take into account the seismic character of the area would ensure that structures could withstand a seismic event. The adequacy of the water pool facility would be documented in an assessment report for each location.

- (2) Man-made Hazards. The water pool facility would be designed to withstand without damage to the spent fuel within the water. Flammable liquids and gases as well as explosive materials would be prohibited in the vicinity of the storage area with the exception of incidental quantities of flammable solvents necessary to support operations. Combustible materials such as wood, paper, and plastic would be kept to a minimum in the spent nuclear fuel storage areas.

the water pool facility.

The fuel spacing within the water pool would be designed to prevent critical ensure that it would safely survive credible man-made accidents for each site. Other man-made hazards such as truck accidents, airplane crashes, and crane accidents would also be addressed in the safety assessment report.

D.1.6 Shipyard and Prototype Locations

This section describes conceptual locations at the shipyard and prototype site facilities could be located to service refuelings and defuelings of naval ships. Table D-1 provides land requirements for each storage method at each location, the construction cost and the associated operating cost.

D.1.6.1 Land Requirements. This section provides a summary of the land required for each of

the storage methods at each of the locations where refueling and defueling are planned through 2035.

These locations are the Portsmouth Naval Shipyard, the Puget Sound Naval Shipyard, Pearl Harbor Naval Shipyard, the Norfolk Naval Shipyard, and the Kesselring Site. Information of these sites is provided in Figures D-3 through D-7, indicating a possible storage facility at these facilities.

Figure D-3. Conceptual location of the interim storage site at Puget Sound Naval Shipyard at each of the locations where storage of naval spent nuclear fuel could be located. Table D-1 provides the number of containers and land required could be slightly less than identified as a result of actions taken during the transition period. As shown in Table D-1, storage of containers on railcars would typically require dedication of the most land. Table D-1. Square feet of land required for storage facility.

| Location | Number of Immobile Dry Storage Containers (1) | Number of Shipping Containers | Immobile Dry Storage Containers (2) (ft ²) | Shipping Containers on Concrete Pad (3) (ft ²) | Shipping Container on Railcar (ft ²) |
|--------------|---|-------------------------------|--|--|--|
| Portsmouth | 27-51 | 61 | 10,000-19,000 | 18,000 | 72,000 |
| Puget Sound | 153-206 | 219 | 57,000-77,000 | 64,000 | 260,000 |
| Pearl Harbor | 21-30 | 42 | 8,000-11,000 | 12,000 | 50,000 |
| Norfolk | 132-219 | 247 | 49,000-82,000 | 72,000 | 293,000 |
| Kesselring | 5-6 | 6 | 1,900-2,000 | 1,700 | 7,100 |

- (1) Range in required number of containers is due to options in conceptual design.
- (2) The immobile dry storage arrangement uses the containers stored on a concrete pad with one container diameter separation between adjacent containers. Each row is separated by a 15-foot wide accessway. Range in required land area is due to options in conceptual design.
- (3) The shipping container arrangement uses the containers stored on a concrete pad between adjacent containers. Each row is separated by a 15-foot wide accessway.
- (4) The water pool facility consists of a building that contains adequate space to store fuel and facilities (approximately 17,000 ft²) and a water pool with adjacent work area to accommodate the amount of spent nuclear fuel expected to be stored in the facility.

D.1.6.2 Site Construction, Container, and Operating Costs. This section provides estimated

costs associated with each alternative for storing spent nuclear fuel at the shipyard. The major cost factors include facility construction or site preparation costs, container and operating costs over the lifetime of the facility. Cost estimates are based on 1990 prices.

Table D-2 provides a summary of the estimated construction costs for each storage method at each shipyard and prototype location. The construction costs for immobile and shipping containers on concrete pads and shipping containers on railcars include estimated costs for concrete materials, rails (for railcars), or cranes for lifting and handling containers or (for concrete pad storage). The majority of the construction costs for concrete pads are for the concrete pad itself. Table D-2. Estimated site construction costs (millions of dollars).

| Location | Immobile Dry Storage Containers on Concrete Pad | Shipping Containers on Concrete Pad | Shipping Containers on Railcars | Construction and Installation of Water Pools |
|--------------|--|---|---------------------------------------|--|
| Portsmouth | 11-12 | 10 | 2 | 96 |
| Puget Sound | 15-16 | 13 | 5 | 141 |
| Pearl Harbor | 10-11 | 9 | 1 | 95 |
| Norfolk | 14-17 | 14 | 6 | 135 |
| Kesselring | 10 | 8 | 1(1) | 89 |
| Total | 60-66 | 54 | 15 | 556 |

(1) Estimate does not include costs associated with establishing railroad extension the storage site.

are associated with the need for a high-capacity crane. Water pool construction cost estimates of costs for construction of the water pool, building structure, and associated equipment. The table shows that construction costs for a water pool facility exceed alternatives, and that shipping containers on railcars involves the lowest construction costs. The water pool facility construction costs represent a complete facility ready to handle interim storage. The construction costs in Table D-2 for the other storage mode completed site construction without the cost of the containers (see Table D-3) to handle nuclear fuel.

Table D-3 provides a summary of the estimated costs to build shipping containers and immobile dry storage containers through 2035. The table shows that the immobile dry storage containers are the least expensive containers, and that the cost to build shipping containers on concrete pads is slightly lower than to rest on railcars. The difference in cost between shipping container options is due to the cost of a dedicated railcar during storage. Shipping container costs in Table D-3 would be reduced by about 13 percent due to actions taken during the transition period (these actions are described in Section 3.8) to ship containers from ECF. Consequently, the total costs for shipping containers on concrete pads and shipping containers on railcars considering the transition period would be about 2615 and 2760 million dollars, respectively.

Table D-3. Estimated container cost (millions of dollars).

| Location | Immobile Dry Storage Containers on Concrete Pad(1) | Shipping Containers on Concrete Pad | Shipping Container Railcars(|
|--------------|---|---|------------------------------------|
| Portsmouth | 55-100 | 319 | 337 |
| Puget Sound | 314-406 | 1145 | 1209 |
| Pearl Harbor | 43-59 | 220 | 232 |
| Norfolk | 271-431 | 1292 | 1363 |
| Kesselring | 10-12 | 31 | 33 |
| Total | 693-1008 | 3007 | 3174 |

(1) Range in container costs due to options in conceptual designs (see Sections D.1. and D.2). The end of the range represents container costs for the maximum fuel loading option (without containers).

(2) Includes the cost of an equal number of railcars and containers required for this option.

Table D-4 provides the estimated costs to operate a naval spent nuclear fuel facility. Operating costs include estimates of cost for personnel to monitor the facility, handle fuel when it arrives at the facility, and maintain the facility. These estimates do not include costs associated with eventual preparation of spent fuel for shipment to a site for disposition. Preparation costs cannot be estimated at this time because the method for preparing spent fuel has not been defined. Table D-4 shows that the lowest operating costs are associated with shipping containers on concrete pads and that water pool storage requires the highest operating costs. Table D-4. Estimated operating costs through the year 2035 (millions of dollars).

| Location | Immobile Dry Storage Containers on Concrete Pad | Shipping Containers on Concrete Pad | Shipping Containers on Railcars(2) | Water Pool |
|----------|--|---|--|------------|
|----------|--|---|--|------------|

| | | | | |
|--------------|----|----|----|---------|
| Portsmouth | 11 | 3 | 8 | 180 |
| Puget Sound | 23 | 4 | 24 | 206 |
| Pearl Harbor | 11 | 3 | 6 | 180 |
| Norfolk | 21 | 4 | 27 | 206 |
| Kesselring | 9 | 2 | 3 | 124 |
| Total | 75 | 16 | 68 | 896 (1) |

(1) For comparison, the estimated operating cost (personnel to monitor and handle fuel for the ICPP Building 666 for the same period is 232 million dollars.

(2) Includes cost to replace or refurbish railcar after prolonged storage.

D.1.6.3 Total Construction and Operating Costs. Table D-5 is a compilation of the data

contained in Tables D-1 through D-4, and calculated based on the entire 40-year per Record of Decision (1995 through 2035). This table shows that the total costs associated with immobile dry storage containers are the lowest of all the storage options considered. Storage at Puget Sound and Norfolk where the largest amounts of spent fuel would be stored in these cases, the total costs for using water pool storage are within the same range as immobile dry container storage.

Table D-5. Total costs through the year 2035 (millions of dollars).

| Location | Immobile Dry Storage Containers on Concrete Pad (1) | Shipping Containers on Concrete Pad | Shipping Containers on Railcars | Water Pool |
|--------------|--|---|---------------------------------------|------------|
| Portsmouth | 77-123 | 332 | 347 | 276 |
| Puget Sound | 352-445 | 1162 | 1238 | 347 |
| Pearl Harbor | 64-81 | 232 | 239 | 275 |
| Norfolk | 306-469 | 1310 | 1396 | 341 |
| Kesselring | 29-31 | 41 | 37 | 213 |
| Total Cost | 828-1149 | 3077 | 3257 | 1452 |

(1) Range in total costs due to options in conceptual design (see Section D.1.3.1) with the maximum loading concept.

D.1.7 Time Required to Implement Each Storage Method

If the Record of Decision were to choose one of the alternatives involving spent nuclear fuel at shipyards and prototype sites, some period of time would be required for the decision to fully implement the selected storage alternative. This section examines the time required to implement each storage method.

D.1.7.1 Container Storage. Implementation of the alternatives involving use of immobile dry

storage containers and shipping containers could be viewed as a three-phase process. The first phase would cover the time required to design the container or container modification, to the design, to approve the container, to establish contracts for container fabrication of the first container. During this phase, the shipyards and prototype sites where the containers would be stored would also construct or modify the container storage location as appropriate. For immobile dry storage containers, this phase would take about 5 years, required to design and accept the container design, 1 year is needed for approval, and 2 years are required to build the container. For containers designed for both storage and shipping, this process would take about 5 years, based on 1 year to design the modifications, the container, and 3 years to build the container.

The second phase would involve establishing funding. This will take approximately 1 year to complete. The third phase of the implementation period would involve fabrication of the required containers. The estimate of the number of containers is based on the projected naval vessel refuelings and current estimates of the amount of spent nuclear fuel to be stored in the containers. Although production rates for immobile dry storage containers are unknown, they can be approximated from existing production rates for shipping containers. With current manufacturing capabilities, 3 years are required to build

container, and the manufacturing capacity is about six containers per year. This p would need to be accelerated to 18 to 24 containers per year by increasing the numb manufacturers and by making fabrication process improvements. If the production ra dry storage containers and shipping containers is the same as that of M-140 contain rates can be increased as noted above, the supply of immobile dry storage or shippi would meet the demand for these containers at some point after the first several ye transition period, when an insufficient number of containers would be available to fuel planned to be removed from U.S. Navy nuclear-powered vessels, some other means naval spent nuclear fuel would be needed. As described in Section 3.8 of this EIS, a transition period of 3 years of shipping followed by 3 years of allowing naval sp be stored in shipping containers at shipyards would provide the necessary storage s

D.1.7.2 Water Pool Storage. If 6 to 9 years would be required to design, approve, and construct

a water pool facility and this process would be initiated for each location within Record of Decision, water pools would be available for storage of naval spent nucle 10 years following the Record of Decision. During the transition period, when wate under construction at selected locations, some other means of spent nuclear fuel st needed, such as the method described in Section 3.8.

D.1.8 Summary

Table D-6 summarizes the major advantages and disadvantages of the spent nucl storage alternatives previously discussed in this attachment.

Table D-6. Comparison of naval spent nuclear fuel storage alternatives.

| Storage Mode | Advantages | Disadvantages |
|-----------------------------|--|--|
| 1. Shipping Container | | |
| A. Storage on Railcars | <ol style="list-style-type: none"> 1. Least amount of container handling after arrival at storage location. 2. Eliminates the need to remove spent fuel modules from the transfer container upon arrival at the storage site. | <ol style="list-style-type: none"> 1. Railcars must be refurbished or replac after prolonged stora 2. Requires the largest area of the storage o except for Kesselring 3. Shipping containers a more expensive than immobile dry storage containers and water (water pools cost more when small fuel quantit are stored such as at Kesselring). |
| B. Storage on Concrete Pads | <ol style="list-style-type: none"> 1. Eliminates the need to remove spent fuel modules from the transfer container upon arrival at the storage site. 2. Concrete pads are less expensive than railcar storage if railcars must be replaced or refurbished. | <ol style="list-style-type: none"> 1. More container handli required compared to railcar storage optio containers will not n be removed from railc 2. Higher total cost tha immobile dry storage containers and water (*when large quantiti fuel are stored). |

Table D-6 (Cont).

| Storage Mode | Advantages | Disadvantages |
|------------------------------------|---|---|
| 2. Immobile Dry Storage Containers | <ol style="list-style-type: none"> 1. Lowest total costs of all the storage options. | <ol style="list-style-type: none"> 1. The maximum fuel load concept requires that containers be filled water for cooling pur for several years aft |

removal from the reac
This requires additio
maintenance and sligh
increases risk of low
contamination spillag
during accidents.

- | | | |
|-----------------------|---|--|
| 3. Water Pool Storage | <ol style="list-style-type: none"> 1. Has a lower total cost than shipping containers, except for Pearl Harbor and Kesselring which have less containers. 2. Provides opportunity for conducting visual examinations. | <ol style="list-style-type: none"> 2. Must remove spent fue from transfer contain load it into immobile container. 1. Has the highest opera costs of all the stor options. 2. Must remove spent fue from transfer contain load into water pool. |
|-----------------------|---|--|

D.2 INSPECT HIGH PRIORITY FUEL AT PUGET SOUND NAVAL

SHIPYARD

D.2.1 Introduction

This section of the attachment discusses the alternative of inspecting a limi spent nuclear fuel at Puget Sound Naval Shipyard (hereafter referred to as Puget So information on nuclear fuel performance for use in the development of advanced nucl The inspections would be performed at the shipyard's existing Water Pit Facility. of fuel inspected would be stored at Puget Sound following inspection, and all othe be stored in a facility at or near the refueling or defueling sites until the time storage becomes available.

D.2.2 Water Pit Facility Description

The Water Pit Facility is located at the west side of Dry Dock 5, within the Puget Sound. This zone consists of facilities involved in ship construction and re and conversions. The area is bounded by Decatur Avenue on the north, the waterfron the Naval Supply Center on the west, and the main gate on the east. The Water Pit approximately 411 meters (1350 feet) from the nearest shipyard public property boun D-8 illustrates the layout of the Water Pit Facility.

The Water Pit Facility was originally constructed to provide the shipyard wit to refuel nuclear-powered aircraft carriers, with the work for the first such refue expected to commence in approximately 2006. To date, the facility water pool has b refueling equipment demonstrations and testing.

Figure D-8. Puget Sound Naval Shipyard Water Pit Facility. The following key feat original aircraft-carrier refueling mission. Because of these design features, the considered suitable for limited naval spent fuel inspection operations.

1. A water pool for disassembly, assembly, and holding of fuel cells. The la water pool is described below.
2. A work area for unpackaging, inspection, and preparation of new fuel clust associated equipment
3. An area for loading of shipping containers
4. A general use work area to support miscellaneous refueling support operati

The Water Pit Facility is divided into two distinctive structures. The high radiologically controlled area containing the water pool and general work areas dis structure is designed to withstand the effects of design basis natural phenomena an

failures of adjoining or adjacent structures without damage to the water pool or co water pool. The high bay walls are constructed of concrete to a height of 3.7 mete ground level. The second structure is the Personnel Support Building which houses support areas. This structure is designed to meet the require-ments of established naval facilities standardized criteria for structural design.

The water pool measures 7.3 meters (24 feet) wide x 20.4 meters (67 feet) lon (36.5 feet) deep with a water depth of 10.5 meters (34.5 feet). It includes four w side of the pool at the east end to support refueling operations and a fuel holding of the pool. Three of the four work areas are a nominal 2.1 meters (7 feet) x 2.1 the fourth area is a nominal 2.6 meters (8.5 feet) x 2.1 meters (7 feet). The tran center of the pool is provided for all fuel and non-fuel movements. The water pool provisions for isolation gates for each work area, for the fuel holding area, and f isolation gate arrangement provides the capability to separate the various areas of required. The dry pit, measuring 7.3 meters (24 feet) wide x 4.9 meters (16 feet) (36.5 feet) deep, permits expansion of the water pool as needed.

D.2.3 Limited Inspection Operations

If future naval spent fuel examinations could not be accomplished at current capacity which was available would be used to best advantage. Only naval spent nuc identified as having the greatest scientific value would be selected for detailed e Generally, this is spent nuclear fuel which is the first of a kind design or which special interest.

Naval nuclear-powered ships would continue to be refueled and defueled at var across the country. Most of the spent fuel would be stored in a facility at or nea defueling sites until the time that permanent geologic storage becomes available. identified as high priority would be transported by railcar to Puget Sound in stand shipping containers. Following its receipt in the Water Pit Facility's railcar wor container would be prepared for fuel cell removal (dust cover removed, leveled, fil containment installed, access plug removed). The fuel cells would be removed from container, one at a time, and transferred to the water pool in a shielded transfer would be discharged into the pool and placed in the holding racks to await examinat completion of examination work, the spent fuel would be stored at Puget Sound as de Section D.1. Storage facilities would have to be designed and certified to accommo sections resulting from spent fuel examinations as well as intact modules.

The following major items of water pool equipment (or equivalent) are conside to support a high-priority naval spent nuclear fuel examination program. Also nece relatively small and portable cameras and light sources for visual inspections. Th support those spent fuel examinations currently performed in the ECF water pools at described in Section B.4.1 of Attachment B and summarized below.

EQUIPMENT

| ITEM | PURPOSE |
|------------------------------------|---|
| Bandsaw/ Upender | Remove non-fuel structurals above & below fuel region to provide access for inspection and to rotate cells between vertical and horizontal orientations |
| Universal Inspection Station | Measure fuel cell dimensions |
| Vertical Inspection Gage | Trace contour of surfaces of fuel cell assemblies and control rods |
| Milling Machine | Section fuel cells into subassemblies, preassemblies, and elements for other examinations |

Based on floor space requirements, the Water Pit Facility water pool and dry accommodate spent nuclear fuel examinations without removal of work area partition without removal of the aircraft carrier refueling equipment. As a result, Puget So longer have the capability to refuel nuclear-powered aircraft carriers. Expansion Facility to accommodate simultaneous refueling and examination operations is undesi proximity of other shipyard facilities.

Puget Sound does not have a shielded cell examination capability. Two option considered for implementing such a capability:

1. Transfer fuel sections from Puget Sound to a shielded cell facility at another Reactors site such as the Knolls Atomic Power Laboratory near Schenectady, or the Bettis Atomic Power Laboratory near Pittsburgh, Pennsylvania. This requires additional shipments of spent fuel sections across the country. They would be transported in shipping casks which would have to be certified for this purpose.
2. Construct shielded cells at Puget Sound. These cells would necessarily be at a distance from the Water Pit Facility since sufficient space is not available at the facility or adjacent to it in the industrial zone of the shipyard. In order of transferring items for examination between the water pool and the shielded cells would have to be implemented. Shielded cask movements via truck and cart via underground tunnel are two possible means of transfer. This option is chosen because it involves construction of a new facility but does not provide direct communication between the water pool and shielded cells.

Based on the above discussion, the alternative of examining a limited amount of nuclear fuel would include a full range of water pool visual and dimensional inspection at the Sound Water Pit Facility and a full range of shielded cell examinations at another facility. This alternative would therefore include all INEL-ECF capabilities as described in the EIS.

B.4.3 of Attachment B.

D.2.4 Advantages and Disadvantages of this Alternative

Advantages

1. Portions of the naval spent nuclear fuel examination program could be moved to INEL-ECF without having to construct new facilities. A full range of water pool inspections could be accomplished at Puget Sound. A full range of shielded cell examinations could be accomplished at another Naval Reactors site.

Disadvantages

1. The small size of the water pool complicates placement of inspection equipment. As a result, the equipment would be limited in nature and would require removal of pool work area partition walls and removal of aircraft carrier refueling equipment. As a result, Puget Sound would no longer have the capability to refuel nuclear-aircraft carriers.
2. Transferring items for examination between the water pool and shielded cells would involve additional spent fuel shipments across the country and would require certification of a container for this purpose.

D.2.5 Facility Support Systems

The systems which were intended to support the aircraft carrier refuelings were the limited naval spent fuel inspection efforts. These include the water pool fluid circulation and ventilation systems, and the normal and emergency electrical power systems.

D.2.6 Radiation Sources

The primary sources of radiation in the Water Pit Facility would be the spent nuclear fuel associated irradiated components which are handled during inspection operations. Radiation from the fission products which reside in the fuel region of the depleted clusters is contained by the fuel cladding. The cladding around the fuel region would not be penetrated by cutting or sectioning operation in the Water Pit Facility. Irradiated non-fuel components are sources of radiation, as are corrosion products which reside on all external surfaces. Operations could cause some of the corrosion products to become detached from the surfaces. Therefore, in addition to direct radiation, contamination must be considered in the radiation sources.

The water pool water is treated by the filtration and purification system to reduce waterborne radioactivity as low as reasonably achievable, typically less than 1 x 10⁻⁶ Ci/ml. This level of activity is below the concentration limit in 10CFR20, Attachment A for liquid effluents released to the general environment. The vessels and piping in the pool then become potential radiation sources. The water must be considered a source even though the radiation level will be very low. The waterborne radioactive material causes equipment

to become radiation sources, the water pool floor to become contaminated, and a rad ring to form on the walls of the water pool at the water surface. Even considering sources contributing to the ambient radiation level in the water pool area, the con exercised will ensure that the overall source is minimal and the occupational expos as reasonably achievable.

There would normally be no airborne radioactivity generated by the handling o the water pool. However, very low levels of airborne activity (approximately 1×1 Co-60/ml) have been detected near the surfaces of other water pools. This level of the concentration limit in 10CFR20, Attachment B, Table 2 for airborne effluents re general environment. The presence of even low-level airborne contamination will ev the ventilation system ductwork and HEPA filters becoming sources of radiation. Th over a very long period of time and the radiation levels would be controlled to a v noted above, the controls which are exercised will ensure that the occupational exp low as reasonably achievable.

D.2.7 Radiological Protection Features

The facility is designed to protect workers and the general public from radio Controls are such that workers receive much less than the allowable limits for radi radioactivity. The ventilation system is designed to mitigate the consequences of of radionuclides within the Water Pit Facility building and to limit the atmospheri stack. The double-walled (reinforced concrete, stainless steel liner) water pool i leakage under design earthquake force loading conditions. The radioactive fluid sy zero liquid discharge to the environment during Water Pit Facility operations.

D.2.8 Estimated On-Site Dose Assessment

The occupational radiation exposure for workers performing limited spent fuel the Water Pit Facility is expected to be consistent with that of ECF workers perfor operations at INEL. As discussed in Section 5.2.12.1, radiation exposures to ECF w have averaged approximately 100 mrem per year. The person-rem per year for the Wat will vary with the manning level which is dependent on the spent fuel inspection ac the facility. However, the maximum manning level is anticipated not to exceed 60 p

D.2.9 Seismic Design

Structural loadings due to seismic activity were determined as follows. Buil response spectra for the horizontal and vertical directions were obtained from a th damping mass spring model of the high bay which included soil-structure interaction 0.35 g ground acceleration value resulting from the seismic design analysis. The h superstructure and substructure were analyzed using the floor response spectra in s element computer models. The superstructure model was subjected to structural load a 113.5-metric ton (125-ton) load lifted by the large overhead crane. The combined loads with the seismic loads were applied to the substructure model at the column b The substructure model was subjected to the design earthquake response spectra. Th repeated for other combinations of structural loads with wind or tornado loads. Me checked and designed for the maximum stress from any of the loading combinations. water pool is designed to contain the pool water under design earthquake force load

ATTACHMENT E - DESCRIPTION OF RECEIPT, HANDLING, AND EXAMINATION OF

NAVAL SPENT NUCLEAR FUEL AT ALTERNATE DOE FACILITIES

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ATTACHMENT E

DESCRIPTION OF RECEIPT, HANDLING, AND EXAMINATION OF NAVAL SPENT NUCLEAR FUEL AT ALTERNATE DOE FACILITIES

E.1 DISCUSSION

This attachment describes the options for establishing new or modified facilities essentially duplicate the capabilities of the existing Expended Core Facility (ECF) National Engineering Laboratory (INEL). Also discussed herein are the differences facility, which is described in detail in Attachment B.

The capabilities of the ECF at INEL include detailed examinations of spent nuclear reactors and test specimens from the Advanced Test Reactor (ATR) at the INEL Area. It would be possible to provide ECF capabilities at an alternate DOE facility Site, Hanford Site, Oak Ridge Reservation, or Nevada Test Site) by constructing an facility. At Savannah River or Hanford, ECF capabilities could also be provided by existing facility. The preferred locations for siting an ECF at Savannah River, Hanford, and the Nevada Test Site are described in Sections 4.3.1, 4.4.1, 4.5.1, and 4.6.1, main advantage of new construction is that the facility can provide all capabilities at the ECF at INEL without limitations. The new construction water pool and shield would be constructed in such a manner as to duplicate, as much as possible, the capabilities of the ECF at INEL. The existing ECF is highly capable, having been designed to accomplish required by the Naval Nuclear Propulsion Program. Key disadvantages of new construction however, are high cost and the time necessary to initiate and complete construction.

Modification of an existing facility at Savannah River or Hanford which has all the features that are required in a functional ECF would enable reductions in cost full capability, depending on how many facility modifications are required. A disadvantage is that some of the methods currently in use at the ECF at INEL may also require more effectively and promptly utilize an existing facility, and such modifications may duplicate capabilities of the examination facility. The existing facility that can be made at the River Site is the Barnwell Nuclear Fuel Plant (hereafter referred to as the Barnwell unused and available following acquisition from its present private corporate owner facility on the Hanford Site is the Fuels and Materials Examination Facility (FMEF) and available immediately. Sections E.2 and E.3 describe the modifications to existing current processes that would be needed to provide the complete range of ECF capabilities at the Barnwell Plant and the FMEF. Section E.4 provides a discussion of how naval spent specimen examination work would proceed through the interim period as this work is transferred from the ECF at INEL to the ECF location at the alternate DOE facility.

Receipt and handling of naval spent fuel at the new ECF location at the alternate DOE facility would be similar to receipt and handling of spent fuel at the ECF at INEL as described in Attachment B. Following all examinations at the new ECF, most of the spent fuel loaded in the water pool into shipping casks for transport to the long-term fuel storage facility. The spent fuel would remain at this location until the time that disposition is possible.

The new ECF would also duplicate the capabilities of the ECF at INEL with respect to assembly, disassembly, and examination of ATR irradiation test specimens.

E.2 USE OF THE BARNWELL PLANT AT SAVANNAH

RIVER

FOR ECF WORK

The Barnwell Plant is not owned by DOE but could be acquired and incorporated Savannah River Site property. It has a water pool complex with about 433 square meters (square feet) of surface area (see Figure E-1) that can be utilized with minor modification for unloading of naval fuel transport casks in a manner virtually identical to that employed at INEL. An overhead crane running the length of the water pool would have to be added providing naval spent nuclear fuel and test specimen examination capabilities comparable at INEL would entail an expansion of the Barnwell Plant water pool to at least two times the size. The design of the Barnwell Plant facility provides for such an expansion in which while the existing water pool remains functional in a reduced capacity mode.

Figure E-1. Plan view of the Barnwell Plant Fuel Receiving and Storage Station. The plant using a combination of the three remote maintenance cells and the eight sample cells. Material would be transferred from the water pool to the remote maintenance conveyor. The crane equipment maintenance gallery and the upper level of the remote maintenance cells are connected by a shielded door; these cells are connected to the remote maintenance below by hatches (see Figure E-2). Additional work stations (viewing window and manways) would have to be added to service these cells. The remote maintenance cells are comparable to sample and analytical cells above via a waste chute which would have to be upgraded to provide transfer capability between these cell areas. Methods would have to be developed for material movement from one shielded cell elevation to another. The combined length of the ECF cells at INEL is less than 57.9 meters (190 feet). The combined length of the Barnwell maintenance cells and sample and analytical cells is greater than 67.1 meters (220 feet). Sufficient cell work space should be available. There are also five contact maintenance cells available, although at present they have no workstations and are not connected to the other cell area, or to the water pool. An alternative to the Barnwell Plant water pool would be to use the contact maintenance cells for some of the operations presently performed in the ECF water pool at INEL. Varying amounts of existing equipment and piping in the Barnwell shielded cells would have to be removed and disposed.

Once modified, the Barnwell Plant would provide the full range of water pool cell examination capabilities. However, the arrangement of the cells in the fuel handling facility make material movement within the facility more difficult than material movement at INEL. As a result, throughput in the Barnwell Plant could be adversely affected.

E.3 USE OF THE FUELS AND MATERIALS EXAMINATION FACILITY

AT HANFORD FOR ECF WORK

The FMEF on the DOE Hanford Site in Washington currently has a large shielded complex that is suitable for ECF-type shielded cell operations with several modifications. Modifications primarily entail the logistics associated with installing the equipment for transporting items for examination to and from this equipment.

Figure E-2. Elevation looking north in the Barnwell Plant fuel handling area. At the facility capabilities would be to establish a dry cell facility. The FMEF main process cell, sample cell, and upper process cell were evaluated for such a facility (see Figure E-3). Material would be transferred from shielded casks in the shipping and receiving area to a decontamination cell via a ceiling port. At present, there are only small penetrations into the decontamination cell and main process cell; this would have to be upgraded to facilitate transfer. The combined surface area of the three cells is about 706 square meters compared to at least 866 square meters (9320 square feet) for the conceptual expanded water pool discussed previously. This suggests that the full ECF water pool capabilities provided in the dry cell facility. In addition, one or more of the process cells included in the shielded cell complex (see next paragraph). Removal of decay heat and irradiation test specimens in temporary dry storage would have to be evaluated. Duplication of ECF spent fuel and test specimen examination capabilities at FM would require construction of a new water pool at least two times the present size of the Barnwell Plant. The location of the pool and the means for transferring items between the pool and the complex would have to be evaluated.

It is envisioned that the full ECF shielded cell capabilities could be provided

combination of the main process cell and the 14 process support cells. The main pr connected to the process support cells below by hatches (see Figure E-3). There ap sufficient workstations (viewing window and manipulator ports) servicing all cells. have to be developed for material movement from one shielded cell elevation to the combined length of the FMEF main process cell and process support cells is greater (250 feet), so that sufficient cell work space should be available. The decontamin process cell would be available in support of shielded cell operations. The FMEF s essentially empty.

Once modified, the FMEF would provide the full range of water pool and shield examination capabilities. However, the arrangement of the cells in the fuel handli separation of the water pool and shielded cells would make material movement within more difficult than material movement at the ECF at INEL. As a result, throughput could be adversely affected.

E.4 INTERIM OPERATIONAL PERIOD

Figure E-3. FMEF fuel handling area.

A transitional period will exist between the date that the Record of Decision date that the alternative selected can be fully implemented (unless the selected al ECF operations at INEL). This transition period would be approximately 6 years. I that all ECF work be completely transferred to an alternate DOE facility, then acti be taken to minimize the disruption in examination capability for naval spent nucle test specimens. This section discusses how this will be accomplished if the altern option is selected in the Record of Decision.

The Barnwell Plant would have to be acquired by the DOE from its present priv owners. It is estimated that less than \$800 million in acquisition, modification, would complete the Barnwell Plant for ECF usage.

The FMEF at Hanford is already owned by the DOE but it appears to require a g amount of design effort to be a fully functional ECF since a large water pool would constructed and tied in to the shielded cell complex in order to initiate fuel rece that less than \$800 million in modification and construction costs would complete t usage.

During the transitional period between the Record of Decision and full implem selected alternative, shipments of naval spent nuclear fuel to the ECF at INEL woul pending construction of storage and examination facilities at the new site. All na would then be transferred to the new site.

ATTACHMENT F - ANALYSIS OF NORMAL OPERATIONS AND ACCIDENT

CONDITIONS

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ATTACHMENT F

ANALYSIS OF NORMAL OPERATIONS AND ACCIDENT CONDITIONS

This attachment presents estimated environmental consequences, event probability product of probability and consequence) for both normal operations and postulated a related to the storage and examination of naval spent nuclear fuel. Normal operations evaluated to estimate the potential for releases of both radioactive material and results of these analyses are presented in terms of the health effects to facility predicted due to the release of radioactive materials and toxic chemicals into the environmental factors are also presented, based on the amount of land which could be postulated accidents.

Analysis results are presented for several different Department of Energy (DOE) shipyard locations which are being considered as alternative sites for future naval storage and examination. The DOE facilities evaluated include the Idaho National Engineering (INEL), Savannah River Site, Hanford Site, Nevada Test Site, Oak Ridge Reservation (as Oak Ridge), and Kenneth A. Kesselring Site. Puget Sound Naval Shipyard, Pearl Harbor Naval Shipyard, Norfolk Naval Shipyard, and Portsmouth Naval Shipyard have also been evaluated for spent nuclear fuel operations.

SUMMARY

Analyses of normal operations and design basis and beyond design basis hypotheses were performed to estimate the potential consequences due to release of radioactive chemicals. The analysis results for radiological operations have been summarized by alternatives being considered in the Environmental Impact Statement.

Historical Accidents

The Naval Nuclear Propulsion Program has an outstanding nuclear safety record of reactor-years of operation and more than 300 refuelings and defuelings of Naval reactors. There has been no nuclear reactor accident, criticality accident, transportation accident, or having a significant effect on the environment.

Summary of Naval Spent Nuclear Fuel (SNF) Alternatives

| Alternative | Description of SNF Activity |
|------------------|---|
| No Action | SNF retained at shipyards and Kesselring. Dry storage in containers only. |
| Decentralization | SNF retained at shipyards and Kesselring. Either dry containers or |
| No Examination | water pool storage would be used. |

| | |
|---|---|
| Decentralization Limited Examination | SNF retained at shipyards and Kesselring. Either dry containers or water pool storage would be used. Limited SNF shipments to Sound Naval Shipyard for examination. |
| Decentralization Full Examination | All SNF shipped to INEL-ECF for examination. All SNF returned to origin for storage in either dry containers or water pools. |
| Planning Basis | SNF would be received, examined, and stored at INEL as in years. The proposed dry cell facility would be completed at ECF. |
| Regionalization or Centralization | SNF would be received, examined, and stored at either INEL Hanford, Savannah River, Nevada Test Site, or Oak Ridge. |

Normal Operations

Table F-1 presents the estimated number of fatal cancers per year to the gene within a 50-mile radius of each facility due to radiological releases from normal operations. This table was calculated using the methods described in Section F.1.3. The numbers are very low at all locations and for all alternatives.

The ISC2 computer code (EPA 1992b) was used to estimate the concentration of released during normal operations. The results show that for INEL, Hanford, Savannah Test Site, the Barnwell Plant, and Oak Ridge, no ambient air quality standards would therefore, no adverse effects are expected. Heating boilers and emergency diesel generators at the Navy shipyard locations and thus selection of these alternate locations would measurable increase in emissions.

Hypothetical Accident Evaluations

Several hypothetical accidents were analyzed at each facility for each of the results are summarized in Tables F-2 and F-3. The results in these tables were calculated using the methods described in Section F.1.3. Both fatal cancers from the maximum foreseeable location and the most severe risk from a facility accident at each location are presented as the product of the consequences of an event multiplied by the probability of that event associated with the accidents analyzed have not been added together in order to avoid the impression that all risks have been calculated. The risks presented in this appendix range of accidents which might make a detectable contribution to overall risk and a would not be expected to result in increases in calculated risk. The facility accident highest risk is a drained water pool at INEL, Hanford, Puget Sound, Portsmouth, and Savannah River, Pearl Harbor, Norfolk, the Nevada Test Site, and Oak Ridge, an air storage area or a dry cell facility results in the greatest risk. As was the case in the evaluation, the accident risk is very low at all locations and for all alternatives.

Table F-4 presents a summary of the risk of fatal cancers by alternative for the most severe facility accident for each alternative. Consistent with the detailed table shows that all alternatives and all locations associated with spent fuel examination.

Tables F-5 through F-8 present a summary by alternative of the impacts from a nuclear fuel facility radiological accidents which were analyzed.

A shipping accident in Puget Sound, at a location in the shipping lane approximately 10 miles from Seattle, was also analyzed using the methods described in this Attachment. This analysis results in a fire onboard the ship which involves spent nuclear fuel shipping containers. Compared to the facility accidents analyzed at Puget Sound Naval Shipyard, this shipping accident lower risk of fatal cancers than the most severe facility accident at the shipyard.

The EPI computer code (Homann 1988) was used to estimate the concentration of released in the event of two postulated accident conditions. One postulated accident is a spill and fire at ECF and the alternate DOE sites and the other postulated accident is a fire at ECF, the alternate DOE sites, and the shipyard locations. The chemical analysis is presented in Table F-1. Number of fatal cancers per year from normal operations (fatalities per 50-mile radius of site).

DRY STORAGE AT NAVAL NUCLEAR PROPULSION PROGRAM SITES, WATER POOL STORAGE AT DOE SITES

| No Action | Decentralization-No Examination | Decentralization-Puget Sound Exam | Decentralization-INEL Exam | Planning Regional Centralization-INEL |
|-----------|---------------------------------|-----------------------------------|----------------------------|---------------------------------------|
|-----------|---------------------------------|-----------------------------------|----------------------------|---------------------------------------|

| | | | | | |
|------------|-------------------|--------------------------|--------------------------|--------------------------|----------|
| INEL | 0.00 | 0.00 | 0.00 | 8.50 x 10 ⁻⁷ | 8.50 x 1 |
| Hanford | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| Savannah | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| River | | | | | |
| Nevada | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| Test Site | | | | | |
| Oak Ridge | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| Puget | 1.20 x | 1.20 x 10 ⁻⁶ | 6.62 x | 1.20 x 10 ⁻⁶ | 0.00 |
| Sound | 10 ⁻⁶ | | 10 ^{-5**} | | |
| Pearl | 9.30 x | 9.30 x 10 ⁻⁹ | 9.30 x 10 ⁻⁹ | 9.30 x 10 ⁻⁹ | 0.00 |
| Harbor | 10 ⁻⁹ | | | | |
| Portsmouth | 2.30 x | 2.30 x 10 ⁻⁷ | 2.30 x 10 ⁻⁷ | 2.30 x 10 ⁻⁷ | 0.00 |
| | 10 ⁻⁷ | | | | |
| Norfolk | 2.10 x | 2.10 x 10 ⁻⁵ | 2.10 x 10 ⁻⁵ | 2.10 x 10 ⁻⁵ | 0.00 |
| | 10 ⁻⁵ | | | | |
| Kesselr- | 4.10 x | 4.10 x 10 ⁻¹² | 4.10 x 10 ⁻¹² | 4.10 x 10 ⁻¹² | 0.00 |
| ing | 10 ⁻¹² | | | | |
| Total | 2.24 x | 2.24 x 10 ⁻⁵ | 8.74 x 10 ⁻⁵ | 2.33 x 10 ⁻⁵ | 8.50 x 1 |
| | 10 ⁻⁵ | | | | |

WATER POOL STORAGE AT ALL SITES*

| | | | | | |
|------------|-------------------|-------------------------|-------------------------|-------------------------|----------|
| INEL | 0.00 | 0.00 | 0.00 | 8.50 x 10 ⁻⁷ | 8.50 x 1 |
| Hanford | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| Savannah | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| River | | | | | |
| Nevada | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| Test Site | | | | | |
| Oak Ridge | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| Puget | 1.20 x | 6.50 x 10 ⁻⁵ | 6.50 x 10 ⁻⁵ | 6.50 x 10 ⁻⁵ | 0.00 |
| Sound | 10 ⁻⁶ | | | | |
| Pearl | 9.30 x | 7.00 x 10 ⁻⁵ | 7.00 x 10 ⁻⁵ | 7.00 x 10 ⁻⁵ | 0.00 |
| Harbor | 10 ⁻⁹ | | | | |
| Portsmouth | 2.30 x | 2.30 x 10 ⁻⁵ | 2.30 x 10 ⁻⁵ | 2.30 x 10 ⁻⁵ | 0.00 |
| | 10 ⁻⁷ | | | | |
| Norfolk | 2.10 x | 1.40 x 10 ⁻⁴ | 1.40 x 10 ⁻⁴ | 1.40 x 10 ⁻⁴ | 0.00 |
| | 10 ⁻⁵ | | | | |
| Kesselring | 4.10 x | 4.10 x 10 ⁻⁵ | 4.10 x 10 ⁻⁵ | 4.10 x 10 ⁻⁵ | 0.00 |
| | 10 ⁻¹² | | | | |
| Total | 2.24 x | 3.39 x 10 ⁻⁴ | 3.39 x 10 ⁻⁴ | 3.40 x 10 ⁻⁴ | 8.50 x 1 |
| | 10 ⁻⁵ | | | | |

*Under No Action alternative, dry storage at Naval Nuclear Propulsion Program site

**Includes dry storage and water pool examination under this alternative

Table F-2. Number of fatal cancers from a maximum foreseeable accident (fatalities accident over a 50-year period to general population within a 50-mile radius of si

DRY STORAGE AT NAVAL NUCLEAR PROPULSION PROGRAM SITES, WATER POOL STORAGE AT DOE SI

| | No Action | Decentral i-zation- No Examination | Decentral i-zation- Puget Sound Exam | Decentral i-zation- INEL Exam | Planning Basis/ Regional i-zation Centrali zation- INEL |
|--|--------------|---|---|-------------------------------------|---|
|--|--------------|---|---|-------------------------------------|---|

| | | | | | |
|------|------|------|------|--------|--------|
| INEL | 0.00 | 0.00 | 0.00 | 1.70 x | 1.70 x |
|------|------|------|------|--------|--------|

| | | | | | |
|----------------------------------|---------------|----------------|-----------------|----------------|----------------|
| | | | | 10-2 | 10-2 |
| Hanford | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| Savannah River | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| Nevada Test Site | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| Oak Ridge | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| Puget Sound | 1.7 x 10-2 | 1.7 x 10-2 | 5.1 x 10-1** | 1.7 x 10-2 | 0.00 |
| Pearl Harbor | 2.60 x 101 | 2.60 x 101 | 2.60 x 101 | 2.60 x 101 | 0.00 |
| Portsmouth | 9.00 | 9.00 | 9.00 | 9.00 | 0.00 |
| Norfolk | 1.6 x 101 | 1.6 x 101 | 1.6 x 101 | 1.6 x 101 | 0.00 |
| Kesselring | 7.50 | 7.50 | 7.50 | 7.50 | 0.00 |
| Max | 2.60 x 101 | 2.60 x 101 | 2.60 x 101 | 2.60 x 101 | 1.70 x 10-2 |
| WATER POOL STORAGE AT ALL SITES* | | | | | |
| INEL | 0.00 | 0.00 | 0.00 | 1.70 x 10-2 | 1.70 x 10-2 |
| Hanford | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| Savannah River | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| Nevada Test Site | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| Oak Ridge | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| Puget Sound | 1.7 x 10-2 | 5.1 x 10-1 | 5.1 x 10-1 | 5.1 x 10-1 | 0.00 |
| Pearl Harbor | 2.60 x 101 | 1.10 | 1.10 | 1.10 | 0.00 |
| Portsmouth | 9.00 | 3.40 x 10-1 | 3.40 x 10-1 | 3.40 x 10-1 | 0.00 |
| Norfolk | 1.6 x 101 | 6.0 x 10-1 | 6.0 x 10-1 | 6.0 x 10-1 | 0.00 |
| Kesselring | 7.50 | 2.50 x 10-1 | 2.50 x 10-1 | 2.50 x 10-1 | 0.00 |
| Max | 2.60 x 101 | 1.10 | 1.10 | 1.10 | 1.70 x 10-2 |

*Under No Action alternative, dry storage at Naval Nuclear Propulsion Program site

**Includes dry storage and water pool examination under this alternative
Table F-3. Most severe risk from a facility accident (probability of fatalities per
accident to general population within a 50-mile radius of site).

DRY STORAGE AT NAVAL NUCLEAR PROPULSION PROGRAM SITES, WATER POOL STORAGE AT DOE SI

| | No Action | Decentral i-zation- No Examination | Decentral i-zation- Puget Sound Exam | Decentral i-zation- INEL Exam | Plannin Basis/ Regiona i-zatio Central ization- INEL |
|---------------------|--------------|---|---|-------------------------------------|--|
| INEL | 0.00 | 0.00 | 0.00 | 1.70 x 10-7 | 1.70 x 10-7 |
| Hanford | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| Savannah River | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| Nevada Test Site | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| Oak Ridge | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |

| | | | | | |
|----------------------------------|----------------|----------------|------------------|----------------|----------------|
| Puget Sound | 1.7 x 10-7 | 1.7 x 10-7 | 5.10 x 10-6** | 1.7 x 10-7 | 0.00 |
| Pearl Harbor | 2.60 x 10-4 | 2.60 x 10-4 | 2.60 x 10-4 | 2.60 x 10-4 | 0.00 |
| Portsmouth | 9.00 x 10-7 | 9.00 x 10-7 | 9.00 x 10-7 | 9.00 x 10-7 | 0.00 |
| Norfolk | 1.6 x 10-5 | 1.6 x 10-5 | 1.6 x 10-5 | 1.6 x 10-5 | 0.00 |
| Kesselring | 7.50 x 10-7 | 7.50 x 10-7 | 7.50 x 10-7 | 7.50 x 10-7 | 0.00 |
| Max | 2.60 x 10-4 | 2.60 x 10-4 | 2.60 x 10-4 | 2.60 x 10-4 | 1.70 x 10-7 |
| WATER POOL STORAGE AT ALL SITES* | | | | | |
| INEL | 0.00 | 0.00 | 0.00 | 1.70 x 10-7 | 1.70 x 10-7 |
| Hanford | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| Savannah River | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| Nevada Test Site | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| Oak Ridge | 0.00 | 0.00 | 0.00 | 0.00 | 0.00 |
| Puget Sound | 1.7 x 10-7 | 5.1 x 10-6 | 5.1 x 10-6 | 5.1 x 10-6 | 0.00 |
| Pearl Harbor | 2.60 x 10-4 | 1.10 x 10-5 | 1.10 x 10-5 | 1.10 x 10-5 | 0.00 |
| Portsmouth | 9.00 x 10-7 | 3.40 x 10-6 | 3.40 x 10-6 | 3.40 x 10-6 | 0.00 |
| Norfolk | 1.6 x 10-5 | 6.0 x 10-6 | 6.0 x 10-6 | 6.0 x 10-6 | 0.00 |
| Kesselring | 7.50 x 10-7 | 2.50 x 10-6 | 2.50 x 10-6 | 2.50 x 10-6 | 0.00 |
| Max | 2.60 x 10-4 | 1.10 x 10-5 | 1.10 x 10-5 | 1.10 x 10-5 | 1.70 x 10-7 |

*Under No Action alternative, dry storage at Naval Nuclear Propulsion Program site

**Includes dry storage and water pool examination under this alternative

Table F-4. Risk of fatal cancers by alternative (probability of fatalities per year accident to general population within a 50-mile radius of site).

| | No Action | Decentrali- za- tion-No Exami- nation | Decentrali- z- ation- Puget Sound Exam | Decentrali- z- ation- INEL Exam | Planning Basis/ Regiona- liza- tion Centraliz a- tion- INEL | Re za io Ce za io Ha |
|--|----------------|---|--|--|---|--|
| Normal Opera- tions Risk Dry Storage At Navy Sites, Water Pool Storage At DOE Sites | 2.24 x 10-5 | 2.24 x 10-5 | 8.74 x 10-5 | 2.33 x 10-5 | 8.50 x 10-7 | 4. |
| Normal Opera- tions Risk Wa- ter | 2.24 x 10-5 | 3.39 x 10-4 | 3.39 x 10-4 | 3.40 x 10-4 | 8.50 x 10-7 | 4. |

| | | | | | | |
|--|-------------------------|-------------------------|-------------------------|-------------------------|-------------------------|----|
| Pool Storage At All Sites | | | | | | |
| Most Severe Risk From A Facility Accident Dry Storage At Naval Nuclear Propulsion Program Sites, Water Pool Storage At DOE Sites | 2.60 x 10 ⁻⁴ | 2.60 x 10 ⁻⁴ | 2.60 x 10 ⁻⁴ | 2.60 x 10 ⁻⁴ | 1.70 x 10 ⁻⁷ | 4. |
| | (1) | (1) | (1) | (1) | (2) | (|
| Most Severe Risk From A Facility Accident Water Pool Storage At All Sites | 2.60 x 10 ⁻⁴ | 1.10 x 10 ⁻⁵ | 1.10 x 10 ⁻⁵ | 1.10 x 10 ⁻⁵ | 1.70 x 10 ⁻⁷ | 4. |
| | (1) | (2) | (2) | (2) | (2) | (|

(1) Accident initiator - Airplane crash

(2) Accident initiator - Drained water pool

Table F-5. Impacts from naval spent nuclear fuel facility radiological accidents f
No Action alternative.

| Accident Description | Probability (per year) | Con-sequences to Public (fatalities per accident) | Risk to Public (fatalities) | Dose to Worker (rem) | Dose to MOI (rem) |
|----------------------|------------------------|---|-----------------------------|----------------------|-------------------|
|----------------------|------------------------|---|-----------------------------|----------------------|-------------------|

DRY STORAGE ACCIDENTS

Mechanical Damage

| | | | | | |
|--------------|------------------------|------------------------|------------------------|------------------------|------------------------|
| Puget Sound | 1.0 x 10 ⁻⁵ | 1.7 x 10 ⁻² | 1.7 x 10 ⁻⁷ | 5.6 x 10 ⁻² | 3.9 x 10 ⁻² |
| Pearl Harbor | 1.0 x 10 ⁻⁵ | 3.0 x 10 ⁻² | 3.0 x 10 ⁻⁷ | 5.6 x 10 ⁻² | 2.1 x 10 ⁻² |
| Norfolk | 1.0 x 10 ⁻⁵ | 1.8 x 10 ⁻² | 1.8 x 10 ⁻⁷ | 5.6 x 10 ⁻² | 8.1 x 10 ⁻² |
| Portsmouth | 1.0 x 10 ⁻⁵ | 1.0 x 10 ⁻² | 1.0 x 10 ⁻⁷ | 5.6 x 10 ⁻² | 4.2 x 10 ⁻² |
| Kesselring | 1.0 x 10 ⁻⁵ | 7.4 x 10 ⁻³ | 7.4 x 10 ⁻⁸ | 5.6 x 10 ⁻² | 8.1 x 10 ⁻³ |

Airplane Crash

| | | | | | |
|--------------|------------------------|-----|------------------------|----|-----|
| Pearl Harbor | 1.0 x 10 ⁻⁵ | 26 | 2.6 x 10 ⁻⁴ | 92 | 19 |
| Norfolk | 1.0 x 10 ⁻⁶ | 16 | 1.6 x 10 ⁻⁵ | 92 | 72 |
| Portsmouth | 1.0 x 10 ⁻⁷ | 9.0 | 9.0 x 10 ⁻⁷ | 92 | 38 |
| Kesselring | 1.0 x 10 ⁻⁷ | 7.5 | 7.5 x 10 ⁻⁷ | 92 | 7.7 |

Table F-6. Impacts from naval spent nuclear fuel facility radiological accidents f
Decentralization alternatives.

| Accident | Probability | Con-sequences to Public | Risk to Public | Dose to W- | Dose to MOI |
|----------|-------------|-------------------------|----------------|------------|-------------|
|----------|-------------|-------------------------|----------------|------------|-------------|

| Description | (per year) | (fatalities per accident) | (fatalities) | ork- er (rem) | (rem) |
|--|------------|------------------------------|--------------|---------------------|-------|
| WET STORAGE AND EXAMINATION ACCIDENTS | | | | | |

*Information applicable only for full examinations at INEL.

Drained Water Pool

| | | | | | |
|--------------|------------------------|------------------------|------------------------|-----|------------------------|
| *INEL | 1.0 x 10 ⁻⁵ | 1.7 x 10 ⁻² | 1.7 x 10 ⁻⁷ | 2.1 | 1.7 x 10 ⁻² |
| Puget Sound | 1.0 x 10 ⁻⁵ | 5.1 x 10 ⁻¹ | 5.1 x 10 ⁻⁶ | 2.1 | 1.4 |
| Pearl Harbor | 1.0 x 10 ⁻⁵ | 1.1 | 1.1 x 10 ⁻⁵ | 2.1 | 7.9 x 10 ⁻¹ |
| Norfolk | 1.0 x 10 ⁻⁵ | 6.0 x 10 ⁻¹ | 6.0 x 10 ⁻⁶ | 2.1 | 3.0 |
| Portsmouth | 1.0 x 10 ⁻⁵ | 3.4 x 10 ⁻¹ | 3.4 x 10 ⁻⁶ | 2.1 | 1.6 |
| Kesselring | 1.0 x 10 ⁻⁵ | 2.5 x 10 ⁻¹ | 2.5 x 10 ⁻⁶ | 2.1 | 2.9 x 10 ⁻¹ |

Accidental Criticality

| | | | | | |
|--------------|------------------------|------------------------|------------------------|-----|------------------------|
| *INEL | 1.0 x 10 ⁻⁵ | 6.4 x 10 ⁻³ | 6.4 x 10 ⁻⁸ | 8.0 | 9.2 x 10 ⁻³ |
| Puget Sound | 1.0 x 10 ⁻⁵ | 2.8 x 10 ⁻¹ | 2.8 x 10 ⁻⁶ | 8.0 | 1.3 |
| Pearl Harbor | 1.0 x 10 ⁻⁵ | 6.0 x 10 ⁻¹ | 6.0 x 10 ⁻⁶ | 8.0 | 6.7 x 10 ⁻¹ |
| Norfolk | 1.0 x 10 ⁻⁵ | 3.5 x 10 ⁻¹ | 3.5 x 10 ⁻⁶ | 8.0 | 2.7 |
| Portsmouth | 1.0 x 10 ⁻⁵ | 1.5 x 10 ⁻¹ | 1.5 x 10 ⁻⁶ | 8.0 | 1.4 |
| Kesselring | 1.0 x 10 ⁻⁵ | 1.1 x 10 ⁻¹ | 1.1 x 10 ⁻⁶ | 8.0 | 2.3 x 10 ⁻¹ |

Mechanical Damage

| | | | | | |
|--------------|------------------------|------------------------|-------------------------|------------------------|------------------------|
| *INEL | 1.0 x 10 ⁻⁵ | 5.3 x 10 ⁻⁶ | 5.3 x 10 ⁻¹¹ | 5.2 x 10 ⁻⁴ | 2.6 x 10 ⁻⁶ |
| Puget Sound | 1.0 x 10 ⁻⁵ | 7.2 x 10 ⁻⁵ | 7.2 x 10 ⁻¹⁰ | 5.2 x 10 ⁻⁴ | 1.7 x 10 ⁻⁴ |
| Pearl Harbor | 1.0 x 10 ⁻⁵ | 1.5 x 10 ⁻⁴ | 1.5 x 10 ⁻⁹ | 5.2 x 10 ⁻⁴ | 9.3 x 10 ⁻⁵ |
| Norfolk | 1.0 x 10 ⁻⁵ | 8.0 x 10 ⁻⁵ | 8.0 x 10 ⁻¹⁰ | 5.2 x 10 ⁻⁴ | 3.5 x 10 ⁻⁴ |
| Portsmouth | 1.0 x 10 ⁻⁵ | 5.6 x 10 ⁻⁵ | 5.6 x 10 ⁻¹⁰ | 5.2 x 10 ⁻⁴ | 1.9 x 10 ⁻⁴ |
| Kesselring | 1.0 x 10 ⁻⁵ | 6.0 x 10 ⁻⁵ | 6.0 x 10 ⁻¹⁰ | 5.2 x 10 ⁻⁴ | 3.6 x 10 ⁻⁵ |

Airplane Crash

| | | | | | |
|--------------|------------------------|------------------------|------------------------|------------------------|------------------------|
| Pearl Harbor | 2.0 x 10 ⁻⁵ | 4.6 x 10 ⁻² | 9.2 x 10 ⁻⁷ | 1.6 x 10 ⁻¹ | 2.8 x 10 ⁻² |
| Norfolk | 4.0 x 10 ⁻⁷ | 2.4 x 10 ⁻² | 9.6 x 10 ⁻⁹ | 1.6 x 10 ⁻¹ | 1.1 x 10 ⁻¹ |
| Kesselring | 2.0 x 10 ⁻⁷ | 1.8 x 10 ⁻² | 3.6 x 10 ⁻⁹ | 1.6 x 10 ⁻¹ | 1.1 x 10 ⁻² |

HEPA Filter Fire

| | | | | | |
|--------------|------------------------|------------------------|------------------------|------------------------|------------------------|
| *INEL | 5.0 x 10 ⁻⁴ | 5.3 x 10 ⁻⁵ | 2.7 x 10 ⁻⁸ | 2.4 x 10 ⁻³ | 2.5 x 10 ⁻⁵ |
| Puget Sound | 5.0 x 10 ⁻⁴ | 6.4 x 10 ⁻⁴ | 3.2 x 10 ⁻⁷ | 2.4 x 10 ⁻³ | 1.6 x 10 ⁻³ |
| Pearl Harbor | 5.0 x 10 ⁻⁴ | 1.2 x 10 ⁻³ | 6.0 x 10 ⁻⁷ | 2.4 x 10 ⁻³ | 8.7 x 10 ⁻⁴ |
| Norfolk | 5.0 x 10 ⁻⁴ | 6.9 x 10 ⁻⁴ | 3.5 x 10 ⁻⁷ | 2.4 x 10 ⁻³ | 3.3 x 10 ⁻³ |

WET STORAGE AND EXAMINATION ACCIDENTS

*Information applicable only for full examinations at INEL.

| | | | | | |
|------------|------------------------|------------------------|------------------------|------------------------|------------------------|
| Portsmouth | 5.0 x 10 ⁻⁴ | 3.9 x 10 ⁻⁴ | 2.0 x 10 ⁻⁷ | 2.4 x 10 ⁻³ | 1.7 x 10 ⁻³ |
| Kesselring | 5.0 x 10 ⁻⁴ | 3.3 x 10 ⁻⁴ | 1.7 x 10 ⁻⁷ | 2.4 x 10 ⁻³ | 3.5 x 10 ⁻⁴ |

Minor Water Pool Leak

| | | | | | |
|--------------|------------------------|-------------------------|-------------------------|-----|-------------------------|
| *INEL | 1.0 x 10 ⁻¹ | 1.3 x 10 ⁻⁸ | 1.3 x 10 ⁻⁹ | N/A | 2.5 x 10 ⁻⁹ |
| Puget Sound | 1.0 x 10 ⁻¹ | 4.2 x 10 ⁻⁹ | 4.2 x 10 ⁻¹⁰ | N/A | 3.2 x 10 ⁻¹⁰ |
| Pearl Harbor | 1.0 x 10 ⁻¹ | 4.6 x 10 ⁻¹⁰ | 4.6 x 10 ⁻¹¹ | N/A | 1.3 x 10 ⁻¹⁰ |
| Norfolk | 1.0 x 10 ⁻¹ | 1.8 x 10 ⁻⁹ | 1.8 x 10 ⁻¹⁰ | N/A | 2.7 x 10 ⁻¹⁰ |
| Portsmouth | 1.0 x 10 ⁻¹ | 1.4 x 10 ⁻⁹ | 1.4 x 10 ⁻¹⁰ | N/A | 1.3 x 10 ⁻¹⁰ |
| Kesselring | 1.0 x 10 ⁻¹ | 8.5 x 10 ⁻⁹ | 8.5 x 10 ⁻¹⁰ | N/A | 6.0 x 10 ⁻⁹ |

DRY STORAGE ACCIDENTS

Mechanical Damage

| | | | | | |
|--------------|------------------------|------------------------|------------------------|------------------------|------------------------|
| Puget Sound | 1.0 x 10 ⁻⁵ | 1.7 x 10 ⁻² | 1.7 x 10 ⁻⁷ | 5.6 x 10 ⁻² | 3.9 x 10 ⁻² |
| Pearl Harbor | 1.0 x 10 ⁻⁵ | 3.0 x 10 ⁻² | 3.0 x 10 ⁻⁷ | 5.6 x 10 ⁻² | 2.1 x 10 ⁻² |
| Norfolk | 1.0 x 10 ⁻⁵ | 1.8 x 10 ⁻² | 1.8 x 10 ⁻⁷ | 5.6 x 10 ⁻² | 8.1 x 10 ⁻² |
| Portsmouth | 1.0 x 10 ⁻⁵ | 1.0 x 10 ⁻² | 1.0 x 10 ⁻⁷ | 5.6 x 10 ⁻² | 4.2 x 10 ⁻² |
| Kesselring | 1.0 x 10 ⁻⁵ | 7.4 x 10 ⁻³ | 7.4 x 10 ⁻⁸ | 5.6 x 10 ⁻² | 8.1 x 10 ⁻³ |

Airplane Crash

| | | | | | |
|--------------|------------------------|-----|------------------------|----|-----|
| Pearl Harbor | 1.0 x 10 ⁻⁵ | 26 | 2.6 x 10 ⁻⁴ | 92 | 19 |
| Norfolk | 1.0 x 10 ⁻⁶ | 16 | 1.6 x 10 ⁻⁵ | 92 | 72 |
| Portsmouth | 1.0 x 10 ⁻⁷ | 9.0 | 9.0 x 10 ⁻⁷ | 92 | 38 |
| Kesselring | 1.0 x 10 ⁻⁷ | 7.5 | 7.5 x 10 ⁻⁷ | 92 | 7.7 |

DRY CELL ACCIDENTS

Mechanical Damage

| | | | | | |
|-------|------------------------|------------------------|------------------------|------------------------|------------------------|
| *INEL | 1.0 x 10 ⁻⁴ | 3.5 x 10 ⁻⁴ | 3.5 x 10 ⁻⁸ | 1.0 x 10 ⁻¹ | 2.2 x 10 ⁻⁴ |
|-------|------------------------|------------------------|------------------------|------------------------|------------------------|

Loss of Shielding

| | | | | | |
|-------|------------------------|-------------------------|-------------------------|------------------------|-------------------------|
| *INEL | 1.0 x 10 ⁻⁵ | 3.0 x 10 ⁻¹⁹ | 3.0 x 10 ⁻²⁴ | 7.2 x 10 ⁻⁵ | 9.3 x 10 ⁻¹⁷ |
|-------|------------------------|-------------------------|-------------------------|------------------------|-------------------------|

Table F-7. Impacts from naval spent nuclear fuel facility radiological accidents f
Planning Basis, Centralization at INEL, and Regionalization at INEL alternatives.

| Accident Description | Probability (per year) | Con- sequences to Public (fatalities per accident) | Risk to Public (fatalities) | Dose to W- ork- er (rem) | Dose to MOI (rem) |
|-------------------------|---------------------------|--|--------------------------------|-----------------------------------|----------------------|
|-------------------------|---------------------------|--|--------------------------------|-----------------------------------|----------------------|

WET STORAGE AND EXAMINATION
ACCIDENTS

Drained Water Pool

| | | | | | |
|------|------------------------|------------------------|------------------------|-----|------------------------|
| INEL | 1.0 x 10 ⁻⁵ | 1.7 x 10 ⁻² | 1.7 x 10 ⁻⁷ | 2.1 | 1.7 x 10 ⁻² |
|------|------------------------|------------------------|------------------------|-----|------------------------|

Accidental Criticality

| | | | | | |
|------|------------------------|------------------------|------------------------|-----|------------------------|
| INEL | 1.0 x 10 ⁻⁵ | 6.4 x 10 ⁻³ | 6.4 x 10 ⁻⁸ | 8.0 | 9.2 x 10 ⁻³ |
|------|------------------------|------------------------|------------------------|-----|------------------------|

Mechanical Damage

| | | | | | |
|------|------------------------|------------------------|-------------------------|------------------------|------------------------|
| INEL | 1.0 x 10 ⁻⁵ | 5.3 x 10 ⁻⁶ | 5.3 x 10 ⁻¹¹ | 5.2 x 10 ⁻⁴ | 2.6 x 10 ⁻⁶ |
|------|------------------------|------------------------|-------------------------|------------------------|------------------------|

HEPA Filter Fire

| | | | | | |
|------|------------------------|------------------------|------------------------|------------------------|------------------------|
| INEL | 5.0 x 10 ⁻⁴ | 5.3 x 10 ⁻⁵ | 2.7 x 10 ⁻⁸ | 2.4 x 10 ⁻³ | 2.5 x 10 ⁻⁵ |
|------|------------------------|------------------------|------------------------|------------------------|------------------------|

Minor Water Pool Leak

| | | | | | |
|------|------------------------|------------------------|------------------------|-----|------------------------|
| INEL | 1.0 x 10 ⁻¹ | 1.3 x 10 ⁻⁸ | 1.3 x 10 ⁻⁹ | N/A | 2.5 x 10 ⁻⁹ |
|------|------------------------|------------------------|------------------------|-----|------------------------|

DRY STORAGE ACCIDENTS

Mechanical Damage

| | | | | | |
|------|------------------------|------------------------|------------------------|------------------------|------------------------|
| INEL | 1.0 x 10 ⁻⁵ | 4.9 x 10 ⁻⁴ | 4.9 x 10 ⁻⁹ | 5.6 x 10 ⁻² | 4.6 x 10 ⁻⁴ |
|------|------------------------|------------------------|------------------------|------------------------|------------------------|

DRY CELL ACCIDENTS

Mechanical Damage

| | | | | | |
|------|------------------------|------------------------|------------------------|------------------------|------------------------|
| INEL | 1.0 x 10 ⁻⁴ | 3.5 x 10 ⁻⁴ | 3.5 x 10 ⁻⁸ | 1.0 x 10 ⁻¹ | 2.2 x 10 ⁻⁴ |
|------|------------------------|------------------------|------------------------|------------------------|------------------------|

Loss of Shielding

| | | | | | |
|------|------------------------|-------------------------|-------------------------|------------------------|-------------------------|
| INEL | 1.0 x 10 ⁻⁵ | 3.0 x 10 ⁻¹⁹ | 3.0 x 10 ⁻²⁴ | 7.2 x 10 ⁻⁵ | 9.3 x 10 ⁻¹⁷ |
|------|------------------------|-------------------------|-------------------------|------------------------|-------------------------|

Table F-8. Impacts from naval spent nuclear fuel facility radiological accidents f
Regionalization or Centralization at other DOE sites alternatives.
Information applicable only to DOE site selected for Regionalization or Centralizat

| Accident Description | Probability (per year) | Con- sequences to Public (fatalities per accident) | Risk to Public (fatalities) | Dose to W- ork- er (rem) | Dose to MOI (rem) |
|-------------------------|---------------------------|--|--------------------------------|-----------------------------------|----------------------|
|-------------------------|---------------------------|--|--------------------------------|-----------------------------------|----------------------|

WET STORAGE AND EXAMINATION
ACCIDENTS

Drained Water Pool

| | | | | | |
|---------------|------------------------|------------------------|------------------------|-----|------------------------|
| Savannah Rive | 1.0 x 10 ⁻⁵ | 1.1 x 10 ⁻¹ | 1.1 x 10 ⁻⁶ | 2.1 | 1.6 x 10 ⁻² |
| Hanford | 1.0 x 10 ⁻⁵ | 4.7 x 10 ⁻² | 4.7 x 10 ⁻⁷ | 2.1 | 6.3 x 10 ⁻³ |
| Nevada Test S | 1.0 x 10 ⁻⁵ | 1.9 x 10 ⁻³ | 1.9 x 10 ⁻⁸ | 2.1 | 3.3 x 10 ⁻² |
| Oak Ridge | 1.0 x 10 ⁻⁵ | 1.8 x 10 ⁻¹ | 1.8 x 10 ⁻⁶ | 2.1 | 5.2 |

Accidental Criticality

| | | | | | |
|---------------|------------------------|------------------------|------------------------|-----|------------------------|
| Savannah Rive | 1.0 x 10 ⁻⁵ | 4.5 x 10 ⁻² | 4.5 x 10 ⁻⁷ | 8.0 | 9.4 x 10 ⁻³ |
| Hanford | 1.0 x 10 ⁻⁵ | 1.6 x 10 ⁻² | 1.6 x 10 ⁻⁷ | 8.0 | 2.8 x 10 ⁻³ |
| Nevada Test S | 1.0 x 10 ⁻⁵ | 7.0 x 10 ⁻⁴ | 7.0 x 10 ⁻⁹ | 8.0 | 2.0 x 10 ⁻² |
| Oak Ridge | 1.0 x 10 ⁻⁵ | 8.8 x 10 ⁻² | 8.8 x 10 ⁻⁷ | 8.0 | 4.7 |

Mechanical Damage

| | | | | | |
|---------------|------------------------|------------------------|-------------------------|------------------------|------------------------|
| Savannah Rive | 1.0 x 10 ⁻⁵ | 2.0 x 10 ⁻⁵ | 2.0 x 10 ⁻¹⁰ | 5.2 x 10 ⁻⁴ | 2.2 x 10 ⁻⁶ |
| Hanford | 1.0 x 10 ⁻⁵ | 8.6 x 10 ⁻⁶ | 8.6 x 10 ⁻¹¹ | 5.2 x 10 ⁻⁴ | 9.8 x 10 ⁻⁷ |
| Nevada Test S | 1.0 x 10 ⁻⁵ | 5.6 x 10 ⁻⁷ | 5.6 x 10 ⁻¹² | 5.2 x 10 ⁻⁴ | 4.6 x 10 ⁻⁶ |
| Oak Ridge | 1.0 x 10 ⁻⁵ | 3.4 x 10 ⁻⁵ | 3.4 x 10 ⁻¹⁰ | 5.2 x 10 ⁻⁴ | 5.9 x 10 ⁻⁴ |

Airplane Crash

| | | | | | |
|---------------|------------------------|------------------------|-------------------------|------------------------|------------------------|
| Savannah Rive | 2.0 x 10 ⁻⁶ | 6.1 x 10 ⁻³ | 1.2 x 10 ⁻⁸ | 1.6 x 10 ⁻¹ | 6.4 x 10 ⁻⁴ |
| Oak Ridge | 1.0 x 10 ⁻⁶ | 1.0 x 10 ⁻² | 1.0 x 10 ⁻⁸ | 1.6 x 10 ⁻¹ | 1.8 x 10 ⁻¹ |
| Nevada Test S | 4.0 x 10 ⁻⁷ | 1.7 x 10 ⁻⁴ | 6.8 x 10 ⁻¹¹ | 1.6 x 10 ⁻¹ | 1.3 x 10 ⁻³ |

HEPA Filter Fire

| | | | | | |
|---------------|------------------------|------------------------|------------------------|------------------------|------------------------|
| Savannah Rive | 5.0 x 10 ⁻⁴ | 1.3 x 10 ⁻⁴ | 6.5 x 10 ⁻⁸ | 2.4 x 10 ⁻³ | 2.1 x 10 ⁻⁵ |
| Hanford | 5.0 x 10 ⁻⁴ | 5.3 x 10 ⁻⁵ | 2.7 x 10 ⁻⁸ | 2.4 x 10 ⁻³ | 7.0 x 10 ⁻⁶ |
| Nevada Test S | 5.0 x 10 ⁻⁴ | 5.7 x 10 ⁻⁶ | 2.9 x 10 ⁻⁹ | 2.4 x 10 ⁻³ | 4.3 x 10 ⁻⁵ |
| Oak Ridge | 5.0 x 10 ⁻⁴ | 2.2 x 10 ⁻⁴ | 1.1 x 10 ⁻⁷ | 2.4 x 10 ⁻³ | 5.7 x 10 ⁻³ |

Minor Water Leak

| | | | | | |
|---------------|------------------------|-------------------------|-------------------------|-----|-------------------------|
| Savannah Rive | 1.0 x 10 ⁻¹ | 1.3 x 10 ⁻⁹ | 1.3 x 10 ⁻¹⁰ | N/A | 7.9 x 10 ⁻¹⁰ |
| Hanford | 1.0 x 10 ⁻¹ | 1.7 x 10 ⁻¹⁰ | 1.7 x 10 ⁻¹¹ | N/A | 9.9 x 10 ⁻¹² |
| Nevada Test S | 1.0 x 10 ⁻¹ | 1.4 x 10 ⁻⁹ | 1.4 x 10 ⁻¹⁰ | N/A | 2.5 x 10 ⁻⁹ |
| Oak Ridge | 1.0 x 10 ⁻¹ | 3.9 x 10 ⁻⁹ | 3.9 x 10 ⁻¹⁰ | N/A | 1.5 x 10 ⁻⁹ |

DRY STORAGE ACCIDENTS

Mechanical Damage

| | | | | | |
|------------------|----------------------|----------------------|-----------------------|----------------------|----------------------|
| Savannah River | 1.0×10^{-5} | 3.0×10^{-3} | 3.0×10^{-8} | 5.6×10^{-2} | 4.9×10^{-4} |
| Hanford | 1.0×10^{-5} | 1.3×10^{-3} | 1.3×10^{-8} | 5.6×10^{-2} | 1.7×10^{-4} |
| Nevada Test Site | 1.0×10^{-5} | 5.3×10^{-5} | 5.3×10^{-10} | 5.6×10^{-2} | 8.8×10^{-4} |
| Oak Ridge | 1.0×10^{-5} | 5.1×10^{-3} | 5.1×10^{-8} | 5.6×10^{-2} | 1.4×10^{-1} |

Airplane Crash

| | | | | | |
|----------------|----------------------|-----|----------------------|----|----------------------|
| Savannah River | 3.0×10^{-7} | 2.8 | 8.4×10^{-7} | 92 | 4.7×10^{-1} |
| Oak Ridge | 3.0×10^{-7} | 4.7 | 1.4×10^{-6} | 92 | 120 |

DRY CELL ACCIDENTS**Mechanical Damage**

| | | | | | |
|------------------|----------------------|----------------------|----------------------|----------------------|----------------------|
| Savannah River | 1.0×10^{-4} | 1.4×10^{-3} | 1.4×10^{-7} | 1.0×10^{-1} | 2.4×10^{-4} |
| Hanford | 1.0×10^{-4} | 5.3×10^{-4} | 5.3×10^{-8} | 1.0×10^{-1} | 7.1×10^{-5} |
| Nevada Test Site | 1.0×10^{-4} | 3.7×10^{-5} | 3.7×10^{-9} | 1.0×10^{-1} | 4.0×10^{-4} |
| Oak Ridge | 1.0×10^{-4} | 2.5×10^{-3} | 2.5×10^{-7} | 1.0×10^{-1} | 5.8×10^{-2} |

Loss of Shielding

| | | | | | |
|------------------|----------------------|-----------------------|-----------------------|----------------------|-----------------------|
| Savannah River | 1.0×10^{-5} | 3.0×10^{-16} | 3.0×10^{-21} | 7.2×10^{-5} | 6.7×10^{-15} |
| Hanford | 1.0×10^{-5} | 4.9×10^{-24} | 4.9×10^{-29} | 7.2×10^{-5} | 3.3×10^{-23} |
| Nevada Test Site | 1.0×10^{-5} | 3.7×10^{-37} | 3.7×10^{-42} | 7.2×10^{-5} | 6.3×10^{-11} |
| Oak Ridge | 1.0×10^{-5} | 7.5×10^{-6} | 7.5×10^{-11} | 7.2×10^{-5} | 1.2×10^{-2} |

Airplane Crash

| | | | | | |
|------------------|----------------------|----------------------|----------------------|-----|----------------------|
| Savannah River | 2.0×10^{-6} | 4.8 | 9.6×10^{-6} | 160 | 8.2×10^{-1} |
| Oak Ridge | 1.0×10^{-6} | 8.4 | 8.4×10^{-6} | 160 | 350 |
| Nevada Test Site | 4.0×10^{-7} | 1.8×10^{-1} | 7.2×10^{-8} | 160 | 1.6 |

concentrations were then compared against Emergency Release Planning Guide (ERPG) 1 of evaluating their effects. ERPG values are specific for each substance and provide airborne concentration thresholds above which one can reasonably observe adverse effects. An ERPG-1 level could result in a very mild effect whereas exposure to an ERPG-3 level could result in a life-threatening health effect. For the postulated accident involving a chemical spill, personnel (worker) could be exposed to concentrations of hydrochloric acid, phosgene, sodium hydroxide above ERPG-3 levels which indicates a potential for long-term health effects. However, no member of the general public located off-site would be expected to be exposed above ERPG-3 except for Oak Ridge where sulfuric acid and sodium hydroxide concentrations could exceed ERPG-3. For the postulated accident involving a diesel fuel fire, on-site personnel could be exposed to concentrations of sulfur dioxide and oxides of nitrogen above ERPG-3 levels. However, the general public located off-site would be expected to be exposed to levels above ERPG-3 at the shipyard location (Norfolk) where nitric oxide concentrations could exceed ERPG-3 under meteorological conditions. However, for both postulated accidents, the accident an evacuation of on-site or off-site personnel and it is expected that chemical exposures would be limited to ERPG-3 levels because actions such as evacuation would be used to reduce the effect on workers.

Fugitive Dust Analysis

The FDM computer code was used to estimate the fugitive dust concentrations from the construction of a water pool facility at the alternate locations. It was estimated that fugitive dust would not result in any adverse effects for any of the alternate locations.

Other Impacts

The radiological impact of accidents on the environs of a facility was determined the area that could be contaminated following such an event. Calculations using various conditions were performed for each accident scenario. These calculations determine contamination which causes only a small increase in background radiation from natural sources. For most facilities and most accidents, the contaminated area was confined to the site. For a few cases, the casualty scenarios did result in contaminated land; however, the total land contaminated for those scenarios (inside and outside the boundary) was less than 207 acres. The impact of this contamination would be temporary while the area remediation efforts are completed.

F.1 RADIOLOGICAL ISSUES FROM NAVAL SPENT NUCLEAR FUEL

INSPECTIONS AND STORAGE

Naval spent nuclear fuel is currently examined and stored at the Naval Reactor Expanded Core Facility (ECF) at the DOE Idaho National Engineering Laboratory (INEL). INEL-ECF is a large laboratory facility used to receive, examine, and ship naval spent irradiated test specimen assemblies. Enclosed work areas at INEL-ECF include an air reinforced concrete water pools which permit visual observation of naval spent nuclear fuel handling and inspection while shielding workers from radiation. Adjacent to the water pools are cells used for operations which must be performed dry. One of the water pools contains that will link the water pools with a proposed Dry Cell Project, which would provide preparation of spent fuel in a dry, enclosed environment.

The proposed Dry Cell Facility will consist of a shielded, radiologically controlled structural steel and concrete with remotely operated equipment necessary to examine

The Organization for Economic Co-operation and Development (OECD) of the Nuclear Energy Agency (NEA) reported that extensive safety analysis has shown that pool storage of a very safe option which can last for decades (NEA 1993). The external hazards, such as aircraft crashes, are potential threats for these facilities (loss of coolant) but additional shielding can cope with these hazards. Dry storage has not yet generally been used on a very large scale but it is anticipated that long-term storage in adequate canisters against earthquakes and aircraft crashes.

Several technologies are being used currently for the storage of spent nuclear fuel at sites away from reactors. Both wet (pool) storage facilities and dry storage facilities (containers) are used on a commercial scale.

The safety of spent nuclear fuel storage has been extensively evaluated. The U.S. Nuclear Regulatory Commission (NRC) reported in the "Waste Confidence Decision" of 1984 that there is assurance that spent nuclear fuel can be stored safely and without significant environmental consequences in spent nuclear fuel storage installations (NUREG 1984). For both dry storage and pool storage, the NRC stated its belief that current storage technologies are capable of providing safe storage beyond the active lifetime of the reactor facility. The NRC also concluded that the risk of an accident or sabotage at a spent nuclear fuel storage facility with radiological consequences is extremely remote.

Considerable experience has been gained in the transport of spent nuclear fuel elements in consequent safety-related development of suitable transportation casks. This experience has made it possible to develop a concept for dry storage of spent nuclear fuel elements within transportation containers generally have not been the transportation casks themselves.

The concept of a cask which could be used for both transportation and storage in the United States in the framework of a policy of dry storage in Independent Spent Fuel Installations (ISFIs) (CFR 1993). According to this policy, the reactor operators are entitled to store spent nuclear fuel elements, which have cooled in a pool for at least one year after discharge from the reactor, in licensed containers under dry conditions for 20 years or more. A number of storage facilities have received official approval for that purpose.

F.1.1 Normal Operations

Current practice for examination of naval spent nuclear fuel at ECF includes visual examination, measurement of key dimensions, and loading into a shipping cask. Temporary storage of spent nuclear fuel at INEL

since fuel is, at times, received into the facility faster than it can be examined facility. In addition, a small amount of spent fuel is selected for retention as a reference and examination. Routine releases to the atmosphere were evaluated at all measured releases from INEL-ECF. Each location was evaluated using releases equivalent to INEL-ECF. Each location's specific population and meteorology were then used to predict consequences.

F.1.1.1 Water Pool Storage. Wet storage is a highly developed technique and it is the standard

method used worldwide for storage of spent fuel. While in wet storage pools, temperature radiation fluxes are lower than in the reactor, so there is no intrinsic driving force for a major fraction of the radioactive materials contained in the stored spent fuel.

The Zircaloy cladding of naval spent nuclear fuel is an efficient barrier against release during handling and storage of spent fuel. Given adequate control of water corrosion in water during the long-term storage conditions of fuel assemblies. At all times, the fuel is covered with a tightly adhering oxide layer formed at high temperature factor that inhibits further corrosion during storage.

Direct exposure to radiation of persons working in storage facilities can occur in activities as handling of fuel casks and fuel assemblies, handling of contaminated maintenance work. Experience shows that, in common with other fuel cycle facilities, increased occupational exposure arises when any maintenance or unusual operations are performed. Increased exposures can, however, generally be minimized by good planning, adequate shielding of critical components, paying particular attention to the design of those items that are contaminated from the point of view of repair and maintenance, and by the use of local equipment decontamination procedures. Systems and components that are important to include:

- pool water cooling and makeup systems;
- filter equipment for purification of pool water;
- ventilation systems;
- equipment for temperature, water level, and leakage measurement in the pool;
- hoists and handling systems for fuel assemblies; and
- equipment for handling and storage of other wastes.

Shielding from radiation is normally assured by providing a minimum depth of fuel elements in storage to reduce the exposure rates. Fuel transfer mechanisms have mechanical stops to prevent the inadvertent raising of fuel to the water surface. A structure is needed in order to guarantee adequate containment of the pool water, but water resulting in a substantial reduction of the shielding layer is unlikely to in exposures to personnel above operational limits since adequate countermeasures can be taken.

Storage of naval spent nuclear fuel in water pools is an alternative being evaluated at Navy shipyard locations discussed above. Source terms for all locations were based on those reported by INEL-ECF in the past. Exposures due to downwind dispersion, water level radiation were calculated.

F.1.1.2 Dry Storage. Many thousands of spent fuel assemblies of different types have been stored for

periods of time ranging from a couple of years to over 30 years in more than 20 different facilities. In general, the spent fuel behavior during storage has been excellent. No problems of dry storage on the integrity of the spent fuel have been detected (NEA 1993).

The dry storage of spent fuel is being used to a limited extent in several countries. In the United States, fuel was stored in dry wells at the INEL. Dry wells were used for the storage of spent fuel at the Nevada Test Site as part of a large dry storage demonstration program. Climax deep dry wells (600 meters below the surface in granite) in 1979. In 1983, they underwent extensive non-destructive and destructive characterization. No problems or changes were identified (NEA 1993).

Designs of metal casks for use in spent fuel storage have been in existence since the 1950s. The casks are generally equipped with a double-lid system to ensure safe containment. The casks have been subjected to a variety of tests and demonstrations since the early 1960s and consolidated fuel.

The DOE sponsored the demonstration of the storage of fuel in metal casks at the INEL facility in 1984 and 1985. The DOE entered into a cooperative agreement with Virginia Electric Power's utility, to demonstrate the use of three types of metal casks. The Virginia Electric Power Station has been licensed by the NRC for storage of spent fuel in metal casks.

Results of demonstration activities have shown the following (NEA 1993):

- radiation and thermal levels resulting from metal cask storage have been
- no fuel failure has occurred during demonstration storage;
- no secondary wastes have arisen from the storage operation.

Storage of naval spent nuclear fuel in storage or shipping containers is an a evaluated at all locations. Since no airborne releases are expected from routine d the biological effects of direct radiation exposure to the on-site personnel and th

F.1.1.3 Dry Cell Operations. The handling of naval spent nuclear fuel for research and development

purposes in dry cells like the proposed Dry Cell Project was evaluated at selected health effects due to routine airborne releases and direct radiation exposure were

F.1.2 Screening/Selection of Accidents for Detailed Examination

Accidents were considered for inclusion in detailed analyses if they were exp substantially to risk (defined as the product of the probability of occurrence of t consequence of the accident). Accidents were categorized into three types as eithe Design Basis Accidents, or Beyond Design Basis Accidents. These categories are cha probability of occurrence as described further in Section F.1.3.7. Construction an included in these categories.

In selecting accidents to include in detailed analyses, several consideration Initiating events were reviewed including natural phenomena (earthquakes, volcanic hurricanes and other natural events) and human initiated events (human error, equip explosions, plane crashes, transportation accidents, and terrorism). Guiding princ such as: the radioactive materials involved must be available in a dispersible for mechanism available for release of such materials from the facility; and, there mus available for off-site dispersion of the released materials. The pathways whereby can be affected from the nuclear aspects of spent fuel operations are direct exposu inhalation of radioactive materials, or ingestion of radioactive materials. Recogn processes and pathways, accidents involving the following basic phenomena were iden

- loss of shielding of radioactive materials,
- release of radioactive products to the environment due to overheating of
- release of radioactive products to the environment due to mechanical sho inadvertent breaching of fuel cladding or containment,
- an unplanned criticality,
- transportation accidents.

After the basic phenomena were identified, other references were consulted to important accidents were considered. These included safety analysis reports, court environmental impact statements, and summary documents such as the "Final Generic E Impact Statement on Handling and Storage of Spent Light Water Reactor Power Reactor 1979a) and "The Safety of the Nuclear Fuel Cycle" (NEA 1993).

Examining the kinds of accidents which could result in release of radioactive environment or an increase in radiation levels shows that they can only occur if a severe conditions. Some types of accidents, such as procedure violations, spills o containing radioactive particles, or most other types of common human error, may oc than the more severe accidents analyzed. However, they do not involve enough radio radiation to result in a significant release to the environment or a meaningful inc Stated another way, the very low consequences associated with these events produce those for the accidents analyzed, even when combined with a higher probability of o Consequently, they have not been included in the results presented in this Environm Statement.

Acts of terrorism are expected to result in consequences which are bounded by accidents which were evaluated. Naval spent nuclear fuel is not considered to be a due to the bulk of the fuel and containers and due to the high radiation fields inv spent nuclear fuel. However, terrorist attacks on naval spent nuclear fuel during The massive structure of the shipping containers used for naval spent nuclear fuel target of a terrorist attack. No such attacks have occurred in the nearly 40 years have now travelled about 2 million kilometers. Thus, the probability of a terroris judged to be no more than the probability of a rail accident which is listed in Sec Attachment A to Appendix D of this Environmental Impact Statement. The consequence

attack are also judged to be no more severe than those listed for transportation accidents. The same conclusions reached for transportation accidents apply to the risk to the external containers from terrorist attack during a shipment. In addition, during shipment, fuel containers are accompanied by escorts who remain in contact with headquarters. In an emergency, state and federal resources would be quickly summoned to stabilize the situation.

For an act of war, sabotage, or terrorist attack, it is likely the risk would be less for the airplane crash because it should be less probable that a force would exist to disperse radioactive products into the atmosphere from a weapon as compared to the motive force of the fire in the case of an airplane crash. For example, attacks on containers using anti-tank weapons would be less severe than the accidents analyzed because: (a) anti-tank weapons would cause a dent in the metal of a container, unlike that which is assumed from the airplane crash (a 10-foot diameter engine rotor); (b) there is no explosive material inside the container, so a tank would not be hit by such a weapon (in a tank attack, the tank shells inside the tank would be no fire to disperse the radioactivity that is released when the container is hit); (c) in an aircraft crash where the jet fuel will burn creating such a fire. The rugged design of the containers, thick walls of water pools, combined with the shock-absorbing nature of water with the effects of other types of explosive charges. It is not credible that a terrorist attack would result in a criticality or meltdown of spent nuclear fuel; however, in Section F.1.4.2.1.2, the hypothetical criticality accident is presented. The risks associated with an accident are less than those associated with a drained water pool or an airplane crash into dry storage.

The effect of a terrorist attack or an act of sabotage is expected to be consistent with the limiting accident discussed at each facility under each alternative. For example, an accident involving naval spent nuclear fuel is described in this attachment to be a shipping container at the Pearl Harbor Naval Shipyard. This accident would lead to a release of radioactive material over the next 50 years in the population within 50 miles of the shipyard. Since there is one chance in 100,000 per year, the risk would be 0.00026 latent fatal cancer fatalities per year. In other words, about one chance in 4,000 of a single latent fatal cancer fatality over 50 years shared among the approximately 820,000 people residing within 50 miles of the shipyard. The shipyard is expected to have over 2,000 cancer fatalities from all causes every year. For an act of sabotage or terrorist attack, it is likely the risk would be lower than calculated because it is less probable that a force would exist to disperse radioactive products into the atmosphere from a weapon as compared to the motive force of the fire assumed in the case of an airplane crash.

Accidents initiated at nearby facilities, by other activities unrelated to spent nuclear fuel storage, or during construction of an ECF or dry cell type of facility, would not be more severe than the sequences of events described. This is because naval spent nuclear fuel examination or in storage under the conditions of the alternatives evaluated would be under continuous conditions or uninterrupted operator attention to prevent overheating, failure of containment, or shielding. Therefore, evacuation in response to an accident at some other facility would not be a safety issue. This inherent safety, combined with the distance between naval spent nuclear fuel facilities and any other activities which might suffer a catastrophic accident, means that the accidents at these facilities would not produce conditions at a naval spent nuclear fuel facility which would be more severe than for any hypothetical synergistic combination of events resulting from accidents at these facilities. Therefore, such analyses have not been included in this evaluation.

The existence of common cause accidents at a facility has been considered. If a spent nuclear fuel facility is located at a particular Navy site. However, it is not possible, like an earthquake, to produce more than one accident at some sites causing a simultaneous release of radioactive material into the atmosphere or an increase in radiation level. However, the probability of two or more accidents having maximum consequences concurrently is less than the probability of the individual events. For example, if the Naval Reactors Facility at INEL, a crane might fail causing damage to stored spent nuclear fuel, might drain, and shielding for the Dry Cell might be damaged. The impacts for this accident would be estimated by summing the consequences. A combined total of 2.8×10^{-2} fatal cancer fatalities per year. Similarly, consequences from spent nuclear fuel facilities within a DOE site could be conservatively estimated site wide impacts. But again, the probability of a common cause accident is lower than the probability of the individual accidents. The number of consequences of impact will vary between facilities due to separation distances.

Several accident scenarios were developed for the handling and storage of naval spent nuclear fuel. All potential accidents were not evaluated, but cases which are considered to be other reasonable accidents were analyzed. Each of these accident scenarios was evaluated at the same locations using identical source terms. Like the evaluations for normal operations, meteorology data specific to each site were used to estimate site specific health effects.

F.1.2.1 Water Pool Storage. Six hypothetical accident scenarios were evaluated for naval spent nuclear fuel

nuclear fuel stored in water pools. These hypothetical sequences of events include pool caused by an earthquake, an accidental criticality, mechanical damage due to o failure, an airplane crash into the water pool facility, a fire in a high efficienc filter, and minor water pool leakage. Radiation exposure to on-site individuals, a boundary, and the general population was estimated for airborne releases of radioac and direct radiation exposure.

F.1.2.2 Dry Storage. Two hypothetical accident scenarios were evaluated for naval spent nuclear fuel

stored in shipping containers. The first scenario postulates that a wind-driven mi casks, with mechanical damage causing a release of corrosion products into the envi hypothetical scenario is based on an airplane crash into the dry storage area. Onc exposure to on-site individuals, an individual at the site boundary, and the genera estimated for airborne releases, water releases, and direct radiation exposure.

F.1.2.3 Dry Cell Operations. Three hypothetical accidents were evaluated for naval spent nuclear fuel

handled in dry cells at several locations. These scenarios include cutting into th mechanical damage during examination work, partial loss of concrete shielding due t an airplane crash into the dry cell facility. Once again, radiation exposure to on individual at the site boundary, and the general population was estimated for airbo releases, and direct radiation exposure.

F.1.2.4 Shipboard Fire Involving Shipping Containers. Attachment A describes the historical

practice of shipping naval spent nuclear fuel from Pearl Harbor Naval Shipyard to P Shipyard by ship where the containers are then transported to ECF by rail. Since 1 shipments containing a total of 20 shipping containers. Even though there have not involving these shipments, hypothetical accidents were evaluated near the Pearl Har shipyards. The scenario involves a collision of the spent nuclear fuel ship with a in a fire. The radiation exposure to nearby individuals and the general population airborne and water releases.

F.1.3 Analysis Methods for Evaluation of Radiation Exposure

F.1.3.1 General. An evaluation of normal operations and hypothetical accidents at the existing and

proposed sites was performed to assess the possible radiation exposure to individua radioactive materials. The analyses are based on the same operations carried out a locations and the same accidents at any of the sites evaluated. With this approach compare the incremental effect of the proposed alternative actions or the different postulated accidents at the different sites. These locations include four naval sh Norfolk, Puget Sound, and Pearl Harbor), five Department of Energy facilities (INEL Hanford, Nevada Test Site, and Oak Ridge), and the Kesselring Site.

F.1.3.2 Exposures to be Calculated. Radiation exposure to the following different individuals and

the general population is calculated for normal operation of the spent fuel facilit conditions:

- Worker (Worker). An individual located 100 meters (330 feet) from the r material release point. (The impact of accidents on close-in workers is numerically but is discussed qualitatively for each accident in Section attachment.)
- Maximally exposed collocated worker (MCW). At DOE locations, a theoreti individual located at whichever is the greater of 0.4 mile from the faci or 75% of the distance to the nearest independent facility area. The MC evaluated if the site boundary is closer than the MCW location. Thus, at

locations and the Kesselring Site, the MCW is not specifically evaluated

- Maximally exposed off-site individual (MOI). A theoretical individual 1 site or shipyard boundary receiving the maximum exposure. At the Savannah two separate MOI locations were evaluated depending upon whether the site is constructed on the Savannah River Site or is located at the existing Fuel Plant (hereafter referred to as the Barnwell Plant) which is adjacent to the Savannah River Site. At Hanford, two separate MOI locations were also evaluated whether a new facility is constructed in the 200 Area or modifications to the Fuels and Materials Examination Facility (FMEF) which is located in the 200 Area.
- Nearest public access individual (NPA). At larger DOE sites, highways upon which the public may cross the federal reservation which includes the facility where nuclear fuel operations could be conducted. Consequently, these analyses evaluate the exposure to a theoretical motorist who might be stranded on a highway at the time of an accident. Based on experience from emergency response teams would be able to evacuate such an individual within 30 minutes so this was the exposure time used in the calculations. At naval shipyards where public access highways exist, but military personnel, civilian employees and their families, including some who reside on the base, may be located outside the industrial area boundary but inside the confines of the military base. They might be at their homes, in buildings, or on the roadways of the base at the time of an accident or at any time throughout the year for the evaluation of normal operations. In the event of a severe accident they would be evacuated under military control of the base, so this time was used in accident calculations. A value was calculated for the Kesselring Site and the Nevada Test Site because of public roads which cross these sites, there are no residents, and there are no accesses.
- Maximally exposed individual at nearby communities is evaluated for accidents.
- General population within a 50-mile radius of the facility.

Exposure is calculated to result from direct radiation from the facility and contamination released to the air. Normal releases directly to the water pathway or to bodies of water, and contamination of the water at all times from fallout of airborne contamination. The releases to the air might result in exposure pathways described as follows:

- External direct exposure from immersion in the airborne radioactive material (immersion)
- External direct exposure from radioactive material deposited on the ground surface
- Internal exposure from inhalation of radioactive aerosols and suspended particulates (inhalation)
- Internal exposure from ingestion of terrestrial food and animal products
- Exposure from contaminated water (water release).

The radiation exposure is calculated by the computer programs discussed in Section 3.1 in the manner recommended by the International Commission on Radiological Protection (ICRP 1979). Weighting factors are used for various body organs to calculate a "committed equivalent" (CEDE) from radiation inside the body due to inhalation or ingestion. Equivalent dose equivalents (CDEs) are calculated for organs such as the lungs, stomach, small intestine, lower large intestine, bone surface red bone marrow, testes, ovaries, muscles, kidneys, liver etc. The CEDE value is the summation of the CDEs to the specific organ relative risk to that organ compared to an equivalent whole-body exposure.

The programs also calculate an effective dose equivalent (EDE) for the external exposure pathways (immersion in the radioactive material, exposure to ground contamination) and for the internal exposure pathways. The sum of the EDE from external pathways and internal pathways is called the "total effective dose equivalent" (TEDE) in this Environment.

(EIS) and is also calculated by the programs. The TEDE reported in the results section TEDE's from air, water, and direct radiation exposures.

The exposure from ingestion of terrestrial food and animal products is calculated. However, it is expected that continued consumption of contaminated food products by humans will be suspended after a Protective Action Guideline is reached. In 1991, the Environment recommended protective action guidelines in the range of 1 to 5 rem whole-body exposure on a consistent analysis basis, no reduction of exposure due to a Protective Action Guideline in the analysis. This would result in a conservative approach which may slightly overestimate effects within an exposed population, but allows for consistent comparisons between communities.

Table F.1.3.2-1 identifies selected nearby communities for each site for which exposures for a maximally exposed individual were calculated. In all cases, the MOI is less than maximum exposure at any nearby community. Calculations were performed for the evaluation of exposures for areas representative of the range of communities within 50 m analyzed. The selection of these communities was not intended to indicate that other communities are important. Other communities of interest in the vicinity of the sites in addition to a number of communities in Maine and New Hampshire near the Portsmouth Naval Shipyard, Portsmouth, Durham, Eliot, Greenland, Kittery, New Castle, North Hampton, Ogunquit, Berwick.

Table F.1.3.2-1. Nearby communities for each site.

| | |
|------------------|--|
| INEL | Howe, Atomic City, Arco, Blackfoot, Idaho Falls |
| Savannah River | Snelling, Barnwell, Jackson, Aiken, Allendale, Augusta |
| Hanford | Othello, Richland, Prosser, Pasco, Yakima, Umatilla |
| Nevada Test Site | Beatty, Pahrump, Las Vegas |
| Oak Ridge | Oak Ridge, Harriman, Rockwood, Knoxville, Jefferson City |
| Puget Sound | Seattle, Tacoma, Olympia, Port Angeles |
| Pearl Harbor | Pearl City, Aiea, Pacific Palisades, Ewa Beach, Honolulu |
| Norfolk | Newport News, Hampton, Suffolk, Virginia Beach, Williamsburg |
| Portsmouth | Dover, Exeter, Hampton Beach, Sanford, Nashua, Lowell, Cambridge, Boston |
| Kesselring | Ballston Spa, Saratoga Springs, Amsterdam, Schenectady |

Table F.1.3.2-2 presents an example of the detailed exposure calculation results performed. The table shows the possible exposure pathways and individuals analyzed.

F.1.3.3 Evaluation of Health Effects. Health effects are calculated from the exposure results. The

risk factors used for calculations of health effects are taken from Publication 60 of the International Commission on Radiological Protection (ICRP 1991). Table F.1.3.3-1 lists the approach for the analysis of both the normal operations and the hypothetical accident scenarios.

Cancer fatalities were used to summarize and compare the results in this Environmental Statement since this effect was viewed to be of the greatest interest to most people. From Table F.1.3.3-1, the number of total health effects (deaths, non-fatal cancers, genetic effects on human health) may be easily obtained by multiplying the latent cancer fatalities which is the ratio of 7.3/5.0.

The numerical estimates of cancer deaths and other health detriments presented in the practice of linear extrapolation from the nominal risk estimate for lifetime to radiation. Other methods of extrapolation to the low-dose region could yield higher or lower estimates of cancer deaths. Studies of human populations exposed at low doses are presented in Table F.1.3.2-2. Summary of exposure calculation results.

| Location | Inhalation | Air | | Ground | | Total Airborne |
|----------|------------------------|------------------------|------------------------|------------------------|------------------|------------------------|
| | | Immersion | Surface | Immersion | Surface | Release |
| | CEDE (rem) | EDE (rem) | EDE (rem) | EDE (rem) | EDE (rem) | EDE (rem) |
| Worker | 5.4 x 10 ⁻¹ | 6.5 x 10 ⁻⁴ | 7.9 x 10 ⁻¹ | N/A | N/A | 1.3 |
| MCW | 4.8 x 10 ⁻⁴ | 8.6 x 10 ⁻⁷ | 3.4 x 10 ⁻⁴ | N/A | N/A | 8.2 x 10 ⁻⁴ |
| NPA | 1.4 x 10 ⁻⁴ | 3.2 x 10 ⁻⁷ | 5.2 x 10 ⁻⁵ | N/A | N/A | 1.9 x 10 ⁻⁴ |
| MOI | 6.1 x 10 ⁻⁴ | 1.2 x 10 ⁻⁶ | 7.8 x 10 ⁻⁴ | 3.1 x 10 ⁻⁴ | 10 ⁻⁴ | 1.7 x 10 ⁻³ |

Exposure to Maximally Exposed Individual at Nearby Communities (rem)

| | | | | | |
|-------------|---------------|---------------|---------------|---------------|---------------|
| Arco | 5.2 x 10-5 | 1.3 x 10-7 | 6.4 x 10-5 | 3.1 x 10-5 | 1.5 x 10-4 |
| (30600m) | | | | | |
| Howe | 9.8 x 10-5 | 1.8 x 10-7 | 1.2 x 10-4 | 5.6 x 10-5 | 2.7 x 10-4 |
| (16100m) | | | | | |
| Idaho Falls | 3.1 x 10-6 | 5.2 x 10-9 | 3.6 x 10-6 | 2.0 x 10-6 | 8.7 x 10-6 |
| (72400m) | | | | | |
| Blackfoot | 4.8 x 10-6 | 3.3 x 10-9 | 5.2 x 10-6 | 3.4 x 10-6 | 1.3 x 10-5 |
| (68100m) | | | | | |
| Atomic City | 2.9 x 10-5 | 1.0 x 10-7 | 3.6 x 10-5 | 1.6 x 10-5 | 8.1 x 10-5 |
| (24200m) | | | | | |

Exposure to Population within 50-mile Radius (person-rem)

| | | | | | |
|----------------------|---------------|---------------|---------------|---------------|---------------|
| Population of 115690 | 1.1 x 10-1 | 6.1 x 10-5 | 1.5 x 10-1 | 4.5 x 10-2 | 3.0 x 10-1 |
|----------------------|---------------|---------------|---------------|---------------|---------------|

Table F.1.3.3-1. Risk estimators for health effects from ionizing radiation.
Risk Factor (probability per rem)*

| Effect | Nuclide | Worker | General Population |
|-----------------------------|---------|------------|--------------------|
| Fatal cancer (all organs) | All | 4.0 x 10-4 | 5.0 x 10-4 |
| Weighted non-fatal cancer** | All | 8.0 x 10-5 | 1.0 x 10-4 |
| Weighted genetic effects** | All | 8.0 x 10-5 | 1.3 x 10-4 |
| Weighted total effects** | All | 5.6 x 10-4 | 7.3 x 10-4 |

* For high individual exposures (y20 rem), the above risk factors are multiplied by a factor of two.

General population exposures were not modified because the large drop in exposure with increasing distances results in average exposure rates well below 20 rem.

** In determining a means of assessing health effects from radiation exposure, the ICRP has developed a weighting method for non-fatal cancers and genetic effects to obtain a total weighted effect, or "health detriment".

inadequate to demonstrate the actual level of risk. There is scientific uncertainty about cancer risk in the low-dose region below the range of epidemiologic observation, and the possibility of no risk cannot be excluded (CIRRPC 1992). In this appendix, the doses have been provided in all cases to allow independent evaluation using any relation between exposure and health effects.

F.1.3.4 Population. Population distributions specific to each site were used for the evaluations. The

population distributions were obtained from 1990 United States Census data. The po was obtained in 16 compass directions and 5 equal radial distances from the likely spent nuclear fuel site to a 50-mile total distance.

F.1.3.5 Meteorology. For the navy shipyards, Savannah River, and Kesselring Sites, the

meteorological data used in the analyses were obtained from the SCRAM bulletin board INEL, Hanford, Nevada Test Site, and Oak Ridge, site tower meteorological data were SCRAM bulletin board is operated by the Support Center for Regulatory Air Models with Environmental Protection Agency, Office of Air Quality Planning and Standards. The meteorological data files are comprised of data acquired from the National Climatic SCRAM data for 4 or 5 years were used with programs from the bulletin board to develop data in the STability ARray (STAR) format which is a joint frequency distribution of intervals, 16 wind directions, and 6 stability categories. The STAR data were reformat required by the GENII program, described below, for evaluation of normal operations

The STAR data were also used to calculate the 50% and 95% meteorological conditions for accident analyses. The 50% condition represents the average meteorological conditions defined as that for which more severe conditions with respect to accident consequences 50% of the time. The 95% condition represents the meteorological conditions which highest calculated exposures. This is defined as that condition which is not exceeded time or is the worst combination of weather stability class and wind speed. Each one evaluated for 16 wind directions.

For each location, the nearest available SCRAM data was used to represent the site being evaluated. Table F.1.3.5-1 shows the pertinent data for the meteorological Table F.1.3.5-1. Meteorological data applicability.

| Site | Data From | Data Years |
|------------------|---------------------|------------|
| Portsmouth | Portland ME Airport | 1985-1989 |
| Norfolk | Norfolk VA Airport | 1985-1989 |
| Puget Sound | SEATAC Airport | 1985-1989 |
| Pearl Harbor | Honolulu Airport | 1985-1989 |
| INEL | NRF Tower | 1987-1991 |
| Kesselring | Albany NY Airport | 1985-1989 |
| Savannah River | Augusta GA Airport | 1984-1987 |
| Hanford | 200 Area Tower | 1983-1990 |
| Nevada Test Site | Desert Rock Tower | 1990 |
| Oak Ridge | Y-12 West Tower | 1990 |

F.1.3.6 Computer Programs. Five computer programs were used to evaluate the radiation exposures

to the specified individuals and general population.

F.1.3.6.1 GENII. The code used for the environmental and transport and exposure assessment

calculations for normal operations was GENII (Napier et al. 1988). This code was developed Northwest Laboratory by Battelle Memorial Institute to incorporate the internal dose recommended by the International Commission on Radiological Protection in Publication 30 (ICRP 1979) into environmental pathway analysis models in use at Laboratory.

Although GENII can be used to model both acute and chronic releases to the atmosphere, the chronic option was used in the normal operations evaluation reflecting long-term average released radioactive contaminants. For the chronic evaluations, the code also uses conditions averaged over each sector to reflect exposure to long-term average concentration ingestion calculation used the modeling approach that exposed individuals within 50 consumed 30% of milk products and 10% of all products grown locally where the people

F.1.3.6.2 RSAC-5. The computer code RSAC-5 was developed by Westinghouse Idaho

Nuclear Co, Inc., for the DOE-ID Operations Office and is in the public domain (Westinghouse calculates the consequences of the release of radionuclides to the atmosphere. It calculates each fission product nuclide from a nuclear event to be input individually or to be the code. RSAC-5 calculates potential radiation exposures to maximally exposed individual population groups via inhalation, ingestion, exposure to radionuclides deposited on immersion-

exposure in airborne radioactive material, and radiation from a cloud of radioactive material meteorological capabilities include Gaussian plume dispersion for Pasquill-Gifford release scenario modeling allows reduction of nuclides by chemical group or element and buildup during transport through operations, facilities, and the environment.

of filters or other cleanup systems. Population exposures are the product of the exposure and the number of people in the affected population.

F.1.3.6.3 ORIGEN. ORIGEN (Croff 1980) is a computer code system for calculating the

buildup and decay of radioactive materials (fission products, actinides, and active input was modeled to describe the naval nuclear fuel system and incorporates cross-distinct to naval fuels.

F.1.3.6.4 SPAN. SPAN (Wallace 1972) is the computer code which was used to calculate the

direct radiation levels. Attenuation from air was included in the calculated radiation the unit person exposure per sector, SPAN was used to integrate the radiation level radiation levels calculated at various distances were used as the source to represent falloff in the sector, and a total radiation level for each sector was calculated. radiation level for each sector was then divided by the sector volume, resulting in exposure for any point within the sector.

F.1.3.6.5 WATER RELEASE. WATER RELEASE is an unpublished computer code used to

calculate exposures to humans arising from radionuclides which have been introduced vicinity of the proposed spent nuclear fuel storage and examination facilities. This provides a brief description of the key points associated with obtaining these estimates which were considered to be introduced into the water at a site were postulated to uniformly in the water in the immediate vicinity of the site during the time period were introduced. There are two processes by which radionuclides might enter the water liquid discharge or via airborne discharge. For liquid discharges, a fraction of the might enter the water accessed by humans each year by infiltrating the ground to then traveling either to wells or surface water. For airborne discharges, some fraction radionuclides might enter the water by deposition from the air. For both of these radionuclides that might enter the water used by humans has been postulated to enter immediately, except for NRF and the Nevada Test Site. For NRF and the Nevada Test Site postulated that 20 years pass before the radionuclides might enter the water accessed by humans is based upon the fact that water must percolate into the ground and reach groundwater contamination must travel with the water in the aquifer to a point where it can be a well at Atomic City. An assessment of the infiltration rate of radionuclides between about 200 years are needed for them to pass into the aquifer (Smith 1994). Also, the flow is at a rate of 5 to 20 feet per day. Therefore, 20 years was used as the time for humans at INEL. Similarly, at the Nevada Test Site surface water is not present so aquifers which are more than 600 feet deep. Hence, 20 years was also used at this site.

Once the radionuclides have been introduced into the water at a site, they were transported to locations where they might affect man either directly as via immersion indirectly as via ingestion of food. During this transport period, these radionuclides various mechanisms which may reduce their concentration in the water such as radionuclides in larger volumes of water, removal by sedimentation, etc. The pathways considered which radionuclides in the water at a site might reach man are immersion, exposure boating and equipment exposure, and consumption of drinking water, fish, crustacea, animals, vegetables and fruits, root crops, milk and eggs, and domesticated animals when the radionuclides have left the water environment and are being transported to man, they may be subjected to both concentration and removal mechanisms which will their effect upon man. These mechanisms include concentration in the surface deposition pathways; decay during periods between harvesting a crop and its ingestion by man; activity due to harvesting, handling, and cleaning of a foodstuff.

For each of the sites at which storage or examination of spent nuclear fuel is estimates were made for the exposures which the total population affected by releases receive and for the exposures which a maximally exposed individual may receive from releases. The exposures to the population affected at a given site were obtained by exposures received by an average individual in the vicinity of that site and multiplied the number of people that are affected. The exposure to a maximally exposed individual maximum exposures and consumption rates which any individual at that site may experience the probabilities associated with just one individual actually following all the major specific pathways which are applicable at a given site are dependent upon the site,

average or a maximum individual to each of the pathways is different for each of the exposures associated with the drinking water pathway are not considered for the short-term radionuclides basically end up in salt water prior to their becoming available to man. On the other hand, the radionuclides introduced at the DOE and prototype sites can enter the drinking water pathway after a delay period. An initial delay occurs while the radionuclides seep into the ground before entering the aquifer. The delay continues while the radionuclides travel through the pathway and ultimately yield exposures to man. The total exposure to the population at a given site is the resultant sum of the exposure commitments from all the pathways applicable at that site.

F.1.3.7 Categorization of Accidents.

F.1.3.7.1 Abnormal Events. Abnormal Events are unplanned or improper events which result

in little or no consequence. Abnormal events include industrial accidents and accidents in operations such as skin contamination with radioactive materials, spills of radioactive materials, or direct radiation due to improper placement of shielding. The occurrence of these events has been anticipated and mitigative procedures are in place which promptly detect and evaluate and limit the effects of these events on individuals. As a result, there is little concern about the population from these events. Such events are considered to occur in the probability range of 10⁻³ to 10⁻⁶ per year. The probability referred to here is the total probability of occurrence and the event occurs (e.g., plane crash) times other probabilities required for the consequences to be included in this range, results are presented for both the 50% meteorological condition (average meteorology) and the 95% meteorological condition.

F.1.3.7.2 Design Basis Accident Range. Accidents which have a probability of occurrence

in the range of 10⁻³ to 10⁻⁶ per year are included in the range called the Design Basis Accident. This terminology "design basis accident," which normally refers to facilities to be constructed, is used in the "evaluation" basis accident which applies to existing facilities. For accidents in this range, results are presented for both the 50% meteorological condition (average meteorology) and the 95% meteorological condition. Risk calculations for accidents in this range utilize the consequences associated with the 95% meteorological conditions.

F.1.3.7.3 Beyond Design Basis Accidents. This range includes accidents which are less

likely to occur than the design basis accidents but which may have very large or catastrophic consequences. Accidents included in this range typically have a total probability range of 10⁻⁶ to 10⁻⁷ per year. Accidents which are less likely than 10⁻⁷ per year are not included since it is expected they do not contribute in any substantial way to the risk. For accidents in this range, consequences are presented for 50% and 95% meteorological conditions. Risk calculations for accidents in this range utilize the consequences associated with the 95% meteorological conditions.

F.1.3.8 Evaluation of Impacted Area

The impacted area surrounding a facility following an accident was determined and evaluated. The impacted area was defined as that area in which the plume deposited to such a degree that an individual standing on the boundary of the fallout area would receive approximately 0.01 mrem/hr of exposure. If this individual spends 24 hours a day a person would receive about 88 mrem per year from the ground surface shine. This is the 100 mrem/year limit of 10CFR20.

To best characterize the affected areas for each casualty, a typical 50% meteorological condition (Pasquill-Gifford Class D, wind speed 10 mph) was applied to each accident scenario. Results for ground surface dose were interpolated to determine the distance downwind where the dose had dropped to approximately 88 mrem per year based on 24 hours per day exposure. For the class chosen, the plume remains within a single 22.5-degree sector. The area affected was determined as the entire sector contaminated to the calculated downwind distance. For each facility accident analyzed and the contaminated footprint associated with the accident, the following information was determined:

| | | |
|------------------|-----------------|---|
| Footprint Length | Footprint Area* | Sites with Footprint Beyond Facility |
|------------------|-----------------|---|

| | | | |
|-------------------------------|-------|---------|--|
| Accident Scenario (miles) | | (acres) | Boundary |
| Drained Water Pool | 0.29 | 11 | Norfolk, Oak Ridge, Portsmouth |
| Criticality | 0.25 | 8 | Norfolk, Oak Ridge, Portsmouth |
| Wet Storage Mechanical Damage | <0.06 | <0.5 | none |
| Wet Storage | <0.06 | <0.5 | none |
| Airplane Crash | | | |
| Dry Storage Mechanical Damage | <0.06 | <0.5 | none |
| Dry Storage | 0.91 | 106 | Pearl Harbor, Norfolk, Oak Ridge, Portsmouth |
| Airplane Crash | | | |
| Dry Cell Mechanical Damage | <0.06 | <0.5 | none |
| HEPA Filter Fire | <0.06 | <0.5 | none |
| Dry Cell | 1.27 | 207 | Oak Ridge |
| Airplane Crash | | | |

*Based on contamination of a single sector.

Although the plume would be contained within a single sector, the direction of the plume is unknown. Therefore, each site was examined for impacts in all directions around the distance equal to the footprint length. Since the accidents do occur over a short acreage of the sector quoted is still an accurate indication of the total contamination. The potential impacts for each site is contained in Tables F.1.3.8-2 through -11.

Table F.1.3.8-2. Secondary impacts of facility accidents at Puget Sound Naval Shipyard

| Site | Significant Accidents in Decreasing Severity | Biotic Resources | Water Resources | Economic Impacts | National Defense |
|----------------------------|---|--|--|---|---|
| Puget Sound Naval Shipyard | 1. Dry Storage Plane Crash | Plants and animals on the site and around the site will experience no long term impacts. | The water used for drinking and industrial purposes is monitored and use may be temporarily suspended during cleanup operations. Some recreational activities may also be temporarily suspended. No enduring impacts are expected. | A small number of individuals may experience temporary job loss due to temporary restrictions on farming, fishing and other support activities near the facility during cleanup operations. Some costs would also be incurred for the actual cleanup operation. | Naval at the site could contain and clean up these full |
| | 2. Drained Water Pool | | | | |
| | 3. Criticality and all other radiological accidents | | | | |

Table F.1.3.8-3. Secondary impacts of facility accidents at Pearl Harbor Naval Shipyard

| Site | Significant Accidents in Decreasing Severity | Biotic Resources | Water Resources | Economic Impacts | National Defense |
|------|--|------------------|-----------------|------------------|------------------|
|------|--|------------------|-----------------|------------------|------------------|

| | | | | | |
|-----------------------------|-------------------------------------|--|--|---|--|
| Pearl Harbor Naval Shipyard | 1. Dry Storage Plane Crash | Plants and animals on the site and around the site will experience no long term impacts. | The water used for drinking industrial purposes is monitored and use may be temporarily suspended during cleanup operations. Some recreational activities may also be temporarily suspended. No enduring impacts are expected. | A small number of individuals may experience temporary job loss due to temporary restrictions on farming, fishing and other support activities near the facility during cleanup operations. Some costs would also be incurred for the actual cleanup operation. | Naval personnel at the site could be contacted during accident. Cleanup would store these full |
| | 2. All other radiological accidents | | | | |

Table F.1.3.8-4. Secondary impacts of facility accidents at Norfolk Naval Shipyard

| Site | Significant Accidents in Decreasing Severity | Biotic Resources | Water Resources | Economic Impacts | National Defense |
|------|--|------------------|-----------------|------------------|------------------|
|------|--|------------------|-----------------|------------------|------------------|

| | | | | | |
|------------------------|---------------------------------------|--|---|---|---|
| Norfolk Naval Shipyard | 1. Dry Storage Plane Crash | Plants and animals on the site and around the site will experience no long term impacts. | The water used for drinking industrial purposes is monitored and use may be temporarily suspended during cleanup operations. Some | A small number of individuals may experience temporary job loss due to temporary restrictions on farming, fishing and other support activities near the facility during | Naval personnel at the site could be contacted during accident. Cleanup would |
| | 2. Drained Water Pool and Criticality | | | | |

| | | |
|-------------------------------------|--|------------------|
| 3. All other radiological accidents | recreational cleanup activities operations. may also be Some costs temporarily would also suspended. be incurred No enduring for the impacts are actual expected. cleanup operation. | store these full |
|-------------------------------------|--|------------------|

Table F.1.3.8-5. Secondary impacts of facility accidents at Portsmouth Naval Shipyard

| Site | Significant Accidents in Decreasing Severity | Biotic Resources | Water Resources | Economic Impacts | National Defense | Naval Shipyard | Environmental Contention |
|---------------------------|---|--|--|---|--|----------------|---|
| Portsmouth Naval Shipyard | 1. Dry Storage Plane Crash | | The water used for drinking and industrial purposes is monitored and use may be temporarily suspended during cleanup operations. Some recreational activities may also be temporarily suspended. No enduring impacts are expected. | A small number of individuals may experience temporary job loss due to temporary restrictions on farming, fishing and other support activities near the facility during cleanup operations. Some costs would also be incurred for the actual cleanup operation. | | | 1. A approximately 10 might cleanup Contention could be extended 0.6 beyond closest boundary. |
| | 2. Drained Water Pool | Plants and animals on the site and around the site will experience no long term impacts. | | | Naval vessels at the shipyard could be temporarily contaminated during the accident. Cleanup operations would restore these ships to full readiness. | | 2. Contention would the ship boundary would limited to appropriate mate 10 a |
| | 3. Criticality and all other radiological accidents | | | | | | 3. Contention with ship boundary Table F.1. the could cont |

nate

| Table F.1.3.8-6. Secondary impacts of facility accidents at Oak Ridge Reservation. | | | | | | |
|--|---|---|--|---|----------------|--|
| Site | Significant Accidents in Decreasing Severity | Biotic Resources | Water Resources | Economic Impacts | Natio Defen | |
| Oak Ridge Reservation | 1. Dry Cell Air Plane Crash | | | A small number of individuals may experience temporary job loss due to temporary restrictions on farming, fishing and other support activities near the facility during cleanup operations. Some costs would also be incurred for the actual cleanup operation. | | |
| | 2. Dry Storage Plane Crash | Plants and animals on the site and around the site will experience no long term impacts. | monitored and use may be temporari- ly suspended during cleanup operations. Some recreational activities may also be temporarily suspended. No enduring impacts are expected. | | No impa | |
| | 3. Drained Water Pool and Criticality | | | | | |
| | 4. All other radiological accidents | | | | | |

| Table F.1.3.8-7. Secondary impacts of facility accidents at Savannah River Site. | | | | | | |
|--|---|---|---|---|----------------|--|
| Site | Significant Accidents in Decreasing Severity | Biotic Resources | Water Resources | Economic Impacts | Natio Defen | |
| Savannah River Site | All Radiological Accidents | | | A small number of individuals may experience temporary job loss due to temporary restrictions on farming, fishing and other | | |
| | | Plants and animals on the site and around the | monitored and use may be temporari- ly suspended | | No impa | |

site will during support
experience cleanup activities
no long operations. near the
term Some facility
impacts. recreational during
activities cleanup
may also be operations.
temporarily Some costs
suspended. would also
No enduring be incurred
impacts are for the
expected. actual
cleanup
operation.

Table F.1.3.8-8. Secondary impacts of facility accidents at Nevada Test Site.

| Site | Significant Accidents in Decreasing Severity | Biotic Resources | Water Resources | Economic Impacts | Natio Defen |
|---------------------|---|---|---|--|----------------|
| Nevada Test Site | All Radiological Accidents | Plants and animals on the site and around the site will experience no long term impacts. | The water used for drinking and industrial purposes is monitored and use may be temporari- ly suspended during cleanup operations. No enduring impacts are expected. | A small number of individuals may experience temporary job loss due to temporary restrictions on support activities near the facility during cleanup operations. Some costs would also be incurred for the actual cleanup operation. | No impa |

Table F.1.3.8-9. Secondary impacts of facility accidents at Idaho National Enginee

| Site | Significant Accidents in Decreasing Severity | Biotic Resources | Water Resources | Economic Impacts | Natio Defen |
|--|---|--|--|---|----------------|
| Idaho National Engineering Laboratory | All Radiological Accidents | Plants and animals on the site and around the site will | The water used for drinking and industrial purposes is monitored and use may be temporari- ly suspended during | A small number of individuals may experience temporary job loss due to temporary restrictions on support activities near the facility | No impa |

experience cleanup during
no long operations. cleanup
term No enduring operations.
impacts. impacts are Some costs
expected. would also
be incurred
for the
actual
cleanup
operation.

Table F.1.3.8-10. Secondary impacts of facility accidents at Hanford Site.

| Site | Significant Accidents in Decreasing Severity | Biotic Resources | Water Resources | Economic Impacts | Natio Defen |
|-----------------|---|---|---|--|----------------|
| Hanford Site | All Radiological Accidents | Plants and animals on the site and around the site will experience no long term impacts. | The water used for drinking and industrial purposes is monitored and use may be temporari- ly suspended during cleanup operations. Some recre- ational activi- ties may also be temporari- ly suspended. No enduring impacts are expected. | A small number of individuals may experience temporary job loss due to temporary restrictions on support activities near the facility during cleanup operations. Some costs would also be incurred for the actual cleanup operation. | No impa |

Table F.1.3.8-11. Secondary impacts of facility accidents at Kenneth A. Kesselring Site.

| Site | Significant Accidents in Decreasing Severity | Biotic Resources | Water Resources | Economic Impacts | Natio Defen |
|----------------------------------|---|--|---|---|----------------|
| Kenneth A. Kesselring Site | 1. Dry Storage Plane Crash 2. Drained Water Pool and | Plants and animals on the site and around the site will experience | The water used for drinking and industrial purposes is monitored and use may be temporari- ly suspended during cleanup operations. | A small number of individuals may experience temporary job loss due to temporary restrictions on support activities near the facility during | No impa |

| | | | |
|--|-----------------------------|--|---|
| all other radiological accidents | no long term impacts. | Some recre- ational activi- ties may also be temporari- ly suspended. No enduring impacts are expected. | cleanup operations. Some costs would also be incurred for the actual cleanup operation. |
|--|-----------------------------|--|---|

F.1.3.9 Emergency Preparedness and Mitigative Measures.

F.1.3.9.1 Emergency Preparedness Emergency plans are in effect at shipyards and

prototype sites to ensure that workers and the public would be properly protected in an accident. In addition, emergency plans are in effect for accidents involving the use of radioactive materials. These response plans include the activation of emergency response by the site and a site emergency control center, as well as activation of a command center with Naval Reactors Headquarters and supporting laboratories. The long standing emergency program that exists within the Naval Nuclear Propulsion Program includes the ability to provide comprehensive and extensive emergency response resources of each naval site and provide coordination with appropriate civil authorities. In addition to the Naval Nuclear resources, extensive federal emergency response resources are available as needed to provide an emergency response.

Emergency response measures include provisions for immediate response to any accident at a shipyard or prototype site, identification of the accident conditions, and communication with local authorities providing radiological data and recommendations for any appropriate protective action. In the event of an accident involving radioactive or toxic materials, workers in the vicinity of the accident are promptly evacuated from the immediate area. This evacuation can typically be accomplished within a short time and would reduce the hazard to workers.

Regularly scheduled exercises are conducted periodically at each site in order to test the ability to respond to accidents. These exercises include realistic tests of people and communications involved in all aspects of the plans, and the plans are regularly revised to incorporate experience gained from the exercises. These exercises also periodically test the adequacy of interactions with local hospitals and emergency personnel and state and federal agencies.

F.1.3.9.2 Mitigative Factors. For members of the general public residing at the site boundary

or beyond, no credit is taken for any preventive or mitigative actions that would reduce the exposure of these individuals. These individuals are calculated as being exposed to the entire contaminated plume from the accident site. Similarly no action is taken to prevent these people from being exposed to day-to-day routine and ingestion of terrestrial food and animal products continue to be discussed in Section F.1.3, action would be taken to prevent the public from exceeding the guideline, if needed. No reduction of exposure due to these actions are accounted for. The public is assumed to spend approximately 30% of the day within their homes or offices. The exposure to ground surface radiation is therefore reduced appropriately on a yearly basis.

Individuals that reside or work on site, or those that may be traversing the site, would be evacuated from the affected area within 2 hours. This is based on the availability of personnel at all locations to oversee the removal of residents, collocated workers, and travel in an emergency manner. Periodic training and evaluation of the security personnel is conducted to ensure that actions are taken during an actual casualty. Therefore, residents, collocated workers, and others would be exposed to the entire contaminated plume as it travels downwind for a period not to exceed 2 hours. Similarly, the radiation dose from the deposited radioactive materials would be limited. No ingestion of contamination is calculated for these individuals.

Facility workers all undergo training to take quick, decisive action during an emergency. Individuals quickly evacuate the area and move to previously defined "relocation" areas. Workers could be exposed to a full 5 minutes of the radioactive plume as they move to the relocation centers. Once the immediate threat of the plume has moved off-site and downwind, the exposure ends.

instructed to walk to vehicles waiting to evacuate them from the site. An addition required to evacuate the workers from the contaminated area and therefore the workers 20 minutes of ground shine. No ingestion of contamination is calculated for these

The following summary provides the individual exposure times utilized in the presented in Section F.1.4.2.

Estimated Time an Individual Might be Exposed

| | Worker (100 m) | Collocated Worker (MCW) and Nearest Public Access (NPA) 100% of release time up to 120 min. | Individual at Nearest Site Boundary (MOI) 100% of release time |
|------------------------------|----------------|---|--|
| To Plume | 5 min. | 120 min. | |
| To Fallout on Ground Surface | 20 min. | 120 min. | 0.7 yr |
| To Food | N/A | N/A | 1 yr |

F.1.3.10 Perspective on Calculations of Cancer Fatalities and Risk

The topics of human health effects caused by radiation and the risks associated operations or postulated accidents associated with spent nuclear fuel management are times throughout this Environmental Impact Statement. It is important to understand how they are used in order to understand the information presented in this document have some frame of reference or comparison for understanding how the risks compare life.

The method used to calculate the risk of any impact is fundamental to all of presented and follows standard accepted practices. The first step is to determine specific event will occur. For example, the probability that a routine task, such as be performed sometime during a year of normal operations at a facility would be 1. action would certainly occur. The probability that an accident might occur is less because accidents occur only occasionally and some of the more severe accidents, such as earthquake, might occur at any location only once in hundreds, thousands, or millions

Once the probability of an event has been determined, the next step is to probabilities of the event being considered might be. One important measure of consequences EIS is the number of human fatalities from cancer induced by radiation. This was the document deals with radioactive materials. The number of cancer fatalities that might routine operation or any postulated accident can be calculated using a standard technique amount of radiation exposure that might occur from all conceivable pathways and the who might be affected (refer to Section F.1.3.3).

A couple of examples should serve to illustrate the calculation of risk. In the case of dying in a motor vehicle accident can be computed from the likelihood of an individual automobile accident and the consequences or number of fatalities per accident. The motor vehicle accidents during 1992 in the United States resulting in about 40,000 deaths. Thus, the probability of a person being in an automobile accident is 10,000,000 accidents approximately 250,000,000 persons in the United States, or 0.04 per year. The number of accidents, 0.004 (40,000 deaths divided by 10,000,000 accidents), is less than 1 since not cause fatalities. Multiplying the probability of the accident (0.04 per year) by the number of accidents (0.004 deaths per accident) by the number of years the person is exposed to (considered to be an average lifetime) gives the risk for any individual being killed in an accident. From this calculation, the overall risk of someone dying in a motor vehicle accident chance in 87 over their lifetime.

A second example illustrates the calculation of risk for another event which involves fuels, such as natural gas or coal, contain naturally occurring radioactive materials in the air during combustion. This radioactivity in the air finds its way into our bodies as we breathe. This radioactivity has been estimated to produce about 0.5 millirem per year the average American each year (NCRP 1987). The probability of this happening is equal to these fuels are burned every day all over the country. The number of fatal cancers per year is calculated by taking 0.5 millirem per year times the 72 years average lifetime times the 0.0005 fatal cancers estimated to be caused by each rem 72 years x 0.0005 fatal cancers per rem = 0.000018 fatal cancers per individual lifetime probability (1.0) times the consequences (0.000018 cancer fatalities) which equals 55,000 of death from this cause over a lifetime.

These risks and others from everyday life can be used to gain a perspective on the alternatives in this EIS. As illustrated, the risk of death from cancer from released daily from combustion of fossil fuels is about 1 chance in 55,000 for the

further comparison, the naturally occurring radioactive materials in agricultural f 1 to 2 millirem per year to an average American's exposure to radiation (NCRP 1987) similar to the one in the preceding paragraph shows that the use of fertilizer to p United States results in a risk of death from cancer between 1 chance in 12,500 and Finally, the average American's risk of dying from cancer from all causes is 1 chan lifetime. These risks can be compared, for example, to the average individual risk 1 billion for a resident in the vicinity of the INEL developing a fatal cancer due tions at the Expanded Core Facility (see the data in Section F.1.4.1).

A frame of reference for the risks from accidents associated with spent nucle alternatives can be developed in the same way. For an average resident in the vici individual risk of death from cancer caused by the water leaking from the Expanded large earthquake would be approximately 1 chance in 9 billion. This individual ris dividing the risk value to the population within 50 miles (1.7×10^{-7} fatalities pe Table F-3) by the total population of 115,690 and multiplying by an average life sp risk can be compared to the risks of death from other accidental causes to gain a p the risk of death in a motor vehicle accident was calculated earlier to be about 1 the risk of death for the average American from fires is approximately 1 chance in accidental poisoning the risk is about 1 chance in 1000 (Crouch 1982).

F.1.4 Analysis Results

F.1.4.1 Normal Operations. The purpose of this analysis is to determine the hypothetical health

effects on workers and the public due to routine handling of naval spent nuclear fu from facilities involved in routine handling of naval spent nuclear fuel are small comparable DOE and commercial nuclear facilities. Records of routine releases due were used as source terms for all locations to estimate what effects these types of and the public. Site-specific meteorological and population data were used at each analyzed. For normal operations at the Naval Reactors Facility (NRF and Oak Ridge) nearest public access (NPA) individual is not estimated due to the short period of individual would spend on-site while driving on the public access road. At Hanford the Washington Public Power Supply System Plant, and at Savannah River at the U.S. Office. The NPA at shipyard locations is defined in Section F.1.3.2.

F.1.4.1.1 Water Pool Examination and Storage Source Terms. The evaluation of

normal water pool operations was performed using two different source terms. In on term was utilized which included both the incremental release of radioactive materi alternative spent nuclear fuel storage actions and the release from other ongoing N Identical source terms were used for the evaluation of radiation exposure due to th materials during normal operations of wet storage and spent fuel examinations. The release from the INEL-ECF was used to evaluate these operations. Since the INEL-EC extremely low, this upper limit approach is not unduly conservative for the wet sto expected to have a lower release. Table F.1.4.1.1-1 shows the 1991 INEL-ECF releas release rate at Kesselring and NRF (including both INEL-ECF and prototypes), and th representing Naval Reactors operations at naval shipyards. The release rate repres based on upper bound data from Navy operations contained in Naval Nuclear Propulsio Report NT-94-1 (NNPP 1994). With no current Naval Reactors facilities at Savannah Oak Ridge, or the Nevada Test Site, the current release for each of these sites is Table F.1.4.1.1-1. Airborne releases from current Naval Reactors operations.

| Location | | Annual Releases (Ci/year) | |
|------------|-------|---------------------------|-----------------------------|
| INEL-ECF | H-3 | 9.35×10^{-2} | Y-90 5.5×10^{-6} |
| | C-14 | 7.0×10^{-1} | I-131 4.82×10^{-6} |
| | Sr-90 | 5.5×10^{-6} | Kr-85 3.0×10^{-1} |
| NRF | H-3 | 9.35×10^{-2} | Sr-90 2.45×10^{-5} |
| | C-14 | 8.0×10^{-1} | Y-90 2.45×10^{-5} |
| | Ar-41 | 2.7×10^{-1} | I-131 6.3×10^{-6} |
| | Co-60 | 1.6×10^{-6} | Cs-137 6.3×10^{-6} |
| | Kr-85 | 3.0×10^{-1} | |
| Kesselring | H-3 | 1.0×10^{-1} | Kr-85 1.0×10^{-3} |
| | C-14 | 4.0×10^{-1} | I-131 5.0×10^{-4} |
| | Ar-41 | 1.4 | Cs-137 5.0×10^{-4} |

| | | | | |
|-------------------------------|--------|------------------------|--------|------------------------|
| | Co-60 | 1.0 x 10 ⁻³ | | |
| Savannah River, Hanford, none | | | | |
| Nevada Test Site, Oak Ridge | | | | |
| Portsmouth, Norfolk | H-3 | 1.0 x 10 ⁻³ | Kr-87 | 5.0 x 10 ⁻² |
| Puget Sound, | C-14 | 1.0 x 10 ⁻¹ | Kr-88 | 2.0 x 10 ⁻² |
| Pearl Harbor | Ar-41 | 4.1 x 10 ⁻¹ | Xe131m | 5.0 x 10 ⁻³ |
| | Co-60 | 1.0 x 10 ⁻³ | Xe133m | 1.0 x 10 ⁻² |
| | Kr-83m | 2.0 x 10 ⁻² | Xe-133 | 2.1 x 10 ⁻¹ |
| | Kr-85m | 2.4 x 10 ⁻² | Xe-135 | 2.5 x 10 ⁻¹ |
| | Kr-85 | 1.0 x 10 ⁻³ | | |

The evaluation of continuing Naval Reactors activities combined with the proposed a for naval spent nuclear fuel is based on the combined airborne release source terms F.1.4.1.1-2. This table presents a summation of the INEL-ECF source term and the c Reactors operations source terms from Table F.1.4.1.1-1 for each location. Beginni shutdown of the S5G prototype, the NRF releases will only result from the INEL-ECF, is shown in the table.

The other analysis utilized the same source term at all locations. The INEL- Table F.1.4.1.1-1 was used to compare the incremental health effects due to providi or examination facilities at each location.

Both analyses also considered the impact on health effects of direct radiatio pool facility and the deposition of radionuclides onto the ground and into water su Sections F.1.3.6.4 and F.1.3.6.5.

Table F.1.4.1.1-2. Airborne releases used in the analysis of water pool activities Naval Reactors operations.

| Location | Annual Releases (Ci/year) | | | |
|--|---------------------------|--------------------------|--------|-------------------------|
| NRF, Savannah River, Hanford, Nevada Test Site, Oak Ridge Kesselring | H-3 | 9.35 x 10 ⁻² | Y-90 | 5.5 x 10 ⁻⁶ |
| | C-14 | 7.0 x 10 ⁻¹ | I-131 | 4.82 x 10 ⁻⁶ |
| | Sr-90 | 5.5 x 10 ⁻⁶ | Kr-85 | 3.0 x 10 ⁻¹ |
| | H-3 | 1.935 x 10 ⁻¹ | Sr-90 | 5.5 x 10 ⁻⁶ |
| | C-14 | 1.1 | Y-90 | 5.5 x 10 ⁻⁶ |
| | Ar-41 | 1.4 | I-131 | 5.0 x 10 ⁻⁴ |
| Portsmouth, Norfolk Puget Sound, Pearl Harbor | Kr-85 | 3.0 x 10 ⁻¹ | Cs-137 | 5.0 x 10 ⁻⁴ |
| | Co-60 | 1.0 x 10 ⁻³ | | |
| | H-3 | 9.45 x 10 ⁻² | Kr-88 | 2.0 x 10 ⁻² |
| | C-14 | 8.0 x 10 ⁻¹ | Sr-90 | 5.5 x 10 ⁻⁶ |
| | Ar-41 | 4.1 x 10 ⁻¹ | Y-90 | 5.5 x 10 ⁻⁶ |
| | Co-60 | 1.0 x 10 ⁻³ | I-131 | 4.8 x 10 ⁻⁶ |
| | Kr-83m | 2.0 x 10 ⁻² | Xe131m | 5.0 x 10 ⁻³ |
| | Kr-85m | 2.4 x 10 ⁻² | Xe133m | 1.0 x 10 ⁻² |
| | Kr-85 | 3.0 x 10 ⁻¹ | Xe-133 | 2.1 x 10 ⁻¹ |
| | Kr-87 | 5.0 x 10 ⁻² | Xe-135 | 2.5 x 10 ⁻¹ |

F.1.4.1.2 Dry Storage Source Terms. Another operation analyzed was the storage of naval

spent nuclear fuel in shipping containers or storage casks in a safe array at NRF, Kesselring locations. It is postulated that shielding and physical boundaries are with existing regulations to protect facility workers. There are expected to be no releases from the dry storage activity. The source will consist of an array of fil Supplementary shielding would be provided as needed to ensure that there would be n increase in radiation levels at the perimeter of the industrial area and that radia industrial area but outside the storage area would not require occupational radiati for workers. Each location analyzed would have a different number of storage casks received over time, shielding will be provided to limit radiation exposure rates as Distance falloff for radiation levels was determined using SPAN computer calculatio Section F.1.3.6.4.

F.1.4.1.3 Dry Cell Facility Source Terms. The normal airborne release source terms

utilized for the dry cell facility analyses are identical to the INEL-ECF releases expected that these values bound the actual releases from the proposed facility. A from the water pool analysis was utilized for the direct radiation calculations. T the proposed facility design, expected fuel examination capacity, and shielding cal

airborne releases, source terms for water deposition were identical to those utilized in the analysis.

F.1.4.1.4 Water Pool Storage. This section presents tabulated radiation exposure results for

the wet storage option. The following summary provides an indication of the incremental increase in radiation exposure due to the addition of an ECF-type facility.

Summary of Exposure Calculation Results

For Normal Operations - Water Pool Examination or Storage only

At All Sites

| | INEL/N- RF | Savannah River | Hanford | Puget Sound | Pearl Har- bor |
|---------------------------------|----------------------|------------------------|-------------------------|----------------------|----------------------|
| Worker EDE (rem) | 7.1×10^{-5} | 9.1×10^{-5} | 8.9×10^{-5} | 9.4×10^{-5} | 1.1×10^{-5} |
| MOI EDE (rem) | 2.5×10^{-7} | 4.8×10^{-7} | 2.4×10^{-7} | 8.7×10^{-5} | 2.0×10^{-5} |
| | | 3.8×10^{-6} * | 4.4×10^{-7} ** | | |
| NPA EDE (rem) | N/A | 2.1×10^{-8} | 1.3×10^{-8} | 6.2×10^{-4} | 5.2×10^{-4} |
| Total EDE (person-rem) | 1.7×10^{-3} | 3.6×10^{-2} | 8.0×10^{-3} | 1.3×10^{-1} | 1.4×10^{-1} |
| Number of Fatal Can- cers | 8.5×10^{-7} | 1.8×10^{-5} | 4.0×10^{-6} | 6.5×10^{-5} | 7.0×10^{-5} |
| * MOI (Barnwell Plant) | | | | | |
| ** MOI (FMEF) | | | | | |

| | Norfolk | Portsm- outh | Kesselring | Nevada Test Site | Oak R |
|---------------------------------|----------------------|----------------------|----------------------|----------------------|----------------------|
| Worker EDE (rem) | 6.9×10^{-5} | 7.7×10^{-5} | 8.5×10^{-5} | 4.6×10^{-5} | 1.2×10^{-5} |
| MOI EDE (rem) | 1.1×10^{-4} | 4.4×10^{-5} | 6.8×10^{-6} | 3.4×10^{-7} | 1.0×10^{-7} |
| NPA EDE (rem) | 6.8×10^{-5} | 3.3×10^{-4} | N/A | N/A | N/A |
| Total EDE (person-rem) | 2.8×10^{-1} | 4.5×10^{-2} | 8.2×10^{-2} | 1.8×10^{-4} | 1.0×10^{-4} |
| Number of Fatal Can- cers | 1.4×10^{-4} | 2.3×10^{-5} | 4.1×10^{-5} | 9.0×10^{-8} | 5.0×10^{-8} |

Evaluations of environmental impacts at DOE sites are presented in Volume 1, C, and F. The radiological impacts at these sites are quite low in that fatal cancer population within 50 miles from normal operations are well below 1.0. Further, impacts at shipyards and prototype sites are addressed in Appendix D and also are well below 1.0. In addition of the above small values to those which already exist at a site result in quite small.

The following summary provides the exposure calculation results for water pool examination plus all ongoing Naval Reactors operations at each site.

Summary of Exposure Calculation Results

For Normal Operations - Water Pool Examination or Storage

plus all ongoing Naval Reactors operations

At all sites

| | INEL/N- RF | Savannah River | Hanford | Puget Sound | Pearl Har- bor |
|---------------------------------|----------------------|------------------------|-------------------------|----------------------|----------------------|
| Worker EDE (rem) | 7.1×10^{-5} | 9.1×10^{-5} | 8.9×10^{-5} | 1.2×10^{-4} | 1.4×10^{-4} |
| MOI EDE (rem) | 2.5×10^{-7} | 4.8×10^{-7} | 2.4×10^{-7} | 1.0×10^{-4} | 2.3×10^{-4} |
| | | 3.8×10^{-6} * | 4.4×10^{-7} ** | | |
| NPA EDE (rem) | N/A | 2.1×10^{-8} | 1.3×10^{-8} | 7.2×10^{-4} | 5.8×10^{-4} |
| Total EDE (person-rem) | 1.7×10^{-3} | 3.6×10^{-2} | 8.0×10^{-3} | 1.5×10^{-1} | 1.7×10^{-1} |
| Number of Fatal Can- cers | 8.5×10^{-7} | 1.8×10^{-5} | 4.0×10^{-6} | 7.6×10^{-5} | 8.5×10^{-5} |
| * MOI (Barnwell Plant) | | | | | |

** MOI (FMEF)

| | Norfolk | Portsm- outh | Kesselring | Nevada Test Site | Oak R |
|---------------------------------|----------------------|----------------------|----------------------|----------------------|----------------------|
| Worker EDE (rem) | 8.4×10^{-5} | 9.7×10^{-5} | 1.4×10^{-4} | 4.6×10^{-5} | 1.2×10^{-5} |
| MOI EDE (rem) | 1.2×10^{-4} | 5.0×10^{-5} | 1.2×10^{-5} | 3.4×10^{-7} | 1.0×10^{-7} |
| NPA EDE (rem) | 7.4×10^{-5} | 3.5×10^{-4} | N/A | N/A | N/A |
| Total EDE (person-rem) | 3.4×10^{-1} | 5.5×10^{-2} | 1.4×10^{-1} | 1.8×10^{-4} | 1.0×10^{-4} |
| Number of Fatal Can- cers | 1.7×10^{-4} | 2.7×10^{-5} | 7.2×10^{-5} | 9.0×10^{-8} | 5.0×10^{-8} |

Tables F.1.4.1.4-1 through -10 present the detailed results of using the source F.1.4.1-2 to determine the radiation exposures. These tables thus depict the result examination operation is added to existing, current, continuing Naval Reactors operation and Navy shipyards.

Table F.1.4.1.4-1. Summary of Exposure Calculation Results.

For Normal Operations - Water Pool Examination plus all ongoing Naval Reactors operation At INEL

| Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|-----------------------|----------------------------------|
| Worker | 7.1×10^{-5} | 2.8×10^{-8} |
| MCW | 4.2×10^{-8} | 1.7×10^{-11} |
| MOI | 2.5×10^{-7} | 1.3×10^{-10} |

| | |
|--|---------------------------------|
| Exposure to Population within 50-mile Radius (person-rem) | Number of Fatal Can- cers |
|--|---------------------------------|

| | |
|-------------------------------|---|
| Popula- tion of 115,690 | 1.7×10^{-3} 8.5×10^{-7} |
|-------------------------------|---|

Table F.1.4.1.4-2. Summary of Exposure Calculation Results.

For Normal Operations - Water Pool Examination plus all ongoing Naval Reactors operation At Savannah River

| Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|----------------------|-----------------------|----------------------------------|
| Worker | 9.1×10^{-5} | 3.6×10^{-8} |
| MCW | 1.4×10^{-6} | 6.5×10^{-10} |
| MOI (New ECF)* | 4.8×10^{-7} | 2.4×10^{-10} |
| MOI (Barnwell Plant) | 3.8×10^{-6} | 1.9×10^{-9} |
| NPA | 2.1×10^{-8} | 1.1×10^{-11} |

| | |
|--|---------------------------------|
| Exposure to Population within 50-mile Radius (person-rem) | Number of Fatal Can- cers |
|--|---------------------------------|

| | |
|-------------------------------|---|
| Popula- tion of 579,541 | 3.6×10^{-2} 1.8×10^{-5} |
|-------------------------------|---|

* MOI (New ECF) applies if spent fuel facility is constructed on the Savannah River

**MOI (Barnwell Plant) applies if spent fuel facility is constructed at Barnwell Nuclear Plant

Table F.1.4.1.4-3. Summary of Exposure Calculation Results.

For Normal Operations - Water Pool Examination plus all ongoing Naval Reactors operation At Hanford

| Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|----------------|-----------------------|----------------------------------|
| Worker | 8.9×10^{-5} | 3.6×10^{-8} |
| MCW | 1.6×10^{-6} | 6.4×10^{-10} |
| MOI (New ECF)* | 2.4×10^{-7} | 1.2×10^{-10} |

| | | |
|--------------|------------------------|-------------------------|
| MOI (FMEF)** | 4.4 x 10 ⁻⁷ | 2.2 x 10 ⁻¹⁰ |
| NPA | 1.3 x 10 ⁻⁸ | 6.5 x 10 ⁻¹² |

| | | |
|--|--|---------------------------------|
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
|--|--|---------------------------------|

| | | |
|-------------------------------|------------------------|------------------------|
| Popula- tion of 375,860 | 8.0 x 10 ⁻³ | 4.0 x 10 ⁻⁶ |
|-------------------------------|------------------------|------------------------|

* MOI (New ECF) applies if spent fuel facility is constructed at the 200 area on t
 **MOI (FMEF) applies if spent fuel facility is constructed at the Fuels and Materia
 Facility.

Table F.1.4.1.4-4. Summary of Exposure Calculation Results.

For Normal Operations - Water Pool Storage plus all ongoing Naval Reactors operatio
 At Puget Sound

| | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|------------------------|----------------------------------|
| Loca- tion | | |
| Worker | 1.2 x 10 ⁻⁴ | 4.8 x 10 ⁻⁸ |
| MOI | 1.0 x 10 ⁻⁴ | 5.1 x 10 ⁻⁸ |
| NPA | 7.2 x 10 ⁻⁴ | 3.6 x 10 ⁻⁷ |

| | | |
|--|------------------------|---------------------------------|
| Exposure to Popula- tion within 50-mile Radius (per- son-rem) | | Number of Fatal Can- cers |
| Popula- tion of 2,975,810 | 1.5 x 10 ⁻¹ | 7.6 x 10 ⁻⁵ |

Table F.1.4.1.4-5. Summary of Exposure Calculation Results.

For Normal Operations - Water Pool Storage plus all ongoing Naval Reactors operatio
 At Pearl Harbor

| | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|------------------------|----------------------------------|
| Loca- tion | | |
| Worker | 1.4 x 10 ⁻⁴ | 5.6 x 10 ⁻⁸ |
| MOI | 2.3 x 10 ⁻⁵ | 1.1 x 10 ⁻⁸ |
| NPA | 5.8 x 10 ⁻⁴ | 2.9 x 10 ⁻⁷ |

| | | |
|--|------------------------|---------------------------------|
| Exposure to Popula- tion within 50-mile Radius (per- son-rem) | | Number of Fatal Can- cers |
| Popula- tion of 817,385 | 1.7 x 10 ⁻¹ | 8.5 x 10 ⁻⁵ |

Table F.1.4.1.4-6. Summary of Exposure Calculation Results.

For Normal Operations - Water Pool Storage plus all ongoing Naval Reactors operatio
 At Norfolk

| | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|------------------------|----------------------------------|
| Loca- tion | | |
| Worker | 8.4 x 10 ⁻⁵ | 3.4 x 10 ⁻⁸ |
| MOI | 1.2 x 10 ⁻⁴ | 6.1 x 10 ⁻⁸ |
| NPA | 7.4 x 10 ⁻⁵ | 3.7 x 10 ⁻⁸ |

| | | |
|--|--|---------------------------------|
| Exposure to Popula- tion within 50-mile Radius (per- son-rem) | | Number of Fatal Can- cers |
| Popula- | | |

tion of 3.4 x 10⁻¹ 1.7 x 10⁻⁴
1,539,002

Table F.1.4.1.4-7. Summary of Exposure Calculation Results.

For Normal Operations - Water Pool Storage plus all ongoing Naval Reactors operation At Portsmouth

| Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|------------------------|----------------------------------|
| Worker | 9.7 x 10 ⁻⁵ | 3.9 x 10 ⁻⁸ |
| MOI | 5.0 x 10 ⁻⁵ | 2.5 x 10 ⁻⁸ |
| NPA | 3.5 x 10 ⁻⁴ | 1.7 x 10 ⁻⁷ |

Exposure to Population within
50-mile Radius (per-
son-rem)
Popula-
tion of 5.5 x 10⁻² 2.7 x 10⁻⁵
2,432,627

Table F.1.4.1.4-8. Summary of Exposure Calculation Results.

For Normal Operations - Water Pool Storage plus all ongoing Naval Reactors operation At Kesselring

| Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|------------------------|----------------------------------|
| Worker | 1.4 x 10 ⁻⁴ | 5.6 x 10 ⁻⁸ |
| MOI | 1.2 x 10 ⁻⁵ | 5.8 x 10 ⁻⁹ |

Exposure to Population within
50-mile Radius (per-
son-rem)
Popula-
tion of 1.4 x 10⁻¹ 7.2 x 10⁻⁵
1,148,587

Table F.1.4.1.4-9. Summary of Exposure Calculation Results.

For Normal Operations - Water Pool Examination plus all ongoing Naval Reactors operation At Nevada Test Site

| Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|------------------------|----------------------------------|
| Worker | 4.6 x 10 ⁻⁵ | 1.8 x 10 ⁻⁸ |
| MCW | 3.7 x 10 ⁻⁹ | 1.5 x 10 ⁻¹² |
| MOI | 3.4 x 10 ⁻⁷ | 1.7 x 10 ⁻¹⁰ |

Exposure to Population within
50-mile Radius (person-rem)
Number of
Fatal Can-
cers

Popula-
tion of 1.8 x 10⁻⁴ 9.0 x 10⁻⁸
13,792

Table F.1.4.1.4-10. Summary of Exposure Calculation Results.

For Normal Operations - Water Pool Examination plus all ongoing Naval Reactors operation At Oak Ridge

| Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|------------------------|----------------------------------|
| Worker | 1.2 x 10 ⁻⁴ | 4.8 x 10 ⁻⁸ |
| MCW | 1.3 x 10 ⁻⁷ | 5.1 x 10 ⁻¹¹ |
| MOI | 1.0 x 10 ⁻⁴ | 5.1 x 10 ⁻⁸ |

| Exposure to Population within 50-mile Radius (person-rem) | Number of Fatal Can- cers | |
|--|---------------------------------|------------------------|
| Popula- tion of 871,531 | 1.0 x 10 ⁻¹ | 5.0 x 10 ⁻⁵ |

F.1.4.1.5 Dry Storage. This section presents tabulated radiation exposure results for the dry

storage option at INEL, Navy shipyard sites, and the Kesselring Site. Dry storage River, the Nevada Test Site, and Oak Ridge is not included in this section as it is 1, Appendices A, C, and F, respectively. The following summary provides an indicat tal change at each location due to the addition of dry storage areas. The health e spent fuel is largest at the Navy shipyards and is extremely small at all DOE locat Summary of Exposure Calculation Results

For Normal Operations - Dry Storage only
At all sites

| | INEL | Puget Sound | Pearl Harbor | Norfolk | Portsm- outh |
|-------------------------------|------------------------|------------------------|------------------------|------------------------|------------------------|
| Worker EDE (rem) | 1.1 x 10 ⁻² | 5.4 x 10 ⁻³ | 2.1 x 10 ⁻³ | 5.8 x 10 ⁻³ | 2.7 x 10 ⁻³ |
| MOI EDE (rem) | 6.5 x 10 ⁻¹ | 8.9 x 10 ⁻⁵ | 1.5 x 10 ⁻⁶ | 2.9 x 10 ⁻³ | 5.6 x 10 ⁻⁵ |
| NPA EDE (rem) | N/A | 7.4 x 10 ⁻³ | 2.3 x 10 ⁻² | 2.9 x 10 ⁻³ | 2.2 x 10 ⁻² |
| Total EDE (person-rem) | 1.7 x 10 ⁻¹ | 2.4 x 10 ⁻³ | 1.9 x 10 ⁻⁵ | 4.3 x 10 ⁻² | 4.6 x 10 ⁻⁴ |
| Number of Fatal Cancers | 8.6 x 10 ⁻¹ | 1.2 x 10 ⁻⁶ | 9.3 x 10 ⁻⁹ | 2.1 x 10 ⁻⁵ | 2.3 x 10 ⁻⁷ |

Tables F.1.4.1.5-1 through -6 present the results if a dry storage area is ad continuing Naval Reactors operations at all locations.

Table F.1.4.1.5-1. Summary of Exposure Calculation Results.

For Normal Operations - Dry Storage plus all ongoing Naval Reactors operations
At INEL

| Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|-------------------------|----------------------------------|
| Worker | 1.1 x 10 ⁻² | 4.4 x 10 ⁻⁶ |
| MOI | 1.1 x 10 ⁻¹⁰ | 5.5 x 10 ⁻¹⁴ |
| NPA | 6.5 x 10 ⁻¹⁴ | 3.3 x 10 ⁻¹⁷ |

| Exposure to Popula- tion within 50-mile Radius (per- son-rem) | Number of Fatal Can- cers | |
|--|---------------------------------|-------------------------|
| Popula- tion of 115,690 | 1.7 x 10 ⁻¹² | 8.6 x 10 ⁻¹⁶ |

Table F.1.4.1.5-2. Summary of Exposure Calculation Results.

For Normal Operations - Dry Storage plus all ongoing Naval Reactors operations
At Puget Sound

| Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|------------------------|----------------------------------|
| Worker | 5.4 x 10 ⁻³ | 2.2 x 10 ⁻⁶ |
| MOI | 1.1 x 10 ⁻⁴ | 5.3 x 10 ⁻⁸ |
| NPA | 7.5 x 10 ⁻³ | 3.8 x 10 ⁻⁶ |

| | | |
|--|------------------------|---------------------------------|
| Exposure to Popula- tion within 50-mile Radius (per- son-rem) | | Number of Fatal Can- cers |
| Popula- tion of | 3.6 x 10 ⁻² | 1.8 x 10 ⁻⁵ |
| 2,975,810 | | |

Table F.1.4.1.5-3. Summary of Exposure Calculation Results.

For Normal Operations - Dry Storage plus all ongoing Naval Reactors operations
At Pearl Harbor

| | | |
|---------------|------------------------|----------------------------------|
| | Total EDE (rem) | Likelihood of Fatal Cancer |
| Loca- tion | | |
| Worker | 2.1 x 10 ⁻³ | 8.5 x 10 ⁻⁷ |
| MOI | 5.3 x 10 ⁻⁶ | 2.7 x 10 ⁻⁹ |
| NPA | 2.3 x 10 ⁻² | 1.2 x 10 ⁻⁵ |

| | | |
|--|------------------------|---------------------------------|
| Exposure to Popula- tion within 50-mile Radius (per- son-rem) | | Number of Fatal Can- cers |
| Popula- tion of | 3.3 x 10 ⁻² | 1.7 x 10 ⁻⁵ |
| 817,385 | | |

Table F.1.4.1.5-4. Summary of Exposure Calculation Results.

For Normal Operations - Dry Storage plus all ongoing Naval Reactors operations
At Norfolk

| | | |
|---------------|------------------------|----------------------------------|
| | Total EDE (rem) | Likelihood of Fatal Cancer |
| Loca- tion | | |
| Worker | 5.8 x 10 ⁻³ | 2.3 x 10 ⁻⁶ |
| MOI | 2.9 x 10 ⁻³ | 1.5 x 10 ⁻⁶ |
| NPA | 2.9 x 10 ⁻³ | 1.5 x 10 ⁻⁶ |

| | | |
|--|------------------------|---------------------------------|
| Exposure to Popula- tion within 50-mile Radius (per- son-rem) | | Number of Fatal Can- cers |
| Popula- tion of | 9.7 x 10 ⁻² | 4.9 x 10 ⁻⁵ |
| 1,539,002 | | |

Table F.1.4.1.5-5. Summary of Exposure Calculation Results.

For Normal Operations - Dry Storage plus all ongoing Naval Reactors operations
At Portsmouth

| | | |
|---------------|------------------------|----------------------------------|
| | Total EDE (rem) | Likelihood of Fatal Cancer |
| Loca- tion | | |
| Worker | 2.7 x 10 ⁻³ | 1.1 x 10 ⁻⁶ |
| MOI | 6.3 x 10 ⁻⁵ | 3.1 x 10 ⁻⁸ |
| NPA | 2.2 x 10 ⁻² | 1.1 x 10 ⁻⁵ |

| | | |
|--|------------------------|---------------------------------|
| Exposure to Popula- tion within 50-mile Radius (per- son-rem) | | Number of Fatal Can- cers |
| Popula- tion of | 9.2 x 10 ⁻³ | 4.6 x 10 ⁻⁶ |
| 2,432,627 | | |

Table F.1.4.1.5-6. Summary of Exposure Calculation Results.

For Normal Operations - Dry Storage plus all ongoing Naval Reactors operations
At Kesselring

| Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|-----------------------|----------------------------------|
| Worker | 6.6×10^{-4} | 2.7×10^{-7} |
| MOI | 5.1×10^{-6} | 2.6×10^{-9} |

| Exposure to Popula- tion within 50-mile Radius (per- son-rem) | Number of Fatal Can- cers |
|--|---|
| Popula- tion of 1,148,587 | 5.7×10^{-2} 2.9×10^{-5} |

F.1.4.1.6 Dry Cell Operations. This section presents tabulated radiation exposure results for

the dry cell operations option. Since a facility like the proposed dry cell would alternatives which include examination of all naval spent fuel, this analysis was o INEL, Savannah River, Hanford, the Nevada Test Site, and Oak Ridge locations. The provides an indication of the incremental change at each location due to the additi The calculated health effect to the general population is roughly proportional to t population with Oak Ridge being the worst and Nevada Test Site being the best.

Summary of Exposure Calculation Results

For Normal Operations - Dry Cell Operations

At all sites

| | INEL/N- RF | Savannah River | Hanford | Nevada Test Site | Oak |
|---------------------------------|----------------------|------------------------|-------------------------|----------------------|-----|
| Worker EDE (rem) | 6.3×10^{-5} | 8.3×10^{-5} | 8.1×10^{-5} | 3.5×10^{-5} | 1.1 |
| MOI EDE (rem) | 2.5×10^{-7} | 4.8×10^{-7} | 2.4×10^{-7} | 3.4×10^{-7} | 8.9 |
| | | 3.8×10^{-6} * | 4.4×10^{-7} ** | | |
| NPA EDE (rem) | N/A | 2.1×10^{-8} | 1.3×10^{-8} | N/A | N/A |
| Total EDE (person-rem) | 1.7×10^{-3} | 3.6×10^{-2} | 8.0×10^{-3} | 1.8×10^{-4} | 1.0 |
| Number of Fatal Can- cers | 8.5×10^{-7} | 1.8×10^{-5} | 4.0×10^{-6} | 9.0×10^{-8} | 5.0 |

* MOI (Barnwell Plant)

** MOI (FMEF)

Tables F.1.4.1.6-1 through -5 present the detailed analysis results.

Table F.1.4.1.6-1. Summary of Exposure Calculation Results.

For Normal Operations - Dry Cell Operations

At INEL

| Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|-----------------------|----------------------------------|
| Worker | 6.3×10^{-5} | 2.5×10^{-8} |
| MCW | 4.2×10^{-8} | 1.7×10^{-11} |
| MOI | 2.5×10^{-7} | 1.3×10^{-10} |

| Exposure to Popula- tion within 50-mile Radius (per- son-rem) | Number of Fatal Can- cers |
|--|---|
| Popula- tion of 115,690 | 1.7×10^{-3} 8.5×10^{-7} |

Table F.1.4.1.6-2. Summary of Exposure Calculation Results.

For Normal Operations - Dry Cell Operations

At Savannah River

| Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|----------------------|-----------------------|----------------------------------|
| Worker | 8.3×10^{-5} | 3.3×10^{-8} |
| MCW | 1.3×10^{-6} | 5.3×10^{-10} |
| MOI (New ECF) * | 4.8×10^{-7} | 2.4×10^{-10} |
| MOI (Barnwell Plant) | 3.8×10^{-6} | 1.9×10^{-9} |
| NPA | 2.1×10^{-8} | 1.1×10^{-11} |

Exposure to Population within Number of
50-mile Radius (person-rem) Fatal Can-
cers

| | | |
|-------------------------------|----------------------|----------------------|
| Popula- tion of 579,541 | 3.6×10^{-2} | 1.8×10^{-5} |
|-------------------------------|----------------------|----------------------|

* MOI (New ECF) applies if spent fuel facility is constructed on the Savannah Rive
**MOI (Barnwell Plant) applies if spent fuel facility is constructed at Barnwell Nu
Table F.1.4.1.6-3. Summary of Exposure Calculation Results.
For Normal Operations - Dry Cell Operations
At Hanford

| Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|-----------------|-----------------------|----------------------------------|
| Worker | 8.1×10^{-5} | 3.2×10^{-8} |
| MCW | 1.5×10^{-6} | 6.1×10^{-10} |
| MOI (New ECF) * | 2.4×10^{-7} | 1.2×10^{-10} |
| MOI (FMEF) ** | 4.4×10^{-7} | 2.2×10^{-10} |
| NPA | 1.3×10^{-8} | 6.5×10^{-12} |

Exposure to Population within Number of
50-mile Radius (person-rem) Fatal Can-
cers

| | | |
|-------------------------------|----------------------|----------------------|
| Popula- tion of 375,800 | 8.0×10^{-3} | 4.0×10^{-6} |
|-------------------------------|----------------------|----------------------|

* MOI (New ECF) applies if spent fuel facility is constructed at the 200 area on t
**MOI (FMEF) applies if spent fuel facility is constructed at the Fuels and Materia
Facility.
Table F.1.4.1.6-4. Summary of Exposure Calculation Results.
For Normal Operations - Dry Cell Operations
At Nevada Test Site

| Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|-----------------------|----------------------------------|
| Worker | 3.5×10^{-5} | 1.5×10^{-8} |
| MCW | 3.7×10^{-9} | 1.5×10^{-12} |
| MOI | 3.4×10^{-7} | 1.7×10^{-10} |

Exposure to Population within Number of
50-mile Radius (person-rem) Fatal Can-
cers

| | | |
|------------------------------|----------------------|----------------------|
| Popula- tion of 13,792 | 1.8×10^{-4} | 9.0×10^{-8} |
|------------------------------|----------------------|----------------------|

Table F.1.4.1.6-5. Summary of Exposure Calculation Results.
For Normal Operations - Dry Cell Operations
At Oak Ridge

| Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|-----------------------|----------------------------------|
|---------------|-----------------------|----------------------------------|

| | | |
|--------|------------------------|-------------------------|
| Worker | 1.1 x 10 ⁻⁴ | 4.4 x 10 ⁻⁸ |
| MCW | 1.1 x 10 ⁻⁷ | 4.6 x 10 ⁻¹¹ |
| MOI | 8.9 x 10 ⁻⁵ | 4.5 x 10 ⁻⁸ |

Exposure to Population within Number of
50-mile Radius (person-rem) Fatal Can-
cers

Popula-
tion of
871,531

1.0 x 10⁻¹ 5.0 x 10⁻⁵

F.1.4.2 Accident Evaluation. The analysis of airborne releases from hypothetical accidents is

evaluated with RSAC-5. Unless stated otherwise, the following conditions were used calculations with RSAC-5. In most cases, these conditions are taken directly as de Meteorological Data

- Wind speed, direction, and Pasquill stability are taken from 50% and 95% See Section F.1.3.5 for a discussion of meteorological conditions.
 - The release is calculated as occurring at ground level (0 m).
 - Mixing layer height is 400 meters (1320 feet). Airborne materials freeel atmosphere near ground level in what is known as the mixing depth. A st above the mixing depth which restricts vertical diffusion.
 - Wet deposition is zero (no rain occurs to accelerate deposition and redu affected).
 - Dry deposition of the cloud is modeled. During movement of the radioact fraction of the plume is deposited on the ground due to gravitational fo available for exposure by ground surface radiation and ingestion.
 - The quantity of deposited radioactive material is proportional to the ma speed. The following dry deposition velocities (m/s) were used:
solids = 0.001 halogens = 0.01 noble gases = 0.0
cesium = 0.001 ruthenium = 0.001.
 - If radioactive releases occur through a stack, then additional plume dis accounted for by calculating a jet plume rise. In this analysis, jet pl
 - When released gases have a heat content, the plume can disperse more qui calculation, buoyant plume effects are ignored.
- Inhalation Data
- Breathing rate is 3.33 x 10⁻⁴ cubic meters per second (cu m/s) for worke NPA; 2.66 x 10⁻⁴ cu m/s for people at site boundary and beyond.
 - Particle size is 1.0 micron.
 - The internal exposure period is 50 years for individual organs and tissu radionuclides committed.
 - Exposure to the entire plume for the general public. The worker, MCW, a exposed as discussed in Section F.1.3.9.
 - Inhalation exposure factors based on ICRP 30.

Ground Surface Exposure

- Exposed to contaminated soil for 1 year for the general public. See Sec additional details.
- Building shielding factor is 0.7 which exposes the individual to contami hours a day.

Ingestion Data

- Ingestion numbers will be reduced by a factor of 10 to account for only

consumed being grown locally (such as in a person's garden).

- The following changes from RSAC-5 defaults were used:

Annual Dietary Consumption Rates:

177 Kg/yr Stored Vegetables (produce)
 18.3 Kg/yr Fresh Vegetables (leafy)
 94 Kg/yr Meat
 112 L/yr Milk.

F.1.4.2.1 Water Pool Storage. In the analysis of a spent fuel storage pool, a number of

possible disturbances and minor accidents have been postulated. A prerequisite for radioactive material to the environment under more severe accident conditions is the cladding of a fairly large amount of stored fuel, with an accompanying release of fission product particles of radioactive material from the fuel. Several conceivable mechanisms in this situation are the possibility that the fuel overheats so that the fuel cladding loses massive mechanical impact on the stored fuel.

The only way for the fuel to overheat would be to lose enough pool water such that the stored fuel ceases and the fuel temperature increases to fission product release temperature. The pool water could be lost by leakage at a rate in excess of the makeup system. A catastrophic event like an earthquake causes severe damage to the structure of the pool water from the pool structure would be a slow phenomenon with only gradually increasing water level which corrective measures can be taken in due time. Additionally, a thermal analysis demonstrates that fuel overheating is not possible in the event of a drained water pool.

The circumstances in which an event could lead to severe mechanical loading of the pool have been identified as:

- accidents during handling of heavy items, such as a lifting device failure
- external events (earthquake, tornado, flood, aircraft crash, etc.) which cause structural failure.

Prevention of inadvertent, uncontrolled nuclear chain reactions is generally achieved by the arrangement of the racks for the fuel, primarily by diminishing the chances for a chain reaction by placing fuel element bundles far enough apart to eliminate the possibility. Special attention is given to avoid accidental criticality which might be experienced in fuel transport and handling of spent nuclear reaction is prevented during fuel handling by applying the principle of three foot rule, element, module, or container at a time. In addition, fuel handling rules are developed to ensure that criticality cannot occur. The double accident criterion is applied to ensure that following two severe, concurrent, unrelated accidents. Thus, three fuel handling actions are required to reach an uncontrolled nuclear chain reaction.

F.1.4.2.1.1 Drained Water Pool.

F.1.4.2.1.1.1 Description of Conditions. In this hypothetical accident scenario, a

catastrophic event, like an earthquake, causes severe damage to the structure of the pool resulting in a complete loss of pool water. A thermal analysis of spent fuel in a water pool was performed to demonstrate that clad failure or fuel melting is not possible in the event of an accident. Air circulation through the fuel racks and fuel units was shown to be sufficient to prevent fuel failure in the unlikely event of complete loss of pool water. However, the loss of pool water results in increased direct radiation and a release of corrosion products.

F.1.4.2.1.1.2 Source Term. Conditions used in developing the source term are as follows:

- 300 naval fuel units would be in the water pool.
- The thermal analysis demonstrates that no fission product release would occur in an accident.
- The amount of corrosion products on the fuel units is based on best estimate.

- The release to the environment would occur at a constant rate over a 15-
 - One percent of the original corrosion products from the fuel units might atmo-
sphere due to thermal air currents. Additionally, 10% of the corrosion could be released to the environment with the pool water.
 - The following amounts of corrosion product nuclides might be released to atmosphere. As noted above, the release to the water environment is 10 values. This listing includes nuclides that result in at least 99% of t
 - No filtration by High Efficiency Particulate Air (HEPA) filters is assum
- | Nuclide | Curies |
|---------|----------------------|
| Co-60 | 3.6 |
| Fe-55 | 6.6 |
| Co-58 | 1.3 |
| Mn-54 | 2.2×10^{-1} |
| Fe-59 | 1.9×10^{-2} |

F.1.4.2.1.1.3 Results. The following table summarizes the public health risk to the general

population that might result from the hypothetical drained water pool accident at e number of fatal cancers would be expected to occur over a 50-year period. "Risk" i number of fatal cancers times the probability of occurrence. The results are prese accident with 50% and 95% meteorology. For INEL, the evaluation basis earthquake r peak ground acceleration at the ECF (Rizzo 1994). This is based on the event being earthquake epicenter and involving a surface rupture length of 34 kilometers. Usin spectra, which is appropriate for a risk oriented analysis, the analyses of the str indicate that damage sufficient to cause the pool to drain would not occur if the p several sections of the water pool were empty, a crack could develop in the area be floor of some of the older sections of the water pool. However, the INEL-ECF water always filled. Sections of the pool are only drained if maintenance work is necess Taking into account the probability of the initiating seismic event (1×10^{-4} per y and the probability the earthquake will occur with a section of the pool drained, t occurrence of an event leading to draining of the pool is estimated to be in the ra year. A value of 10^{-5} was used to develop the risk results in the table.

A beyond design basis seismic event was also considered. For INEL, this beyo earthquake is based on a scenario that results in a peak ground acceleration at the (Rizzo 1994). Analysis of this event has shown that some cracks could develop. Th beyond design basis event is estimated to be in the range of 10^{-6} to 10^{-7} per year the initiating seismic event (2×10^{-5} to 6×10^{-5}), and the probability of failure would be taken to prevent the pool from draining. A value of 10^{-6} was selected to beyond design basis event. Any cracks developed as a result of either a design bas basis seismic event are expected to be small and mitigative actions could be taken draining. Analysis has shown that air cooling is sufficient to maintain fuel integ drained. No overheating of fuel would occur; hence, no fission products would be r were completely drained. The consequences calculated stem from the release of radi within the pool water and would be the same for the design basis and beyond design Since the consequences are the same, the following table uses the accident probabill seismic event since that results in the larger risk.

For locations other than INEL, water pools might need to be constructed. For was expected that the design approaches would be similar to or better than were use the INEL-ECF. Therefore, a probability value of 10^{-5} per year was also used at the probability that a design basis seismic event would lead to draining of a water poo based on site specific population data and meteorology.

Drained Water Pool Summary

| Site | Maximal- ly ex- posed off-site individu- al (MOI) (rem) | No. of fatal cancers if acci- dent occurs | Risk per year |
|------|---|---|----------------------|
| INEL | 1.7×10^{-2} | 1.7×10^{-2} | 1.7×10^{-7} |

| | | | |
|------------------|----------------------|----------------------|----------------------|
| Savannah River | 1.6×10^{-2} | 1.1×10^{-1} | 1.1×10^{-6} |
| Hanford | 6.3×10^{-3} | 4.7×10^{-2} | 4.7×10^{-7} |
| Puget Sound | 1.4 | 5.1×10^{-1} | 5.1×10^{-6} |
| Pearl Harbor | 7.9×10^{-1} | 1.1 | 1.1×10^{-5} |
| Norfolk | 3.0 | 6.0×10^{-1} | 6.0×10^{-6} |
| Portsmouth | 1.6 | 3.4×10^{-1} | 3.4×10^{-6} |
| Kesselring | 2.9×10^{-1} | 2.5×10^{-1} | 2.5×10^{-6} |
| Nevada Test Site | 3.3×10^{-2} | 1.9×10^{-3} | 1.9×10^{-8} |
| Oak Ridge | 5.2 | 1.8×10^{-1} | 1.8×10^{-6} |

The risk for this hypothetical accident is generally more severe at Navy shipyards sites. At all sites, this accident results in the highest risk of the wet storage

For the hypothetical drained water pool scenario, the radioactive plume might contamination of the ground to a downwind distance of 0.29 mile. This would yield by the accident of approximately 11 acres. The calculated downwind distance would the boundaries of all sites under evaluation with the exception of Oak Ridge and No Table F.1.4.2.1.1-1. Summary of Exposure Calculation Results.

For Wet Storage - Drained Water Pool
At INEL

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 7.5×10^{-1} | 3.0×10^{-4} |
| MCW | 6.9×10^{-4} | 2.7×10^{-7} |
| NPA | 3.9×10^{-4} | 2.0×10^{-7} |
| MOI | 2.8×10^{-3} | 1.4×10^{-6} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 115690 | 6.7 | 3.3×10^{-3} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 2.1 | 8.3×10^{-4} |
| MCW | 7.6×10^{-3} | 3.0×10^{-6} |
| NPA | 2.3×10^{-3} | 1.2×10^{-6} |
| MOI | 1.7×10^{-2} | 8.5×10^{-6} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 115690 | 3.5×10^1 | 1.7×10^{-2} |

Table F.1.4.2.1.1-2. Summary of Exposure Calculation Results.

For Wet Storage - Drained Water Pool
At Savannah River

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 3.4×10^{-1} | 1.3×10^{-4} |
| MCW | 2.0×10^{-2} | 7.9×10^{-6} |
| NPA | 2.5×10^{-4} | 1.3×10^{-7} |
| MOI (New ECF) | 3.5×10^{-3} | 1.8×10^{-6} |
| MOI (Barnwell) | 1.3×10^{-2} | 6.3×10^{-6} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- |

cers

Population of
579541 2.4 x 10¹ 1.2 x 10⁻²

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|------------------------|----------------------------------|
| Worker | 2.1 | 8.3 x 10 ⁻⁴ |
| MCW | 2.5 x 10 ⁻¹ | 1.0 x 10 ⁻⁴ |
| NPA | 4.3 x 10 ⁻³ | 2.1 x 10 ⁻⁶ |
| MOI (New ECF) | 1.6 x 10 ⁻² | 8.0 x 10 ⁻⁶ |
| MOI (Barnwell) | 1.4 x 10 ⁻¹ | 7.2 x 10 ⁻⁵ |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |

Population of
579541 2.2 x 10² 1.1 x 10⁻¹

Table F.1.4.2.1.1-3. Summary of Exposure Calculation Results.
For Wet Storage - Drained Water Pool
At Hanford

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|------------------------|----------------------------------|
| Worker | 3.4 x 10 ⁻¹ | 1.3 x 10 ⁻⁴ |
| MCW | 2.6 x 10 ⁻² | 1.0 x 10 ⁻⁵ |
| NPA | 3.0 x 10 ⁻⁴ | 1.5 x 10 ⁻⁷ |
| MOI (New ECF) | 8.3 x 10 ⁻⁴ | 4.2 x 10 ⁻⁷ |
| MOI (FMEF) | 1.7 x 10 ⁻³ | 8.6 x 10 ⁻⁷ |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |

Population of
375860 4.8 2.4 x 10⁻³

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|------------------------|----------------------------------|
| Worker | 2.1 | 8.3 x 10 ⁻⁴ |
| MCW | 1.6 x 10 ⁻¹ | 6.6 x 10 ⁻⁵ |
| NPA | 4.8 x 10 ⁻³ | 2.4 x 10 ⁻⁶ |
| MOI (New ECF) | 6.3 x 10 ⁻³ | 3.2 x 10 ⁻⁶ |
| MOI (FMEF) | 2.2 x 10 ⁻² | 1.1 x 10 ⁻⁵ |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |

Population of
375860 9.4 x 10¹ 4.7 x 10⁻²

Table F.1.4.2.1.1-4. Summary of Exposure Calculation Results.
For Wet Storage - Drained Water Pool
At Puget Sound

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|----------|------------------------|----------------------------------|
| Worker | 1.8 x 10 ⁻¹ | 7.3 x 10 ⁻⁵ |

| | | |
|--|----------------------|---------------------------------|
| MCW | N/A | N/A |
| NPA | 2.2×10^{-1} | 1.1×10^{-4} |
| MOI | 1.2×10^{-1} | 6.0×10^{-5} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 2975810 | 1.7×10^2 | 8.2×10^{-2} |

95% METEOROLOGY

| | | |
|--|-------------------|---------------------------------|
| | Total EDE | Likelihood |
| Location | (rem) | of Fatal |
| Worker | 2.1 | Cancer |
| MCW | N/A | 8.3×10^{-4} |
| NPA | 2.6 | N/A |
| MOI | 1.4 | 1.3×10^{-3} |
| Exposure to Population within 50-mile Radius (person-rem) | | 7.2×10^{-4} |
| | | Number of Fatal Can- cers |
| Population of 2975810 | 1.0×10^3 | 5.1×10^{-1} |

Table F.1.4.2.1.1-5. Summary of Exposure Calculation Results.
For Wet Storage - Drained Water Pool
At Pearl Harbor

50% METEOROLOGY

| | | |
|--|----------------------|---------------------------------|
| | Total EDE | Likelihood |
| Location | (rem) | of Fatal |
| Worker | 7.5×10^{-1} | Cancer |
| MCW | N/A | 3.0×10^{-4} |
| NPA | 1.9×10^{-1} | N/A |
| MOI | 2.0×10^{-1} | 9.7×10^{-5} |
| Exposure to Population within 50-mile Radius (person-rem) | | 9.8×10^{-5} |
| | | Number of Fatal Can- cers |
| Population of 817385 | 8.0×10^2 | 4.0×10^{-1} |

95% METEOROLOGY

| | | |
|--|----------------------|---------------------------------|
| | Total EDE | Likelihood |
| Location | (rem) | of Fatal |
| Worker | 2.1 | Cancer |
| MCW | N/A | 8.3×10^{-4} |
| NPA | 6.3 | N/A |
| MOI | 7.9×10^{-1} | 3.1×10^{-3} |
| Exposure to Population within 50-mile Radius (person-rem) | | 3.9×10^{-4} |
| | | Number of Fatal Can- cers |
| Population of 817385 | 2.2×10^3 | 1.1 |

Table F.1.4.2.1.1-6. Summary of Exposure Calculation Results.
For Wet Storage - Drained Water Pool
At Norfolk

50% METEOROLOGY

| | | |
|--|-----------|------------|
| | Total EDE | Likelihood |
| | | of Fatal |

| Location | (rem) | Cancer |
|---|----------------------|-------------------------|
| Worker | 1.8×10^{-1} | 7.4×10^{-5} |
| MCW | N/A | N/A |
| NPA | 4.6×10^{-2} | 2.3×10^{-5} |
| MOI | 2.8×10^{-1} | 1.4×10^{-4} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Cancers |
| Population of 1539002 | 1.5×10^2 | 7.7×10^{-2} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|----------------------|----------------------------|
| Worker | 2.1 | 8.3×10^{-4} |
| MCW | N/A | N/A |
| NPA | 5.3×10^{-1} | 2.7×10^{-4} |
| MOI | 3.0 | 1.5×10^{-3} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Cancers |
| Population of 1539002 | 1.2×10^3 | 6.0×10^{-1} |

Table F.1.4.2.1.1-7. Summary of Exposure Calculation Results.
For Wet Storage - Drained Water Pool
At Portsmouth

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|----------------------|----------------------------|
| Worker | 1.8×10^{-1} | 7.3×10^{-5} |
| MCW | N/A | N/A |
| NPA | 4.4×10^{-2} | 2.2×10^{-5} |
| MOI | 1.3×10^{-1} | 6.4×10^{-5} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Cancers |
| Population of 2432627 | 6.5×10^1 | 3.2×10^{-2} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|----------------------|----------------------------|
| Worker | 2.1 | 8.3×10^{-4} |
| MCW | N/A | N/A |
| NPA | 9.8×10^{-1} | 4.9×10^{-4} |
| MOI | 1.6 | 7.9×10^{-4} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Cancers |
| Population of 2432627 | 6.7×10^2 | 3.4×10^{-1} |

Table F.1.4.2.1.1-8. Summary of Exposure Calculation Results.
For Wet Storage - Drained Water Pool
At Kesselring

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 1.8×10^{-1} | 7.4×10^{-5} |
| MCW | N/A | N/A |
| NPA | N/A | N/A |
| MOI | 2.0×10^{-2} | 1.0×10^{-5} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 1148587 | 7.1×10^1 | 3.6×10^{-2} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 2.1 | 8.3×10^{-4} |
| MCW | N/A | N/A |
| NPA | N/A | N/A |
| MOI | 2.9×10^{-1} | 1.5×10^{-4} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 1148587 | 5.0×10^2 | 2.5×10^{-1} |

Table F.1.4.2.1.1-9. Summary of Exposure Calculation Results.
For Wet Storage - Drained Water Pool
At Nevada Test Site

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 1.2×10^{-1} | 4.8×10^{-5} |
| MCW | 9.3×10^{-5} | 3.7×10^{-8} |
| NPA | N/A | N/A |
| MOI | 1.5×10^{-3} | 7.5×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 13792 | 3.2×10^{-1} | 1.6×10^{-4} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 2.1 | 8.3×10^{-4} |
| MCW | 5.4×10^{-3} | 2.2×10^{-6} |
| NPA | N/A | N/A |
| MOI | 3.3×10^{-2} | 1.7×10^{-5} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 13792 | 3.7 | 1.9×10^{-3} |

Table F.1.4.2.1.1-10. Summary of Exposure Calculation Results.
For Wet Storage - Drained Water Pool
At Oak Ridge

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|----------------------|----------------------------|
| Worker | 7.5×10^{-1} | 3.0×10^{-4} |
| MCW | 2.0×10^{-2} | 7.9×10^{-6} |
| NPA | 2.6×10^{-1} | 1.3×10^{-4} |
| MOI | 8.2×10^{-1} | 4.1×10^{-4} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Cancers |
| Population of 871531 | 7.1×10^1 | 3.6×10^{-2} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|----------------------|----------------------------|
| Worker | 2.1 | 8.3×10^{-4} |
| MCW | 1.2×10^{-1} | 4.8×10^{-5} |
| NPA | 1.6 | 8.2×10^{-4} |
| MOI | 5.2 | 2.6×10^{-3} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Cancers |
| Population of 871531 | 3.5×10^2 | 1.8×10^{-1} |

F.1.4.2.1.2 Accidental Criticality.**F.1.4.2.1.2.1 Description of Conditions. In this hypothetical accident scenario, an accidental uncon-**

trolled chain reaction producing 1×10^{19} fissions is postulated. The criticality the water pool which is not emptied by the event and does not subsequently empty. products includes those specified in Regulatory Guide 3.34 (NUREG 1979b) from the fission products remaining in the fuel as a result of the original use. Removal of pool water is included.

F.1.4.2.1.2.2 Source Term. Conditions used in developing the source term are as follows:

- The fraction of the fission products released to the building is 100% of 25% of the halogens, 0.1% of the ruthenium (Elder et al. 1986), and 0.05 and remaining solids.
- The original inventory of fission products from two naval fuel units are release in addition to those created by the criticality event.
- A High Efficiency Particulate Air (HEPA) filter removes 99.9% of the sol products from the plume.
- The release to the environment occurs at a constant rate over a 15-minute period. This is conservative as compared to the 8-hour release allowed in Regulatory Guide 3.34.
- The following amounts of radionuclides are released to the environment. This listing includes nuclides that result in at least 99% of the

| Nuclide | Curies | Nuclide | Curies |
|---------|-------------------|---------|--------------------|
| Te-133 | 3.4×10^3 | I-132 | 1.7×10^1 |
| I-134 | 3.5×10^2 | Sr-90 | 1.94×10^1 |
| I-135 | 1.2×10^2 | Y-91m | 4.3×10^1 |

| | | | |
|--------|-------------------------|---------|---------|
| Cs-138 | 1.6 x 10 ⁻⁴ | Rb-88 | 1.7 x 1 |
| Rb-89 | 6.05 x 10 ⁻⁴ | Y-91 | 1.1 x 1 |
| Pu-238 | 3.7 x 10 ⁻⁴ | Cs-139 | 7.3 x 1 |
| Br-84 | 2.3 x 10 ² | Ba-142 | 4.8 x 1 |
| I-133 | 2.4 x 10 ⁰ | Y-93 | 1.3 x 1 |
| Sr-91 | 5.4 x 10 ⁻⁶ | Ba-137m | 1.9 x 1 |
| Sr-92 | 2.4 x 10 ⁻⁴ | Ru-106 | 7.6 x 1 |
| Ba-139 | 6.9 x 10 ⁻⁶ | Zr-95 | 1.4 x 1 |
| Ba-141 | 8.8 x 10 ⁻⁴ | Sr-89 | 7.01 x |
| I-129 | 5.1 x 10 ⁻³ | Eu-154 | 1.3 x 1 |
| I-131 | 3.2 x 10 ⁻¹ | | |
| H-3 | 1.42 x 10 ² | | |
| Cs-134 | 1.5 x 10 ⁻² | | |
| Ba-140 | 2.5 x 10 ⁻⁵ | | |
| I-136 | 1.1 x 10 ⁴ | | |
| Cs-137 | 2.0 x 10 ⁻² | | |
| Ce-144 | 4.5 x 10 ⁻² | | |
| Nb-95 | 2.7 x 10 ⁻² | | |
| Rb-90 | 2.2 x 10 ⁻² | | |

F.1.4.2.1.2.3 Results. The following table summarizes the public health risk to the general

population that would result from the hypothetical criticality accident at each local site. Fatal cancers would be expected to occur over a 50-year period. "Risk" is defined as the number of fatal cancers times the probability of occurrence. An accidental criticality during spent nuclear fuel operations is extremely unlikely. There are no known events of this type which have involved the handling of fuel modules either in or out of water. Due to the need for a neutron moderator, large quantities of naval fuels would be required to achieve criticality in a dry storage cask. Procedures in water in conjunction with required physical barriers ensure that a criticality does not occur. This criterion specifies that the fuel will not attain a critical condition even if multiple unrelated accidents occur at the same time. The DOE criticality control requirements include a contingency criterion which specifies that a second unlikely and unrelated accident must occur for a critical condition to result. To satisfy the NNPP double accident criterion, naval fuel handling operations are conducted in the following manner:

- No more than one module is to be handled in one area at a time.
- If two modules are capable of achieving a critical condition, separation must be maintained by a positive barrier between them which is locked in place.
- If three modules are required to achieve criticality, a physical barrier must be locked and is required to be placed between them.
- If four or more modules are needed to achieve criticality, no barriers are required and the modules are to remain separated.

Based on the above requirements, at least three distinct errors are needed to achieve a criticality. For example, bringing two or more modules in close proximity is always required to achieve a criticality. Failure to maintain separation constitutes an error. Secondly, failure to recognize and use the barriers required also constitutes an error. A human error rate of 10⁻³ per operation (Swain) is taken as the probability of error for trained personnel. Further, because all fuel handling operations are checked by an independent verifier, an additional factor of 10⁻¹ may be taken for each independent error. For naval fuel handling, an error in which two modules are brought into a subcritical state is a violation of a fundamental requirement. Compliance with this requirement alone does not ensure a subcritical state is maintained. Therefore, the bringing of two or more modules to a subcritical state is considered separate and independent of all other errors. Because a second error must occur to achieve a criticality, an additional reduction in the probability is warranted. For example, if a barrier is required and has been checked, a second value of 10⁻⁴ is appropriate for a total value of 10⁻⁵. This probability is taken as the likelihood of a criticality for movement of a single fuel module. For an estimated 1,000 fuel handling operations a year, a value of 10⁻⁵ per year has been used in the risk assessment of accidental criticality.

Accidental Criticality Summary

| | |
|---------------------|---------------------|
| Maximal-ly expected | No. of fatal cancer |
|---------------------|---------------------|

| Site | posed off-site individu- al (MOI) (rem) | if acci- dent occurs | Risk per year |
|------------------|---|-------------------------|----------------------|
| INEL | 9.2×10^{-3} | 6.4×10^{-3} | 6.4×10^{-8} |
| Savannah River | 9.4×10^{-3} | 4.5×10^{-2} | 4.5×10^{-7} |
| Hanford | 2.8×10^{-3} | 1.6×10^{-2} | 1.6×10^{-7} |
| Puget Sound | 1.3 | 2.8×10^{-1} | 2.8×10^{-6} |
| Pearl Harbor | 6.7×10^{-1} | 6.0×10^{-1} | 6.0×10^{-6} |
| Norfolk | 2.7 | 3.5×10^{-1} | 3.5×10^{-6} |
| Portsmouth | 1.4 | 1.5×10^{-1} | 1.5×10^{-6} |
| Kesselring | 2.3×10^{-1} | 1.1×10^{-1} | 1.1×10^{-6} |
| Nevada Test Site | 2.0×10^{-2} | 7.0×10^{-4} | 7.0×10^{-9} |
| Oak Ridge | 4.7 | 8.8×10^{-2} | 8.8×10^{-7} |

The risk for this hypothetical accident is more severe at Navy shipyards than all sites, this accident results in the second highest risk of the wet storage acci

For the hypothetical criticality accident scenario, the radioactive plume mig of the ground to a downwind distance of 0.25 mile. This would yield a total area i of approximately 8 acres. The calculated downwind distance would be contained with all sites under evaluation with the exception of Oak Ridge and Norfolk.

Table F.1.4.2.1.2-1. Summary of Exposure Calculation Results.

For Wet Storage - Accidental Criticality
At INEL

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 3.0 | 1.2×10^{-3} |
| MCW | 1.3×10^{-3} | 5.1×10^{-7} |
| NPA | 5.9×10^{-4} | 2.9×10^{-7} |
| MOI | 2.0×10^{-3} | 1.0×10^{-6} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 115690 | 5.5 | 2.8×10^{-3} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 8.0 | 3.2×10^{-3} |
| MCW | 1.3×10^{-2} | 5.0×10^{-6} |
| NPA | 2.8×10^{-3} | 1.4×10^{-6} |
| MOI | 9.2×10^{-3} | 4.6×10^{-6} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 115690 | 1.3×10^1 | 6.4×10^{-3} |

Table F.1.4.2.1.2-2. Summary of Exposure Calculation Results.

For Wet Storage - Accidental Criticality
At Savannah River

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|----------|--------------------|----------------------------------|
| Worker | 1.3 | 5.3×10^{-4} |

| | | |
|--|-------------------------|----------------------|
| MCW | 6.8×10^{-2} | 2.7×10^{-5} |
| NPA | 7.4×10^{-4} | 3.7×10^{-7} |
| MOI (New (ECF) | 3.3×10^{-3} | 1.6×10^{-6} |
| MOI (Barnwell) | 1.2×10^{-2} | 5.9×10^{-6} |
| Exposure to Population within 50-mile Radius (person-rem) | Number of Fatal Can- | c- |
| Population of 579541 | 2.2×10^1 | 1.1×10^{-2} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|-------------------------|----------------------------------|
| Worker | 8.0 | 3.2×10^{-3} |
| MCW | 7.9×10^{-1} | 3.1×10^{-4} |
| NPA | 6.4×10^{-3} | 3.2×10^{-6} |
| MOI (New ECF) | 9.4×10^{-3} | 4.7×10^{-6} |
| MOI (Barnwell) | 1.1×10^{-1} | 5.3×10^{-5} |
| Exposure to Population within 50-mile Radius (person-rem) | Number of Fatal Can- | c- |
| Population of 579541 | 8.9×10^1 | 4.5×10^{-2} |

Table F.1.4.2.1.2-3. Summary of Exposure Calculation Results.
For Wet Storage - Accidental Criticality
At Hanford

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|-------------------------|----------------------------------|
| Worker | 1.3 | 5.3×10^{-4} |
| MCW | 8.9×10^{-2} | 3.5×10^{-5} |
| NPA | 6.6×10^{-4} | 3.3×10^{-7} |
| MOI (New (ECF) | 4.7×10^{-4} | 2.4×10^{-7} |
| MOI (FMEF) | 1.3×10^{-3} | 6.7×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | Number of Fatal Can- | c- |
| Population of 375860 | 2.2 | 1.1×10^{-3} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|-------------------------|----------------------------------|
| Worker | 8.0 | 3.2×10^{-3} |
| MCW | 4.9×10^{-1} | 2.0×10^{-4} |
| NPA | 6.9×10^{-3} | 3.5×10^{-6} |
| MOI (New ECF) | 2.8×10^{-3} | 1.4×10^{-6} |
| MOI (FMEF) | 1.2×10^{-2} | 6.1×10^{-6} |
| Exposure to Population within 50-mile Radius (person-rem) | Number of Fatal Can- | c- |
| Population of 375860 | 3.1×10^1 | 1.6×10^{-2} |

Table F.1.4.2.1.2-4. Summary of Exposure Calculation Results.
For Wet Storage - Accidental Criticality
At Puget Sound

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 7.2×10^{-1} | 2.9×10^{-4} |
| MCW | N/A | N/A |
| NPA | 7.7×10^{-1} | 3.8×10^{-4} |
| MOI | 1.1×10^{-1} | 5.6×10^{-5} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 2975810 | 2.3×10^2 | 1.1×10^{-1} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|--------------------|----------------------------------|
| Worker | 8.0 | 3.2×10^{-3} |
| MCW | N/A | N/A |
| NPA | 8.8 | 4.4×10^{-3} |
| MOI | 1.3 | 6.3×10^{-4} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 2975810 | 5.6×10^2 | 2.8×10^{-1} |

Table F.1.4.2.1.2-5. Summary of Exposure Calculation Results.
For Wet Storage - Accidental Criticality
At Pearl Harbor

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 3.0 | 1.2×10^{-3} |
| MCW | N/A | N/A |
| NPA | 7.0×10^{-1} | 3.5×10^{-4} |
| MOI | 1.8×10^{-1} | 8.9×10^{-5} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 817385 | 5.6×10^2 | 2.8×10^{-1} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 8.0 | 3.2×10^{-3} |
| MCW | N/A | N/A |
| NPA | 2.2×10^1 | 2.2×10^{-2} |
| MOI | 6.7×10^{-1} | 3.4×10^{-4} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 817385 | 1.2×10^3 | 6.0×10^{-1} |

Table F.1.4.2.1.2-6. Summary of Exposure Calculation Results.
For Wet Storage - Accidental Criticality

At Norfolk

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 7.4×10^{-1} | 2.9×10^{-4} |
| MCW | N/A | N/A |
| NPA | 1.6×10^{-1} | 8.2×10^{-5} |
| MOI | 2.7×10^{-1} | 1.3×10^{-4} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 1539002 | 1.6×10^2 | 8.1×10^{-2} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|--------------------|----------------------------------|
| Worker | 8.0 | 3.2×10^{-3} |
| MCW | N/A | N/A |
| NPA | 1.8 | 8.8×10^{-4} |
| MOI | 2.7 | 1.4×10^{-3} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 1539002 | 7.0×10^2 | 3.5×10^{-1} |

Table F.1.4.2.1.2-7. Summary of Exposure Calculation Results.
For Wet Storage - Accidental Criticality
At Portsmouth

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 7.2×10^{-1} | 2.9×10^{-4} |
| MCW | N/A | N/A |
| NPA | 1.5×10^{-1} | 7.7×10^{-5} |
| MOI | 1.2×10^{-1} | 5.9×10^{-5} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 2432627 | 7.9×10^1 | 4.0×10^{-2} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|--------------------|----------------------------------|
| Worker | 8.0 | 3.2×10^{-3} |
| MCW | N/A | N/A |
| NPA | 3.3 | 1.6×10^{-3} |
| MOI | 1.4 | 7.0×10^{-4} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 2432627 | 2.9×10^2 | 1.5×10^{-1} |

Table F.1.4.2.1.2-8. Summary of Exposure Calculation Results.
For Wet Storage - Accidental Criticality
At Kesselring

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 7.4×10^{-1} | 2.9×10^{-4} |
| MCW | N/A | N/A |
| NPA | N/A | N/A |
| MOI | 1.9×10^{-2} | 9.7×10^{-6} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 1148587 | 5.6×10^1 | 2.8×10^{-2} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 8.0 | 3.2×10^{-3} |
| MCW | N/A | N/A |
| NPA | N/A | N/A |
| MOI | 2.3×10^{-1} | 1.2×10^{-4} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 1148587 | 2.2×10^2 | 1.1×10^{-1} |

Table F.1.4.2.1.2-9. Summary of Exposure Calculation Results.
For Wet Storage - Accidental Criticality
At Nevada Test Site

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 4.8×10^{-1} | 1.9×10^{-4} |
| MCW | 2.1×10^{-4} | 8.0×10^{-8} |
| NPA | N/A | N/A |
| MOI | 1.5×10^{-3} | 7.3×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 13792 | 4.3×10^{-1} | 2.2×10^{-4} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 8.0 | 3.2×10^{-3} |
| MCW | 8.1×10^{-3} | 3.3×10^{-6} |
| NPA | N/A | N/A |
| MOI | 2.0×10^{-2} | 9.9×10^{-6} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |

Population of
13792 1.4 7.0×10^{-4}
Table F.1.4.2.1.2-10. Summary of Exposure Calculation Results.
For Wet Storage - Accidental Criticality
At Oak Ridge

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 3.0 | 1.2×10^{-3} |
| MCW | 6.6×10^{-2} | 2.6×10^{-5} |
| NPA | 9.1×10^{-1} | 4.6×10^{-4} |
| MOI | 7.6×10^{-1} | 3.8×10^{-4} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 871531 | 7.4×10^1 | 3.7×10^{-2} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 8.0 | 3.2×10^{-3} |
| MCW | 3.6×10^{-1} | 1.4×10^{-4} |
| NPA | 5.6 | 2.8×10^{-3} |
| MOI | 4.7 | 2.4×10^{-3} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 871531 | 1.8×10^2 | 8.8×10^{-2} |

F.1.4.2.1.3 Mechanical Damage from Operator Error, Crane Failure, or Similar

Accidents

F.1.4.2.1.3.1 Description of Conditions. Accidental mechanical damage to spent fuel was

evaluated. The hypothetical accident included damage to one fuel unit, allowing fi the elements to escape through the clad failures. All gas and some volatile and so calculated to be released to the pool. The release fractions are consistent with s Regulatory Guide 1.4. Due to the presence of pool water, no solids would be releas the facility.

F.1.4.2.1.3.2 Source Term. Conditions used in developing the source term are as follows:

- One fuel unit is damaged because only one fuel unit would be handled at storage facility design prevents damage to stored units from such events
- One percent of the fuel is damaged and those fission products are availa
- All (100%) of the noble gases are released to the environment.
- Approximately 25% of the halogens are released to the pool and 90% of th products are absorbed in the water as they rise through the pool water. the halogens are released to the air inside the facility.
- Due to the gaseous nature of the released fission products, installed HE

not remove them once they are released to the air in the building.

- The release to the environment occurs at a constant rate over a 15-minute
- There is no particulate fission product release to the atmosphere due to pool water.
 - The following amounts of radionuclides could be released to the environment. This listing includes nuclides that result in at least 99% of the possible

| Nuclides | Curies |
|----------|-----------------------|
| H-3 | 1.42 |
| I-129 | 2.52×10^{-6} |
| I-131 | 5.37×10^{-5} |

F.1.4.2.1.3.3 Results. The following table summarizes the public health risk to the general

population that would result from the hypothetical mechanical damage accident at each number of fatal cancers would be expected to occur over a 50-year period. "Risk" is number of fatal cancers times the probability of occurrence. The probability of the damage is small based on the conservative fuel handling rules. At the INEL-ECF, if a drop of a heavy container into a storage rack could crush the rack and the stored fuel are never moved over the storage rack area. The heavy containers are brought only to the receiving area to discharge a single fuel unit. The spent fuel is removed from the next fuel unit is brought into the receiving area. Therefore, two errors must occur to be possible. The first is that fuel is improperly left in the discharge station while over the discharge station. The second is that the cask must accidentally fall from the crane must fail. The probability of failure associated with crane failure has been further, the crane failure must also occur in the right location and the drop must have sufficient energy is available to damage both the discharge station structural and an additional factor of 10^{-2} has been taken for this event, giving the total probability of a cask in the right location. Allowing a fuel unit to remain in the stand requires a fuel handling procedure call for the fuel unit to be removed from the stand and to a storage location away from the receiving area. In addition, because independent of for all fuel movement, an error by a verifier is also required. Therefore, based on (Swain and Guttman 1983), the likelihood of this error is taken as 10^{-4} per year. The probability of a cask drop on a fuel unit is taken as 10^{-8} per year per fuel movement. The estimated rate of 1,000 fuel movements per year, the overall probability is taken as 10^{-5} events per year.

Wet Storage Mechanical Damage Summary

| Site | Maximal-ly exposed off-site individual (MOI) (rem) | No. of fatal cancers if accident occurs | Risk per year |
|------------------|--|---|-----------------------|
| INEL | 2.6×10^{-6} | 5.3×10^{-6} | 5.3×10^{-11} |
| Savannah River | 2.2×10^{-6} | 2.0×10^{-5} | 2.0×10^{-10} |
| Hanford | 9.8×10^{-7} | 8.6×10^{-6} | 8.6×10^{-11} |
| Puget Sound | 1.7×10^{-4} | 7.2×10^{-5} | 7.2×10^{-10} |
| Pearl Harbor | 9.3×10^{-5} | 1.5×10^{-4} | 1.5×10^{-9} |
| Norfolk | 3.5×10^{-4} | 8.0×10^{-5} | 8.0×10^{-10} |
| Portsmouth | 1.9×10^{-4} | 5.6×10^{-5} | 5.6×10^{-10} |
| Kesselring | 3.6×10^{-5} | 6.0×10^{-5} | 6.0×10^{-10} |
| Nevada Test Site | 4.6×10^{-6} | 5.6×10^{-7} | 5.6×10^{-12} |
| Oak Ridge | 5.9×10^{-4} | 3.4×10^{-5} | 3.4×10^{-10} |

The risk for this hypothetical accident is generally more severe at Navy ship sites. At all sites, this accident results in the lowest or next to the lowest risk evaluated.

For the hypothetical wet storage mechanical damage accident scenario, the radionuclides might cause contamination of the ground to a downwind distance of less than 0.06 miles and a total area impacted by the accident of less than 0.5 acre. The calculated downwind

contained within the boundaries of all sites under evaluation.
 Table F.1.4.2.1.3-1. Summary of Exposure Calculation Results.
 For Wet Storage - Mechanical Damage
 At INEL

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 1.9×10^{-4} | 7.6×10^{-8} |
| MCW | 2.5×10^{-7} | 9.6×10^{-11} |
| NPA | 1.5×10^{-7} | 7.4×10^{-11} |
| MOI | 5.7×10^{-7} | 2.9×10^{-10} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 115690 | 5.0×10^{-3} | 2.5×10^{-6} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 5.2×10^{-4} | 2.1×10^{-7} |
| MCW | 2.4×10^{-6} | 9.6×10^{-10} |
| NPA | 8.3×10^{-7} | 4.2×10^{-10} |
| MOI | 2.6×10^{-6} | 1.3×10^{-9} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 115690 | 1.1×10^{-2} | 5.3×10^{-6} |

Table F.1.4.2.1.3-2. Summary of Exposure Calculation Results.
 For Wet Storage - Mechanical Damage
 At Savannah River

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 8.4×10^{-5} | 3.4×10^{-8} |
| MCW | 5.2×10^{-6} | 2.1×10^{-9} |
| NPA | 9.1×10^{-8} | 4.5×10^{-11} |
| MOI (New ECF) | 3.9×10^{-7} | 1.9×10^{-10} |
| MOI (Barnwell) | 1.5×10^{-6} | 7.4×10^{-10} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 579541 | 7.1×10^{-3} | 3.5×10^{-6} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|----------------|----------------------|----------------------------------|
| Worker | 5.2×10^{-4} | 2.1×10^{-7} |
| MCW | 6.7×10^{-5} | 2.6×10^{-8} |
| NPA | 1.4×10^{-6} | 7.2×10^{-10} |
| MOI (New ECF) | 2.2×10^{-6} | 1.1×10^{-9} |
| MOI (Barnwell) | 1.8×10^{-5} | 9.0×10^{-9} |

Exposure to Population within 50-mile Radius (person-rem) Number of Fatal Cancers

Population of 579541 4.1×10^{-2} 2.0×10^{-5}

Table F.1.4.2.1.3-3. Summary of Exposure Calculation Results.
For Wet Storage - Mechanical Damage
At Hanford

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|----------------------|----------------------------|
| Worker | 8.4×10^{-5} | 3.4×10^{-8} |
| MCW | 7.1×10^{-6} | 2.9×10^{-9} |
| NPA | 1.0×10^{-7} | 5.1×10^{-11} |
| MOI (New ECF) | 1.3×10^{-7} | 6.5×10^{-11} |
| MOI (FMEF) | 2.4×10^{-7} | 1.2×10^{-10} |

Exposure to Population within 50-mile Radius (person-rem) Number of Fatal Cancers

Population of 375860 9.4×10^{-4} 4.7×10^{-7}

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|----------------------|----------------------------|
| Worker | 5.2×10^{-4} | 2.1×10^{-7} |
| MCW | 4.4×10^{-5} | 1.8×10^{-8} |
| NPA | 1.6×10^{-6} | 7.9×10^{-10} |
| MOI (New ECF) | 9.8×10^{-7} | 4.9×10^{-10} |
| MOI (FMEF) | 3.1×10^{-6} | 1.5×10^{-9} |

Exposure to Population within 50-mile Radius (person-rem) Number of Fatal Cancers

Population of 375860 1.7×10^{-2} 8.6×10^{-6}

Table F.1.4.2.1.3-4. Summary of Exposure Calculation Results.
For Wet Storage - Mechanical Damage
At Puget Sound

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|----------|----------------------|----------------------------|
| Worker | 4.6×10^{-5} | 1.8×10^{-8} |
| MCW | N/A | N/A |
| NPA | 5.5×10^{-5} | 2.7×10^{-8} |
| MOI | 1.3×10^{-5} | 6.7×10^{-9} |

Exposure to Population within 50-mile Radius (person-rem) Number of Fatal Cancers

Population of 2975810 6.0×10^{-3} 3.0×10^{-6}

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|----------|-----------------|----------------------------|
|----------|-----------------|----------------------------|

| | | |
|--|------------------------|---------------------------------|
| Worker | 5.2 x 10 ⁻⁴ | 2.1 x 10 ⁻⁷ |
| MCW | N/A | N/A |
| NPA | 6.5 x 10 ⁻⁴ | 3.2 x 10 ⁻⁷ |
| MOI | 1.7 x 10 ⁻⁴ | 8.4 x 10 ⁻⁸ |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |

| | | |
|--------------------------|------------------------|------------------------|
| Population of 2975810 | 1.5 x 10 ⁻¹ | 7.2 x 10 ⁻⁵ |
|--------------------------|------------------------|------------------------|

Table F.1.4.2.1.3-5. Summary of Exposure Calculation Results.
For Wet Storage - Mechanical Damage
At Pearl Harbor

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|------------------------|----------------------------------|
| Worker | 1.9 x 10 ⁻⁴ | 7.6 x 10 ⁻⁸ |
| MCW | N/A | N/A |
| NPA | 4.9 x 10 ⁻⁵ | 2.4 x 10 ⁻⁸ |
| MOI | 2.3 x 10 ⁻⁵ | 1.2 x 10 ⁻⁸ |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 817385 | 1.1 x 10 ⁻¹ | 5.6 x 10 ⁻⁵ |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|------------------------|----------------------------------|
| Worker | 5.2 x 10 ⁻⁴ | 2.1 x 10 ⁻⁷ |
| MCW | N/A | N/A |
| NPA | 1.6 x 10 ⁻³ | 7.9 x 10 ⁻⁷ |
| MOI | 9.3 x 10 ⁻⁵ | 4.6 x 10 ⁻⁸ |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 817385 | 3.1 x 10 ⁻¹ | 1.5 x 10 ⁻⁴ |

Table F.1.4.2.1.3-6. Summary of Exposure Calculation Results.
For Wet Storage - Mechanical Damage
At Norfolk

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|------------------------|----------------------------------|
| Worker | 4.6 x 10 ⁻⁵ | 1.9 x 10 ⁻⁸ |
| MCW | N/A | N/A |
| NPA | 1.2 x 10 ⁻⁵ | 6.0 x 10 ⁻⁹ |
| MOI | 3.2 x 10 ⁻⁵ | 1.6 x 10 ⁻⁸ |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 1539002 | 1.4 x 10 ⁻² | 7.0 x 10 ⁻⁶ |

95% METEOROLOGY

Likelihood

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 5.2×10^{-4} | 2.1×10^{-7} |
| MCW | N/A | N/A |
| NPA | 1.4×10^{-4} | 7.0×10^{-8} |
| MOI | 3.5×10^{-4} | 1.7×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 1539002 | 1.6×10^{-1} | 8.0×10^{-5} |

Table F.1.4.2.1.3-7. Summary of Exposure Calculation Results.
For Wet Storage - Mechanical Damage
At Portsmouth

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 4.6×10^{-5} | 1.8×10^{-8} |
| MCW | N/A | N/A |
| NPA | 1.1×10^{-5} | 5.6×10^{-9} |
| MOI | 1.5×10^{-5} | 7.4×10^{-9} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 2432627 | 3.8×10^{-3} | 1.9×10^{-6} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 5.2×10^{-4} | 2.1×10^{-7} |
| MCW | N/A | N/A |
| NPA | 2.5×10^{-4} | 1.3×10^{-7} |
| MOI | 1.9×10^{-4} | 9.3×10^{-8} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 2432627 | 1.1×10^{-1} | 5.6×10^{-5} |

Table F.1.4.2.1.3-8. Summary of Exposure Calculation Results.
For Wet Storage - Mechanical Damage
At Kesselring

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 4.6×10^{-5} | 1.9×10^{-8} |
| MCW | N/A | N/A |
| NPA | N/A | N/A |
| MOI | 3.2×10^{-6} | 1.6×10^{-9} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 1148587 | 4.7×10^{-2} | 2.3×10^{-5} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|----------------------|----------------------------|
| Worker | 5.2×10^{-4} | 2.1×10^{-7} |
| MCW | N/A | N/A |
| NPA | N/A | N/A |
| MOI | 3.6×10^{-5} | 1.8×10^{-8} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Cancers |
| Population of 1148587 | 1.2×10^{-1} | 6.0×10^{-5} |

Table F.1.4.2.1.3-9. Summary of Exposure Calculation Results.
For Wet Storage - Mechanical Damage
At Nevada Test Site

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|----------------------|----------------------------|
| Worker | 3.0×10^{-5} | 1.2×10^{-8} |
| MCW | 3.0×10^{-8} | 1.5×10^{-11} |
| NPA | N/A | N/A |
| MOI | 3.8×10^{-7} | 1.9×10^{-10} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Cancers |
| Population of 13792 | 4.5×10^{-4} | 2.3×10^{-7} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|----------------------|----------------------------|
| Worker | 5.2×10^{-4} | 2.1×10^{-7} |
| MCW | 1.8×10^{-6} | 7.1×10^{-10} |
| NPA | N/A | N/A |
| MOI | 4.6×10^{-6} | 2.3×10^{-9} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Cancers |
| Population of 13792 | 1.1×10^{-3} | 5.6×10^{-7} |

Table F.1.4.2.1.3-10. Summary of Exposure Calculation Results.
For Wet Storage - Mechanical Damage
At Oak Ridge

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|----------------------|----------------------------|
| Worker | 1.9×10^{-4} | 7.6×10^{-8} |
| MCW | 5.4×10^{-6} | 2.2×10^{-9} |
| NPA | 6.6×10^{-5} | 3.3×10^{-8} |
| MOI | 9.3×10^{-5} | 4.7×10^{-8} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Cancers |
| Population of 871531 | 2.0×10^{-2} | 1.0×10^{-5} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 5.2×10^{-4} | 2.1×10^{-7} |
| MCW | 3.3×10^{-5} | 1.3×10^{-8} |
| NPA | 4.2×10^{-4} | 2.1×10^{-7} |
| MOI | 5.9×10^{-4} | 3.0×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 871531 | 6.7×10^{-2} | 3.4×10^{-5} |

F.1.4.2.1.4 Airplane Crash.**F.1.4.2.1.4.1 Description of Conditions. Impact into water pools by aircraft with resulting**

damage to the naval fuel units stored inside the pool was evaluated. Based on the occurrence, as discussed in Section F.3, specific analyses were only performed for Nevada Test Site, Oak Ridge, Pearl Harbor, Norfolk, and Kesselring locations. At a likelihood of occurrence is less than 10^{-7} per year. The hypothetical accident involves units stored at the water pool. Fission products and corrosion products are released to the water pool; however, the pool water is not released to the environment. An airplane crash would not produce enough force to cause the pool to leak because the walls of the pool are constructed of thick, reinforced concrete with earth surrounding them, making them unlikely to fail. In addition, it was judged unlikely that an airplane would impact the water pool at an angle that would expose the floor of the pool or the walls of the pool below the water level to the presence of pool water results in only a release of gaseous fission products to the atmosphere.

F.1.4.2.1.4.2 Source Term. Conditions used in developing the source term are as follows:

- One percent of the fission products from each of the fuel units stored in the pool are available for release.
- Of the available fission products, 100% of the noble gases and 25% of the fission products are released to the pool water. Due to the presence of pool water, a reduction in release by a factor of 10 prior to release to the atmosphere occurs.
- No solid fission products or corrosion products are released to the environment in the continued presence of pool water.
- The release to the environment occurs at a constant rate over a 15-minute period.
- 300 naval fuel units would be in the water pool.
- No filtration by HEPA filters is assumed.
- The following amounts of radionuclides could be released to the environment. This listing includes nuclides that result in at least 99% of the release.

| Nuclide | Curies |
|---------|-----------------------|
| I-129 | 7.59×10^{-4} |
| I-131 | 1.61×10^{-2} |
| H-3 | 4.28×10^2 |

F.1.4.2.1.4.3 Results. The following table summarizes the public health risk to the general

population that would result from the hypothetical airplane crash accident at each location. The number of fatal cancers would be expected to occur over a 50-year period. "Risk" is defined as the number of fatal cancers per person per year.

cancers times the probability of occurrence.

| Site | Probability of accident per year | Water Pool Airplane Crash Summary | |
|---------------------|-------------------------------------|--|--|
| | | Maximal- ly ex- posed off-site individu- al (MOI) (rem) | No. of fatal cancers if acci- dent occurs |
| Savannah River | 2×10^{-6} | 6.4×10^{-4} | 6.1×10^{-3} |
| Pearl Harbor | 2×10^{-5} | 2.8×10^{-2} | 4.6×10^{-2} |
| Norfolk | 4×10^{-7} | 1.1×10^{-1} | 2.4×10^{-2} |
| Kesselring | 2×10^{-7} | 1.1×10^{-2} | 1.8×10^{-2} |
| Nevada Test Site | 4×10^{-7} | 1.3×10^{-3} | 1.7×10^{-4} |
| Oak Ridge | 1×10^{-6} | 1.8×10^{-1} | 1.0×10^{-2} |

The risk for this hypothetical accident is most severe at Pearl Harbor. For probabilities less than 10^{-7} per year, consequences were not calculated since it is not substantially contribute to the risk.

For the hypothetical airplane crash into a wet storage facility accident scene plume might result in contamination of the ground to a downwind distance of less than would yield a total area impacted by the accident of less than 0.5 acre. The calculation would be contained within the boundaries of all sites that are at risk for this accident. Table F.1.4.2.1.4-1. Summary of Exposure Calculation Results.

For Wet Storage - Airplane Crash
At Savannah River

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 2.5×10^{-2} | 1.0×10^{-5} |
| MCW | 1.6×10^{-3} | 6.3×10^{-7} |
| NPA | 2.8×10^{-5} | 1.4×10^{-8} |
| MOI | 1.1×10^{-4} | 5.5×10^{-8} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 579541 | 2.2 | 1.1×10^{-3} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 1.6×10^{-1} | 6.3×10^{-5} |
| MCW | 2.0×10^{-2} | 8.0×10^{-6} |
| NPA | 4.3×10^{-4} | 2.2×10^{-7} |
| MOI | 6.4×10^{-4} | 3.2×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 579541 | 1.2×10^1 | 6.1×10^{-3} |

Table F.1.4.2.1.4-2. Summary of Exposure Calculation Results.
For Wet Storage - Airplane Crash
At Pearl Harbor

50% METEOROLOGY

Likelihood

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 5.7×10^{-2} | 2.3×10^{-5} |
| MCW | N/A | N/A |
| NPA | 1.5×10^{-2} | 7.3×10^{-6} |
| MOI | 6.9×10^{-3} | 3.5×10^{-6} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 817385 | 3.3×10^1 | 1.7×10^{-2} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 1.6×10^{-1} | 6.3×10^{-5} |
| MCW | N/A | N/A |
| NPA | 4.7×10^{-1} | 2.4×10^{-4} |
| MOI | 2.8×10^{-2} | 1.4×10^{-5} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 817385 | 9.2×10^1 | 4.6×10^{-2} |

Table F.1.4.2.1.4-3. Summary of Exposure Calculation Results.
For Wet Storage - Airplane Crash
At Norfolk

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 1.4×10^{-2} | 5.6×10^{-6} |
| MCW | N/A | N/A |
| NPA | 3.6×10^{-3} | 1.8×10^{-6} |
| MOI | 9.6×10^{-3} | 4.8×10^{-6} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 1539002 | 4.2 | 2.1×10^{-3} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 1.6×10^{-1} | 6.3×10^{-5} |
| MCW | N/A | N/A |
| NPA | 4.2×10^{-2} | 2.1×10^{-5} |
| MOI | 1.1×10^{-1} | 5.3×10^{-5} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 1539002 | 4.8×10^1 | 2.4×10^{-2} |

Table F.1.4.2.1.4-4. Summary of Exposure Calculation Results.
For Wet Storage - Airplane Crash
At Kesselring

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 1.4×10^{-2} | 5.6×10^{-6} |
| MCW | N/A | N/A |
| NPA | N/A | N/A |
| MOI | 9.5×10^{-4} | 4.8×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 1148587 | 1.4×10^1 | 7.1×10^{-3} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 1.6×10^{-1} | 6.3×10^{-5} |
| MCW | N/A | N/A |
| NPA | N/A | N/A |
| MOI | 1.1×10^{-2} | 5.4×10^{-6} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 1148587 | 3.6×10^1 | 1.8×10^{-2} |

Table F.1.4.2.1.4-5. Summary of Exposure Calculation Results.
For Wet Storage - Airplane Crash
At Nevada Test Site

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 9.0×10^{-3} | 3.6×10^{-6} |
| MCW | 9.1×10^{-6} | 3.7×10^{-9} |
| NPA | N/A | N/A |
| MOI | 5.5×10^{-5} | 2.8×10^{-8} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 13792 | 1.3×10^{-1} | 6.5×10^{-5} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 1.6×10^{-1} | 6.4×10^{-5} |
| MCW | 5.3×10^{-4} | 2.2×10^{-7} |
| NPA | N/A | N/A |
| MOI | 1.3×10^{-3} | 6.5×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |

Population of
13792 3.3×10^{-1} 1.7×10^{-4}

Table F.1.4.2.1.4-6. Summary of Exposure Calculation Results.
For Wet Storage - Airplane Crash
At Oak Ridge

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 5.7×10^{-2} | 2.3×10^{-5} |
| MCW | 1.6×10^{-3} | 6.5×10^{-7} |
| NPA | 2.0×10^{-2} | 9.9×10^{-6} |
| MOI | 2.8×10^{-2} | 1.4×10^{-5} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 871531 | 6.0 | 3.0×10^{-3} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 1.6×10^{-1} | 6.3×10^{-5} |
| MCW | 9.9×10^{-3} | 3.9×10^{-6} |
| NPA | 1.3×10^{-1} | 6.3×10^{-5} |
| MOI | 1.8×10^{-1} | 8.9×10^{-5} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 871531 | 2.0×10^1 | 1.0×10^{-2} |

F.1.4.2.1.5 HEPA Filter Fire.**F.1.4.2.1.5.1 Description of Conditions. In this hypothetical accident scenario, a fire in the**

ECF High Efficiency Particulate Air (HEPA) filter banks is postulated. This accident scenario involves the ignition of a flammable mixture released upstream of the system or by an external fire that spreads to this system. Although the risks associated with this accident are relatively low, they are analyzed to bound the higher probability, lower consequence type accident category. The release fractions associated with this accident were conservatively chosen so that a HEPA filter failure due to crushing or impact was also bounded.

F.1.4.2.1.5.2 Source Term. Conditions used in developing the source term are as follows:

- The original inventory of fission products in the filters is based on the unabated ECF releases over a 5-year period.
- One percent of the radionuclide inventory present on the filters becomes released during the fire. Release fractions for HEPA filters are small because the filter material containing glass fibers which would melt during a fire and trap the radionuclides. Measurements from experiments show that one hundredth of 1% of the material in HEPA filters could be released during a fire, but 1% has been analyzed to allow for uncertainties in the final results of an individual filter.
- The release to the environment occurs at a constant rate over a 15-minute period.
- There is no increase in direct radiation due to this accident.
- The following amounts of radionuclides could be released to the environment. This

listing includes nuclides that result in at least 99% of the possible ex

- No filtration by HEPA filters is assumed.

| Nuclide | Curies | Nuclide | Curies |
|---------|-----------------------|---------|-----------------------|
| Cs-137 | 1.46×10^{-3} | Co-60 | 2.09×10^{-3} |
| Cs-134 | 2.04×10^{-4} | Sr-90 | 8.90×10^{-4} |
| Ba-137M | 6.26×10^{-6} | Y-90 | 8.90×10^{-4} |
| Fe-55 | 2.32×10^{-3} | Eu-154 | 9.80×10^{-5} |
| Ni-63 | 2.98×10^{-3} | | |

F.1.4.2.1.5.3 Results. The following table summarizes the public health risk to the general

population that would result from the hypothetical HEPA filter fire accident at each of the sites. The number of fatal cancers would be expected to occur over a 50-year period. "Risk" is defined as the number of fatal cancers times the probability of occurrence. The probability of a fire in a HEPA filter system is assigned to be the probability of other fires spreading to the HEPA filter system. As discussed in the previous section, the probability of 5 x 10⁻³ is assigned to chemical fires. The probability of HEPA filter fire is assigned to be the probability of a chemical fire since chemicals would not be stored in the immediate vicinity of the filter. Additionally, HEPA filters are not inherently volatile or explosive. It is estimated that the probability of an existing chemical fire to spread to the HEPA filter is less than 0.1. This results in a value of 5 x 10⁻⁴ for a HEPA filter fire. A value of 5 x 10⁻⁴ was used to develop the

HEPA Filter Fire Summary

| Site | Maximal-ly exposed off-site individual (MOI) (rem) | No. of fatal cancer if accident occurs | Risk per year |
|------------------|--|--|----------------------|
| INEL | 2.5×10^{-5} | 5.3×10^{-5} | 2.7×10^{-8} |
| Savannah River | 2.1×10^{-5} | 1.3×10^{-4} | 6.5×10^{-8} |
| Hanford | 7.0×10^{-6} | 5.3×10^{-5} | 2.7×10^{-8} |
| Puget Sound | 1.6×10^{-3} | 6.4×10^{-4} | 3.2×10^{-7} |
| Pearl Harbor | 8.7×10^{-4} | 1.2×10^{-3} | 6.0×10^{-7} |
| Norfolk | 3.3×10^{-3} | 6.9×10^{-4} | 3.5×10^{-7} |
| Portsmouth | 1.7×10^{-3} | 3.9×10^{-4} | 2.0×10^{-7} |
| Kesselring | 3.5×10^{-4} | 3.3×10^{-4} | 1.7×10^{-7} |
| Nevada Test Site | 4.3×10^{-5} | 5.7×10^{-6} | 2.9×10^{-9} |
| Oak Ridge | 5.7×10^{-3} | 2.2×10^{-4} | 1.1×10^{-7} |

The risk for this hypothetical accident is generally more severe at the Navy DOE sites.

For the hypothetical HEPA filter fire accident scenario, the radioactive plume contamination of the ground to a downwind distance of less than 0.06 mile. This would be impacted by the accident of less than 0.5 acre. The calculated downwind distance would be within the boundaries of all sites under evaluation.

Table F.1.4.2.1.5-1. Summary of Exposure Calculation Results.

For Wet Storage - HEPA Filter Fire
At INEL

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|----------------------|----------------------------|
| Worker | 8.7×10^{-4} | 3.5×10^{-7} |
| MCW | 7.9×10^{-7} | 3.2×10^{-10} |
| NPA | 4.5×10^{-7} | 2.2×10^{-10} |
| MOI | 9.9×10^{-6} | 5.0×10^{-9} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Cancers |
| Population of 115690 | 7.6×10^{-2} | 3.8×10^{-5} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 2.4×10^{-3} | 9.6×10^{-7} |
| MCW | 8.8×10^{-6} | 3.5×10^{-9} |
| NPA | 2.7×10^{-6} | 1.4×10^{-9} |
| MOI | 2.5×10^{-5} | 1.3×10^{-8} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 115690 | 1.1×10^{-1} | 5.3×10^{-5} |

Table F.1.4.2.1.5-2. Summary of Exposure Calculation Results.
For Wet Storage - HEPA Filter Fire
At Savannah River

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 3.9×10^{-4} | 1.5×10^{-7} |
| MCW | 2.3×10^{-5} | 8.8×10^{-9} |
| NPA | 2.9×10^{-7} | 1.4×10^{-10} |
| MOI (New ECF) | 7.2×10^{-6} | 3.6×10^{-9} |
| MOI (Barnwell) | 1.7×10^{-5} | 8.6×10^{-9} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 579541 | 4.1×10^{-2} | 2.0×10^{-5} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 2.4×10^{-3} | 9.6×10^{-7} |
| MCW | 2.9×10^{-4} | 1.1×10^{-7} |
| NPA | 4.9×10^{-6} | 2.5×10^{-9} |
| MOI (New ECF) | 2.1×10^{-5} | 1.0×10^{-8} |
| MOI (Barnwell) | 1.6×10^{-4} | 8.1×10^{-8} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 579541 | 2.5×10^{-1} | 1.3×10^{-4} |

Table F.1.4.2.1.5-3. Summary of Exposure Calculation Results.
For Wet Storage - HEPA Filter Fire
At Hanford

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|----------------------|----------------------------------|
| Worker | 3.9×10^{-4} | 1.5×10^{-7} |
| MCW | 3.0×10^{-5} | 1.2×10^{-8} |
| NPA | 3.5×10^{-7} | 1.8×10^{-10} |
| MOI (New ECF) | 9.6×10^{-7} | 4.8×10^{-10} |
| MOI (FMEF) | 1.9×10^{-6} | 9.7×10^{-10} |

| | |
|--|---|
| Exposure to Population within 50-mile Radius (person-rem) | Number of Fatal Can- cers |
| Population of 375860 | 6.7 x 10 ⁻³ 3.4 x 10 ⁻⁶ |

95% METEOROLOGY

| | | |
|--|------------------------|----------------------------------|
| | Total EDE (rem) | Likelihood of Fatal Cancer |
| Location | | |
| Worker | 2.4 x 10 ⁻³ | 9.6 x 10 ⁻⁷ |
| MCW | 1.9 x 10 ⁻⁴ | 7.5 x 10 ⁻⁸ |
| NPA | 5.5 x 10 ⁻⁶ | 2.7 x 10 ⁻⁹ |
| MOI (New ECF) | 7.0 x 10 ⁻⁶ | 3.5 x 10 ⁻⁹ |
| MOI (FMEF) | 2.4 x 10 ⁻⁵ | 1.2 x 10 ⁻⁸ |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |

| | |
|-------------------------|---|
| Population of 375860 | 1.1 x 10 ⁻¹ 5.3 x 10 ⁻⁵ |
|-------------------------|---|

Table F.1.4.2.1.5-4. Summary of Exposure Calculation Results.
For Wet Storage - HEPA Filter Fire
At Puget Sound

50% METEOROLOGY

| | | |
|--|------------------------|----------------------------------|
| | Total EDE (rem) | Likelihood of Fatal Cancer |
| Location | | |
| Worker | 2.1 x 10 ⁻⁴ | 8.4 x 10 ⁻⁸ |
| MCW | N/A | N/A |
| NPA | 2.5 x 10 ⁻⁴ | 1.2 x 10 ⁻⁷ |
| MOI | 1.4 x 10 ⁻⁴ | 6.8 x 10 ⁻⁸ |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |

| | |
|--------------------------|---|
| Population of 2975810 | 3.4 x 10 ⁻¹ 1.7 x 10 ⁻⁴ |
|--------------------------|---|

95% METEOROLOGY

| | | |
|--|------------------------|----------------------------------|
| | Total EDE (rem) | Likelihood of Fatal Cancer |
| Location | | |
| Worker | 2.4 x 10 ⁻³ | 9.6 x 10 ⁻⁷ |
| MCW | N/A | N/A |
| NPA | 2.9 x 10 ⁻³ | 1.5 x 10 ⁻⁶ |
| MOI | 1.6 x 10 ⁻³ | 8.0 x 10 ⁻⁷ |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |

| | |
|--------------------------|----------------------------|
| Population of 2975810 | 1.3 6.4 x 10 ⁻⁴ |
|--------------------------|----------------------------|

Table F.1.4.2.1.5-5. Summary of Exposure Calculation Results.
For Wet Storage - HEPA Filter Fire
At Pearl Harbor

50% METEOROLOGY

| | | |
|----------|------------------------|----------------------------------|
| | Total EDE (rem) | Likelihood of Fatal Cancer |
| Location | | |
| Worker | 8.7 x 10 ⁻⁴ | 3.5 x 10 ⁻⁷ |

| | | |
|--|----------------------|---------------------------------|
| MCW | N/A | N/A |
| NPA | 2.2×10^{-4} | 1.1×10^{-7} |
| MOI | 2.2×10^{-4} | 1.1×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 817385 | 9.0×10^{-1} | 4.5×10^{-4} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 2.4×10^{-3} | 9.6×10^{-7} |
| MCW | N/A | N/A |
| NPA | 7.2×10^{-3} | 3.6×10^{-6} |
| MOI | 8.7×10^{-4} | 4.3×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 817385 | 2.4 | 1.2×10^{-3} |

Table F.1.4.2.1.5-6. Summary of Exposure Calculation Results.
For Wet Storage - HEPA Filter Fire
At Norfolk

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 2.1×10^{-4} | 8.5×10^{-8} |
| MCW | N/A | N/A |
| NPA | 5.3×10^{-5} | 2.7×10^{-8} |
| MOI | 3.2×10^{-4} | 1.6×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 1539002 | 2.3×10^{-1} | 1.2×10^{-4} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 2.4×10^{-3} | 9.6×10^{-7} |
| MCW | N/A | N/A |
| NPA | 6.2×10^{-4} | 3.1×10^{-7} |
| MOI | 3.3×10^{-3} | 1.7×10^{-6} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 1539002 | 1.4 | 6.9×10^{-4} |

Table F.1.4.2.1.5-7. Summary of Exposure Calculation Results.
For Wet Storage - HEPA Filter Fire
At Portsmouth

50% METEOROLOGY

| Total EDE | Likelihood of Fatal |
|-----------|------------------------|
|-----------|------------------------|

| Location | (rem) | Cancer |
|---|----------------------|-------------------------|
| Worker | 2.1×10^{-4} | 8.4×10^{-8} |
| MCW | N/A | N/A |
| NPA | 5.0×10^{-5} | 2.5×10^{-8} |
| MOI | 1.4×10^{-4} | 7.2×10^{-8} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Cancers |
| Population of 2432627 | 1.2×10^{-1} | 6.0×10^{-5} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|----------------------|----------------------------|
| Worker | 2.4×10^{-3} | 9.6×10^{-7} |
| MCW | N/A | N/A |
| NPA | 1.1×10^{-3} | 5.6×10^{-7} |
| MOI | 1.7×10^{-3} | 8.7×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Cancers |
| Population of 2432627 | 7.9×10^{-1} | 3.9×10^{-4} |

Table F.1.4.2.1.5-8. Summary of Exposure Calculation Results.
For Wet Storage - HEPA Filter Fire
At Kesselring

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|----------------------|----------------------------|
| Worker | 2.1×10^{-4} | 8.5×10^{-8} |
| MCW | N/A | N/A |
| NPA | N/A | N/A |
| MOI | 5.5×10^{-5} | 2.7×10^{-8} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Cancers |
| Population of 1148587 | 2.0×10^{-1} | 9.8×10^{-5} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|----------------------|----------------------------|
| Worker | 2.4×10^{-3} | 9.6×10^{-7} |
| MCW | N/A | N/A |
| NPA | N/A | N/A |
| MOI | 3.5×10^{-4} | 1.8×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Cancers |
| Population of 1148587 | 6.7×10^{-1} | 3.3×10^{-4} |

Table F.1.4.2.1.5-9. Summary of Exposure Calculation Results.
For Wet Storage - HEPA Filter Fire
At Nevada Test Site

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 1.4×10^{-4} | 5.5×10^{-8} |
| MCW | 1.1×10^{-7} | 4.2×10^{-11} |
| NPA | N/A | N/A |
| MOI | 8.5×10^{-6} | 4.2×10^{-9} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 13792 | 7.6×10^{-3} | 3.8×10^{-6} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 2.4×10^{-3} | 9.6×10^{-7} |
| MCW | 6.2×10^{-6} | 2.5×10^{-9} |
| NPA | N/A | N/A |
| MOI | 4.3×10^{-5} | 2.2×10^{-8} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 13792 | 1.1×10^{-2} | 5.7×10^{-6} |

Table F.1.4.2.1.5-10. Summary of Exposure Calculation Results.
For Wet Storage - HEPA Filter Fire
At Oak Ridge

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 8.7×10^{-4} | 3.5×10^{-7} |
| MCW | 2.3×10^{-5} | 8.8×10^{-9} |
| NPA | 3.0×10^{-4} | 1.5×10^{-7} |
| MOI | 9.0×10^{-4} | 4.5×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 871531 | 1.2×10^{-1} | 6.0×10^{-5} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 2.4×10^{-3} | 9.6×10^{-7} |
| MCW | 1.4×10^{-4} | 5.6×10^{-8} |
| NPA | 1.9×10^{-3} | 9.4×10^{-7} |
| MOI | 5.7×10^{-3} | 2.9×10^{-6} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 871531 | 4.3×10^{-1} | 2.2×10^{-4} |

F.1.4.2.1.6 Minor Water Pool Leakage.

F.1.4.2.1.6.1 Description of Conditions. In this hypothetical accident scenario, a minor

leak develops in the water pool resulting in a gradual discharge to the environment uncovering any spent nuclear fuel in the water pool, since the leak is so small the water level is maintained in the water pool. Since a strict accounting of water in the water pool is maintained, the magnitude of this leak would be less than 4,400 g 4,400 gallons per year value is the maximum amount of water which might leak out of before periodic review of the water balance would detect a leak.

F.1.4.2.1.6.2 Source Term. There is no airborne release above normal levels in this

hypothetical accident scenario. The radionuclide inventory in the leaking water is analysis of ECF water pool water. The isotopes that were analyzed for but not detected were below the minimum detection limit.

| Nuclide | Sample Results (-Ci/mL) | 10CFR20 Effluent Limit (-Ci/mL) | Annual Limit (Ci/yr) |
|---------|----------------------------|---------------------------------------|-------------------------|
| H-3 | 2.0 x 10 ⁻⁴ | 1.0 x 10 ⁻³ | 3.3 |
| Mn-54 | 2.5 x 10 ⁻⁸ | 3.0 x 10 ⁻⁵ | 4.1 |
| Fe-55 | 1.0 x 10 ⁻⁸ * | 1.0 x 10 ⁻⁴ | 1.6 |
| Co-58 | 7.0 x 10 ⁻⁸ | 2.0 x 10 ⁻⁵ | 1.1 |
| Co-60 | 1.6 x 10 ⁻⁵ | 3.0 x 10 ⁻⁶ | 2.6 |
| Ni-63 | 2.3 x 10 ⁻⁷ | 1.0 x 10 ⁻⁴ | 3.8 |
| Sr-90 | 4.0 x 10 ⁻⁹ | 5.0 x 10 ⁻⁷ | 6.5 |
| Y-90 | 4.0 x 10 ⁻⁹ | 7.0 x 10 ⁻⁶ | 6.5 |
| I-129 | 4.0 x 10 ⁻⁷ * | 2.0 x 10 ⁻⁷ | 6.5 |
| Cs-137 | 4.2 x 10 ⁻⁸ | 1.0 x 10 ⁻⁶ | 6.9 |

* These radionuclides were not detected in the ECF water. The numbers quoted are the detection limit of the analysis.

It should be noted that the sample results for the water pool indicate that t below the Code of Federal Regulations limits for liquid effluent in 10CFR20 with th The level of I-129 used in the calculations was based on the minimum detection limi level exceeds the effluent limit; however, I-129 was not actually detected in the w has comparable water solubility to I-129 and exists in spent nuclear fuel at about higher than I-129, it is inferred from the detected level of Sr-90 that the actual the 10CFR20 effluent limit.

F.1.4.2.1.6.3 Results. The following table summarizes the public health risk to the general

population that might result from the hypothetical minor water pool leak at each of the 10 locations. The number of fatal cancers would be expected to occur over a 50-year period. "Risk" is defined as the number of cancers times the probability of occurrence. The probability of a leak developing at each location was estimated from the results of the leak frequency analysis.

| Minor Water Pool Leakage Summary | | | |
|----------------------------------|---|--|-----------------------|
| Site | Maximal- ly ex- posed off-site individu- al (MOI) (rem) | No. of fatal cancer if acci- dent occurs | Risk per ye |
| INEL | 2.5×10^{-9} | 1.3×10^{-8} | 1.3×10^{-9} |
| Savannah River | 7.9×10^{-10} | 1.3×10^{-9} | 1.3×10^{-10} |
| Hanford | 9.9×10^{-12} | 1.7×10^{-10} | 1.7×10^{-11} |
| Puget Sound | 3.2×10^{-10} | 4.2×10^{-9} | 4.2×10^{-10} |
| Pearl Harbor | 1.3×10^{-10} | 4.6×10^{-10} | 4.6×10^{-11} |
| Norfolk | 2.7×10^{-10} | 1.8×10^{-9} | 1.8×10^{-10} |
| Portsmouth | 1.3×10^{-10} | 1.4×10^{-9} | 1.4×10^{-10} |

| | | | |
|------------------|----------------------|----------------------|-----------------------|
| Kesselring | 6.0×10^{-9} | 8.5×10^{-9} | 8.5×10^{-10} |
| Nevada Test Site | 2.5×10^{-9} | 1.4×10^{-9} | 1.4×10^{-10} |
| Oak Ridge | 1.5×10^{-9} | 3.9×10^{-9} | 3.9×10^{-10} |

At all sites except the Nevada Test Site, this accident results in the lowest the wet storage accidents evaluated.

Table F.1.4.2.1.6-1. Summary of Exposure Calculation Results.

For Wet Storage - Minor Water Pool Leakage

At INEL

| Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|-----------------------|----------------------------------|
| Worker | N/A | N/A |
| MCW | 1.6×10^{-13} | 6.4×10^{-17} |
| NPA | 1.6×10^{-13} | 8.0×10^{-17} |
| MOI | 2.5×10^{-9} | 1.3×10^{-12} |

Exposure to Population within 50-mile Radius (person-rem) Number of Fatal Can-
cers

Popula-
tion of 2.6×10^{-5} 1.3×10^{-8}
115690

Table F.1.4.2.1.6-2. Summary of Exposure Calculation Results.

For Wet Storage - Minor Water Pool Leakage

At Savannah River

| Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|-----------------------|----------------------------------|
| Worker | N/A | N/A |
| MCW | 4.8×10^{-13} | 1.9×10^{-16} |
| NPA | 4.8×10^{-13} | 2.4×10^{-16} |
| MOI | 7.9×10^{-10} | 4.0×10^{-13} |

Exposure to Population within 50-mile Radius (person-rem) Number of Fatal Can-
cers

Popula-
tion of 2.5×10^{-6} 1.3×10^{-9}
579541

Table F.1.4.2.1.6-3. Summary of Exposure Calculation Results.

For Wet Storage - Minor Water Pool Leakage

At Hanford

| Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|-----------------------|----------------------------------|
| Worker | N/A | N/A |
| MCW | 8.3×10^{-15} | 3.3×10^{-18} |
| NPA | 8.3×10^{-15} | 4.2×10^{-18} |
| MOI | 9.9×10^{-12} | 5.0×10^{-15} |

Exposure to Population within 50-mile Radius (person-rem) Number of Fatal Can-
cers

Popula-
tion of 3.3×10^{-7} 1.7×10^{-10}
375860

Table F.1.4.2.1.6-4. Summary of Exposure Calculation Results.

For Wet Storage - Minor Water Pool Leakage

At Puget Sound

| Total EDE | Likelihood of Fatal |
|-----------|------------------------|
|-----------|------------------------|

| | | |
|---------------|-----------------------|-----------------------|
| Loca- tion | (rem) | Cancer |
| Worker | N/A | N/A |
| MCW | N/A | N/A |
| NPA | 1.2×10^{-11} | 6.0×10^{-15} |
| MOI | 3.2×10^{-10} | 1.6×10^{-13} |

| | |
|--|---------------------------------|
| Exposure to Population within 50-mile Radius (person-rem) | Number of Fatal Can- cers |
|--|---------------------------------|

| | | |
|-------------------------------|----------------------|----------------------|
| Popula- tion of 2975810 | 8.4×10^{-6} | 4.2×10^{-9} |
|-------------------------------|----------------------|----------------------|

Table F.1.4.2.1.6-5. Summary of Exposure Calculation Results.
For Wet Storage - Minor Water Pool Leakage
At Pearl Harbor

| | | |
|---------------|-----------------------|----------------------------------|
| | Total EDE (rem) | Likelihood of Fatal Cancer |
| Loca- tion | | |
| Worker | N/A | N/A |
| MCW | N/A | N/A |
| NPA | 4.8×10^{-12} | 2.4×10^{-15} |
| MOI | 1.3×10^{-10} | 6.5×10^{-14} |

| | |
|--|---------------------------------|
| Exposure to Population within 50-mile Radius (person-rem) | Number of Fatal Can- cers |
|--|---------------------------------|

| | | |
|------------------------------|----------------------|-----------------------|
| Popula- tion of 817385 | 9.2×10^{-7} | 4.6×10^{-10} |
|------------------------------|----------------------|-----------------------|

Table F.1.4.2.1.6-6. Summary of Exposure Calculation Results.
For Wet Storage - Minor Water Pool Leakage
At Norfolk

| | | |
|---------------|-----------------------|----------------------------------|
| | Total EDE (rem) | Likelihood of Fatal Cancer |
| Loca- tion | | |
| Worker | N/A | N/A |
| MCW | N/A | N/A |
| NPA | 9.9×10^{-12} | 5.0×10^{-15} |
| MOI | 2.7×10^{-10} | 1.4×10^{-13} |

| | |
|--|---------------------------------|
| Exposure to Population within 50-mile Radius (person-rem) | Number of Fatal Can- cers |
|--|---------------------------------|

| | | |
|-------------------------------|----------------------|----------------------|
| Popula- tion of 1539002 | 3.6×10^{-6} | 1.8×10^{-9} |
|-------------------------------|----------------------|----------------------|

Table F.1.4.2.1.6-7. Summary of Exposure Calculation Results.
For Wet Storage - Minor Water Pool Leakage
At Portsmouth

| | | |
|---------------|-----------------------|----------------------------------|
| | Total EDE (rem) | Likelihood of Fatal Cancer |
| Loca- tion | | |
| Worker | N/A | N/A |
| MCW | N/A | N/A |
| NPA | 4.8×10^{-12} | 2.4×10^{-15} |
| MOI | 1.3×10^{-10} | 6.5×10^{-14} |

| | |
|--|---------------------------------|
| Exposure to Population within 50-mile Radius (person-rem) | Number of Fatal Can- cers |
|--|---------------------------------|

Popula-
tion of
2432627

2.7 x 10⁻⁶ 1.4 x 10⁻⁹

Table F.1.4.2.1.6-8. Summary of Exposure Calculation Results.
For Wet Storage - Minor Water Pool Leakage
At Kesselring

| Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|------------------------|----------------------------------|
| Worker | N/A | N/A |
| MCW | N/A | N/A |
| NPA | N/A | N/A |
| MOI | 6.0 x 10 ⁻⁹ | 3.0 x 10 ⁻¹² |

Exposure to Population within 50-mile Radius (person-rem) Number of
Fatal Can-
cers

Popula-
tion of
1148587

1.7 x 10⁻⁵ 8.5 x 10⁻⁹

Table F.1.4.2.1.6-9. Summary of Exposure Calculation Results.
For Wet Storage - Minor Water Pool Leakage
At Nevada Test Site

| Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|-------------------------|----------------------------------|
| Worker | N/A | N/A |
| MCW | 1.6 x 10 ⁻¹³ | 6.4 x 10 ⁻¹⁷ |
| NPA | 1.6 x 10 ⁻¹³ | 8.0 x 10 ⁻¹⁷ |
| MOI | 2.5 x 10 ⁻⁹ | 1.3 x 10 ⁻¹² |

Exposure to Population within 50-mile Radius (person-rem) Number of
Fatal Can-
cers

Popula-
tion of
13792

2.7 x 10⁻⁶ 1.4 x 10⁻⁹

Table F.1.4.2.1.6-10. Summary of Exposure Calculation Results.
For Wet Storage - Minor Water Pool Leakage
At Oak Ridge

| Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|---------------|-------------------------|----------------------------------|
| Worker | N/A | N/A |
| MCW | 9.4 x 10 ⁻¹³ | 3.8 x 10 ⁻¹⁶ |
| NPA | 9.4 x 10 ⁻¹³ | 4.7 x 10 ⁻¹⁶ |
| MOI | 1.5 x 10 ⁻⁹ | 7.5 x 10 ⁻¹³ |

Exposure to Population within 50-mile Radius (person-rem) Number of
Fatal Can-
cers

Popula-
tion of
871531

7.7 x 10⁻⁶ 3.9 x 10⁻⁹

F.1.4.2.2 Dry Storage.

F.1.4.2.2.1 Wind-driven Missile Impact into Storage Casks with Mechanical

Damage.

F.1.4.2.2.1.1 Description of Conditions. In this hypothetical accident, no fuel damage

would result from any impact because of the strength of the containers used. Dry s experience a major wind storm or tornado which could propel a large object into a s causing the container seal to be breached. However, container analysis for this si container is strong enough to prevent crushing of the spent nuclear fuel and releas

Winds produced by tornados are higher than hurricane winds and thus the impac would be travelling with higher velocity and would have higher kinetic energy. Eve velocity, analysis has shown that the missile would not penetrate the container. T penetration at the lower velocity of a hurricane (212 miles per hour) would be even probability of penetration for a missile propelled by the winds of a tornado (trave hurricanes can have high winds, hurricane winds normally cannot generate the very l missiles analyzed for tornados. While hurricanes may occur more frequently than to risk from a hurricane is lower because the container would not be penetrated.

The analysis of wind damage using missiles propelled by the winds of tornados done for design of nuclear power plants. Hurricanes very infrequently have winds t such missiles, so the analyses provided for tornados provide an upper limit for the Examination of damage caused by recent severe hurricanes shows that robust structur hurricanes.

F.1.4.2.2.1.2 Source Term. Conditions used in developing the source term are as follows:

- The source term is based on best estimate spent nuclear fuel corrosion p
- One percent of the original corrosion products associated with the fuel from the cask to the atmosphere. This is based on experimental measurem fraction of corrosion products loosened from naval spent nuclear fuel by vibration and the fact that a wind-driven missile would not penetrate th damage the fuel inside. Only loose corrosion products would be avail- able for release from the container, and any release from the container would have to occ convoluted path through the damaged seal.
- The release to the environment occurs at a constant rate over a 15-minut
- There is no increase in direct radiation due to this accident.
- The following amounts of radionuclides could be released to the environ- ment. This listing includes nuclides that result in at least 99% of the possible ex

| Nuclide | Curies |
|---------|-----------------------|
| Co-60 | 9.58×10^{-2} |
| Fe-55 | 1.76×10^{-1} |
| Co-58 | 3.54×10^{-2} |
| Mn-54 | 5.98×10^{-3} |
| Fe-59 | 5.11×10^{-4} |

F.1.4.2.2.1.3 Results. The following table summarizes the public health risk to the general

population that would result from the hypothetical wind-driven missile accident at number of fatal cancers would be expected to occur over a 50-year period. "Risk" i number of fatal cancers times the probability of occurrence. The probability of co due to the very strong container design. The dry storage containers are expected t shipping containers so that they would not be penetrated by environmentally caused would not be affected. However, an analysis was performed for a case in which the missile might topple a container on a railcar and cause unseating of the container radioactive material in the form of corrosion products.

The probability of the occurrence of a tornado was obtained using the data in WASH-1300 (AEC 1974). The maximum likelihood of a tornado occurrence at all storag

being evaluated in the continental United States is 10^{-3} per year. The probability the tornado striking a container and causing the damage analyzed has been estimated. Thus, the total probability of a wind-driven missile damaging a container is less than the probability of 10^{-5} per year was used in the risk assessment.

Dry Storage Mechanical Damage Summary

| Site | Maximal-ly exposed off-site individual (MOI) (rem) | No. of fatal cancer if accident occurs | Risk per year |
|------------------|--|--|-----------------------|
| INEL | 4.6×10^{-4} | 4.9×10^{-4} | 4.9×10^{-9} |
| Savannah River | 4.9×10^{-4} | 3.0×10^{-3} | 3.0×10^{-8} |
| Hanford | 1.7×10^{-4} | 1.3×10^{-3} | 1.3×10^{-8} |
| Puget Sound | 3.9×10^{-2} | 1.7×10^{-2} | 1.7×10^{-7} |
| Pearl Harbor | 2.1×10^{-2} | 3.0×10^{-2} | 3.0×10^{-7} |
| Norfolk | 8.1×10^{-2} | 1.8×10^{-2} | 1.8×10^{-7} |
| Portsmouth | 4.2×10^{-2} | 1.0×10^{-2} | 1.0×10^{-7} |
| Kesselring | 8.1×10^{-3} | 7.4×10^{-3} | 7.4×10^{-8} |
| Nevada Test Site | 8.8×10^{-4} | 5.3×10^{-5} | 5.3×10^{-10} |
| Oak Ridge | 1.4×10^{-1} | 5.1×10^{-3} | 5.1×10^{-8} |

The risk for this hypothetical accident is generally more severe at Navy ship sites. This accident results in the lowest risk of the two dry storage accidents evaluated.

For the hypothetical wind-driven missile accident scenario, the radioactive contamination of the ground to a downwind distance of less than 0.06 mile. This was not impacted by the accident of less than 0.5 acre. The calculated downwind distance was within the boundaries of all sites under evaluation.

Table F.1.4.2.2.1-1. Summary of Exposure Calculation Results.

For Dry Storage - Mechanical Damage
At INEL

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|----------------------|----------------------------|
| Worker | 2.0×10^{-2} | 8.0×10^{-6} |
| MCW | 1.8×10^{-5} | 9.2×10^{-9} |
| NPA | 1.0×10^{-5} | 5.2×10^{-9} |
| MOI | 8.0×10^{-5} | 4.0×10^{-8} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Cancers |
| Population of 115690 | 2.3×10^{-1} | 1.2×10^{-4} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|----------------------|----------------------------|
| Worker | 5.6×10^{-2} | 2.2×10^{-5} |
| MCW | 2.0×10^{-4} | 1.0×10^{-7} |
| NPA | 6.3×10^{-5} | 3.1×10^{-8} |
| MOI | 4.6×10^{-4} | 2.3×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Cancers |
| Population of 115690 | 9.8×10^{-1} | 4.9×10^{-4} |

Table F.1.4.2.2.1-2. Summary of Exposure Calculation Results.

For Dry Storage - Mechanical Damage
At Savannah River

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 8.9×10^{-3} | 3.6×10^{-6} |
| MCW | 5.3×10^{-4} | 2.1×10^{-7} |
| NPA | 6.7×10^{-6} | 3.4×10^{-9} |
| MOI (New ECF) | 1.6×10^{-4} | 8.1×10^{-8} |
| MOI (Barnwell) | 4.0×10^{-4} | 2.0×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 579541 | 9.4×10^{-1} | 4.7×10^{-4} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 5.6×10^{-2} | 2.2×10^{-5} |
| MCW | 6.7×10^{-3} | 2.6×10^{-6} |
| NPA | 1.1×10^{-4} | 5.7×10^{-8} |
| MOI (New ECF) | 4.9×10^{-4} | 2.5×10^{-7} |
| MOI (Barnwell) | 3.9×10^{-3} | 2.0×10^{-6} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 579541 | 6.1 | 3.0×10^{-3} |

Table F.1.4.2.2.1-3. Summary of Exposure Calculation Results.
For Dry Storage - Mechanical Damage
At Hanford

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 8.9×10^{-3} | 3.6×10^{-6} |
| MCW | 7.0×10^{-4} | 2.8×10^{-7} |
| NPA | 8.1×10^{-6} | 4.1×10^{-9} |
| MOI (New ECF) | 2.3×10^{-5} | 1.1×10^{-8} |
| MOI (FMEF) | 4.6×10^{-5} | 2.3×10^{-8} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 375860 | 1.4×10^{-1} | 7.0×10^{-5} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 5.6×10^{-2} | 2.2×10^{-5} |
| MCW | 4.4×10^{-3} | 1.8×10^{-6} |
| NPA | 1.3×10^{-4} | 6.3×10^{-8} |
| MOI (New ECF) | 1.7×10^{-4} | 8.4×10^{-8} |
| MOI (FMEF) | 5.9×10^{-4} | 2.9×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- |

Population of
375860 2.5 1.3×10^{-3}
Table F.1.4.2.2.1-4. Summary of Exposure Calculation Results.
For Dry Storage - Mechanical Damage
At Puget Sound

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 4.9×10^{-3} | 1.9×10^{-6} |
| MCW | N/A | N/A |
| NPA | 5.7×10^{-3} | 2.9×10^{-6} |
| MOI | 3.5×10^{-3} | 1.7×10^{-6} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 2975810 | 1.2×10^1 | 5.8×10^{-3} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 5.6×10^{-2} | 2.2×10^{-5} |
| MCW | N/A | N/A |
| NPA | 6.8×10^{-2} | 3.4×10^{-5} |
| MOI | 3.9×10^{-2} | 1.9×10^{-5} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 2975810 | 3.4×10^1 | 1.7×10^{-2} |

Table F.1.4.2.2.1-5. Summary of Exposure Calculation Results.
For Dry Storage - Mechanical Damage
At Pearl Harbor

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 2.0×10^{-2} | 8.0×10^{-6} |
| MCW | N/A | N/A |
| NPA | 5.2×10^{-3} | 2.6×10^{-6} |
| MOI | 5.3×10^{-3} | 2.7×10^{-6} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 817385 | 2.2×10^1 | 1.1×10^{-2} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|----------|----------------------|----------------------------------|
| Worker | 5.6×10^{-2} | 2.2×10^{-5} |
| MCW | N/A | N/A |
| NPA | 1.7×10^{-1} | 8.4×10^{-5} |
| MOI | 2.1×10^{-2} | 1.1×10^{-5} |

Exposure to Population within 50-mile Radius (person-rem) Number of Fatal Can-
cancers

Population of 817385 5.9 x 10¹ 3.0 x 10⁻²

Table F.1.4.2.2.1-6. Summary of Exposure Calculation Results.
For Dry Storage - Mechanical Damage
At Norfolk

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|----------|------------------------|----------------------------|
| Worker | 4.9 x 10 ⁻³ | 2.0 x 10 ⁻⁶ |
| MCW | N/A | N/A |
| NPA | 1.2 x 10 ⁻³ | 6.2 x 10 ⁻⁷ |
| MOI | 7.8 x 10 ⁻³ | 3.9 x 10 ⁻⁶ |

Exposure to Population within 50-mile Radius (person-rem) Number of Fatal Can-
cancers

Population of 1539002 7.4 3.7 x 10⁻³

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|----------|------------------------|----------------------------|
| Worker | 5.6 x 10 ⁻² | 2.2 x 10 ⁻⁵ |
| MCW | N/A | N/A |
| NPA | 1.4 x 10 ⁻² | 7.1 x 10 ⁻⁶ |
| MOI | 8.1 x 10 ⁻² | 4.0 x 10 ⁻⁵ |

Exposure to Population within 50-mile Radius (person-rem) Number of Fatal Can-
cancers

Population of 1539002 3.5 x 10¹ 1.8 x 10⁻²

Table F.1.4.2.2.1-7. Summary of Exposure Calculation Results.
For Dry Storage - Mechanical Damage
At Portsmouth

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|----------|------------------------|----------------------------|
| Worker | 4.9 x 10 ⁻³ | 1.9 x 10 ⁻⁶ |
| MCW | N/A | N/A |
| NPA | 1.2 x 10 ⁻³ | 5.8 x 10 ⁻⁷ |
| MOI | 3.5 x 10 ⁻³ | 1.8 x 10 ⁻⁶ |

Exposure to Population within 50-mile Radius (person-rem) Number of Fatal Can-
cancers

Population of 2432627 4.2 2.1 x 10⁻³

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|----------|------------------------|----------------------------|
| Worker | 5.6 x 10 ⁻² | 2.2 x 10 ⁻⁵ |
| MCW | N/A | N/A |

| | | |
|---|-------------------------|------------------------|
| NPA | 2.6 x 10 ⁻² | 1.3 x 10 ⁻⁵ |
| MOI | 4.2 x 10 ⁻² | 2.1 x 10 ⁻⁵ |
| Exposure to Population within 50-mile Radius (person-rem) | Number of Fatal Cancers | |

| | | |
|-----------------------|-----------------------|------------------------|
| Population of 2432627 | 2.0 x 10 ¹ | 1.0 x 10 ⁻² |
|-----------------------|-----------------------|------------------------|

Table F.1.4.2.2.1-8. Summary of Exposure Calculation Results.
For Dry Storage - Mechanical Damage
At Kesselring

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|-------------------------|----------------------------|
| Worker | 4.9 x 10 ⁻³ | 2.0 x 10 ⁻⁶ |
| MCW | N/A | N/A |
| NPA | N/A | N/A |
| MOI | 8.8 x 10 ⁻⁴ | 4.4 x 10 ⁻⁷ |
| Exposure to Population within 50-mile Radius (person-rem) | Number of Fatal Cancers | |
| Population of 1148587 | 3.3 | 1.7 x 10 ⁻³ |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|-------------------------|----------------------------|
| Worker | 5.6 x 10 ⁻² | 2.2 x 10 ⁻⁵ |
| MCW | N/A | N/A |
| NPA | N/A | N/A |
| MOI | 8.1 x 10 ⁻³ | 4.0 x 10 ⁻⁶ |
| Exposure to Population within 50-mile Radius (person-rem) | Number of Fatal Cancers | |
| Population of 1148587 | 1.5 x 10 ¹ | 7.4 x 10 ⁻³ |

Table F.1.4.2.2.1-9. Summary of Exposure Calculation Results.
For Dry Storage - Mechanical Damage
At Nevada Test Site

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|-------------------------|----------------------------|
| Worker | 3.2 x 10 ⁻³ | 1.3 x 10 ⁻⁶ |
| MCW | 2.5 x 10 ⁻⁶ | 9.6 x 10 ⁻¹⁰ |
| NPA | N/A | N/A |
| MOI | 4.5 x 10 ⁻⁵ | 2.2 x 10 ⁻⁸ |
| Exposure to Population within 50-mile Radius (person-rem) | Number of Fatal Cancers | |
| Population of 13792 | 1.5 x 10 ⁻² | 7.3 x 10 ⁻⁶ |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|----------|-----------------|----------------------------|
|----------|-----------------|----------------------------|

| | | |
|--|---------------------------------|----------------------|
| Worker | 5.6×10^{-2} | 2.2×10^{-5} |
| MCW | 1.4×10^{-4} | 5.8×10^{-8} |
| NPA | N/A | N/A |
| MOI | 8.8×10^{-4} | 4.4×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | Number of Fatal Can- cers | |

Population of

13792 1.1×10^{-1} 5.3×10^{-5}

Table F.1.4.2.2.1-10. Summary of Exposure Calculation Results.
For Dry Storage - Mechanical Damage
At Oak Ridge

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|---------------------------------|----------------------------------|
| Worker | 2.0×10^{-2} | 8.0×10^{-6} |
| MCW | 5.3×10^{-4} | 2.1×10^{-7} |
| NPA | 6.9×10^{-3} | 3.4×10^{-6} |
| MOI | 2.2×10^{-2} | 1.1×10^{-5} |
| Exposure to Population within 50-mile Radius (person-rem) | Number of Fatal Can- cers | |
| Population of 871531 | 2.8 | 1.4×10^{-3} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|---------------------------------|----------------------------------|
| Worker | 5.6×10^{-2} | 2.2×10^{-5} |
| MCW | 3.2×10^{-3} | 1.3×10^{-6} |
| NPA | 4.4×10^{-2} | 2.2×10^{-5} |
| MOI | 1.4×10^{-1} | 6.9×10^{-5} |
| Exposure to Population within 50-mile Radius (person-rem) | Number of Fatal Can- cers | |
| Population of 871531 | 1.0×10^1 | 5.1×10^{-3} |

F.1.4.2.2.2 Airplane Crash.

F.1.4.2.2.2.1 Description of Conditions. A hypothetical aircraft accident scenario was

developed for the dry storage option. Based on the probability of occurrence, as d specific analyses were only performed for Savannah River, Oak Ridge, Pearl Harbor, Portsmouth, and Kesselring locations. At other locations, the likelihood of occur year. The accident is postulated to cause damage to a single storage cask. This i containers used to store naval spent nuclear fuel would be very rugged so that only one of an airliner's jet engines would be strong enough and possess enough energy t penetrating a container. From analyses of existing container designs, the rotor of including those from the largest aircraft such as a Boeing 777, Russian Antonov An-C-5, would not penetrate a container during an airliner crash, but, for the purpose calculations were performed for one container damaged to the extent that fission pr products might be released. Due to the severity of the shock, the cask seal might damage to the fuel. The severe mechanical shock results in the release of corrosio environment. The release of fission products also occurs due to the impact and res product release factors are based on overheating testing performed on the naval fue

F.1.4.2.2.2.2 Source Term. Conditions used in developing the source term are as follows:

- One percent of all of the fuel units stored inside the cask are damaged or the resultant fire and those fission products are available for release
- Of the available fission products, 100% of the noble gases, 3% of the cesium, and 0.1% of the remaining solids are released to the environment
- The release to the environment occurs at a constant rate over a 15-minute
- Ten percent of the original corrosion products from the fuel units are released from the cask to the atmosphere.
- The following amount of radionuclides could be released to the environment. This listing includes nuclides that result in at least 99% of the possible exposure.

| Nuclide | Curies |
|---------|-----------------------|
| Cs-134 | 2.57×10^1 |
| Cs-137 | 3.56×10^1 |
| Pu-238 | 5.90×10^{-2} |
| Ba-137M | 3.07 |
| Sr-90 | 3.12 |
| Ce-144 | 7.17 |
| Nb-95 | 4.37 |
| Y-90 | 3.12 |
| Ru-106 | 6.11×10^{-1} |

F.1.4.2.2.3 Results. The following table summarizes the public health risk to the general

population that would result from the hypothetical airplane crash accident at each site. Fatal cancers would be expected to occur over a 50-year period. "Risk" is defined as the number of fatal cancers times the probability of occurrence.

| Dry Storage Airplane Crash Summary | | | | |
|------------------------------------|----------------------------------|--|---|---|
| Site | Probability of accident per year | Maximal-ly exposed off-site individual (MOI) (rem) | No. of fatal cancers if accident occurs | R |
| Savannah River | 3×10^{-7} | 4.7×10^{-1} | 2.8 | 8 |
| Pearl Harbor | 1×10^{-5} | 19 | 26 | 2 |
| Norfolk | 1×10^{-6} | 72 | 16 | 1 |
| Portsmouth | 1×10^{-7} | 38 | 9.0 | 9 |
| Kesselring | 1×10^{-7} | 7.7 | 7.5 | 7 |
| Oak Ridge | 3×10^{-7} | 120 | 4.7 | 1 |

The risk for this hypothetical accident is most severe at Pearl Harbor and Norfolk. The highest risk for any hypothetical accident evaluated at Pearl Harbor and Norfolk. Probabilities less than 10^{-7} per year, consequences were not calculated since it is not substantially contribute to the risk.

For the hypothetical airplane crash into a dry storage cask accident scenario might cause contamination of the ground to a downwind distance of approximately 0.9 miles. This would yield a total area impacted by the accident of about 106 acres. The calculated dose would be contained within the boundaries of the Savannah River and Kesselring sites. The dose would extend beyond the boundaries of Oak Ridge and the shipyards that are at risk. Table F.1.4.2.2.2-1. Summary of Exposure Calculation Results.

Dry Storage - Airplane Crash
At Savannah River

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|----------------------|----------------------------------|
| Worker | 1.5×10^1 | 5.9×10^{-3} |
| MCW | 8.7×10^{-1} | 3.5×10^{-4} |
| NPA | 1.1×10^{-2} | 5.5×10^{-6} |
| MOI | 1.8×10^{-1} | 8.8×10^{-5} |
| Exposure to Population within 50-mile Radius (person-rem) Number of Fatal Can- cers | | |
| Population of 579541 | 9.6×10^2 | 4.8×10^{-1} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|----------------------|----------------------------------|
| Worker | 9.2×10^1 | 7.4×10^{-2} |
| MCW | 1.1×10^1 | 4.4×10^{-3} |
| NPA | 1.9×10^{-1} | 9.5×10^{-5} |
| MOI | 4.7×10^{-1} | 2.3×10^{-4} |
| Exposure to Population within 50-mile Radius (person-rem) Number of Fatal Can- cers | | |
| Population of 579541 | 5.5×10^3 | 2.8 |

Table F.1.4.2.2.2-2. Summary of Exposure Calculation Results.
Dry Storage - Airplane Crash
At Pearl Harbor

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|--------------------|----------------------------------|
| Worker | 3.3×10^1 | 2.7×10^{-2} |
| MCW | N/A | N/A |
| NPA | 8.6 | 4.3×10^{-3} |
| MOI | 4.7 | 2.3×10^{-3} |
| Exposure to Population within 50-mile Radius (person-rem) Number of Fatal Can- cers | | |
| Population of 817385 | 2.0×10^4 | 9.8 |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|--------------------|----------------------------------|
| Worker | 9.2×10^1 | 7.4×10^{-2} |
| MCW | N/A | N/A |
| NPA | 2.8×10^2 | 2.8×10^{-1} |
| MOI | 1.9×10^1 | 9.3×10^{-3} |
| Exposure to Population within 50-mile Radius (person-rem) Number of Fatal Can- cers | | |
| Population of 817385 | 5.2×10^4 | 2.6×10^1 |

Table F.1.4.2.2.2-3. Summary of Exposure Calculation Results.
Dry Storage - Airplane Crash
At Norfolk

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|--------------------|----------------------------------|
| Worker | 8.2 | 3.3×10^{-3} |
| MCW | N/A | N/A |
| NPA | 2.0 | 1.0×10^{-3} |
| MOI | 6.9 | 3.4×10^{-3} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 1539002 | 6.5×10^3 | 3.2 |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|--------------------|----------------------------------|
| Worker | 9.2×10^1 | 7.4×10^{-2} |
| MCW | N/A | N/A |
| NPA | 2.4×10^1 | 2.4×10^{-2} |
| MOI | 7.2×10^1 | 7.2×10^{-2} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 1539002 | 3.1×10^4 | 1.6×10^1 |

Table F.1.4.2.2.2-4. Summary of Exposure Calculation Results.
Dry Storage - Airplane Crash
At Portsmouth

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|--------------------|----------------------------------|
| Worker | 8.1 | 3.2×10^{-3} |
| MCW | N/A | N/A |
| NPA | 1.9 | 9.6×10^{-4} |
| MOI | 3.1 | 1.6×10^{-3} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 2432627 | 3.7×10^3 | 1.9 |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|--------------------|----------------------------------|
| Worker | 9.2×10^1 | 7.4×10^{-2} |
| MCW | N/A | N/A |
| NPA | 4.3×10^1 | 4.3×10^{-2} |
| MOI | 3.8×10^1 | 3.8×10^{-2} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 2432627 | 1.8×10^4 | 9.0 |

Table F.1.4.2.2.2-5. Summary of Exposure Calculation Results.

Dry Storage - Airplane Crash
At Kesselring

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|--------------------|----------------------------------|
| Worker | 8.2 | 3.3×10^{-3} |
| MCW | N/A | N/A |
| NPA | N/A | N/A |
| MOI | 1.3 | 6.6×10^{-4} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 1148587 | 4.8×10^3 | 2.4 |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|--------------------|----------------------------------|
| Worker | 9.2×10^1 | 7.4×10^{-2} |
| MCW | N/A | N/A |
| NPA | N/A | N/A |
| MOI | 7.7 | 3.8×10^{-3} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 1148587 | 1.5×10^4 | 7.5 |

Table F.1.4.2.2.2-6. Summary of Exposure Calculation Results.
Dry Storage - Airplane Crash
At Oak Ridge

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 3.3×10^1 | 2.7×10^{-2} |
| MCW | 8.7×10^{-1} | 3.5×10^{-4} |
| NPA | 1.1×10^1 | 5.7×10^{-3} |
| MOI | 1.9×10^1 | 9.7×10^{-3} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 871531 | 2.9×10^3 | 1.4 |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|--------------------|----------------------------------|
| Worker | 9.2×10^1 | 7.4×10^{-2} |
| MCW | 5.3 | 2.2×10^{-3} |
| NPA | 7.2×10^1 | 7.2×10^{-2} |
| MOI | 1.2×10^2 | 1.2×10^{-1} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of | | |

871531

9.5 x 10³ 4.7**F.1.4.2.3 Dry Cell Operations.****F.1.4.2.3.1 Inadvertent Cutting into Fuel Region or Mechanical Damage.****F.1.4.2.3.1.1 Description of Conditions. Mechanical damage due to handling during**

examination, such as accidentally cutting into the fuel region of an element, was a hypothetical accident results from inadvertent cutting across the fuel region when Zircaloy ends of a fuel unit. All noble gas isotopes within the vicinity of the fuel facility building and escape to the environment. The majority of the volatile and to be retained in the fuel or the facility exhaust filters. The resulting airborne was evaluated. The possible exposure to the workers, individuals living on the site, general population was evaluated.

F.1.4.2.3.1.2 Source Term. Conditions used in developing the source term are as follows:

- One percent of the fission products in the fuel element being handled at the cut site to be available for release.
- All (100%) of the noble gases available for release are released to the
- Twenty-five percent of the halogens available for release are released.
- One percent of the particulate fission products could be released and 99 removed by normally installed HEPA filters.
- Cs and Ru would behave like particulate fission products.
- The release to the environment occurs at a constant rate over a 15-minute
- There is no increase in direct radiation due to this accident.
- The following amounts of radionuclides could be released to the environment. This listing includes nuclides that result in at least 99% of the

| Nuclide | Curies |
|---------|------------------------|
| Pu-238 | 7.2 x 10 ⁻⁵ |
| Cs-134 | 2.9 x 10 ⁻³ |
| Cs-137 | 4 x 10 ⁻³ |
| I-129 | 2.5 x 10 ⁻⁵ |
| Sr-90 | 3.9 x 10 ⁻³ |
| Ce-144 | 9.0 x 10 ⁻³ |
| Nb-95 | 5.4 x 10 ⁻³ |
| I-131 | 5.4 x 10 ⁻⁴ |
| H-3 | 1.42 |
| Y-90 | 3.9 x 10 ⁻³ |
| Ba-137m | 3.8 x 10 ⁻³ |
| Ru-106 | 7.6 x 10 ⁻⁴ |
| Zr-95 | 2.9 x 10 ⁻³ |
| Y-91 | 2.3 x 10 ⁻³ |
| Eu-154 | 2.7 x 10 ⁻⁴ |

F.1.4.2.3.1.3 Results. The following table summarizes the public health risk to the general

population that would result from the hypothetical mechanical damage accident at each number of fatal cancers would be expected to occur over a 50-year period. "Risk" is number of fatal cancers times the probability of occurrence. The probability of damage handling is small. The work on fuel at the INEL-ECF includes removal of the non-fuel end of the fuel unit. This is done in a sawing operation. To cut into the fuel, there is positioning the spent fuel in the cutting apparatus and error in selecting the size

The combined operator and independent checker error probability for cutting of the evaluated to be less than 10^{-7} per cut (Swain and Guttman 1983). Using a conservative cut operations per year results in a fuel cutting probability of less than 10^{-4} per the risk evaluation.

Dry Cell Mechanical Damage Summary

| Site | Maximal-ly exposed off-site individual (MOI) (rem) | No. of fatal cancer if accident occurs | Risk per year |
|------------------|--|--|----------------------|
| INEL | 2.2×10^{-4} | 3.5×10^{-4} | 3.5×10^{-8} |
| Savannah River | 2.4×10^{-4} | 1.4×10^{-3} | 1.4×10^{-7} |
| Hanford | 7.1×10^{-5} | 5.3×10^{-4} | 5.3×10^{-8} |
| Nevada Test Site | 4.0×10^{-4} | 3.7×10^{-5} | 3.7×10^{-9} |
| Oak Ridge | 5.8×10^{-2} | 2.5×10^{-3} | 2.5×10^{-7} |

The risk for this hypothetical accident is roughly proportional to the surrounding Oak Ridge being the worst and the Nevada Test Site being the best.

For the hypothetical dry cell mechanical damage accident scenario, the radioactive result in contamination of the ground to a downwind distance of less than 0.06 mile total area impacted by the accident of less than 0.5 acre. The calculated downwind contained within the boundaries of all DOE sites under evaluation.

Table F.1.4.2.3.1-1. Summary of Exposure Calculation Results.

For Dry Cell Operations - Mechanical Damage
At INEL

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|----------------------|----------------------------|
| Worker | 3.7×10^{-2} | 1.5×10^{-5} |
| MCW | 3.4×10^{-5} | 1.4×10^{-8} |
| NPA | 1.9×10^{-5} | 9.5×10^{-9} |
| MOI | 6.2×10^{-5} | 3.1×10^{-8} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Cancers |
| Population of 115690 | 3.9×10^{-1} | 1.9×10^{-4} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|---|----------------------|----------------------------|
| Worker | 1.0×10^{-1} | 4.1×10^{-5} |
| MCW | 3.7×10^{-4} | 1.5×10^{-7} |
| NPA | 1.1×10^{-4} | 5.7×10^{-8} |
| MOI | 2.2×10^{-4} | 1.1×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Cancers |
| Population of 115690 | 7.0×10^{-1} | 3.5×10^{-4} |

Table F.1.4.2.3.1-2. Summary of Exposure Calculation Results.

For Dry Cell Operations - Mechanical Damage
At Savannah River

50% METEOROLOGY

Likelihood

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 1.6×10^{-2} | 6.6×10^{-6} |
| MCW | 9.6×10^{-4} | 3.8×10^{-7} |
| NPA | 1.2×10^{-5} | 6.1×10^{-9} |
| MOI (New ECF) | 1.0×10^{-4} | 5.1×10^{-8} |
| MOI (Barnwell) | 2.0×10^{-4} | 1.0×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 579541 | 6.2×10^{-1} | 3.1×10^{-4} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 1.0×10^{-1} | 4.1×10^{-5} |
| MCW | 1.2×10^{-2} | 4.9×10^{-6} |
| NPA | 2.1×10^{-4} | 1.0×10^{-7} |
| MOI (New ECF) | 2.4×10^{-4} | 1.2×10^{-7} |
| MOI (Barnwell) | 1.7×10^{-3} | 8.4×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 579541 | 2.8 | 1.4×10^{-3} |

Table F.1.4.2.3.1-3. Summary of Exposure Calculation Results.
For Dry Cell Operations - Mechanical Damage
At Hanford

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 1.6×10^{-2} | 6.6×10^{-6} |
| MCW | 1.3×10^{-3} | 5.1×10^{-7} |
| NPA | 1.5×10^{-5} | 7.4×10^{-9} |
| MOI (New ECF) | 9.8×10^{-6} | 4.9×10^{-9} |
| MOI (FMEF) | 2.0×10^{-5} | 9.9×10^{-9} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 375860 | 6.2×10^{-2} | 3.1×10^{-5} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 1.0×10^{-1} | 4.1×10^{-5} |
| MCW | 8.0×10^{-3} | 3.2×10^{-6} |
| NPA | 2.3×10^{-4} | 1.2×10^{-7} |
| MOI (New ECF) | 7.1×10^{-5} | 3.6×10^{-8} |
| MOI (FMEF) | 2.5×10^{-4} | 1.2×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 375860 | 1.07 | 5.3×10^{-4} |

Table F.1.4.2.3.1-4. Summary of Exposure Calculation Results.

For Dry Cell Operations - Mechanical Damage
At Nevada Test Site

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 5.8×10^{-3} | 2.3×10^{-6} |
| MCW | 4.5×10^{-6} | 1.8×10^{-9} |
| NPA | N/A | N/A |
| MOI | 4.7×10^{-5} | 2.3×10^{-8} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 13792 | 3.6×10^{-2} | 1.8×10^{-5} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 1.0×10^{-1} | 4.1×10^{-5} |
| MCW | 2.6×10^{-4} | 1.0×10^{-7} |
| NPA | N/A | N/A |
| MOI | 4.0×10^{-4} | 2.0×10^{-7} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 13792 | 7.4×10^{-2} | 3.7×10^{-5} |

Table F.1.4.2.3.1-5. Summary of Exposure Calculation Results.
For Dry Cell Operations - Mechanical Damage
At Oak Ridge

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 3.7×10^{-2} | 1.5×10^{-5} |
| MCW | 9.6×10^{-4} | 3.8×10^{-7} |
| NPA | 1.3×10^{-2} | 6.3×10^{-6} |
| MOI | 9.3×10^{-3} | 4.6×10^{-6} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 871531 | 1.9 | 9.5×10^{-4} |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|----------------------|----------------------------------|
| Worker | 1.0×10^{-1} | 4.1×10^{-5} |
| MCW | 5.9×10^{-3} | 2.4×10^{-6} |
| NPA | 8.0×10^{-2} | 4.0×10^{-5} |
| MOI | 5.8×10^{-2} | 2.9×10^{-5} |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of | | |

871531 5.1 2.5×10^{-3}

F.1.4.2.3.2 Partial Loss of Shielding Due to Earthquake.

F.1.4.2.3.2.1 Description of Conditions. A hypothetical earthquake causes the proposed

Dry Cell Facility to lose some portion of its concrete shielding. Direct radiation work force and the general public has been calculated.

F.1.4.2.3.2.2 Source Term. The conditions used to calculate the dry cell direct radiation

levels are as follows:

- For calculational purposes, a total of 50% of the high-density concrete might be removed due to the earthquake. More realistic damage from an earthquake would result in cracks or small openings in the shielding. This bounds damage to the facility.
- Building containment and ventilation systems remain in operation. There is no airborne release to the environment. Calculations have already been performed per F.1.4.2.1.1 for a drained water pool hypothetical accident which bound airborne releases from the dry cell facility should the building containment systems fail.

F.1.4.2.3.2.3 Results. The following table summarizes the public health risk to the general

population that would result from the hypothetical loss of shielding accident at each of the sites. The number of fatal cancers would be expected to occur over a 50-year period. "Risk" is defined as the number of fatal cancers times the probability of occurrence. As discussed in Section F.1.4.2.1.1.3, the risk for the hypothetical accident is estimated to be 10^{-5} per year.

Dry Cell Partial Loss of Shielding Summary

| Site | Maximal-ly exposed off-site individual (MOI) (rem) | No. of fatal cancers if accident occurs | Risk per year |
|------------------|--|---|-----------------------|
| INEL | 9.3×10^{-17} | 3.0×10^{-19} | 3.0×10^{-24} |
| Savannah River | 6.7×10^{-15} | 3.0×10^{-16} | 3.0×10^{-21} |
| Hanford | 3.3×10^{-23} | 4.9×10^{-24} | 4.9×10^{-29} |
| Nevada Test Site | 6.3×10^{-11} | 3.7×10^{-37} | 3.7×10^{-42} |
| Oak Ridge | 1.2×10^{-2} | 7.5×10^{-6} | 7.5×10^{-11} |

At all sites, the risks associated with this accident are the lowest of any a Table F.1.4.2.3.2-1. Summary of Exposure Calculation Results. For Dry Cell Operations - Partial Loss of Shielding At INEL

| Receptor Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|-------------------|-----------------------|----------------------------|
| Worker | 7.2×10^{-5} | 2.9×10^{-8} |
| MCW | 7.5×10^{-13} | 3.0×10^{-16} |
| MOI | 9.3×10^{-17} | 4.9×10^{-20} |

| Exposure to Population within 50-mile Radius (person-rem) | Number of Fatal Cancers |
|---|-------------------------|
| Population of 115,690 | 5.9×10^{-16} |
| | 3.0×10^{-19} |

Table F.1.4.2.3.2-2. Summary of Exposure Calculation Results.
For Dry Cell Operations - Partial Loss of Shielding
At Savannah River

| Recep- tor Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|--------------------------------|-----------------------|----------------------------------|
| Worker | 7.2×10^{-5} | 2.9×10^{-8} |
| MCW | 2.7×10^{-6} | 1.1×10^{-9} |
| MOI (New ECF) | 6.7×10^{-15} | 3.4×10^{-18} |
| MOI (Barnwell Plant) | 2.4×10^{-6} | 1.2×10^{-9} |
| NPA | 7.9×10^{-17} | 4.0×10^{-20} |

Exposure to Population within 50-mile Radius (person-rem) Number of Fatal Cancers
Popula-
tion of 5.9×10^{13} 3.0×10^{-16}
579,541

Table F.1.4.2.3.2-3. Summary of Exposure Calculation Results.
For Dry Cell Operations - Partial Loss of Shielding
At Hanford

| Recep- tor Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|--------------------------------|-----------------------|----------------------------------|
| Worker | 7.2×10^{-5} | 2.9×10^{-8} |
| MCW | 2.7×10^{-6} | 1.1×10^{-9} |
| MOI (New ECF) | 3.3×10^{-23} | 1.7×10^{-26} |
| MOI (FMEF) | 6.7×10^{-15} | 3.4×10^{-18} |
| NPA | 3.9×10^{-25} | 2.0×10^{-28} |

Exposure to Population within 50-mile Radius (person-rem) Number of Fatal Cancers
Popula-
tion of 9.7×10^{21} 4.9×10^{-24}
375,860

Table F.1.4.2.3.2-4. Summary of Exposure Calculation Results.
For Dry Cell Operations - Partial Loss of Shielding
At Nevada Test Site

| Recep- tor Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|--------------------------------|-----------------------|----------------------------------|
| Worker | 7.2×10^{-5} | 2.9×10^{-8} |
| MCW | 7.1×10^{-15} | 2.8×10^{-18} |
| MOI | 6.3×10^{-11} | 3.2×10^{-14} |

Exposure to Population within 50-mile Radius (person-rem) Number of Fatal Can-
c
ers
Popula-
tion of 8.7×10^{33} 4.4×10^{-36}
12,159

Table F.1.4.2.3.2-5. Summary of Exposure Calculation Results.
For Dry Cell Operations - Partial Loss of Shielding
At Oak Ridge

| Recep- tor Loca- tion | Total EDE (rem) | Likelihood of Fatal Cancer |
|--------------------------------|--------------------|----------------------------------|
|--------------------------------|--------------------|----------------------------------|

| | | |
|--------|----------------------|-----------------------|
| Worker | 7.2×10^{-5} | 2.9×10^{-8} |
| MCW | 5.5×10^{-7} | 2.2×10^{-10} |
| MOI | 1.2×10^{-2} | 6.0×10^{-6} |
| NPA | 1.4×10^{-4} | 7.0×10^{-8} |

| | | |
|--|----------------------|---------------------------------|
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Popula- tion of 871,531 | 1.5×10^{-2} | 7.5×10^{-6} |

F.1.4.2.3.3 Airplane Crash Into Dry Cell Facility.

F.1.4.2.3.3.1 Description of Conditions. A hypothetical aircraft accident scenario was

developed for dry cell operations. Based on the probability of occurrence, as disc specific analysis was only performed for Savannah River, the Nevada Test Site, and accident was postulated to cause major damage to the building, resulting in the los filtered exhaust systems. The fuel units inside the dry cell could also be damaged impacts and potential fire. The fission products which might be released are based overheating testing performed on the naval fuel systems. The mechanical impact als release of corrosion products to the environment.

F.1.4.2.3.3.2 Source Term. The development of the radioactive source term for this scenario

is based on the following:

- One percent of the fuel units stored inside of the dry cell might be da impact or resultant fire and those fission products would be available
- Of the fission products available for release, 100% of the noble gases, halogens, 1.1% of the cesium, and 0.1% of the remaining solids could be environment.
- The release to the environment would occur at a constant rate over a 15
- 10% of the available corrosion products could be released to the enviro
- A portion of the concrete shielding is destroyed; however, the resultan minimum of 6 inches of concrete shielding.
- The following amount of radionuclides could be released to the environm includes nuclides that result in at least 99% of the possible exposure.

| | |
|---------|-----------------------|
| Nuclide | Curies |
| Cs-134 | 4.5×10^1 |
| Cs-137 | 6.23×10^1 |
| Pu-238 | 1.03×10^{-1} |
| BA-137M | 5.37 |
| Sr-90 | 5.46 |
| Ce-144 | 1.25×10^1 |
| Nb-95 | 7.65 |
| Y-90 | 5.46 |
| Ru-106 | 1.07 |

F.1.4.2.3.3.3 Results. The following table summarizes the public health risk to the general

population that would result from the hypothetical airplane crash into the dry cell Site. The number of fatal cancers would be expected to occur over a 50-year period the number of fatal cancers times the probability of occurrence.

| Site | Probability of accident per year | Maximal- ly ex- | No. of fatal cancers if acci- |
|------|-------------------------------------|--------------------|-------------------------------------|
|------|-------------------------------------|--------------------|-------------------------------------|

| | | | |
|------------------|----------------------|------------------------|------------------------|
| | | posed off-site | dent occurs |
| | | individu- | |
| | | al (MOI) | |
| | | (rem) | |
| Savannah River | 2 x 10 ⁻⁶ | 8.2 x 10 ⁻¹ | 4.8 |
| Nevada Test Site | 4 x 10 ⁻⁷ | 1.6 | 1.8 x 10 ⁻¹ |
| Oak Ridge | 1 x 10 ⁻⁶ | 350 | 8.4 |

This accident results in the highest risk for any hypothetical accident evaluated at the Nevada Test Site, and Oak Ridge.

For the hypothetical airplane crash into a dry cell accident scenario, the radiological cause contamination of the ground to a downwind distance of approximately 1.3 miles, a total area impacted by the accident of about 207 acres. The calculated downwind area is contained within the boundaries of Savannah River and the Nevada Test Site, but not within the boundaries of Table F.1.4.2.3.3-1. Summary of Exposure Calculation Results.

For Dry Cell Operations - Airplane Crash
At Savannah River

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|------------------------|----------------------------------|
| Worker | 2.6 x 10 ¹ | 2.1 x 10 ⁻² |
| MCW | 1.6 | 6.2 x 10 ⁻⁴ |
| NPA | 1.9 x 10 ⁻² | 9.6 x 10 ⁻⁶ |
| MOI | 3.1 x 10 ⁻¹ | 1.5 x 10 ⁻⁴ |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 579541 | 1.6 x 10 ³ | 8.1 x 10 ⁻¹ |

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|------------------------|----------------------------------|
| Worker | 1.6 x 10 ² | 1.3 x 10 ⁻¹ |
| MCW | 1.9 x 10 ¹ | 7.8 x 10 ⁻³ |
| NPA | 3.3 x 10 ⁻¹ | 1.7 x 10 ⁻⁴ |
| MOI | 8.2 x 10 ⁻¹ | 4.1 x 10 ⁻⁴ |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of 579541 | 9.6 x 10 ³ | 4.8 |

Table F.1.4.2.3.3-2. Summary of Exposure Calculation Results.
For Dry Cell Operations - Airplane Crash
At Nevada Test Site

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|------------------------|----------------------------------|
| Worker | 9.2 | 3.7 x 10 ⁻³ |
| MCW | 7.1 x 10 ⁻³ | 2.9 x 10 ⁻⁶ |
| NPA | N/A | N/A |
| MOI | 2.5 x 10 ⁻¹ | 1.3 x 10 ⁻⁴ |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |
| Population of | | |

13792 2.1 x 10² 1.1 x 10⁻¹

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|------------------------|----------------------------------|
| Worker | 1.6 x 10 ² | 1.3 x 10 ⁻¹ |
| MCW | 4.2 x 10 ⁻¹ | 1.7 x 10 ⁻⁴ |
| NPA | N/A | N/A |
| MOI | 1.6 | 8.0 x 10 ⁻⁴ |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |

Population of

13792 3.5 x 10² 1.8 x 10⁻¹

Table F.1.4.2.3.3-3. Summary of Exposure Calculation Results.
For Dry Cell Operations - Airplane Crash
At Oak Ridge

50% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|-----------------------|----------------------------------|
| Worker | 5.8 x 10 ¹ | 4.7 x 10 ⁻² |
| MCW | 1.5 | 6.2 x 10 ⁻⁴ |
| NPA | 2.2 x 10 ¹ | 2.2 x 10 ⁻² |
| MOI | 1.7 x 10 ² | 1.7 x 10 ⁻¹ |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |

Population of

871531 5.2 x 10³ 2.6

95% METEOROLOGY

| Location | Total EDE (rem) | Likelihood of Fatal Cancer |
|--|-----------------------|----------------------------------|
| Worker | 1.6 x 10 ² | 1.3 x 10 ⁻¹ |
| MCW | 9.3 | 4.7 x 10 ⁻³ |
| NPA | 1.3 x 10 ² | 1.3 x 10 ⁻¹ |
| MOI | 3.5 x 10 ² | 3.5 x 10 ⁻¹ |
| Exposure to Population within 50-mile Radius (person-rem) | | Number of Fatal Can- cers |

Population of

871531 1.7 x 10⁴ 8.4

F.1.4.3 Impact of Accidents on Close-in Workers. An evaluation has been made of the impact to

close-in workers involved in naval spent nuclear fuel management that might occur d radiological accidents postulated in spent fuel handling. This evaluation focused consequences of the accident. Clearly, a limited number of fatalities may occur wh fuel handling only in a secondary manner; i.e., the worker who happened to be in th due to a plane crash, seismic event, crane failure, etc. These secondary effects a following. Rather, only radiological consequences are considered.

F.1.4.3.1 Wet Storage.

F.1.4.3.1.1 Drained Water Pool Due to Seismic Event. No fatalities to workers close to

the scene of the accident would be expected due to radiological consequences. This the large amount of water in a water pool is expected to take several days which pr workers to leave the facility.

F.1.4.3.1.2 Accidental Criticality in a Water Pool Due to Human Error. It is likely no

fatalities would occur. At most, two or three workers may receive some appreciable This is because the criticality would occur under approximately 20 feet of water. would be sufficient to prevent exposure of nearby workers. Expulsion of a cone of criticality might lead to significant exposure to any workers who were directly abo criticality.

F.1.4.3.1.3 Mechanical Damage to Fuel in a Water Pool Due to Operator Error or

Crane Failure. No fatalities to workers would be expected from radiological conseq because the release of the source term is underwater. Attenuation by the water wou products, but release of noble gases would cause a direct radiation exposure to wor releases from the surface of the water pool, radiation alarms would sound requiring workers. Timely evacuation would prevent substantial radiation exposure.

F.1.4.3.1.4 Airplane Crash into Water Pool Storage. No fatalities to workers would be

expected from radiological consequences. This is because any release of radioactiv underwater and radiation alarms would sound requiring evacuation of nearby workers. would prevent substantial radiation exposure.

F.1.4.3.2 Dry Storage.

F.1.4.3.2.1 Wind-driven Missile Impact on Storage Casks. It is likely there would be no

fatalities to workers from radiological consequences. This is because there usuall workers except for brief periods when a container is being placed in the dry storag driven missile is not expected to penetrate a dry storage container, direct radiati nearby workers would not be expected. The container seal could be breached and som released. At most, two or three nearby workers may receive some radiation exposure airborne radioactivity.

F.1.4.3.2.2 Airplane Crash into Dry Storage. It is not likely that any fatalities would occur

to nearby workers due to the radiological consequences of this accident. As in Sec workers are usually not in the dry storage array except when a container is being p most, two or three nearby workers might receive significant radiation exposure from radioactivity since the container seal may be breached. The low probability of the coupled with the probability that workers would be close enough to be affected, cou probability that the wind would be blowing in the direction of the workers, makes i worker would receive substantial radiation exposure.

F.1.4.3.3 Dry Cell Operations.

F.1.4.3.3.1 Inadvertent Cutting into Fuel or Mechanical Damage. No fatalities to

workers would be expected from the radiological consequences of this accident. Thi ventila- tion systems' exhaust from a dry cell is directed to the outside of the building in constructed and away from nearby workers.

F.1.4.3.3.2 Partial Loss of Shielding of a Dry Cell. It is likely that no fatalities would

occur among nearby workers from the radiological consequences of this accident. There is still substantial shielding of radiation from material inside the cell even with the presence of the high-density concrete. However, one or two nearby workers may receive some radiation streaming through a crack in the dry cell if this is the mode of failure. Workers should evacuate quickly when radiation alarms sound.

F.1.4.3.4 Other Accidents.

F.1.4.3.4.1 HEPA Filter Fire. No fatalities would be expected among nearby workers from the

radiological consequences of a fire in a HEPA filter. This is because HEPA filters are located in an area where workers are likely to be working. In addition, the release of radioactive material from a filter fire is not large.

F.1.4.3.4.2 Small Leaks from Water Pools. No fatalities are expected among nearby

workers from the radiological consequences of a small leak from a water pool. The pool is expected to be into the ground through the water pathway. Drinking water supplies are immediately impacted. In addition, the typical concentration of radioactivity in the pool is low.

F.1.4.4 Evaluation of Shipboard Fire Involving Shipping Containers.

F.1.4.4.1 Description of Conditions. In this hypothetical accident scenario, a fire onboard a

ship that is transporting naval spent nuclear fuel in shipping containers from Pearl Harbor is postulated. This accident could be initiated by a collision with another ship. Subsequent fires are postulated to occur in Puget Sound in the center of the shipping channel approximately 2 miles from Seattle. The consequences of a similar accident at Pearl Harbor because of the smaller population and the fact that Pearl Harbor is a restricted area of the sea on the south side, limiting the number of people who might be exposed. The severe radiological consequences of this postulated accident scenario. The toxic chemical consequences of the burning fuel oil are presented in Section F.2.4.2.2.

During shipment, the containers are well protected from direct mechanical damage by collision. The rugged nature of the shipping container and the naval reactor demonstrated by the analysis of airplane crashes which showed that a jet engine did not rupture the container or rupture the fuel. A severe fire is necessary to potentially cause seals and overheat the spent fuel sufficiently to release fission products. Collisions are extremely unlikely. During the hypothetical accident, the fire would need to burn several hours to cause release of fission products or corrosion products to the environment.

F.1.4.4.2 Source Term. Conditions used in developing the source term are as follows:

- Ten percent of all fuel unit cladding inside of two shipping containers contained fission products are available to be released from the fuel units.
- Of the available fission products, 100% of the noble gases, 3% of the heavy metals, the cesium, and 0.1% of the remaining solid fission products are assumed to be released from the container.
- Ten percent of all fission products released to the container are released to the environment and the remainder are adherent on the fuel and cask surfaces.
- Ten percent of the original corrosion products from the fuel units are released to the environment.
- The following amount of radionuclides could be released to the environment: includes nuclides from one container that result in at least 99% of the source term.

| Nuclide | Curies |
|---------|-------------------------|
| Cs-134 | 2.57 x 10 ¹ |
| Cs-137 | 3.56 x 10 ¹ |
| Pu-238 | 5.90 x 10 ⁻² |
| Ba-137M | 3.07 |
| Sr-90 | 3.12 |
| Ce-144 | 7.17 |
| Nb-95 | 4.37 |
| Y-90 | 3.12 |
| Ru-106 | 6.11 x 10 ⁻¹ |

F.1.4.4.3 Results. The following table summarizes the public health risk to the general

population that would result from the hypothetical shipboard fire accident. The nu would be expected to occur over a 50-year period. "Risk" is defined as the number the probability of occurrence.

The probability of occurrence of this hypothetical shipping accident is 6.7 x and was obtained as follows. The probability of a single port entry accident is 1. The probability of a fire, given the occurrence of an accident, is 8 x 10⁻⁴ (DOE 19 two probabilities with the port entry frequency of 21 naval spent nuclear fuel ship years results in a probability of 6.7 x 10⁻⁸ per year. Due to the rugged nature of effectiveness of fire fighting over a several hour period, the probability of fissi environment would be even less.

DOE guidance (DOE 1993b) provides that the consequence of an accident which h of occurrence of less than 1 x 10⁻⁷ per year need not be calculated. However, in v accident expressed in several public comments, the following table is provided list consequence and the risk.

Shipboard Fire Involving Shipping Containers

| In Puget Sound | | 50% Meteorology | | 95% Meteorology | |
|---|--|--|---------------------------|-----------------|--|
| Shipping Lane | | | | | |
| Maximally Exposed Off-site Individual (MOI) | Total EDE (Rem) | Likelihood of Fatal Cancer | Total EDE (Rem) | Li Fa | |
| General Population within 50-mile Radius | 9.3 x 10 ⁻¹ Exposure (Person-Rem) | 4.7 x 10 ⁻⁴ Number of Fatal Cancers | 1.8 Exposure (Person-Rem) | 9. Nu Fa | |
| | 2.27 x 10 ⁴ | 11.4 | 1.03 x 10 ⁵ | 51 | |
| Risk per year | 7.6 x 10 ⁻⁷ | | 3.5 x 10 ⁻⁶ | | |

The risk for this hypothetical accident is slightly lower than that for the most se analyzed at Puget Sound.

For the hypothetical shipboard fire accident, the radioactive plume might cause con downwind distance of less than 1 mile. However, since this area is entirely over w would be quickly diluted by tidal flow and turbulence.

F.1.5 Analysis of Uncertainties

The analyses of the impacts of normal operations and hypothetical accidents as management of naval spent nuclear fuel presented in this Environmental Impact State on conservative calculations. This is necessary because virtually all of the event occurred and most of the impacts of routine operations are so small that they canno of calculations introduces the possibility that the actual impacts may differ from various kinds of uncertainties, such as differences between actual behavior and the equations and the variability of the values of factors used in the calculations. I effects of such variability and uncertainty, the analyses performed for this append four components: the probability that an event, such as an accident, could occur; tive material or radiation that might be released by the event; the calculation of to human beings from the release; and the conversion of the radiation exposure to d tal health effects. Each of these components is discussed separately in the follow operations and accidents.

Each of these components has been analyzed for both routine operations and a discussion in the following sections focuses on accident analyses, but it should be analysis of uncertainties for routine operations is the same, with a few exceptions are certain to occur, so the "probability" of such events is effectively 1.0. Seco for the analyses of routine operations are based on monitoring of current operation Propulsion Program facilities such as the Expended Core Facility at INEL. Consequ the amount of radiation or radioactivity involved are expected to be close to those occur under the alternatives evaluated in this EIS. It is possible that there woul among facilities and that future efforts to keep exposures to workers as low as rea might reduce the source terms further, but the values used in the analyses in this little different from those actually encountered. The effects of routine operation calculated using similar analytical methods and models for determination of radionu environment, pathways to humans, and conversion of exposure to health effects. The of uncertainties in Sections F.1.5.3 and F.1.5.4 applies to the results of analyses well as to postulated accidents.

F.1.5.1 Probabilities of Events. The probability that an accident might occur has been determined

for a number of events which might reasonably be postulated. These probabilities a appendix to calculate the risk, defined as the product of the probability times the postulated accident.

The best methods available have been used to estimate the probabilities for t analysis. For example, a methodology developed by Sandia Laboratories (Sandia 1983 compute the probability that an aircraft might crash into naval spent nuclear fuel uses actual aircraft crash statistics obtained from the Federal Aviation Administra by Sandia to reproduce the observed frequencies as closely as possible. Probabilit were derived from published studies of the frequencies of seismic activity and repr estimates, but these probabilities are subject to some uncertainty due to the relat have occurred at the sites evaluated under the alternatives in this EIS.

The probabilities of a range of accidents which might be caused by human erro included. Such events include accidental criticality caused by handling errors, dr improper operation of cranes, and incorrectly performing machining procedures. For probability of one error in one thousand operations (a frequency of 10^{-3} events per operations performed by a single trained operator following a written procedure. I verification of the action by a second trained operator, this frequency is lowered probabilities are derived from the methodology used by the Nuclear Regulatory Commi assessment of human reliability (Swain 1983).

In many instances, the probabilities assigned to the events reflect the likel event, such as an earthquake or an aircraft crash, might occur. However, for the p the resulting accident was assumed to have quite severe consequenc- es. The probability of such severe consequences is smaller than the probability that the initiating event might occur, severe as used in the analyses possibly occurring only one time in 10 or 100 occur event. The probabilities for most of the analyses in this appendix used only the p event and did not include the further reduction in the probability of the postulate resulting from the severity used. This was done, in part, because the severe conse some cases the initiating events themselves, occur very infrequently, or have never on their frequency is available.

For example, one accident analyzed is the impact on a spent fuel container of by a tornado or other high winds. The sequence of events analyzed included breachi in order to release radioactive material. In reality, the missile would have to be at high enough speed to cause the postulated damage. Similarly, it would have to c the correct location and at the correct angle in order to damage the seal. The pro accident is 10^{-5} per year, the probability that a wind-driven missile might strike include any factor to account for other elements in the sequence required to actual Therefore, the probability of the consequences calculated for this accident would b probability of 10^{-5} per year used in the analysis.

A second example is provided by the analysis of aircraft impact on shipping c storage of naval spent nuclear fuel. In this accident analysis, the impact was ass container to be penetrated if the container were contacted by the aircraft. Howeve fuel shipping containers are of very rugged design, and structural analysis of the naval shipping container is very unlikely to be penetrated by an aircraft crash, ev the airplane. Consequently, the probability that the naval spent nuclear fuel coul fission products might be released is much, much less than the crash probability al

probability assigned to these consequences in this appendix.

A third example is seen in the ship fire accident. In this analysis, it is a carrying naval spent nuclear fuel shipping containers were involved in a very severe occurred, the fire would include the cargo hold where the naval spent nuclear fuel the fire would not be extinguished by the redundant systems provided, and it would sufficient intensity to damage the shipping container and the spent nuclear fuel in radioactive materials from the containment provided. Given that a severe collision probability that all of the necessary conditions would occur and a fire of the required duration would occur in the cargo hold is clearly far less than the probability of

As can be seen from these examples, the actual probability of the consequences analyses are smaller than the values presented in this appendix, at least in part b do not include an additional factor to reflect the accident severity used in the analyses stated in this appendix for most accidents are believed to be at least 10 to would actually occur. However, the same probabilities have been used in the evaluation alternatives considered and all of the risks are small, so the approach used is adequate this EIS.

F.1.5.2 Release of Radioactive Material or Radiation (Source Term). Since the source terms

used in the accident analyses are typically for accidents which have never occurred for uncertainty. All of the accidents analyzed in this EIS are intended to be accidents consequences which are unlikely to be exceeded by any reasonably foreseeable accidents themselves and the sequences of events during the accidents have been chosen source term. For example, systems such as high efficiency particulate filters have inoperative in all cases where the accident might have an opportunity to disable them

The source terms for the hypothetical accident analyses are dependent upon a For there to be an accidental release of radioactivity to the environment, there must storage facility or containment structure. Furthermore, naval spent nuclear fuel must in order for there to be any release of fission products since all fission products naval nuclear fuel. The amount of damage to the external containment or the fuel is severity and the nature of the accident. In the accidents analyzed, there are assumptions containment or the extent of damage to the fuel units which were made to provide a bounding evaluation whose results would not be exceeded by reasonably postulated accidents type.

One example of this is the evaluation of the dry storage container impacted by a missile. Damage to the container by the missile is not expected to occur, but for the seal is assumed to be damaged by the missile impact and corrosion products with assumed to be released through the damaged seal. The uncertainty on the resultant since the probability of a release larger than in the calculation (resulting in a release essentially zero while the possibility of a release of less radioactive material is release if the container seal is not broken). The range of variation, or the uncertainty term for this accident is between +0% and -100%.

Another example is the plane crash into a dry processing facility for naval spent dry processing facility includes a thick concrete shielded cell in which a few naval are processed at a time. The massive concrete shield is provided to protect against radiation but it has the secondary benefit of protecting the fuel units being processed by natural or man-made phenomena. In the unlikely event that an airplane crashed is expected that no damage to the spent fuel would result. Even so, for evaluation of EIS, it is assumed that 1% of the fuel in the dry cell could be damaged and that such enter the dry cell to cause a fire which could cause the release of fission product and destroy the filtration system. Again, the uncertainty range is one-sided since expected, causing the variability or uncertainty to range from +0% to -100%.

All of the source terms used for the evaluation of the accidents were developed in this fashion. Thus, the expected outcome for all of the accidents is that a lower release is expected than is used in the analysis, representing a range of variation of

F.1.5.3 Exposure to Humans. Exposure to the individuals and the general population is evaluated by

integrated computer programs. The methods used model the movement of airborne, ground contamination resulting from the postulated release using five types of pathways to pathways include exposure directly to the radiation from the material in the plume, radiation from contaminated soil or water, inhalation of air containing gases or particulate contaminated water or food. The analyses in this appendix used parameter values which

available estimates or, when best estimate values were not available, are conservative

The Gaussian plume model used in these analyses to represent airborne movement of material is the standard used in virtually all evaluations of environmental effects. Distributions calculated using the Gaussian plume model with test data has shown that they differ by as much as a factor of 5 in some circumstances. In order to ensure that the high as could occur under any set of conditions, in most of the analyses a ground level and no reduction in the airborne concentrations was included for either turbulence or the effect of wind meander which occurs naturally at the low wind speeds accompanying meteorological conditions.

One intentional choice of parameters to ensure that the results would be conservative is the worst case meteorological conditions in the tabulations of the risks and consequences alternatives provided in Chapters 3 and 5. The results for both the most likely meteorological conditions and for the worst case are provided in detailed tables in this attachment and show that meteorological conditions produce exposure estimates which are 2 to 10 times higher than the most likely conditions (depending upon local meteorological conditions). Overall, the Gaussian plume model might introduce an uncertainty of a factor of 5 or less in the use of the worst case meteorological conditions would essentially offset any underestimation.

The direct radiation from the cloud is calculated using a conservative representation as a finite cloud, and, as a result, little uncertainty is introduced in this part of the radiation from contamination which results from particles from the plume deposited on the ground. This depends upon the deposition parameters which are input as best-estimate values. Factors that result in more material on the ground and increased exposure to those closer to the accident site and less material on the ground and decreased exposure for those farther from the accident site, the uncertainty in this parameter would depend upon the population distribution around the accident scene.

The possible exposure to direct radiation from material in surface water and as a result of accidental release directly to the water or fallout from an airborne source. The people involved in activities such as professional fishing, maritime operations, swimming, etc. The calculations took no credit for dilution by river currents or tidal movement and the concentrations in the air were not reduced by the amount of material deposited in the water. Due to the high concentrations used in the calculations and an assumption that every member of the population would be exposed to direct radiation from surface waters, exposure from this pathway is overestimated.

The inhalation pathway evaluation is based on average breathing rates and upon the recommendations by the ICRP (ICRP 1977 and ICRP 1979). Obviously, higher value parameters would increase the estimated exposures and lower values would decrease them. It appears to be little controversy concerning these parameters and the same parameter evaluation of all of the alternatives in this appendix.

The ingestion pathway includes meat, seafood, dairy and crop products, and drinking water. The estimate parameters are used to evaluate the contamination levels in food and water consumption. Consumption rates for individuals are based on observed eating habits and includes the assumption that a conservative 10% of the entire diet of the affected population is from contaminated products. The uncertainties associated with these pathways can obviously be large, but the range of variation is not large and the same values for evaluation of all alternatives.

The drinking water contribution to the ingestion pathway was calculated by assuming that a portion of the radioactive material would become dissolved in the drinking water supply. If fresh surface water provides drinking water, any contamination of the water was assumed to be promptly and no decreases due to radioactive decay were used. At sites where aquifer water is used for drinking water, consumption of water from the aquifer was delayed for the time required for contamination to reach the aquifer and then to reach the nearest drinking water source. If a postulated leak from the Expanded Core Facility, it was assumed that 20 years would elapse before the radioactive material would reach a well drawing from the aquifer and that the material released would enter the aquifer each year. Maximum exposed individuals were assumed to drink only water from the contaminated source and to drink 2 liters of water per day. In the population in general, a conservative fraction of the population was assumed to drink water from affected sources. The concentrations in these calculations are considered conservative because no reduction of the concentration by dilution was included and the population exposed to the affected drinking water is conservatively high.

At sites where irrigation is used, contamination of food crops, livestock, and game was analyzed. The same concentration of radioactive material as in drinking water was assumed for irrigation water. Affected crops, livestock, and game were assumed to receive all water from the irrigation source and applicable biological accumulation factors were used. Human consumption of crops, livestock, and game were used to calculate the exposure from this source. The uncertainty

source is associated with the concentration of contaminants in the irrigation water foods consumed, and the fraction of the population which ingests the affected food.

The population used to determine the effects of postulated accidents in this population within the 22.5-degree sector at each distance within 50 miles downwind spread of the plume for the worst case meteorology does not cover the entire sector there is a conservatism of more than a factor of 2 in the application of the calculation the dose to the population. The population data used were obtained from the 1990 U.S. population growth or decreases in a region could introduce small changes, but the distributions were used for a specific site for evaluation of all alternatives.

Considering all of the factors which might have an appreciable effect on the analyses, any tendency of the Gaussian plume model to underestimate concentrations the use of other parameters which are known to be conservative. Examples of such include the general use of the meteorological conditions which would produce the most the use of the entire population of a 22.5-degree sector. Consequently, this portion appear to contribute little in the way of uncertainty which could cause the results presented in this appendix.

F.1.5.4 Conversion of Exposure to Health Effects. The conversion of amounts of radiation or

radioactive material transmitted to an individual or to population groups requires exposure or dose received by humans caused by inhaling or ingesting radioactive material radiation field. Such calculations are based on a number of factors, including the metabolic processes, such as respiration or excretion, the type of radiation involving various organs, and the age of the individuals involved. The rates of human metabolism characterized at this time and the energies, half-lives, and similar properties of radiation have been measured extensively and are not subject to great debate. Considers introduce little uncertainty into the calculations in this EIS.

However, the number of detrimental health effects which might result from exposure of a group of people to low levels of radiation has been the subject of debate for many years. The Academy of Sciences has conducted several investigations of this matter and its full 181 of its latest study of the health effects of exposure to low levels of radiation BEIR V (NAS 1990), states:

Finally, it must be recognized that derivation of risk estimates for low dose through the use of any type of risk model involves assumptions that remain to be tested. At low doses, a model dependent interpolation is involved between the spontaneous incidence at the lowest doses for which data are available. Since the current models are a linear function of dose, little uncertainty should be introduced by departure from linearity cannot be excluded at low doses below the range of observed departures could be in the direction of either an increased or decreased risk. Epidemiologic data cannot rigorously exclude the existence of a threshold in this range. Thus, the possibility that there may be no risks from exposures comparable to natural background radiation cannot be ruled out. At such low dose rates, it is acknowledged that the lower limit of the range of uncertainty in the risk estimate

The National Academy of Sciences considers that the uncertainty in the lifetime total mortality risk estimates calculated using the linear extrapolation, no threshold model preferred, which is consistent with the model used in this EIS, is approximately a factor of two (an interval of 0.5 to 2 times the calculated estimates).

The calculations of health effects performed in this Environmental Impact Statement are in accordance with the relation recommended by the International Council on Radiation Protection because it is the most up to date by the Council. It is also consistent with the preferred model used by the Academy of Sciences in the BEIR V report and is widely accepted by the scientific community as representing a method which produces estimates of health effects which will not be overestimated. There are some who believe that exposure to low levels of radiation can produce more health effects than would be estimated using the International Council on Radiation Protection relation. A growing number of researchers believe that the International Council on Radiation Protection overestimates the number of detrimental health effects produced by low levels of radiation. The possibility of no effect cannot be excluded (CIRRPC 1992).

Clearly, using a relation developed by one or the other of these groups would produce a smaller estimate of the number of health effects than the values presented in this EIS. A change in the small risks calculated for all of the alternatives would still leave the results of analyses of normal operations and hypothetical accidents in Appendix A. The calculated exposure in addition to the number of health effects in order to permit the use of any relation between radiation exposure and health effects judged appropriate.

F.1.5.5 Summary of Uncertainties. As discussed in the preceding portions of this section, the

calculations in this EIS have generally been performed in such a way that the estimate are unlikely to be exceeded during either normal operations or in the event of an accident. When combined with conservative estimates of the effects of radiation, produce estimates very unlikely to be exceeded. The effects for all alternatives have been calculated in terms of human health and the environment.

The analyses of hypothetical accidents provide more opportunities for uncertainty because the calculations must be based on sequences of events and models of effects that occurred. In this appendix, the goal in selecting the hypothetical accidents analyzed is to produce events which would be as severe or more severe than any which might reasonably be postulated. The models have attempted to provide estimates of probabilities, source terms, pathways for dispersion and exposure, and the effects on the environment which are as realistic as possible. However, in many cases, the very low probabilities of accidents postulated has required the use of models or values for input which produce consequences and risks which are higher than would actually occur because of the degree of uncertainty which will not be exceeded. In summary, it is judged that the risks presented in this EIS to be at least 10 to 100 times larger than what would actually occur.

The use of conservative analyses is not an important problem or disadvantage of the alternatives have been evaluated using the same methods and data, allowing a comparison of the alternatives on the same basis. Furthermore, even using these conservative estimates of risks for all of the alternatives are small, which greatly reduces the significance of the parameters.

F.2 TOXIC CHEMICAL ISSUES AT NAVAL SPENT NUCLEAR FUEL

EXAMINATION AND STORAGE SITES

The INEL-ECF is a large laboratory facility used to receive, examine, and ship spent nuclear fuel and irradiated test specimen assemblies. In order to accomplish these tasks, classified as toxic are involved in a variety of operations and thus a potential exposure to toxic chemicals due to human error and failure or malfunctioning of equipment.

This section provides the results of an evaluation of both normal operations and accidents which could result in toxic chemical releases. This section describes how facilities and equipment are selected for analysis, discusses the computer codes used in the analysis, presents estimates of atmospheric dispersion, defines the hypothetical accidents which would produce the most severe consequences, and estimates the potential health effects. Each alternative population and meteorology were used to produce estimated consequences for each operation and accident.

F.2.1 Toxic Chemical Inventory

Some chemicals classified as toxic are routinely used in a variety of operations at INEL-ECF. Table F.2-1 provides the INEL-ECF Chemical Inventory. This inventory was developed from the Naval Reactors Facility Superfund Amendments and Reauthorization Act of 1992 chemical inventory (INEL 1993). Those chemicals specifically stored at INEL-ECF as well as those used for facility support (e.g., fuel oil, diesel fuel, sodium hydroxide) were included. Chemicals at INEL-ECF that were (a) in excess of reportable quantities (usually 1 pound) on the EPA Title III List were evaluated. The chemicals in the EPA Title III List of Lists are the hazardous chemicals in:

- SARA Section 302 Extremely Hazardous Substances (CFR 1992a)
- CERCLA Hazardous Substances (CFR 1992b)
- SARA Section 313 Toxic Chemicals (CFR 1992c)
- RCRA Hazardous Wastes (CFR 1992d)
- EPA list of 100 extremely hazardous chemicals (FR 1993).

Table F.2-1. INEL-ECF chemical inventory.

| CAS No. | Chemical Name | Weight Total (pounds) | Weight Unit1 (pounds) |
|---|---|-----------------------------|-----------------------------|
| Chemicals Used for Water Pool Operations | | | |
| 60-00-4 | Ethylenediaminetetraacetic Acid (EDTA) (reagent for water analyses) | 46.3 | 1.1 |
| 75-71-8 | Dichlorodifluoromethane (CFC-12) (refrigerant in coolers for pool water) | 30.0 | 30.0 |
| Chemicals Used for Examination Operations | | | |
| 60-29-7 | Ethyl Ether | 5.7 | 5.7 |
| 67-63-0 | Isopropyl Alcohol | 100.6 | 6.6 |
| 123-31-9 | Hydroquinone (photographic film developer) | 65.5 | 3.3 |
| 144-55-8 | Sodium Bicarbonate | 198.0 | 99.0 |
| 302-01-2 | Hydrazine | 3.7 | 1.8 |
| 7664-41-7 | Ammonia ² | 2.8 | 0.28 |
| 7727-37-9 | Diatomic Nitrogen | 643 | 125 |
| Chemicals Used for Facility Support | | | |
| 107-21-1 | Ethylene Glycol (anti-freeze and paint additive) | 516.1 | 514.0 |
| 115-07-1 | Propylene (Propene) | 0.01 | 0.005 |
| 1310-73-2 | Sodium Hydroxide (boiler water pH control) | 43260 | 43260 |
| 7664-93-9 | Sulfuric Acid (boiler and cooling tower water pH control) | 96427 | 96427 |
| 68476-33-5 | Fuel Oil #5 | 776210 | 204270 |
| 68476-34-6 | Diesel Fuel #2 | 14316 | 10735 |
| 72623-83-7 | Hydrotreated Lubricating Oil | 882.6 | 413 |
| Chemical Used for Nuclear Poison | | | |
| 1332-77-0 | Potassium Tetraborate | 17000 | 10 |

1 The quantities in this column represent the amount of chemical stored in the large tank identified in the INEL-ECF chemical inventory.

2 The ammonia is present as ammonium hydroxide.

In order to evaluate the alternate locations, the same inventory of chemicals was used at the DOE sites; namely, the Savannah River Site, the Hanford Site, the N and the Oak Ridge Reservation. In addition, the Barnwell Nuclear Fuel Plant (hereinafter the Barnwell Plant), which is adjacent to the Savannah River Site, was evaluated at the sites. Since the shipyards would not be involved with examination operations (except Sound), of the chemicals listed, only diesel fuel would be available in a substantial form of fuel stored at the shipyards. Although several of the chemicals listed in water treatment chemicals associated with water pool operations and small water pools at the shipyards for fuel storage and inspection, the shipyard would already have water treatment chemicals for other operations at the shipyard. Therefore, an inventory of quantities or types of chemicals at the shipyards was considered to be very small and require evaluation. In addition, even though the Kenneth A. Kesselring Site is not a facility would also not be involved with examination operations. Therefore, this facility was evaluated in the same manner as the shipyards.

F.2.2 Computer Modeling to Estimate Toxic Chemical Exposures

Factors such as locations of affected persons, terrain, meteorological conditions, and characteristics of the chemical inventory are required as input parameters for calculations to determine human exposure from airborne releases of toxic chemicals. This section describes the computer models used to perform exposure estimates. Specific input parameters for the analyses are summarized in the appropriate subsection for normal operations and emergency conditions. The EPIcode was used to evaluate toxic chemical releases resulting from normal operations. The ISC2 code was used to evaluate releases from normal operations.

F.2.2.1 EPIcode (tm). The Emergency Prediction Information Computer Code (EPIcode-) is the

computer code chosen for estimating airborne concentrations resulting from most releases of toxic chemicals (Homann 1988). Like RSAC, EPIcode uses the well-established Gaussian Plume model to calculate the airborne toxic chemical concentrations usually at the same downwind location. The EPIcode library contains information on over 600 toxic substances listed by the Conference of Governmental Industrial Hygienists in the EPIcode Manual. EPIcode also

description of substances not included in the library. A step-by-step flow chart of features (up to the output options) is shown in Figure F.2-1.

Figure F.2-1. Flow sheet for EPIcode (Homann 1988). As shown in Figure F.2-1, the user enters an ambient concentration and a release rate. For releases over a specific term (e.g., term releases), the user specifies the release duration and the total quantity of material released.

Area continuous and area term releases are useful in calculating the effects of spills of volatile liquids. The user must enter the radius of the circle enclosing the spill area. Also entered is the temperature of the pool and ambient temperature to establish a liquid spill. An upwind virtual point source, which results in an initial plume to the effective radius of the area source, is used to model an area release.

By specifying a release quantity, release duration, and release area, the user proposes a release rate per unit spill area. The release quantity is defined as a fraction of the material at risk. The concepts and defined terms are the same as for Gaussian calculations. EPIcode confirms that the volatility of the spilled substance can support the release rate. If the proposed release rate exceeds the saturation conditions at the release point, EPIcode calculates a lower release rate and a corresponding longer release time.

In calculating effective release height, the actual plume height may not be the effective height, e.g., the stack height. Plume rise can occur because of the velocity of the plume and the temperature differential between the stack effluent and the surrounding air. EPIcode calculates both the momentum plume rise and the buoyant plume rise and chooses the greater of the two. Since this effective increase in release height leads to lower concentrations at the receptor, physical release heights were used to calculate the concentrations that the general public would be exposed to during accidental releases of toxic substances. This approach will always provide conservative estimates.

In this application, the standard terrain calculation of EPIcode is always used. Concentrations were calculated using both 95% and 50% meteorological conditions (See Appendix B). The elevation of the affected person is always ground level (0 meters) and, as in RASCAL, the layer height is always 400 meters (1320 feet). The deposition velocities used (See Appendix B) are somewhat different than those of RSAC-5, but they are still conservatively low.

As described in its user manual (Homann 1988), EPIcode also includes the following adjustments:

- Treating a release as instantaneous vs. continuous depending upon the plume type and the specific downwind location being considered
- Correcting the concentration for sampling time
- Adjusting the wind speed for release height
- Depleting the plume as a function of downwind distance
- Adjusting the standard deviations of the crosswind and vertical concentration profiles for area releases.

As output, EPIcode can generate data plots of mean toxic chemical concentrations (for a specified averaging time) as a function of downwind distance. From these graphs an output, the concentrations for the worker at 100 meters (330 feet) (the shortest distance from the release point), for the nearest public access (NPA), for the maximum off-site exposure, and for nearby communities are determined and evaluated for health effects.

EPIcode was selected as the computer code for release analysis of chemicals as compared to Gaussian modeling after comparison with a number of codes, primarily CHARM and ARCHIE. EPIcode was judged more applicable for this application than either the CHARM code or the ARCHIE code.

F.2.2.2 ISC2 Code. The Industrial Source Complex (ISC2) model is a widely used, publicly

available, and accepted EPA regulatory model which employs straight line (i.e., unit Gaussian diffusion to estimate pollutant dispersion (EPA 1992b)). ISC2 is an appropriate model for industrial complexes in rural or urban areas with transport distances less than 50 miles (30 miles). This model employs a standard meteorological data set requiring single speed, wind direction, ambient air temperature, atmospheric stability, and vertical mixing. Also, the ISC2 model is able to account for variations in pollutant concentration influenced by nearby structures.

In addition to the ISC2 model, the MESOPUFF II model was also evaluated. MESOPUFF is a regional (mesoscale) scale model that takes into account a varying wind field. Analyses at the INEL have demonstrated that plumes may undergo many changes in direction due to the varying winds common to the INEL vicinity. The number of changes is partially a function of release time and transport duration. The plume transport and estimation of pollutant concentrations beyond 12 miles (20 kilometers) is best modeled using spatially varying wind data. Since the model is used as a basis for determining or enforcing compliance with regulations, it is used as a basis. The model is also readily available to the public.

Upon review of the ISC2 and MESOPUFF II models, the decision was made to utilize for the dispersion analysis of pollutants emitted from stationary sources. ISC2 is able to accurately predict downwind pollutant concentrations within 30 miles (50 kilometers) into account multiple point and area emission sources, evaluating hourly meteorological data, and determining the effects of nearby structures.

F.2.3 Health Effects

Toxic constituents dispersed during an accident could induce adverse health effects in exposed individuals. This possible impact is assessed by comparing the airborne concentration of each substance at specified downwind locations to standard accident exposure guideline toxicity.

Where available, Emergency Response Planning Guideline (ERPG) values are used for comparison. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (Rusch 1993). ERPG values are specific to each substance, and are derived for each of three general severity levels:

- Exposure to concentrations greater than ERPG-1 values results in an unacceptable likelihood that one would experience mild transient adverse health effects such as a clearly defined objectionable odor.
- Exposure to concentrations greater than ERPG-2 values results in an unacceptable likelihood that one would experience or develop irreversible or other serious effects, or symptoms that could impair one's ability to take protective action.
- Exposure to concentrations greater than ERPG-3 values results in an unacceptable likelihood that one would experience or develop life-threatening health effects.

Where ERPG values have not been derived for a toxic substance, other chemical values are substituted, as follows:

- For ERPG-1, Threshold Limit Value, Time-Weighted Average (TLV-TWA) values (ACGIH 1993) are substituted: The TWA is the time-weighted average concentration for a normal 8-hour workday and a 40-hour workweek, to which nearly all workers repeatedly exposed, day after day, without adverse effect.
- For ERPG-2, Level of Concern values (equal to 0.1 of Immediately Dangerous to Life or Health) are substituted: Level of Concern is defined as the concentration of a hazardous substance in air, above which there may be serious irreversible effects or death as a result of a single exposure for a relatively short period of time (1987).
- For ERPG-3, Immediately Dangerous to Life or Health (IDLH) values are substituted: IDLH is defined as the maximum concentration from which a person could escape 30 minutes without a respirator and without experiencing any effects which would impair the ability to escape or irreversible side effects (NIOSH 1990).

Possible health effects associated with exceeding an ERPG-2 or -3 value are of substance of concern, and must be characterized in that context. When concentrations exceed an ERPG or substitute value, the specific toxicological effects for the chemical are considered in describing possible health effects associated with exceeding a threshold.

ERPG values are based upon a 1-hour exposure of a member of the general population. This EIS, exposures resulting from the release of toxic chemicals during an accident are postulated to occur over a period of 1 hour or less to allow for a direct comparison to ERPG values. This approach provides an additional element of conservatism in the evaluation of releases that last much less than 1 hour.

In addition to comparing the airborne concentrations of each substance to standard exposure guidelines, each substance was evaluated to determine if it has the potential for carcinogenic health impacts. If a particular substance has this potential, the International System (IRIS) (TOXnet 1993) was reviewed and if sufficient toxicological information was available, a future potential likelihood of developing cancer was determined. If IRIS was not available, alternative evaluation methods, including comparison to quality criteria, were substituted.

The impact of normal operations was also evaluated. This impact was assessed by comparing the airborne concentrations of each substance at specified downwind locations to the Air Quality Standards (NAAQS) assigned for each substance. NAAQS consist of national

and secondary ambient air quality standards (CFR 1991). National primary ambient air quality standards define levels of air quality which the EPA judges are necessary, with an adequate margin of safety, to protect the public health. National secondary ambient air quality standards define levels of air quality which the EPA judges are necessary to protect the public welfare from anticipated adverse effects of a pollutant. As a result, the immediate as well as long-term impacts of normal operations was evaluated by comparing the airborne concentrations of each substance to the NAAQS.

F.2.4 Analysis Description and Results

The analysis results for both normal operations and accident conditions are presented for the location analyzed. Detailed estimated concentrations and ERPG levels, expressed in micrograms per cubic meter (mg/m³), are reported in tabular form for a worker, maximally exposed condition (MCW), maximally exposed off-site individual (MOI), and maximally exposed individual at the nearest public access (NPA). A complete description of these individuals is provided in F.1.3.2.

F.2.4.1 Normal Operations.

F.2.4.1.1 Source of Emissions. Emissions resulting from normal operations involving

toxic chemicals listed in Table F.2-1 were evaluated. It was determined that the burning of fuel oil in the facility's boilers and the burning of Number 2 diesel fuel in the facility's diesel generators represented the largest sources of emissions under normal operations. These operations result in the release of oxides of nitrogen (90% nitric oxide and 10% nitrous oxide), sulfur dioxide, particulates (PM-10), lead, and volatile organic compounds (VOCs). The release of these chemicals was evaluated for effects on the on-site workers, MCW, NPA, and MOI.

The emissions that occur due to normal operations at the INEL-ECF were evaluated using the ISC2 code. These releases were also used at the alternate locations (Hanford, Nevada Test Site, Barnwell Plant, and Oak Ridge) for evaluation purposes. Heating and emergency diesel generators already exist at the alternate shipyard locations and therefore these alternate locations would not result in a measurable increase in emissions. Releases from shipyard locations were not considered.

F.2.4.1.2 Conditions and Key Parameters.

- Number 5 fuel oil was burned in facility boilers for space heating.
- Number 2 diesel fuel was burned in facility emergency diesel generators.
- Source term was based on the INEL report on routine yearly releases (NRF 1) included:
 - 1.02 tons per year of carbon monoxide released
 - 9.04 tons per year of oxides of nitrogen released
 - 33.7 tons per year of sulfur dioxide
 - 1.54 tons per year of particulates
 - 5.86 x 10⁻⁴ tons per year of lead
 - 0.18 tons per year of volatile organic compounds.
- Forty percent of the total boiler and emergency diesel generator use for the Reactors Facility was attributed to the INEL-ECF.
- Three point sources (one representing boilers and two representing emergency diesel generators) were used.
- Stack diameters of 1.07 meters (3.5 feet) for boilers and 0.305 meter (1 foot) for emergency diesel generators were used.
- Stack gas exit velocities of 21.8 meters per second (72 feet per second) for boilers and 44.2 meters per second (145 feet per second) for emergency diesel generators were used.
- Stack gas exit temperatures of 505°K for boilers and 794°K for emergency diesel generators were used.
- Worker concentrations were based on 16 sector polar grids. Other affected areas were defined as discrete points.
- DOE site meteorological data were used for evaluations at the Naval Reactors Facility, Hanford, Nevada Test Site, and Oak Ridge. Meteorological data from the climate data base were used for the other locations.

National Weather Service Station were used for evaluations at Savannah Riv Barnwell Plant.

F.2.4.1.3 Results. The airborne concentrations, averaged over the duration of each

exposure, were calculated by ISC2 for the worker, MCW, NPA, and MOI using normal me Tables F.2.4.1-1 through -6 list the downwind concentrations at various locations. concentrations were compared to respective NAAQS values where available. The NAAQS follows:

Carbon monoxide. The national primary ambient air quality standards for carb are 10 mg/m³ for an 8-hour average concentration not to be exceeded more than once 40 mg/m³ for a 1-hour average concentration not to be exceeded more than once per y

Sulfur oxides. The national primary ambient air quality standards for sulfur measured as sulfur dioxide are 0.08 mg/m³ as an annual arithmetic mean and 0.365 mg maximum 24-hour concentration not to be exceeded more than once per year. The nati ambient air quality standards are 1.3 mg/m³ as a maximum 3-hour concentration not t more than once per year.

Nitrogen dioxide. The national primary and secondary ambient air quality sta nitrogen dioxide is 0.1 mg/m³ as an annual arithmetic mean.

Lead. The national primary and secondary ambient air quality standard for le compounds that are measured as elemental lead is 1.5×10^{-3} mg/m³ as a maximum arit averaged over a calendar quarter.

Particulate matter. The national primary and secondary ambient air quality s particulate matter is 0.05 mg/m³ as an annual arithmetic mean and 0.15 mg/m³ as a m 24-hour concentration.

A comparison of the downwind concentrations provided in Tables F.2.4.1-1 thro the NAAQS identified above indicates that no NAAQS is exceeded for normal operation Table F.2.4.1-1. Summary of chemical concentrations for normal operations at the I

CHEMICAL CONCENTRATIONS mg/m³

| | Carbon Monoxide | Sulfur Dioxide | Nitric Oxide | Nitrogen Dioxide | Lead | VOC |
|--------|----------------------|----------------------|----------------------|----------------------|-----------------------|---------------------|
| Worker | 4.6×10^{-5} | 5.5×10^{-4} | 1.9×10^{-4} | 2.1×10^{-5} | 9.0×10^{-9} | $1.9 \times 10^{-}$ |
| MCW | 3.7×10^{-6} | 9.5×10^{-5} | 2.6×10^{-5} | 2.9×10^{-6} | 2.0×10^{-9} | $8.5 \times 10^{-}$ |
| MOI | 7.7×10^{-7} | 2.3×10^{-5} | 5.8×10^{-6} | 6.4×10^{-7} | $<1.0 \times 10^{-9}$ | $1.6 \times 10^{-}$ |
| NPA | 7.7×10^{-7} | 2.3×10^{-5} | 5.8×10^{-6} | 6.4×10^{-7} | $<1.0 \times 10^{-9}$ | $1.6 \times 10^{-}$ |

Table F.2.4.1-2. Summary of chemical concentrations for normal operations at Hanfo

CHEMICAL CONCENTRATIONS mg/m³

| | Carbon Monoxide | Sulfur Dioxide | Nitric Oxide | Nitrogen Dioxide | Lead | VO |
|----------------|----------------------|----------------------|----------------------|----------------------|----------------------|----|
| Worker | 2.9×10^{-5} | 1.5×10^{-4} | 1.3×10^{-4} | 1.0×10^{-5} | 3.0×10^{-9} | 1. |
| MCW | 1.6×10^{-5} | 2.1×10^{-4} | 9.6×10^{-5} | 1.1×10^{-5} | 5.0×10^{-9} | 4. |
| MOI (New ECF)* | 1.0×10^{-6} | 3.2×10^{-5} | 8.0×10^{-6} | 8.9×10^{-7} | 1.0×10^{-9} | 2. |
| MOI (FMEF)** | 1.4×10^{-6} | 4.0×10^{-5} | 1.1×10^{-5} | 1.2×10^{-6} | 1.0×10^{-9} | 3. |
| NPA | 1.3×10^{-6} | 4.1×10^{-5} | 1.0×10^{-5} | 1.1×10^{-6} | 1.0×10^{-9} | 2. |

*MOI (New ECF) applies if spent fuel facility is constructed at the 200 Area on th

**MOI (FMEF) applies if spent fuel facility is constructed at the Fuels and Materia

Table F.2.4.1-3. Summary of chemical concentrations for normal operations at Savan

CHEMICAL CONCENTRATIONS mg/m³

| | Carbon Monoxide | Sulfur Dioxide | Nitric Oxide | Nitrogen Dioxide | Lead | V |
|--------|----------------------|----------------------|----------------------|----------------------|-----------------------|---|
| Worker | 1.5×10^{-5} | 6.4×10^{-5} | 6.4×10^{-5} | 7.1×10^{-6} | 1.0×10^{-9} | 6 |
| MCW | 9.4×10^{-6} | 1.6×10^{-4} | 5.7×10^{-5} | 6.3×10^{-6} | 3.0×10^{-9} | 2 |
| MOI | 1.8×10^{-6} | 4.8×10^{-5} | 1.3×10^{-5} | 1.4×10^{-6} | 1.0×10^{-9} | 3 |
| NPA | 8.6×10^{-7} | 2.4×10^{-5} | 6.3×10^{-6} | 7.0×10^{-7} | $<1.0 \times 10^{-9}$ | 1 |

Table F.2.4.1-4. Summary of chemical concentrations for normal operations at the N

CHEMICAL CONCENTRATIONS mg/m³

| | Carbon Monoxide | Sulfur Dioxide | Nitric Oxide | Nitrogen Dioxide | Lead | V |
|--------|----------------------|----------------------|----------------------|----------------------|-----------------------|---|
| Worker | 9.0×10^{-5} | 3.6×10^{-4} | 4.0×10^{-4} | 4.5×10^{-5} | 7.0×10^{-9} | 3 |
| MCW | 2.5×10^{-7} | 7.3×10^{-6} | 1.9×10^{-6} | 2.1×10^{-7} | $<1.0 \times 10^{-9}$ | 5 |
| MOI | 7.9×10^{-7} | 2.3×10^{-5} | 5.9×10^{-6} | 6.6×10^{-7} | $<1.0 \times 10^{-9}$ | 1 |

Table F.2.4.1-5. Summary of chemical concentrations for normal operations at Oak R

CHEMICAL CONCENTRATIONS mg/m³

Carbon Sulfur Nitric Nitrogen

| | Monoxide | Dioxide | Oxide | Dioxide | Lead | V |
|--------|----------------------|----------------------|----------------------|----------------------|-----------------------|---|
| Worker | 6.4×10^{-5} | 3.0×10^{-4} | 2.8×10^{-4} | 3.1×10^{-5} | 5.0×10^{-9} | 2 |
| MCW | 1.6×10^{-6} | 2.6×10^{-5} | 9.6×10^{-6} | 1.1×10^{-6} | $<1.0 \times 10^{-9}$ | 5 |
| MOI | 1.4×10^{-5} | 2.5×10^{-4} | 8.8×10^{-5} | 9.8×10^{-6} | 4.0×10^{-9} | 4 |
| NPA | 1.9×10^{-5} | 3.1×10^{-4} | 1.1×10^{-4} | 1.2×10^{-5} | 5.0×10^{-9} | 5 |

Table F.2.4.1-6. Summary of chemical concentrations for normal operations at the B

| CHEMICAL CONCENTRATIONS mg/m3 | | | | | | |
|-------------------------------|----------------------|----------------------|----------------------|----------------------|----------------------|---|
| | Carbon Monoxide | Sulfur Dioxide | Nitric Oxide | Nitrogen Dioxide | Lead | V |
| Worker | 1.5×10^{-5} | 6.5×10^{-5} | 6.4×10^{-5} | 7.1×10^{-6} | 1.0×10^{-9} | 6 |
| MCW | 1.9×10^{-6} | 4.7×10^{-5} | 1.3×10^{-5} | 1.5×10^{-6} | 1.0×10^{-9} | 4 |
| MOI | 5.9×10^{-6} | 1.4×10^{-4} | 4.0×10^{-5} | 4.5×10^{-6} | 2.0×10^{-9} | 1 |
| NPA | 5.9×10^{-6} | 1.4×10^{-4} | 4.0×10^{-5} | 4.5×10^{-6} | 2.0×10^{-9} | 1 |

F.2.4.2 Accidents. Spillage of chemicals with a subsequent fire was evaluated for the bounding

accident involving toxic chemicals. The toxic chemicals that could be involved in accident are described in Section F.2.1. As was noted in that section, the extensive chemicals provided in Table F.2-1 would be applicable only at sites involved with fire. The bounding accident evaluated for spent nuclear fuel storage in water pools at shipyard was a diesel fuel spill and fire. A diesel fuel fire involving spent nuclear fuel aboard a ship at sea in Puget Sound was also evaluated.

Evaluation of the chemical spill with fire accident (excluding diesel fuel) at INEL-ECF, Hanford, Savannah River, Nevada Test Site, Oak Ridge, and the Barnwell Plant. Naval spent nuclear fuel examinations may be conducted is presented in Section F.2. of diesel fuel fires at shipyards and aboard ship in Puget Sound, as well as at INEL Savannah River, Nevada Test Site, Barnwell Plant, and Oak Ridge, is described in Section F.2.

These accidents incorporate spillage of the entire amount of a given chemical in a fire. The initiating event might be, for example, an airplane crash or ship collision accident bounds simpler chemical spills, such as handling accidents involving limited Table F.2-1) amounts of a chemical, which were also considered. Consequently, only fire accident are provided. The analyses utilize meteorological (see Section F.1.3) parameters specific to the evaluated location.

The toxic chemicals evaluated in the accident analyses would be used and stored in different areas within the facility. Fuel oils, sulfuric acid, and sodium hydroxide are expected to be located outside facility buildings in storage tanks. Other chemical support and operation would likely be stored in a variety of locations within facility tool rooms, laboratories, craft shops, equipment rooms, chemical mixing areas, hot flammable cabinets. The probability of releasing all or most of these chemicals in such as an airplane crash would be quite low, less than 10^{-7} per year, as supported by historical data. However, the probability of releasing an individual or limited number of chemicals greater than this level and include a consideration of storage locations, types, sizes of containers, and types and frequencies of initiating events. For accidents that could result in chemical release, a probability of 5×10^{-3} per year (Ganti and Krasner 1984) was considered a reasonable upper level. This level was based on the probability that a structural failure could occur.

F.2.4.2.1 Chemical Spill and Fire.

F.2.4.2.1.1 Accident Description. An accident might occur which caused toxic

chemicals to spill, dispersed powdered toxic chemicals, and accelerated the vaporization of chemicals with a subsequent fire. The airborne release resulting from the involvement of available amount of the toxic chemicals was evaluated with respect to the on-site workers, NPA, and MOI.

F.2.4.2.1.2 Source Term. The toxic chemicals involved in this hypothetical accident are

provided in Table F.2-1. The entire amount of the toxic chemical might be involved in the catastrophic nature of this accident.

F.2.4.2.1.3 Conditions and Key Parameters.

- (1) Gases
 - 100% of the gas was released to the atmosphere.
 - Release period was 10 minutes.
 - Release was a point source.
 - Deposition velocity was 0.1 centimeter per second.
- (2) Liquids
 - 100% of the liquid was released to the atmosphere.
 - The liquid was released into a pool of 0.1-inch depth.
 - The liquid was at its boiling point.
 - The release period was the longer of the calculated evaporation time minutes.
 - Release area was equal to the pool area.
 - Deposition velocity was 0.1 centimeter per second.
- (3) Solids
 - 1% of the solid was dispersed into the atmosphere as PM-10.
 - Release period was 10 minutes.
 - Release was a point source.
 - Deposition velocity was 1.0 centimeter per second.
- (4) Specific Chemicals
 - CFC-12 could break down at elevated temperatures into hydrochloric acid (10%) and phosgene (1%) with the remaining (89%) released as CFC-12.
 - The hypothetical sulfuric acid spill would be contained by a berm resulting in a pool release area of 443.2 square feet.
 - The hypothetical spill of sodium hydroxide was in the form of an aqueous solution and was contained by a berm resulting in a pool release area of 443.2 square feet. A 10-minute period was used for this release, and the hydroxide was dispersed as a particulate.
- (5) Meteorology
 - Wind speeds and atmospheric stability classifications used for the calculations were based on both 50% and 95% meteorology (Section F.1.3.5) to estimate downwind concentrations. The 95% meteorology included atmospheric stability classes A through F and wind speeds from 1.1 to 30 miles per hour.
- (6) General
 - Standard rural terrain was used since this most closely resembles the terrain evaluated.
 - Release was calculated to occur at ground level.
 - No evacuation of downwind populations was included, in order to obtain maximum estimates of effects; therefore, exposures were not reduced to account for this action.
 - No credit was taken for building containment or filtration.
 - Biological effects of exposure to each chemical were treated separately. It was done to account for a lack of a current methodology to evaluate the resulting from simultaneous multiple chemical exposures.
 - To determine health impacts, the estimated concentrations were compared against the Emergency Response Planning Guidelines (ERPG) levels 1, 2, and 3, concentration limits or alternates.
 - To determine the likelihood of developing cancer from exposure to hydrocarbons, a slope factor of 1.7×10^1 per mg/kg-day obtained from IRIS (TOXnet 1) was used. In addition, the exposure time was based on the duration of the release and individual breathing rates and sizes were the same as those used in F.1 for radiological accident evaluations using the Radiological Safety Computer Program (RSAC-5) (Wenzel 1993).

F.2.4.2.1.4 Results. The airborne concentrations, averaged over the duration of each

exposure, were calculated using EPIcode for the alternate locations for the worker, MOI for both 50% and 95% meteorology. The airborne concentrations were compared to ERPG values where available. However, ERPG values have not been derived for some of the chemicals. The effects of these substances were assessed by comparison with other chemicals for toxic effects as discussed in Section F.2.3.3.

Tables F.2.4.2-1 through -12 list the downwind concentrations at various locations corresponding ERPG values (or equivalent if TLV-TWA and IDLH concentrations are available). Hydrochloric acid and phosgene, from decomposition of CFC-12, sulfuric acid, and sodium hydroxide dominate the toxic chemical effects for on-site personnel. Concentrations of chemicals above ERPG-3 levels might result in life-threatening effects. However, the ERPG-3 level was exceeded for any member of the general public except for Oak Ridge where acid concentrations could exceed ERPG-3 levels under both 50% and 95% meteorological conditions. For the on-site workers, collocated workers, and any member of the general public could be exposed to toxic chemicals at levels above ERPG-3, it is expected that actual exposures would be much less due to the mitigative measures that would be implemented (F.2.4.3).

Additional information on the toxic properties for the chemicals that dominate is provided below.

Hydrochloric acid is an irritant to the respiratory tract, skin, eyes, and mucous membranes. More severe exposures result in pulmonary edema, and often laryngeal spasm. A concentration of 53 mg/m³ causes irritation of the throat after short exposure. Concentrations of 7 mg/m³ are tolerable for 1 hour; concentrations of 1,500-3,000 mg/m³ are dangerous, even for brief exposures (TOXnet 1993).

Phosgene, also known as carbonyl chloride, is a highly toxic, corrosive liquid with a boiling point of 9.8°C. It is toxic from intakes by inhalation, ingestion, and dermal absorption. Exposure may include contact burns to the skin and eyes, shortness of breath, chest pain, pulmonary edema, and death. At low vapor concentrations, it smells like musty hay. At higher concentrations, it has a sharp and pungent odor. It is a severe irritant to the eyes and can be fatal if inhaled, even for short durations and at low concentrations. Exposure of 5 mg/cm³ can result in immediate irritation of the respiratory tract. 80 mg/m³ may cause irritation within 2 minutes; 100 mg/m³ for as little as 30 minutes is very dangerous; and 360 mg/m³ is fatal for exposures of 30 minutes or less (TOXnet 1993).

Sulfuric acid mist can be strongly irritating to the skin, eyes, mucous membranes, and the respiratory tract. Odor may be detected at concentrations of 1 mg/m³; irritating effects at concentrations of 1.1 mg/m³. Inhalation of concentrations near 3 mg/m³ may cause cough, air passage and choking sensations. At higher concentrations and durations of exposure can cause pulmonary edema, emphysema, and permanent changes in pulmonary function (TOXnet 1993).

Sodium hydroxide dust can be irritating to the upper respiratory system. Irritation may occur at concentrations of 2 mg/m³. At higher concentrations and durations of exposure, inhalation can cause extreme irritation of the respiratory tract and permanent changes in pulmonary function (TOXnet 1993).

Table F.2.4.2-1. Summary of chemical concentrations for chemical spill and fire at 50% METEOR

| | CHEMICAL CONCENTRATIONS mg/m ³ - 50% METEOR | | | |
|--------|--|------------------------|-------------|------------------------|
| | Ethylene | Glycol | Hydrazine | Isopropyl Alcohol |
| | | ERPG-1 127 | ERPG-1 0.13 | ERPG-1 983 |
| | | ERPG-2 * | ERPG-2 10 | ERPG-2 2950 |
| | | ERPG-3 * | ERPG-3 100 | ERPG-3 29500 |
| Worker | 3300 | 49 | 890 | 38 |
| MCW | 2.3 | 1.6 x 10 ⁻² | 0.45 | 1.2 x 10 ⁻² |
| MOI | 1.5 | 1.0 x 10 ⁻² | 0.29 | 7.9 x 10 ⁻³ |
| NPA | 1.6 | 1.1 x 10 ⁻² | 0.30 | 8.3 x 10 ⁻³ |

Table F.2.4.2-2. Summary of chemical concentrations for chemical spill and fire at 95% METEOR

| | CHEMICAL CONCENTRATIONS mg/m ³ - 95% METEOR | | | |
|--------|--|------------------------|-------------|------------------------|
| | Ethylene | Glycol | Hydrazine | Isopropyl Alcohol |
| | | ERPG-1 127 | ERPG-1 0.13 | ERPG-1 983 |
| | | ERPG-2 * | ERPG-2 10 | ERPG-2 2950 |
| | | ERPG-3 * | ERPG-3 100 | ERPG-3 29500 |
| Worker | 4400 | 58 | 2200 | 150 |
| MCW | 7.6 | 4.8 x 10 ⁻² | 2.6 | 8.3 x 10 ⁻² |
| MOI | 3.6 | 2.3 x 10 ⁻² | 1.1 | 3.2 x 10 ⁻² |
| NPA | 3.6 | 2.3 x 10 ⁻² | 1.1 | 3.2 x 10 ⁻² |

*IDLH concentrations are not available; therefore, corresponding ERPG-2 and -3 levels are used.
Table F.2.4.2-3. Summary of chemical concentrations for chemical spill and fire at 50% METEOR

| | Ethylene | Isopropyl |
|--|----------|-----------|
|--|----------|-----------|

| | | Glycol | | Hydrazine | | Alcohol | |
|--------|------|----------------------|-----|-----------|------|----------------------|-------|
| | | ERPG-1 | 127 | ERPG-1 | 0.13 | ERPG-1 | 983 |
| | | ERPG-2 | * | ERPG-2 | 10 | ERPG-2 | 2950 |
| | | ERPG-3 | * | ERPG-3 | 100 | ERPG-3 | 29500 |
| Worker | 1500 | 19 | | 370 | | 14 | |
| MCW | 32 | 0.25 | | 6.6 | | 0.19 | |
| MOI | 1.3 | 8.7×10^{-3} | | 0.24 | | 6.7×10^{-3} | |
| NPA | 1.3 | 8.7×10^{-3} | | 0.24 | | 6.7×10^{-3} | |

Table F.2.4.2-4. Summary of chemical concentrations for chemical spill and fire at
CHEMICAL CONCENTRATIONS mg/m³ - 95% METEOROLOGY

| | | Ethylene | | Glycol | | Hydrazine | | Isopropyl Alcohol | |
|--------|------|----------------------|-----|--------|------|----------------------|-------|-------------------|-------|
| | | ERPG-1 | 127 | ERPG-1 | 0.13 | ERPG-1 | 983 | ERPG-1 | 983 |
| | | ERPG-2 | * | ERPG-2 | 10 | ERPG-2 | 2950 | ERPG-2 | 2950 |
| | | ERPG-3 | * | ERPG-3 | 100 | ERPG-3 | 29500 | ERPG-3 | 29500 |
| Worker | 4400 | 58 | | 2200 | | 150 | | | |
| MCW | 220 | 1.6 | | 85 | | 4.0 | | | |
| MOI | 4.9 | 3.0×10^{-2} | | 1.6 | | 4.7×10^{-2} | | | |
| NPA | 4.9 | 3.0×10^{-2} | | 1.6 | | 4.7×10^{-2} | | | |

*IDLH concentrations are not available; therefore, corresponding ERPG-2 and -3 levels
Table F.2.4.2-5. Summary of chemical concentrations for chemical spill and fire at

| | | Ethylene | | Glycol | | Hydrazine | | Isopropyl Alcohol | |
|-----------------|------|----------------------|-----|----------------------|------|----------------------|-------|-------------------|-------|
| | | ERPG-1 | 127 | ERPG-1 | 0.13 | ERPG-1 | 983 | ERPG-1 | 983 |
| | | ERPG-2 | * | ERPG-2 | 10 | ERPG-2 | 2950 | ERPG-2 | 2950 |
| | | ERPG-3 | * | ERPG-3 | 100 | ERPG-3 | 29500 | ERPG-3 | 29500 |
| Worker | 1500 | 19 | | 370 | | 14 | | | |
| MCW | 46 | 0.36 | | 9.6 | | 0.28 | | | |
| MOI (New ECF)** | 0.73 | 5.1×10^{-3} | | 8.1×10^{-2} | | 3.9×10^{-3} | | | |
| MOI (FMEF)*** | 0.97 | 7.1×10^{-3} | | 0.19 | | 5.4×10^{-3} | | | |
| NPA | 1.5 | 9.9×10^{-3} | | 0.29 | | 7.9×10^{-3} | | | |

Table F.2.4.2-6. Summary of chemical concentrations for chemical spill and fire at
CHEMICAL CONCENTRATIONS mg/m³ - 95% METE

| | | Ethylene | | Glycol | | Hydrazine | | Isopropyl Alcohol | |
|-----------------|------|----------------------|-----|--------|------|----------------------|-------|-------------------|-------|
| | | ERPG-1 | 127 | ERPG-1 | 0.13 | ERPG-1 | 983 | ERPG-1 | 983 |
| | | ERPG-2 | * | ERPG-2 | 10 | ERPG-2 | 2950 | ERPG-2 | 2950 |
| | | ERPG-3 | * | ERPG-3 | 100 | ERPG-3 | 29500 | ERPG-3 | 29500 |
| Worker | 4400 | 58 | | 2200 | | 150 | | | |
| MCW | 150 | 1.1 | | 55 | | 2.5 | | | |
| MOI (New ECF)** | 2.1 | 1.3×10^{-2} | | 0.47 | | 1.3×10^{-2} | | | |
| MOI (FMEF)*** | 5.5 | 3.5×10^{-2} | | 1.8 | | 5.4×10^{-2} | | | |
| NPA | 5.3 | 3.3×10^{-2} | | 1.7 | | 5.1×10^{-2} | | | |

*IDLH concentrations are not available; therefore, corresponding ERPG-2 and -3 levels

**MOI (New ECF) applies if spent fuel facility is constructed at the 200 Area on the

***MOI (FMEF) applies if spent fuel facility is constructed at the Fuels and Materials

Table F.2.4.2-7. Summary of chemical concentrations for chemical spill and fire at

| | | Ethylene | | Glycol | | Hydrazine | | Isopropyl Alcohol | |
|--------|------|----------------------|-----|----------------------|------|----------------------|-------|-------------------|-------|
| | | ERPG-1 | 127 | ERPG-1 | 0.13 | ERPG-1 | 983 | ERPG-1 | 983 |
| | | ERPG-2 | * | ERPG-2 | 10 | ERPG-2 | 2950 | ERPG-2 | 2950 |
| | | ERPG-3 | * | ERPG-3 | 100 | ERPG-3 | 29500 | ERPG-3 | 29500 |
| Worker | 530 | 6.8 | | 130 | | 5.1 | | | |
| MCW | 0.22 | 1.5×10^{-3} | | 4.1×10^{-2} | | 1.1×10^{-3} | | | |
| MOI | 0.74 | 5.4×10^{-3} | | 0.14 | | 4.0×10^{-3} | | | |

Table F.2.4.2-8. Summary of chemical concentrations for chemical spill and fire at
CHEMICAL CONCENTRATIONS mg/m³ - 95% METEOR

| | | Ethylene | | Glycol | | Hydrazine | | Isopropyl Alcohol | |
|--|--|----------|-----|--------|------|-----------|-------|-------------------|-------|
| | | ERPG-1 | 127 | ERPG-1 | 0.13 | ERPG-1 | 983 | ERPG-1 | 983 |
| | | ERPG-2 | * | ERPG-2 | 10 | ERPG-2 | 2950 | ERPG-2 | 2950 |
| | | ERPG-3 | * | ERPG-3 | 100 | ERPG-3 | 29500 | ERPG-3 | 29500 |

| | | | | |
|--------|------|------------------------|------|------------------------|
| Worker | 4400 | 58 | 2200 | 150 |
| MCW | 5.9 | 3.7 x 10 ⁻² | 1.9 | 5.8 x 10 ⁻² |
| MOI | 7.3 | 4.6 x 10 ⁻² | 2.5 | 7.8 x 10 ⁻² |

*IDLH concentrations are not available; therefore, corresponding ERPG-2 and -3 levels are used.

Table F.2.4.2-9. Summary of chemical concentrations for chemical spill and fire at

| CHEMICAL CONCENTRATIONS mg/m ³ - 50% METEOR | | | | |
|--|------------|------|-------------|-------------------|
| | Ethylene | | | Isopropyl Alcohol |
| | Glycol | | Hydrazine | |
| | ERPG-1 127 | | ERPG-1 0.13 | ERPG-1 983 |
| | ERPG-2 * | | ERPG-2 10 | ERPG-2 2950 |
| | ERPG-3 * | | ERPG-3 100 | ERPG-3 29500 |
| Worker | 3300 | 49 | 890 | 38 |
| MCW | 34 | 0.27 | 7.1 | 0.21 |
| MOI | 310 | 2.8 | 68 | 2.1 |
| NPA | 440 | 4.3 | 100 | 3.2 |

Table F.2.4.2-10. Summary of chemical concentrations for chemical spill and fire at

CHEMICAL CONCENTRATIONS mg/m³ - 95% METEOR

| CHEMICAL CONCENTRATIONS mg/m ³ - 95% METEOR | | | | |
|--|------------|------|-------------|-------------------|
| | Ethylene | | | Isopropyl Alcohol |
| | Glycol | | Hydrazine | |
| | ERPG-1 127 | | ERPG-1 0.13 | ERPG-1 983 |
| | ERPG-2 * | | ERPG-2 10 | ERPG-2 2950 |
| | ERPG-3 * | | ERPG-3 100 | ERPG-3 29500 |
| Worker | 4400 | 58 | 2200 | 150 |
| MCW | 110 | 0.75 | 41 | 1.9 |
| MOI | 930 | 8.4 | 400 | 22 |
| NPA | 1300 | 13 | 590 | 33 |

*IDLH concentrations are not available; therefore, corresponding ERPG-2 and -3 levels are used.

Table F.2.4.2-11. Summary of chemical concentrations for chemical spill and fire at

| CHEMICAL CONCENTRATIONS mg/m ³ - 50% METEOR | | | | |
|--|------------|------------------------|-------------|------------------------|
| | Ethylene | | | Isopropyl Alcohol |
| | Glycol | | Hydrazine | |
| | ERPG-1 127 | | ERPG-1 0.13 | ERPG-1 983 |
| | ERPG-2 * | | ERPG-2 10 | ERPG-2 2950 |
| | ERPG-3 * | | ERPG-3 100 | ERPG-3 29500 |
| Worker | 1500 | 19 | 370 | 14 |
| MCW | 0.89 | 6.4 x 10 ⁻³ | 0.17 | 4.9 x 10 ⁻³ |
| MOI | 6.1 | 4.3 x 10 ⁻² | 1.2 | 3.4 x 10 ⁻² |
| NPA | 6.1 | 4.3 x 10 ⁻² | 1.2 | 3.4 x 10 ⁻² |

Table F.2.4.2-12. Summary of chemical concentrations for chemical spill and fire at

CHEMICAL CONCENTRATIONS mg/m³ - 95% METEOR

| CHEMICAL CONCENTRATIONS mg/m ³ - 95% METEOR | | | | |
|--|------------|------------------------|-------------|-------------------|
| | Ethylene | | | Isopropyl Alcohol |
| | Glycol | | Hydrazine | |
| | ERPG-1 127 | | ERPG-1 0.13 | ERPG-1 983 |
| | ERPG-2 * | | ERPG-2 10 | ERPG-2 2950 |
| | ERPG-3 * | | ERPG-3 100 | ERPG-3 29500 |
| Worker | 4400 | 58 | 2200 | 150 |
| MCW | 11 | 6.4 x 10 ⁻² | 3.5 | 0.13 |
| MOI | 28 | 0.18 | 10 | 0.41 |
| NPA | 28 | 0.18 | 10 | 0.41 |

*IDLH concentrations are not available; therefore, corresponding ERPG-2 and -3 levels are used.

In addition to comparing the airborne concentrations to their respective ERPG appropriate values, each substance was evaluated to determine if it has the potential carcinogenic health impacts. It was determined that exposure to hydrazine could result in an increased likelihood for developing cancer. Tables F.2.4.2-13 and F.2.4.2-14 provide the potential likelihood for developing cancer from exposure to hydrazine for the workers at the alternate locations under 50% and 95% meteorological conditions, respectively. Table F.2.4.2-13. Future potential likelihood for developing cancer from hydrazine under 50% meteorology.

| | INEL | Savannah | Hanford* | Nevada | Oak Ridge | B |
|--------|------------------------|------------------------|------------------------|------------------------|------------------------|---|
| | Expended | FacilitRiver | | Test | | P |
| Worker | 9.3 x 10 ⁻⁵ | 3.6 x 10 ⁻⁵ | 3.6 x 10 ⁻⁵ | 1.3 x 10 ⁻⁵ | 9.3 x 10 ⁻⁵ | 3 |
| MCW | 3.0 x 10 ⁻⁸ | 4.8 x 10 ⁻⁷ | 6.8 x 10 ⁻⁷ | 2.8 x 10 ⁻⁹ | 5.1 x 10 ⁻⁷ | 1 |
| MOI | 1.5 x 10 ⁻⁸ | 1.3 x 10 ⁻⁸ | 7.6 x 10 ⁻⁹ | 8.1 x 10 ⁻⁹ | 4.2 x 10 ⁻⁶ | 6 |

Table F.2.4.2-14. Future potential likelihood for developing cancer from hydrazine 95% meteorology.

| | INEL | | | Nevada | | |
|--------|----------------------|----------------------|----------------------|----------------------|----------------------|---|
| | Expended | Savannah | | Test | Oak | B |
| | Core Facilit | River | Hanford* | Site | Ridge | P |
| Worker | 3.8×10^{-4} | 3.8×10^{-4} | 3.8×10^{-4} | 3.8×10^{-4} | 3.8×10^{-4} | 3 |
| MCW | 2.0×10^{-7} | 6.7×10^{-6} | 4.6×10^{-6} | 1.6×10^{-7} | 3.2×10^{-6} | 2 |
| MOI | 7.8×10^{-8} | 1.0×10^{-7} | 4.4×10^{-8} | 1.6×10^{-7} | 2.9×10^{-5} | 6 |

* MOI shown applies to new ECF if spent fuel facility is constructed at the 200 Are Site. A future potential carcinogenic risk of 1.1×10^{-8} (50% meteorology) and 1 meteorology) applies to a spent fuel facility constructed at the Fuels and Materit ion Facility.

F.2.4.2.2 Fire Involving Diesel Fuel.

F.2.4.2.2.1 Accident Description. A catastrophic failure of the diesel fuel storage tank

facility was postulated to occur. This could result in the spilling of the entire and a subsequent fire. The airborne release of toxic chemicals resulting from the with respect to the on-site workers, MCW, NPA, and MOI as applicable for the accide

F.2.4.2.2.2 Source Term. The material involved in this accident was diesel fuel with the

fire generating the following toxic chemicals due to combustion:

- Carbon monoxide
- Oxides of nitrogen (90% nitric oxide and 10% nitrogen dioxide)
- Lead
- Sulfur dioxide.

F.2.4.2.2.3 Conditions and Key Parameters.

- For alternate DOE sites and the Barnwell Plant, the diesel fuel was stored storage tanks.
- For shipyards, the diesel fuel was stored in a portable diesel power unit.
- For the ship accident, the diesel fuel was stored in large tanks adjacent
- For alternate DOE sites and the Barnwell Plant, 1950 gallons of diesel fue spilled.
- For shipyards, 315 gallons of diesel fuel could be spilled.
- For the ship accident, 121,000 gallons of diesel fuel could be spilled.
- For all facilities, the entire quantity of diesel fuel was spilled and ign
- For alternate DOE sites and the Barnwell Plant, the spill area was 261 squ
- For shipyards, the spill area was 66 square feet.
- For the ship accident, the spill area used was 4812 square feet.
- For alternate DOE sites and the Barnwell Plant, the entire amount of diese consumed by the fire over a 2-hour period.
- For shipyards, the entire amount of diesel fuel was consumed by the fire o period.
- For the ship accident, the entire amount of diesel fuel was consumed by th 6-hour period.
- For all facilities, the releases per gallon of fuel burned were as follows
Carbon monoxide = 0.34 pound
Oxides of nitrogen = 1.58 pounds
Lead = 4.2×10^{-6} pound
Sulfur dioxide = 0.105 pound.
- For alternate DOE sites, the Barnwell Plant, and shipyards, the airborne r chemicals occurred at ground level.
- For the ship accident, the airborne release of toxic chemicals occurred at the sea (i.e., at the middle of the flame height above the cargo hatch) fo land-based exposures. For shipboard exposures, a release height of zero w
- For all facilities, standard rural terrain was used and building wake effe considered.

- For all facilities, wind speeds and atmospheric stability classifications both 50% and 95% meteorology (Section F.1.3.5).
- For all facilities, no evacuation of downwind populations occurred and the effects of chemical exposure act uniquely and do not affect the individual way.
- For all facilities, to determine the health impacts, the estimated concentration compared against the Emergency Response Planning Guidelines (ERPG) levels 3 concentration limits or alternates.

F.2.4.2.4 Results. The airborne concentrations, averaged over the duration of each

exposure, were calculated using EPIcode for the combustion products resulting from worker, MCW, NPA, and MOI (as applicable for the accident site) under both 50% and meteorology. The airborne concentrations were compared to respective ERPG values w available. However, ERPG values have not been derived for some of the constituents effects of these constituents were assessed by comparison with other appropriate va effects as discussed in Section F.2.3.3.

Tables F.2.4.2-15 through -38 list the downwind concentrations at various loc corresponding ERPG (or equivalent) values. Results for the diesel fuel fire at fue indicate that the toxic chemical concentrations for sulfur dioxide and oxides of ni ERPG-3 levels for the worker. At Savannah River and Hanford, the MCW also may be e nitric oxide concentration exceeding ERPG-3 levels under 95% meteorological conditi and MOI exposures at all the fuel examination sites would be expected to be below E except for Oak Ridge. At this location under 95% meteorological conditions, the NP be exposed to concentrations of sulfur dioxide and oxides of nitrogen that exceed E concentrations of carbon monoxide that exceed ERPG-2. Under 50% meteorological con Oak Ridge, the NPA and MOI may be exposed to concentrations of nitric oxide that ex and concentrations of sulfur dioxide and nitrogen dioxide that exceed ERPG-2. Resu fuel fire at shipyards show that for the worker and NPA categories, the toxic chemi for sulfur dioxide and oxides of nitrogen may exceed ERPG-3 levels. For the MOI, h concentrations are expected to be less than the ERPG-3 levels with the exception th meteorological conditions the ERPG-3 level for nitric oxide may be exceeded at the Results for the ship diesel fuel fire show that shipboard (worker) concentrations o sulfur dioxide, and oxides of nitrogen may exceed ERPG-3 levels, but the shore (MOI are expected to be less than ERPG-3 levels. For the individuals on board the ship exposed to toxic chemicals at levels above ERPG-3, it is expected that actual toxic would be much less due to the mitigative measures that would be implemented (Sectio

Additional information on the toxic properties for the chemicals that dominat is provided below.

Sulfur dioxide is a colorless gas with a pungent odor. It is a poison, and i skin, and mucous membrane irritant. It chiefly affects the upper respiratory tract higher concentrations, sulfur dioxide causes respiratory paralysis (TOXnet 1993).

Nitric oxide and nitrogen dioxide occur together in dynamic equilibrium. Nit colorless gas, and nitrogen dioxide is a reddish brown gas. Both chemicals are eye membrane irritants and primarily affect the respiratory system. Exposure to 47 mg/ dioxide can cause respiratory irritation and chest pain, 93 mg/m³ can cause lung in 187 mg/m³ can be fatal (TOXnet 1993).

In addition to comparing the airborne concentrations to their respective ERPG appropriate values, each substance was evaluated to determine if it has the potenti carcinogenic impacts. It was determined that exposure to lead could result in an i for developing cancer. However, sufficient information to quantify this likelihood in IRIS. Therefore, the concentrations of lead resulting from the accident were co NAAQS value for lead. For the lead concentrations provided in Tables F.2.4.2-15 th F.2.4.2-38, no NAAQS is exceeded.

F.2.4.3 Mitigative Measures for Toxic Chemicals. Mitigative measures for potential releases of

toxic materials involve administrative controls for personnel protection and emerge personnel protection, controls involve safety review committees for planned activit requirements, safe work permits, and procedures for required clothing (rubber boots shields, eye protection) that can mitigate the effects of potential releases of tox Procedures may also require provisions for prestationing mitigative devices such as and emergency showers. All of the alternate facilities being evaluated employ emer

programs to mitigate impacts of potential toxic chemical accidents to workers and t
Emergency planning, emergency preparedness, and emergency response programs are in
involve established resources such as warning communications, fire departments, and
command centers. The cargo ships used for naval spent nuclear fuel have smoke dete
fighting equipment on board. They also have fire suppression systems in their hold
gas to smother fires. In addition, less freely available oxygen in the ship's carg
slow the combustion rate of the diesel fuel. Port facilities would also have avail
fighting equipment, public warning systems, and emergency response programs.
Table F.2.4.2-15. Summary of chemical concentrations for fire involving diesel fue
Expended Core Facility.

| CHEMICAL CONCENTRATIONS mg/m3 - 50% METEOROLOGY | | | | | | | |
|---|--------------------|------------------------|-----------------|------------------------|-------|----------|--|
| | Carbon Monoxide | Sulfur Dioxide | Nitric Oxide | Nitrogen Dioxide | Lead | | |
| ERP-1 | 29 | 0.79 | 31 | 5.6 | ERP-1 | | |
| ERP-2 | 172 | 7.9 | * | 9.4 | ERP-2 | | |
| ERP-3 | 1720 | 39 | 123 | 94 | ERP-3 | | |
| Worker | 480 | 150 | 2000 | 220 | | 3.9 x 10 | |
| MCW | 0.25 | 7.7 x 10 ⁻² | 1.0 | 0.11 | | 9.5 x 10 | |
| MOI | 0.15 | 4.8 x 10 ⁻² | 0.65 | 7.3 x 10 ⁻² | | 6.1 x 10 | |
| NPA | 0.16 | 5.0 x 10 ⁻² | 0.69 | 7.7 x 10 ⁻² | | 6.1 x 10 | |

Table F.2.4.2-16. Summary of chemical concentrations for fire involving diesel fue
Expended Core Facility.

| CHEMICAL CONCENTRATIONS mg/m3 - 95% METEOROLOGY | | | | | | | |
|---|--------------------|-------------------|-----------------|---------------------|-------|----------|--|
| | Carbon Monoxide | Sulfur Dioxide | Nitric Oxide | Nitrogen Dioxide | Lead | | |
| ERP-1 | 29 | 0.79 | 31 | 5.6 | ERP-1 | | |
| ERP-2 | 172 | 7.9 | * | 9.4 | ERP-2 | | |
| ERP-3 | 1720 | 39 | 123 | 94 | ERP-3 | | |
| Worker | 1200 | 370 | 5100 | 560 | | 4.6 x 10 | |
| MCW | 1.45 | 0.45 | 6.1 | 0.68 | | 3.0 x 10 | |
| MOI | 0.66 | 0.20 | 2.7 | 0.30 | | 4.7 x 10 | |
| NPA | 0.66 | 0.20 | 2.7 | 0.30 | | 4.7 x 10 | |

* ERP-2 level not assigned since one-tenth the IDLH level would be less than the
Table F.2.4.2-17. Summary of chemical concentrations for fire involving diesel fue
Savannah River.

| CHEMICAL CONCENTRATIONS mg/m3 - 50% METEOROLOGY | | | | | | | |
|---|--------------------|------------------------|-----------------|------------------------|-------|----------|--|
| | Carbon Monoxide | Sulfur Dioxide | Nitric Oxide | Nitrogen Dioxide | Lead | | |
| ERP-1 | 29 | 0.79 | 31 | 5.6 | ERP-1 | | |
| ERP-2 | 172 | 7.9 | * | 9.4 | ERP-2 | | |
| ERP-3 | 1720 | 39 | 123 | 94 | ERP-3 | | |
| Worker | 200 | 62 | 850 | 94 | | 2.0 x 10 | |
| MCW | 3.6 | 1.1 | 15 | 1.7 | | 3.6 x 10 | |
| MOI | 0.13 | 4.1 x 10 ⁻² | 0.55 | 6.1 x 10 ⁻² | | 7.5 x 10 | |
| NPA | 0.13 | 4.1 x 10 ⁻² | 0.55 | 6.1 x 10 ⁻² | | 7.5 x 10 | |

Table F.2.4.2-18. Summary of chemical concentrations for fire involving diesel fue
Savannah River.

| CHEMICAL CONCENTRATIONS mg/m3 - 95% METEOROLOGY | | | | | | | |
|---|--------------------|-------------------|-----------------|---------------------|-------|----------|--|
| | Carbon Monoxide | Sulfur Dioxide | Nitric Oxide | Nitrogen Dioxide | Lead | | |
| ERP-1 | 29 | 0.79 | 31 | 5.6 | ERP-1 | | |
| ERP-2 | 172 | 7.9 | * | 9.4 | ERP-2 | | |
| ERP-3 | 1720 | 39 | 123 | 94 | ERP-3 | | |
| Worker | 1200 | 370 | 5100 | 560 | | 4.6 x 10 | |
| MCW | 49 | 15 | 200 | 23 | | 6.9 x 10 | |
| MOI | 0.90 | 0.28 | 3.8 | 0.42 | | 1.1 x 10 | |
| NPA | 0.90 | 0.28 | 3.8 | 0.42 | | 1.1 x 10 | |

* ERP-2 level not assigned since one-tenth the IDLH level would be less than the
Table F.2.4.2-19. Summary of chemical concentrations for fire involving diesel fue
Hanford.

| CHEMICAL CONCENTRATIONS mg/m3 - 50% METEOROLOGY | | | | | | | |
|---|--------------------|-------------------|-----------------|---------------------|-------|---|--|
| | Carbon Monoxide | Sulfur Dioxide | Nitric Oxide | Nitrogen Dioxide | Lead | | |
| ERP-1 | 29 | 0.79 | 31 | 5.6 | ERP-1 | 0 | |

| | | | | | | | | | | |
|-------------------|--|-----|------------|-----|--------|-----|------------|-----|-----------|---|
| | ERPG-2 | 172 | ERPG-2 | 7.9 | ERPG-2 | * | ERPG-2 | 9.4 | ERPG-2 | 7 |
| | ERPG-3 | | ERPG-3 | 39 | ERPG-3 | 123 | ERPG-3 | 94 | ERPG-3 | 7 |
| | 1720 | | | | | | | | | |
| Worker | 200 | | 62 | | 840 | | 94 | | 2.0 x 10- | |
| MCW | 5.2 | | 1.6 | | 21 | | 2.4 | | 4.1 x 10- | |
| MOI (New ECF)** | 8.3 x 10-2 | | 2.4 x 10-2 | | 0.34 | | 3.7 x 10-2 | | 2.5 x 10- | |
| MOI (FMEF)*** | 0.11 | | 3.3 x 10-2 | | 0.44 | | 4.9 x 10-2 | | 8.1 x 10- | |
| NPA | 0.16 | | 4.8 x 10-2 | | 0.65 | | 7.3 x 10-2 | | 7.6 x 10- | |
| Table F.2.4.2-20. | Summary of chemical concentrations for fire involving diesel fue | | | | | | | | | |
| Hanford. | | | | | | | | | | |

CHEMICAL CONCENTRATIONS mg/m3 - 95% METEOROLOGY

| | | | | | | | | | | |
|-----------------|-----------------|-----|----------------|------|--------------|-----|------------------|-----|-----------|---|
| | Carbon Monoxide | | Sulfur Dioxide | | Nitric Oxide | | Nitrogen Dioxide | | Lead | |
| | ERPG-1 | 29 | ERPG-1 | 0.79 | ERPG-1 | 31 | ERPG-1 | 5.6 | ERPG-1 | 0 |
| | ERPG-2 | 172 | ERPG-2 | 7.9 | ERPG-2 | * | ERPG-2 | 9.4 | ERPG-2 | 7 |
| | ERPG-3 | | ERPG-3 | 39 | ERPG-3 | 123 | ERPG-3 | 94 | ERPG-3 | 7 |
| | 1720 | | | | | | | | | |
| Worker | 1200 | | 370 | | 5100 | | 560 | | 4.6 x 10- | |
| MCW | 32 | | 9.7 | | 130 | | 15 | | 3.9 x 10- | |
| MOI (New ECF)** | 0.34 | | 0.10 | | 1.4 | | 0.15 | | 4.9 x 10- | |
| MOI (FMEF)*** | 1.0 | | 0.32 | | 4.3 | | 0.48 | | 1.5 x 10- | |
| NPA | 0.78 | | 0.24 | | 3.2 | | 0.36 | | 5.0 x 10- | |

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the

** MOI (New ECF) applies if spent fuel facility is constructed at the 200 Area on

*** MOI (FMEF) applies if spent fuel facility is constructed at the Fuels and Mater
Table F.2.4.2-21. Summary of chemical concentrations for fire involving diesel fue
Nevada Test Site.

CHEMICAL CONCENTRATIONS mg/m3 - 50% METEOROLOGY

| | | | | | | | | | | |
|-------------------|--|------|----------------|------|--------------|-----|------------------|-----|----------|--|
| | Carbon Monoxide | | Sulfur Dioxide | | Nitric Oxide | | Nitrogen Dioxide | | Lead | |
| | ERPG-1 | 29 | ERPG-1 | 0.79 | ERPG-1 | 31 | ERPG-1 | 5.6 | ERPG-1 | |
| | ERPG-2 | 172 | ERPG-2 | 7.9 | ERPG-2 | * | ERPG-2 | 9.4 | ERPG-2 | |
| | ERPG-3 | 1720 | ERPG-3 | 39 | ERPG-3 | 123 | ERPG-3 | 94 | ERPG-3 | |
| | 73 | | 22 | | 300 | | 34 | | 8.3 x 10 | |
| Worker | 2.3 x 10-2 | | 7.0 x 10-3 | | 9.6 x 10-2 | | 1.1 x 10-2 | | 2.2 x 10 | |
| MCW | 8.0 x 10-2 | | 2.4 x 10-2 | | 0.33 | | 3.7 x 10-2 | | 7.3 x 10 | |
| MOI | | | | | | | | | | |
| Table F.2.4.2-22. | Summary of chemical concentrations for fire involving diesel fue | | | | | | | | | |
| Nevada Test Site. | | | | | | | | | | |

CHEMICAL CONCENTRATIONS mg/m3 - 95% METEOROLOGY

| | | | | | | | | | | |
|--------|-----------------|------|----------------|------|--------------|-----|------------------|-----|----------|--|
| | Carbon Monoxide | | Sulfur Dioxide | | Nitric Oxide | | Nitrogen Dioxide | | Lead | |
| | ERPG-1 | 29 | ERPG-1 | 0.79 | ERPG-1 | 31 | ERPG-1 | 5.6 | ERPG-1 | |
| | ERPG-2 | 172 | ERPG-2 | 7.9 | ERPG-2 | * | ERPG-2 | 9.4 | ERPG-2 | |
| | ERPG-3 | 1720 | ERPG-3 | 39 | ERPG-3 | 123 | ERPG-3 | 94 | ERPG-3 | |
| | 1200 | | 370 | | 5100 | | 560 | | 4.6 x 10 | |
| Worker | 1.1 | | 0.34 | | 4.6 | | 0.52 | | 1.7 x 10 | |
| MCW | 1.4 | | 0.43 | | 5.9 | | 0.65 | | 2.7 x 10 | |
| MOI | | | | | | | | | | |

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the

Table F.2.4.2-23. Summary of chemical concentrations for fire involving diesel fue
Oak Ridge.

CHEMICAL CONCENTRATIONS mg/m3 - 50% METEOROLOGY

| | | | | | | | | | | |
|-------------------|--|------|----------------|------|--------------|-----|------------------|-----|----------|--|
| | Carbon Monoxide | | Sulfur Dioxide | | Nitric Oxide | | Nitrogen Dioxide | | Lead | |
| | ERPG-1 | 29 | ERPG-1 | 0.79 | ERPG-1 | 31 | ERPG-1 | 5.6 | ERPG-1 | |
| | ERPG-2 | 172 | ERPG-2 | 7.9 | ERPG-2 | * | ERPG-2 | 9.4 | ERPG-2 | |
| | ERPG-3 | 1720 | ERPG-3 | 39 | ERPG-3 | 123 | ERPG-3 | 94 | ERPG-3 | |
| | 480 | | 150 | | 2000 | | 220 | | 3.9 x 10 | |
| Worker | 3.8 | | 1.2 | | 16 | | 1.8 | | 3.0 x 10 | |
| MCW | 37 | | 11 | | 150 | | 18 | | 3.3 x 10 | |
| MOI | 54 | | 17 | | 230 | | 26 | | 5.0 x 10 | |
| NPA | | | | | | | | | | |
| Table F.2.4.2-24. | Summary of chemical concentrations for fire involving diesel fue | | | | | | | | | |
| Oak Ridge. | | | | | | | | | | |

CHEMICAL CONCENTRATIONS mg/m3 - 95% METEOROLOGY

| | | | |
|--------|--------|--------|----------|
| Carbon | Sulfur | Nitric | Nitrogen |
|--------|--------|--------|----------|

| | Monoxide | | Dioxide | | Oxide | | Dioxide | | Lead |
|--------|-------------|--|-------------|--|------------|--|------------|--|----------|
| | ERPG-1 29 | | ERPG-1 0.79 | | ERPG-1 31 | | ERPG-1 5.6 | | ERPG-1 |
| | ERPG-2 172 | | ERPG-2 7.9 | | ERPG-2 * | | ERPG-2 9.4 | | ERPG-2 |
| | ERPG-3 1720 | | ERPG-3 39 | | ERPG-3 123 | | ERPG-3 94 | | ERPG-3 |
| Worker | 1200 | | 370 | | 5100 | | 560 | | 4.6 x 10 |
| MCW | 24 | | 7.3 | | 98 | | 11 | | 2.6 x 10 |
| MOI | 230 | | 70 | | 950 | | 110 | | 5.3 x 10 |
| NPA | 340 | | 100 | | 1400 | | 160 | | 8.7 x 10 |

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the Table F.2.4.2-25. Summary of chemical concentrations for fire involving diesel fuel Barnwell Plant.

| CHEMICAL CONCENTRATIONS mg/m3 - 50% METEOROLOGY | | | | | | | | | |
|---|------------------------|--|------------------------|--|--------------|--|------------------------|--|----------|
| | Carbon Monoxide | | Sulfur Dioxide | | Nitric Oxide | | Nitrogen Dioxide | | Lead |
| | ERPG-1 29 | | ERPG-1 0.79 | | ERPG-1 31 | | ERPG-1 5.6 | | ERPG-1 |
| | ERPG-2 172 | | ERPG-2 7.9 | | ERPG-2 * | | ERPG-2 9.4 | | ERPG-2 |
| | ERPG-3 1720 | | ERPG-3 39 | | ERPG-3 123 | | ERPG-3 94 | | ERPG-3 |
| Worker | 200 | | 62 | | 840 | | 94 | | 2.0 x 10 |
| MCW | 9.5 x 10 ⁻² | | 2.9 x 10 ⁻² | | 0.40 | | 4.4 x 10 ⁻² | | 9.3 x 10 |
| MOI | 0.65 | | 0.20 | | 2.7 | | 0.30 | | 5.0 x 10 |
| NPA | 0.65 | | 0.20 | | 2.7 | | 0.30 | | 5.0 x 10 |

Table F.2.4.2-26. Summary of chemical concentrations for fire involving diesel fuel Barnwell Plant.

| CHEMICAL CONCENTRATIONS mg/m3 - 95% METEOROLOGY | | | | | | | | | |
|---|-----------------|--|----------------|--|--------------|--|------------------|--|----------|
| | Carbon Monoxide | | Sulfur Dioxide | | Nitric Oxide | | Nitrogen Dioxide | | Lead |
| | ERPG-1 29 | | ERPG-1 0.79 | | ERPG-1 31 | | ERPG-1 5.6 | | ERPG-1 |
| | ERPG-2 172 | | ERPG-2 7.9 | | ERPG-2 * | | ERPG-2 9.4 | | ERPG-2 |
| | ERPG-3 1720 | | ERPG-3 39 | | ERPG-3 123 | | ERPG-3 94 | | ERPG-3 |
| Worker | 1200 | | 370 | | 5100 | | 560 | | 4.6 x 10 |
| MCW | 2.0 | | 0.62 | | 8.4 | | 0.94 | | 5.4 x 10 |
| MOI | 5.8 | | 1.7 | | 24 | | 2.7 | | 3.2 x 10 |
| NPA | 5.8 | | 1.7 | | 24 | | 2.7 | | 3.2 x 10 |

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the Table F.2.4.2-27. Summary of chemical concentrations for fire involving diesel fuel Kenneth A. Kesselring Site.

| CHEMICAL CONCENTRATIONS mg/m3 - 50% METEOROLOGY | | | | | | | | | |
|---|-----------------|--|------------------------|--|--------------|--|------------------|--|----------|
| | Carbon Monoxide | | Sulfur Dioxide | | Nitric Oxide | | Nitrogen Dioxide | | Lead |
| | ERPG-1 29 | | ERPG-1 0.79 | | ERPG-1 31 | | ERPG-1 5.6 | | ERPG-1 |
| | ERPG-2 172 | | ERPG-2 7.9 | | ERPG-2 * | | ERPG-2 9.4 | | ERPG-2 |
| | ERPG-3 1720 | | ERPG-3 39 | | ERPG-3 123 | | ERPG-3 94 | | ERPG-3 |
| Worker | 44 | | 13 | | 180 | | 20 | | 4.8 x 10 |
| MOI | 0.25 | | 7.7 x 10 ⁻² | | 1.0 | | 0.11 | | 2.3 x 10 |
| NPA | 0.25 | | 7.7 x 10 ⁻² | | 1.0 | | 0.11 | | 2.3 x 10 |

Table F.2.4.2-28. Summary of chemical concentrations for fire involving diesel fuel Kenneth A. Kesselring Site.

| CHEMICAL CONCENTRATIONS mg/m3 - 95% METEOROLOGY | | | | | | | | | |
|---|-----------------|--|----------------|--|--------------|--|------------------|--|----------|
| | Carbon Monoxide | | Sulfur Dioxide | | Nitric Oxide | | Nitrogen Dioxide | | Lead |
| | ERPG-1 29 | | ERPG-1 0.79 | | ERPG-1 31 | | ERPG-1 5.6 | | ERPG-1 |
| | ERPG-2 172 | | ERPG-2 7.9 | | ERPG-2 * | | ERPG-2 9.4 | | ERPG-2 |
| | ERPG-3 1720 | | ERPG-3 39 | | ERPG-3 123 | | ERPG-3 94 | | ERPG-3 |
| Worker | 500 | | 150 | | 2100 | | 230 | | 1.9 x 10 |
| MOI | 3.9 | | 1.2 | | 17 | | 1.8 | | 3.1 x 10 |
| NPA | 3.9 | | 1.2 | | 17 | | 1.8 | | 3.1 x 10 |

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the Table F.2.4.2-29. Summary of chemical concentrations for fire involving diesel fuel Norfolk Naval Shipyard.

| CHEMICAL CONCENTRATIONS mg/m3 - 50% METEOROLOGY | | | | | | | | | |
|---|-----------------|--|----------------|--|--------------|--|------------------|--|--------|
| | Carbon Monoxide | | Sulfur Dioxide | | Nitric Oxide | | Nitrogen Dioxide | | Lead |
| | ERPG-1 29 | | ERPG-1 0.79 | | ERPG-1 31 | | ERPG-1 5.6 | | ERPG-1 |
| | ERPG-2 172 | | ERPG-2 7.9 | | ERPG-2 * | | ERPG-2 9.4 | | ERPG-2 |

| | | | | | | | | | |
|--------|--------|------|--------|----|--------|-----|--------|----|----------|
| | ERPG-3 | 1720 | ERPG-3 | 39 | ERPG-3 | 123 | ERPG-3 | 94 | ERPG-3 |
| Worker | 44 | | 13 | | 180 | | 20 | | 4.8 x 10 |
| MOI | 4.3 | | 1.3 | | 18 | | 2.0 | | 4.7 x 10 |
| NPA | 4.3 | | 1.3 | | 18 | | 2.0 | | 4.7 x 10 |

Table F.2.4.2-30. Summary of chemical concentrations for fire involving diesel fuel Norfolk Naval Shipyard.

| CHEMICAL CONCENTRATIONS mg/m3 - 95% METEOROLOGY | | | | | | | | | |
|---|-----------------|------|----------------|------|--------------|-----|------------------|-----|----------|
| | Carbon Monoxide | | Sulfur Dioxide | | Nitric Oxide | | Nitrogen Dioxide | | Lead |
| | ERPG-1 | 29 | ERPG-1 | 0.79 | ERPG-1 | 31 | ERPG-1 | 5.6 | ERPG-1 |
| | ERPG-2 | 172 | ERPG-2 | 7.9 | ERPG-2 | * | ERPG-2 | 9.4 | ERPG-2 |
| | ERPG-3 | 1720 | ERPG-3 | 39 | ERPG-3 | 123 | ERPG-3 | 94 | ERPG-3 |
| Worker | 500 | | 150 | | 2100 | | 230 | | 1.9 x 10 |
| MOI | 47 | | 14 | | 200 | | 22 | | 2.8 x 10 |
| NPA | 47 | | 14 | | 200 | | 22 | | 2.8 x 10 |

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the
Table F.2.4.2-31. Summary of chemical concentrations for fire involving diesel fuel Pearl Harbor Naval Shipyard.

| CHEMICAL CONCENTRATIONS mg/m3 - 50% METEOROLOGY | | | | | | | | | |
|---|-----------------|------|----------------|------|--------------|-----|------------------|-----|----------|
| | Carbon Monoxide | | Sulfur Dioxide | | Nitric Oxide | | Nitrogen Dioxide | | Lead |
| | ERPG-1 | 29 | ERPG-1 | 0.79 | ERPG-1 | 31 | ERPG-1 | 5.6 | ERPG-1 |
| | ERPG-2 | 172 | ERPG-2 | 7.9 | ERPG-2 | * | ERPG-2 | 9.4 | ERPG-2 |
| | ERPG-3 | 1720 | ERPG-3 | 39 | ERPG-3 | 123 | ERPG-3 | 94 | ERPG-3 |
| Worker | 200 | | 61 | | 830 | | 92 | | 1.6 x 10 |
| MOI | 3.3 | | 1.0 | | 13 | | 1.5 | | 1.7 x 10 |
| NPA | 12 | | 3.6 | | 49 | | 5.4 | | 1.4 x 10 |

Table F.2.4.2-32. Summary of chemical concentrations for fire involving diesel fuel Pearl Harbor Naval Shipyard.

| CHEMICAL CONCENTRATIONS mg/m3 - 95% METEOROLOGY | | | | | | | | | |
|---|-----------------|------|----------------|------|--------------|-----|------------------|-----|----------|
| | Carbon Monoxide | | Sulfur Dioxide | | Nitric Oxide | | Nitrogen Dioxide | | Lead |
| | ERPG-1 | 29 | ERPG-1 | 0.79 | ERPG-1 | 31 | ERPG-1 | 5.6 | ERPG-1 |
| | ERPG-2 | 172 | ERPG-2 | 7.9 | ERPG-2 | * | ERPG-2 | 9.4 | ERPG-2 |
| | ERPG-3 | 1720 | ERPG-3 | 39 | ERPG-3 | 123 | ERPG-3 | 94 | ERPG-3 |
| Worker | 500 | | 150 | | 2100 | | 230 | | 1.9 x 10 |
| MOI | 11 | | 3.4 | | 47 | | 5.3 | | 1.4 x 10 |
| NPA | 500 | | 150 | | 2100 | | 230 | | 1.9 x 10 |

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the
Table F.2.4.2-33. Summary of chemical concentrations for fire involving diesel fuel Portsmouth Naval Shipyard.

| CHEMICAL CONCENTRATIONS mg/m3 - 50% METEOROLOGY | | | | | | | | | |
|---|-----------------|------|----------------|------|--------------|-----|------------------|-----|----------|
| | Carbon Monoxide | | Sulfur Dioxide | | Nitric Oxide | | Nitrogen Dioxide | | Lead |
| | ERPG-1 | 29 | ERPG-1 | 0.79 | ERPG-1 | 31 | ERPG-1 | 5.6 | ERPG-1 |
| | ERPG-2 | 172 | ERPG-2 | 7.9 | ERPG-2 | * | ERPG-2 | 9.4 | ERPG-2 |
| | ERPG-3 | 1720 | ERPG-3 | 39 | ERPG-3 | 123 | ERPG-3 | 94 | ERPG-3 |
| Worker | 33 | | 10 | | 140 | | 15 | | 3.6 x 10 |
| MOI | 1.7 | | 0.51 | | 7.0 | | 0.78 | | 1.7 x 10 |
| NPA | 2.7 | | 0.83 | | 11 | | 1.2 | | 3.0 x 10 |

Table F.2.4.2-34. Summary of chemical concentrations for fire involving diesel fuel Portsmouth Naval Shipyard.

| CHEMICAL CONCENTRATIONS mg/m3 - 95% METEOROLOGY | | | | | | | | | |
|---|-----------------|------|----------------|------|--------------|-----|------------------|-----|----------|
| | Carbon Monoxide | | Sulfur Dioxide | | Nitric Oxide | | Nitrogen Dioxide | | Lead |
| | ERPG-1 | 29 | ERPG-1 | 0.79 | ERPG-1 | 31 | ERPG-1 | 5.6 | ERPG-1 |
| | ERPG-2 | 172 | ERPG-2 | 7.9 | ERPG-2 | * | ERPG-2 | 9.4 | ERPG-2 |
| | ERPG-3 | 1720 | ERPG-3 | 39 | ERPG-3 | 123 | ERPG-3 | 94 | ERPG-3 |
| Worker | 500 | | 150 | | 2100 | | 230 | | 1.9 x 10 |
| MOI | 24 | | 7.2 | | 99 | | 11 | | 3.7 x 10 |
| NPA | 73 | | 22 | | 300 | | 34 | | 1.7 x 10 |

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the
Table F.2.4.2-35. Summary of chemical concentrations for fire involving diesel fuel Puget Sound Naval Shipyard.

| CHEMICAL CONCENTRATIONS mg/m3 - 50% METEOROLOGY | | | | | | | |
|---|-----------------|----------------|--------------|--|------------------|--|----------|
| | Carbon Monoxide | Sulfur Dioxide | Nitric Oxide | | Nitrogen Dioxide | | Lead |
| | ERPG-1 29 | ERPG-1 0.79 | ERPG-1 31 | | ERPG-1 5.6 | | ERPG-1 |
| | ERPG-2 172 | ERPG-2 7.9 | ERPG-2 * | | ERPG-2 9.4 | | ERPG-2 |
| | ERPG-3 1720 | ERPG-3 39 | ERPG-3 123 | | ERPG-3 94 | | ERPG-3 |
| Worker | 33 | 10 | 140 | | 15 | | 3.6 x 10 |
| MOI | 1.5 | 0.47 | 6.3 | | 0.71 | | 1.5 x 10 |
| NPA | 13 | 4.0 | 54 | | 6.1 | | 1.4 x 10 |

Table F.2.4.2-36. Summary of chemical concentrations for fire involving diesel fuel at Puget Sound Naval Shipyard.

| CHEMICAL CONCENTRATIONS mg/m3 - 95% METEOROLOGY | | | | | | | |
|---|-----------------|----------------|--------------|--|------------------|--|----------|
| | Carbon Monoxide | Sulfur Dioxide | Nitric Oxide | | Nitrogen Dioxide | | Lead |
| | ERPG-1 29 | ERPG-1 0.79 | ERPG-1 31 | | ERPG-1 5.6 | | ERPG-1 |
| | ERPG-2 172 | ERPG-2 7.9 | ERPG-2 * | | ERPG-2 9.4 | | ERPG-2 |
| | ERPG-3 1720 | ERPG-3 39 | ERPG-3 123 | | ERPG-3 94 | | ERPG-3 |
| Worker | 500 | 150 | 2100 | | 230 | | 1.9 x 10 |
| MOI | 21 | 6.5 | 89 | | 9.8 | | 3.2 x 10 |
| NPA | 200 | 61 | 830 | | 92 | | 5.8 x 10 |

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the
Table F.2.4.2-37. Summary of chemical concentrations for fire involving diesel fuel at Puget Sound.

| CHEMICAL CONCENTRATIONS mg/m3 - 50% METEOROLOGY | | | | | | | |
|---|-----------------|----------------|--------------|--|------------------|--|----------|
| | Carbon Monoxide | Sulfur Dioxide | Nitric Oxide | | Nitrogen Dioxide | | Lead |
| | ERPG-1 29 | ERPG-1 0.79 | ERPG-1 31 | | ERPG-1 5.6 | | ERPG-1 |
| | ERPG-2 172 | ERPG-2 7.9 | ERPG-2 * | | ERPG-2 9.4 | | ERPG-2 |
| | ERPG-3 1720 | ERPG-3 39 | ERPG-3 123 | | ERPG-3 94 | | ERPG-3 |
| Worker | 900 | 280 | 3800 | | 420 | | 9.9 x 10 |
| MOI | 4.0 | 1.2 | 17 | | 1.9 | | 4.1 x 10 |

Table F.2.4.2-38. Summary of chemical concentrations for fire involving diesel fuel at Puget Sound.

| CHEMICAL CONCENTRATIONS mg/m3 - 95% METEOROLOGY | | | | | | | |
|---|-----------------|----------------|--------------|--|------------------|--|----------|
| | Carbon Monoxide | Sulfur Dioxide | Nitric Oxide | | Nitrogen Dioxide | | Lead |
| | ERPG-1 29 | ERPG-1 0.79 | ERPG-1 31 | | ERPG-1 5.6 | | ERPG-1 |
| | ERPG-2 172 | ERPG-2 7.9 | ERPG-2 * | | ERPG-2 9.4 | | ERPG-2 |
| | ERPG-3 1720 | ERPG-3 39 | ERPG-3 123 | | ERPG-3 94 | | ERPG-3 |
| Worker | 9900 | 3100 | 41000 | | 4600 | | 3.8 x 10 |
| MOI | 28 | 8.8 | 120 | | 13 | | 1.7 x 10 |

* ERPG-2 level not assigned since one-tenth the IDLH level would be less than the

F.3 AIRCRAFT CRASH PROBABILITIES

F.3.1 Introduction

The probability of an airplane crashing into a fuel storage area or a file at the various alternate site locations is presented in this section. An airplane crash is of concern since it might result in the release of corrosion products from the storage area or the release of radioactive fission products from the fuel. The method outlined in "A Method for the Calculation of the Probability of Crash of an Aircraft into Structures in Weapon Storage Areas" (Sandia 1983) has been used to predict the crash probabilities for this analysis. This methodology takes into consideration the crash probabilities associated with landing operations at nearby airports and crashes during in-flight operations.

The aircraft crash probability analysis presented herein is based on the exclusion of civilian aircraft and military aircraft crossing the space within a 10-mile radius of the site. The crash probability of general aviation aircraft is not included in this assessment since these aircraft generally do not possess sufficient mass or attain sufficiently high velocities to pose a significant radiological threat in the event that they crash into a fuel storage area or a fuel storage tank. Further, the crash probability contribution due to air travel beyond 10 miles was determined to be very small based on the models and conditions used in this analysis, and therefore

F.3.2 Methodology

The Sandia report provides the methodology which has been used for this assessment (1983). In this report, the following expressions are given for calculating the crash associated with takeoff and landing operations at a given airport runway, and in-flight along a given airway:

$$P_{to} = EN_i * P_{nto} * A * c(a) * e^{-l_{xijl}/0(x,a)} * e^{-l_{yijl}/0(y,a)}$$

$$P_l = EN_{il} * P_{nl} * A * c(a) * e^{-l_{xijl}/0(x,a)} * e^{-l_{yijl}/0(j,a)}$$

$$P_{if} = EN_k * P_{nif} * A * c(if) * e^{-l_{xkjl}/0(x,if)}$$

where: subscript "to" refers to airport takeoff operations

subscript "l" refers to airport landing operations

subscript "if" refers to in-flight operations

N_i = the number of runway operations per year

N_k = the number of in-flight operations per year

P_n = the crash probability per operation given in Table F.3-1

= the perpendicular distance from the centerline of the runway to the target

= the perpendicular distance from the airway to the target in miles

= the perpendicular distance from the end of the runway to the target in miles

$c(a)$ = crash density constant given in Table F.3-2

$c(if)$ = crash density constant given in Table F.3-3

$0(x,a)$ = crash density constant given in Table F.3-2

$0(y,a)$ = crash density constant given in Table F.3-2

$e(x,if)$ = crash density constant given in Table F.3-3

A = effective crash area in square miles.

Table F.3-1. Crash parameter P_n . Table F.3-2. Crash density constants. Table F.3-3. Crash density constants. Table F.3-4. Airport landings and takeoffs per site location per year. The product of the number of operations per year, times the crash probability per operation, times the effective crash area per square mile, times the crash probability density constant, gives the crash probability associated with a given site. To determine the crash probability associated with a given site requires the use of these relationships for each airport runway and for each airway. These individual crash probabilities are then summed to arrive at a total overall crash probability for a site.

In the Sandia report, the effective crash area is identified as the sum of the area of the plane, the effective plan view associated with the target, and the effective crash area (Sandia 1983). The following expression relates these terms and is valid for angles greater than zero. If the crash attitude angle is zero, an airplane would be to the ground at an altitude equal to or greater than the height of the target; the airplane would clear the object and there would be no crash.

$$A = (L + A_c) * (W + 5k + H * \cot \theta)$$

where: L - target length dimension

W - target width dimension

H - target height

A_c - aircraft wingspan

θ - crash attitude angle

Sk - aircraft skid distance.

F.3.3 Site Specific Information

The existence and location of airports and airways within 10 statute miles of the site were obtained from Sectional Aeronautical Maps published by the National Oceanic and Atmospheric Administration (NOAA), and from detailed site specific maps which identify nearby airports (NOAA 1993a; NOAA 1993b; NOAA 1993c; NOAA 1993d; NOAA 1993e; NOAA 1993f; NOAA 1993g; USGS 1983a; USGS 1983b). These same sources of information were also used to obtain the distances from airport runways and airways to the sites of interest. Information on air traffic along airways within this region was obtained from the Federal Aviation Administration (FAA). Airplane holding patterns and approach and departure routes that were identified were converted into equivalent airways for this analysis. Information regarding the number of takeoffs and landings at each airport runway was obtained from the cognate airport manager or base commander, or from the FAA. Tables F.3-4 and F.3-5 summarize airport and airway traffic information that was obtained.

Table F.3-4. Airport landings and takeoffs per site location per year. Table F.3-5. Crash density constants. The crash probability density constant was based on the storage facility footprints identified in Table D-1 of the EIS. Length and width dimensions associated with the target area were calculated from the

treating the storage area as square (i.e., equal length and width dimensions). The storage containers was based on that of an existing M-140 shipping container, and the water pool facility superstructure was based on the approximate height of the Expend at INEL. For the water pool facility, a crash into the building might damage the fuel airplane directly striking it or by the airplane causing sufficient damage to the building of the building structure to collapse and strike the fuel. The crash attitude angle based on the recommended value identified in the Sandia report (Sandia 1983). A red skid distance of 300 feet was used. This skid distance is based on a review of the locations and reflects the fact that nearby buildings, dry docks, or retaining wall the length of the aircraft skid to 300 feet or less prior to impact.

The effective crash area associated with fuel examination at the Expended Core INEL or similar facilities to be constructed at the Barnwell Plant, Hanford, Oak Ridge Test Site, or Savannah River was based on the vulnerable part of the facility being 100 feet wide, and 60 feet high. This represents the portion of the Expended Core Facility the combined dry cell, shielded cell, and water pool as identified in Attachment B. facilities, a crash into the building might damage the fuel either by the airplane by the airplane causing sufficient damage to the building to cause part of the building collapse and strike the fuel. The effective crash area associated with dry storage containers waiting to be handled at these fuel examination facilities is based on that of an existing M-140 shipping container and the modeling approach that two such containers located outside of the fuel processing facility and separated by a reasonably large attitude angle that was used was 15 degrees. For these facilities and containers, a distances of 2200 feet for military high performance aircraft and 1600 feet for large civilian aircraft were used. These skid distances correspond to the maximum expected based on the information presented in the Sandia report (Sandia 1983).

F.3.4 Aircraft Specific Information

Aircraft wingspans which are representative of large civilian aircraft, military aircraft (i.e., tactical fighter and tactical fighter trainer), and large military transport, refueling, and bomber) have been taken into account separately in computing crash probabilities for each site. Wingspans for these three classes of aircraft have average values computed from individual planes within each class. Data from "Aviation Space Technology" served as the basis for determining these wingspans (AWST 1992). Calculated average wingspans were: 40 feet for military high-performance aircraft, military aircraft, and 135 feet for large civilian aircraft. For large military and effective wingspan that was 75% of the average wingspan was used in the probability. This effective wingspan reflects the fact that only the region between the most outer engines has the potential to seriously damage a fuel storage area or a fuel examination

F.3.5 Results

Tables F.3-6 and F.3-7 present the crash probability results for the four methods of storage at shipyards and prototypes and for fuel examination facilities. The probabilities in these tables represent the combined takeoff, landing, and in-flight crash probabilities for each method of fuel storage at each site. Following the DOE NEPA oversight guidance consequences for beyond design basis accidents are calculated where the probability is greater than 10⁻⁷ per year. These consequences are discussed in Section F. 1.4 of this attachment. For probabilities less than 10⁻⁷ per year, calculations of consequences are not included.

The probability calculated for airplane crashes at different facilities located at the DOE site may vary somewhat. This situation exists at INEL where low altitude testing of jet airliners has been conducted near the NOAA tower. This tower is located about 1 mile from the ICPP, and 2.3 miles from ECF. As a result of this difference in distance, the crash probability is expected to be about a factor of two higher at ICPP than at ECF. Further, two different methodologies have been in general use for determination of aircraft accident probabilities. Sandia methodology used in this appendix, a technique developed by the NRC in the 1970s and applied at some facilities. Comparison of the two methods has shown that results can differ by a factor of two to four, with the NRC method generally producing higher probabilities. This difference stems from the somewhat more detailed nature of the Sandia methodology. Therefore, calculated aircraft crash probabilities at ICPP are expected to be about eight times higher than those calculated for ECF.

Crash probabilities fall in the design basis range (i.e., probability of occurrence between 10⁻⁶ and 10⁻⁷ per year).

year) at Pearl Harbor for all types of fuel storage, at Norfolk for fuel storage in on railcars, and at Oak Ridge and Savannah River for the fuel examination facility pool. The radiological consequences associated with an airplane crash into these are in detail in Section F.1.4.

Crash probabilities fall in the beyond design basis range (i.e., probability between $10(-6)$ and $10(-7)$ per year) at Norfolk for fuel storage in immobile dry storage shipping containers on a concrete pad, and in the water pool facility, at Kesselring shipping containers on railcars and in the water pool facility, at Portsmouth for railcars, at the Nevada Test Site for the fuel examination facility dry cell and water examination facility dry storage containers at Oak Ridge and Savannah River. The radiological consequences associated with an airplane crash into these areas are also addressed in Section F.1.4.

Crash probabilities with a likelihood of occurrence less than $10(-7)$ per year since it is expected that they would contribute very very little to the risk. This immobile dry storage and shipping containers on a concrete pad at Kesselring and the water pool facility at Portsmouth, all types of fuel storage at Puget Sound, the fuel facilities at Barnwell, Hanford, and INEL, and the fuel examination facility dry storage at the Nevada Test Site.

F.4 FUGITIVE DUST

Table F.3-6. Crash probabilities for various fuel storage options per site location per year.

Table F.3-7. Crash probabilities for fuel examination facilities per site location per year.

The INEL-ECF is a large laboratory facility used to receive, examine, and ship nuclear fuel and irradiated test specimen assemblies. This section provides the results of fugitive dust emissions that could be generated during the construction of a similar facility at an alternate location (Hanford, Savannah River, the Nevada Test Site, the Barnwell site).

F.4.1 Computer Modeling to Estimate Fugitive Dust Emissions

Factors such as locations of affected persons, terrain, meteorological conditions, and grain size distributions are required as input parameters for calculating particulate concentrations from fugitive dust emissions during construction activities. This section describes the computer model used to perform fugitive dust concentration estimates. Parameters used in this analysis are summarized in Section F.4.2.

The Fugitive Dust Model (FDM) was the computer code chosen to evaluate fugitive dust emissions from construction activities at an alternate DOE location. FDM is a computer quality model specifically designed for estimating fugitive dust emissions from point sources (EPA 1992c).

FDM is designed to work with properly prepared meteorological data such as the RAMMET program or card images of meteorological data in either hourly or Stability format. FDM is based on the well-known Gaussian plume formulation for computing concentrations but the model has been specifically adapted to incorporate an improved gradient transport algorithm. Emissions for each source are apportioned by the user into a series of particle size classes. A gravitational settling velocity and a deposition velocity are subsequently calculated for each class, and dust concentrations and depositions are then calculated for each class by the user.

FDM is the preferred model for estimating conditions resulting from particulate emissions from fugitive sources such as excavation and soil handling. The ISC2 Code (F.2.2.2) can also be used for this purpose; however, FDM was judged to be superior to ISC2 for this evaluation.

F.4.2 Conditions and Key Parameters

- Construction area was 30 acres
- Construction activities occurred over a 3- to 5-year period.

- An emission factor of 2.0 tons per acre-month was used.
- Grain sizes used were as follows:

| Average Diameter (um) | % of Total |
|-----------------------|------------|
| 1.25 | 3 |
| 3.75 | 5 |
| 7.5 | 15 |
| 12.5 | 10 |
| 20.0 | 67 |
- Meteorological conditions used were the 5-year average STAR data sets.
- Roughness heights were 2 centimeters for Hanford and Nevada Test Site
30 centimeters for Savannah River, the Barnwell Plant, and Oak Ridge.

F.4.3 Results

The fugitive dust concentrations were calculated using FDM for the worker, MC and MOI using normal meteorology. Table F.4-1 lists the fugitive dust concentration locations. These airborne concentrations were compared against the TLV-TWA concentration particulates. The TLV-TWA concentration of 10 mg/m³ was not exceeded at any of the locations for fugitive dust that could be generated during construction activities locations. Since these concentrations were extremely low, it can also be concluded would be expected for the alternate shipyard locations since the facilities to be smaller.

F.5 OCCUPATIONAL ACCIDENTS

Table F.4-1. Summary of fugitive dust concentrations for construction activities at

Occupational accidents can occur in the workplace during the construction or operation of an industrial facility. In order to assess the possible extent of occupational accidents and non-construction operations at naval spent nuclear fuel facilities, projections of fatalities and injuries or illnesses were made for each alternative. The projections in this section. The projections are based on average occupational fatality and injury rates published by the DOE (DOE 1993a) for DOE and DOE contractor operations. The incidents that were used in the analyses are provided below. A more detailed discussion of the incidence rates is presented in Volume 1.

Average occupational injury/illness and fatality rates (a)

| | All Labor Categories | | Construction Workers | |
|-------------------------|----------------------|------------|----------------------|--|
| | Total | | Total | |
| | Injury/Illness | Fatalities | Injury/Illness | |
| DOE and Contractors (b) | 3.2 | 0.0032 | 6.2 | |

(a) All incidence rates are given per 100 worker-years

(b) 1988-1992 averages (DOE 1993a)

The term "injury/illness" as used in this analysis corresponds to the DOE definition of a recordable injury or illness. Specifically, an injury or illness case represents any work-related injury or illness, or any work-related injury which would result in loss of consciousness, removal from work, transfer to another job, or medical treatment beyond first aid.

F.5.1 Accident Evaluation

F.5.1.1 Construction. The average number of construction-related fatalities and injury or illnesses

and the 40-year total were calculated. The methods of calculating construction-related injuries or illnesses are presented below.

The number of construction workers that would be required to construct or modify spent nuclear fuel storage and examination facility was calculated for every year that would take place during the period 1995 through 2035. The sum of these workers represents the number of construction workers. The 40-year total of construction fatalities was obtained by multiplying the total number of construction workers by the construction fatality rate.

DOE contractors.

The annual average number of construction workers for each facility was obtained by dividing the total number of construction workers by the number of years that construction was performed. The product of the annual average number of construction workers and the construction injury or illness rate for DOE and DOE contractors was calculated to provide the annual average number of fatalities.

The annual average and 40-year total construction injuries or illnesses were calculated in the same manner as construction fatalities except that the construction injury or illness rate for DOE and DOE contractors was used.

F.5.1.2 Storage and Examination Facility Operations. The average number of fatalities and

injuries or illnesses and the 40-year total fatalities and injuries or illnesses were calculated for the operation of naval spent nuclear fuel storage and examination facilities. The methods for calculating the operational fatalities and injuries or illnesses are presented below.

The accident rates for DOE and DOE contractor operations other than construction were calculated because examination and storage facility operations would more likely be performed by DOE contractor personnel (or Navy personnel in the case of shipyards). The number of workers would be required to operate each naval spent nuclear fuel storage and examination facility was calculated for every year during the period 1995 through 2035 and summed over the 40 years to obtain the total number of workers. The 40-year total of fatalities was obtained by multiplying the total number of workers by the DOE fatality rate.

The annual average number of workers for each facility was obtained by dividing the total number of workers by the number of operational years (40 years). The product of the annual average number of workers and the DOE fatality rate represents the annual average number of fatalities.

The annual average and 40-year total estimated injuries or illnesses associated with storage and examination operations were calculated in the same manner as fatalities associated with construction operations except that the DOE injury or illness accident rate was used.

F.5.2 Results

This section presents tabulated results of calculations of construction and operation fatalities and injuries or illnesses for each alternative. Table F.5-1 provides the projection of construction fatalities and injuries or illnesses for construction activities and storage and examination facilities for each alternative. Tables F.5-2 through F.5-3 present the results of calculations of fatalities and injuries or illnesses for construction activities and storage and examination facilities at naval sites. The results of all calculations show that the number of fatalities and injuries or illnesses for construction activities and storage and examination operations would be low for each alternative.

Table F.5-1. Occupational fatalities and injury/illnesses by alternative-construction activities
Table F.5-2. Occupational fatalities for construction activities at Naval Nuclear Fuel Storage and Examination Facilities
Table F.5-3. Occupational fatalities for storage and examination facility operations
Table F.5-4. Occupational injuries/illnesses for construction activities at Naval Nuclear Fuel Storage and Examination Facilities
Table F.5-5. Occupational injuries/illnesses for storage and examination facility operations

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ATTACHMENT G

COMPARISON OF THE NAVAL SPENT NUCLEAR FUEL STORAGE ENVIRONMENTAL ASSESSMENT AND THIS ENVIRONMENTAL IMPACT STATEMENT

The Naval Nuclear Propulsion Program has prepared an environmental assessment of the long-term storage of naval spent nuclear fuel until the environmental impact statement, appendix is a part, can be completed and an alternative for management of naval spent nuclear fuel selected (Federal Register, Vol. 59, No. 19, 4051, January 8, 1994). The environmental assessment considered alternatives for storing, until June 1995, naval spent nuclear fuel removed from powered vessels and reactor prototypes at several naval sites. The environmental impact statement, which the appendix including this attachment is a part, considers alternatives for storage of naval spent nuclear fuel during a 40-year period beginning in June 1995.

Occasions may arise when comparison of the impacts for naval spent nuclear fuel storage in these two documents may be desired. However, there are some differences between the environmental impact assessment and this appendix which should be recognized because they make such comparisons complicated. Failure to recognize these differences may lead to an erroneous conclusion that the documents are inconsistent or contradictory.

First, and most importantly, the environmental assessment considered only a less than 2 years, needed to conduct the National Environmental Policy Act process a decision on the long-term management of Department of Energy (DOE) spent nuclear process includes preparation of this environmental impact statement. The environmental statement, and therefore this appendix, provides the evaluation of the alternatives managing spent nuclear fuel for 40 years. As a result, this environmental impact statement provides a wider range of alternatives than the environmental assessment, partly because more possible if a longer time is available to implement them and partly because some decisions could be deferred for a short period such as 2 years should not be deferred for a period of 40 years.

The alternatives considered in the environmental impact statement also include sites for management of naval spent nuclear fuel. This provides a wider range of consequences, it also increases the number of potential destinations and the shipments of naval spent nuclear fuel under some alternatives. In the same manner, environmental assessment considered temporary storage of naval spent nuclear fuel at Shipbuilding, storage at Newport News is not included in the alternatives in the environmental statement because that shipyard is not federally owned.

The alternatives considered in the environmental impact statement also include naval spent nuclear fuel in water pools and immobile dry storage casks in addition to shipping containers. There is also an evaluation of alternatives for examination of fuel in the environmental impact statement. These additional storage modes and examination alternatives were not considered in detail in the environmental assessment because the document was short and consequently, the implementation of some of the alternatives have been impractical. For example, water pool storage facilities could not be funded at the shipyards in a period of less than 2 years.

Also, as a natural result of the longer period considered in this environmental statement, a larger number of naval spent nuclear fuel assemblies and additional types of assemblies are included in the analyses. The increase in the amount of naval spent nuclear fuel occurs since a certain number of naval reactors are refueled or defueled each year, number of years more fuel becomes available for storage. Similarly, some newer designs of nuclear propulsion plants will not be refueled for the first time until some time after the types of fuel are not treated in the environmental assessment.

The environmental impact statement addresses some impacts of normal operation accidents not discussed in the environmental assessment because the conditions or circumstances might cause these effects would not occur under the alternatives considered in the assessment. The environmental impact statement also addresses several types of impacts alternative in greater detail than the environmental assessment. This was done because treatment was judged to be appropriate with the broader scope of alternatives in the impact statement.

The methods used to perform the analyses in the environmental impact statement were refined in the time since the environmental assessment was prepared. This occurred because the larger number of naval spent nuclear fuel assemblies analyzed and the wider scope of methods of storage to be evaluated, and partly because additional time was available for refinements. In addition to refinements in the methods for performing the calculations, changes in the calculational models were made in order to establish a high degree of consistency in the analytical methods used for the other DOE sites that are part of the environmental statement. This consistency is appropriate in some cases in order to establish a direct comparison of alternatives. The changes in the calculational methods make a direct analytical results presented in the environmental assessment for naval sites with time difficult.

GLOSSARY

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| activation | The process of making a material radioactive by exposing the material to neutrons, protons, or other nuclear particles. |
| activation products | The radionuclides formed as a result of a material being activated. For example, cobalt-60 is an activation product resulting from the activation of cobalt-59. |
| activity | A measure of the rate at which a material is emitting nuclear radiation. Activity is usually measured in terms of the number of nuclear disintegrations which occur in a quantity of the material per unit of time. The standard unit of activity is the curie (Ci) which is equal to 37 billion (3.7×10^{10}) disintegrations per second. |

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| aggregates | Sand, gravel, or rock which is used in concrete or mortar mix achieve increased strength. |
| airborne emissions | Radioactivity in the form of radioactive particles, gases, or transported by air. |
| alloy | A mixture of two or more metals. |
| aquifer | A water-bearing stratum of permeable rock, sand, or gravel beneath the surface of the earth, which is capable of yielding a well or spring. |
| archaeological areas | Areas of or relating to the scientific study of material remains (fossil relics, artifacts, monuments) of past human life and activity. |
| average individual | An individual who could consume items or occupy areas at a rate which would be typical for the population of interest. |
| base flood | A flood which has a 1-percent chance of occurrence in any given year. Also referred to as a 100-year flood. |
| benthic | Pertaining to the bottom of the ocean. |
| best estimate | An estimate in which the factors used in determining the estimate chosen such that the result approximately represents what would be expected. |
| cladding | A metal casing that surrounds the nuclear fuel. |
| coastal zone | The region along the shore, adjacent to the ocean. A coastal zone usually defined as the region within 3 nautical miles of a shoreline. |
| concentration factor | A factor which is defined as the concentration of an element or radionuclide in an organism or its tissues divided by the concentration directly available from the organism's environment under equilibrium or steady-state conditions. |
| conservative estimate | An estimate in which the factors used in determining the estimate chosen such that the result would be unlikely to be exceeded. |
| containments | Devices as complex as a glove box or as simple as a plastic bag designed to limit the spread of radioactive contamination to as close as possible to the source, and to break the chain of transmission to prevent contaminating other material. |
| core | The central portion of a nuclear reactor containing the nuclear fuel. |
| corrosion | The process denoting the destruction of metal by chemical or electrochemical action. |
| corrosion products | The substances produced by corrosion of a metal. Rust is a common corrosion product resulting from the corrosion of iron. |
| corrosion-resistant alloy | An alloy which corrodes slowly compared to ordinary alloys. Stainless steel is an example of a corrosion-resistant alloy. |
| critical organ | The limiting organ for evaluating exposure to ionizing radiation. A critical organ is determined by the following criteria: (1) that organ which accumulates the greatest concentration of a radioactive substance; (2) the necessity of the organ to the well being of the entire body; (3) the organ most damaged by the entry of a radionuclide into the body; and (4) the organ damaged by the lowest exposure. Usual case (1) is the determining factor for choosing the critical organ. |
| critical pathways | Those pathways which result in the most significant amount of exposure to radiation. |
| cumulative effects | The changes in the health of an individual(s) from the sum of yearly exposures to radiation. |
| curie (Ci) | The curie is the common unit used for expressing the magnitude of radioactive decay in a sample containing radioactive material. Specifically, the curie is that amount of radioactivity equal to 3.7 x 10 ¹⁰ (37 billion) disintegrations per second. This unit gives any indication of the radiological hazard associated with the material. |
| defueling | Removal of all nuclear fuel from a nuclear-powered ship. |
| design earthquake | The maximum intensity earthquake that might occur along the fault to a structure. Structures are built to withstand a design earthquake. |
| diffusion | The process of spreading out or scattering from regions of high concentration to regions of lower concentration. |
| dispersion | The process of scattering or distributing over a large region. |
| dose | A general term which denotes the quantity of radiation or energy absorbed; usually expressed in rems for doses to man. |

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| dose commitment | The total radiation dose accrued by an individual over a specified period of time due to the exposure of the individual to radiation over a given interval of time. This includes the total time the material would reside in the body, if ingested or inhaled (usually expressed in rems). |
| dose commitment conversion factor | A factor which converts the quantity of radioactivity taken into the body to the dose to the individual (usually expressed in rems per curie). |
| dose equivalent | A quantity used to express all radiations on a common scale for calculating the effective absorbed dose. It is defined as the absorbed dose and certain modifying factors and is expressed in rems. |
| dose rate | The amount of radiation dose delivered in a unit amount of time, for example, in rems per hour. |
| dose rate conversion factor | A factor which converts the exposure to a given radiation level to the dose that an individual could receive. It is usually expressed in rems per hour per curie per cubic meter (or square meter). |
| dredge spoil | Bottom sediments or materials that have been excavated from a waterway. |
| ecosystem | A community of plant and animal populations together with the physical environment. An organizational unit which can maintain biological activities independent of other units. |
| element | A chemical substance that cannot be divided into simpler substances by chemical means. A substance whose atoms all have the same atomic number. |
| endangered species | A species or subspecies which is in danger of extinction throughout all or a significant portion of its range. |
| environmental consequences | Changes to the environment as a result of the effects of radioactive materials. |
| epidemiological study | A scientific study that deals with the incidence, distribution, and control of disease in a specified population. |
| exclusion area | An area where access would result in personnel exceeding radiation exposure limits in a very short time. |
| Expended Core Facility (ECF) | A large laboratory facility, located at the Naval Reactors Facility in Idaho, consisting of water pools and shielded cells used to research, examine, and ship naval spent nuclear fuel and irradiated test specimen assemblies. Naval spent nuclear fuel is prepared at storage and shipment to the Idaho Chemical Processing Plant. |
| exposure, external | The subjecting of the outside of the body of an organism to ionizing radiation. |
| exposure, internal | The subjecting of the inside of the body of an organism to ionizing radiation. |
| exposure, occupational | The subjecting of an individual to ionizing radiation in the course of employment. |
| exposure, radiation | The subjecting of a material or organism to ionizing radiation. |
| fauna | Animals. |
| fissile | A material whose nucleus is capable of being split (fissioned) by neutrons of all energies. |
| fission | The splitting of a heavy nucleus into two approximately equal parts which is accompanied by the release of a relatively large amount of energy and generally one or more neutrons. |
| fission products | During operation of a nuclear reactor, heat is produced by the fission (splitting) of "heavy" atoms, such as uranium, plutonium, or thorium. The residue left after the splitting of these "heavy" atoms into intermediate weight atoms generally termed "fission products." Because of the nature of the fission process, many fission products are unstable and, hence, radioactive. |
| floodplain | The lowlands which adjoin inland and coastal waters and related areas and floodprone areas of offshore islands which are covered by water from a 1-percent or greater chance flood in any given year. |
| floodplain/wetlands assessment | An evaluation which consists of a description of a proposed action, a discussion of its effects on the floodplain/wetlands, and a consideration of alternatives. |
| flora | Plants. |

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| fuel | Fissionable material used or useable to produce energy in a reactor. It may also refer to a mixture, such as natural uranium which only part of the atoms are readily fissionable. |
| gamma ray | [Symbol - (gamma)] High-energy, short wavelength electromagnetic radiation. Gamma radiation frequently accompanies beta particle emission and are very penetrating and are stopped most effectively by dense materials such as lead or uranium. They are essentially similar to x-rays but more energetic and originate from the nucleus. Cobalt-60 is a radionuclide that emits gamma rays. |
| geology | The study of the origin, history, materials, and structure of the earth. |
| geophysical survey | An examination of the condition, situation, or value of the earth including the fields of meteorology, hydrology, oceanography, seismology, volcanology, magnetism, radioactivity, and geology. |
| glaciation | The act of having been subjected to glaciers, extreme cold, or ice. |
| groundwater | Water that exists or flows beneath the earth's surface in the saturation zone between soil and rock. |
| half-life, biological | The time required for a biological system, such as an organism, to clear by natural (non-radioactive) processes the amount of a substance that has entered it. |
| half-life, radioactive | The time required for half of the atoms of a radioactive material to decay to another nuclear form. |
| hazardous wastes | Excess chemical material that is dangerous to human health. |
| health detriment | The sum of all fatal cancers, a fraction of the non-fatal cancers proportional to the severity of the cancer types, and all genetic defects. |
| health effect | The occurrence of a fatal cancer, a non-fatal cancer, or a genetic defect. |
| high-efficiency particulate filter | A ventilation system device that can separate a particle size of 0.3 micron from the air into a filter medium at an efficiency of at least 99.97 percent. |
| hydrology | The study of the properties, distribution, and effects of water on the earth's surface, in the soil and underlying rocks, and in the atmosphere. |
| incident-free operations | Routine, day-to-day operations without accidents or other unusual occurrences. Synonymous and interchangeable with normal operations. |
| ion | An atom or molecule which has acquired an electrical charge by gaining or losing electrons. |
| ionizing radiation | Any radiation which displaces electrons from atoms or molecules thereby producing ions. Examples include alpha, beta, and gamma radiation. Exposure to ionizing radiation may produce skin damage. |
| irradiate | To expose to radiation. |
| isotope | One of two or more nuclides which have the same number of protons but have different numbers of neutrons in their nuclei. These are isotopes of an element have the same atomic number but different atomic weights. Isotopes usually have very nearly the same chemical properties but somewhat different physical properties. |
| long-lived radioactivity | Radioactive nuclides which decay slowly, therefore having relatively long half-lives. |
| man-rem | A unit used to measure the radiation exposure to an entire group of people to compare the effects of different amounts of radiation on groups of people. It is obtained by multiplying the average dose equivalent (measured in rems) to a given organ or tissue by the number of persons in the population of interest. |
| maximally exposed individual (MEI) | A theoretical individual who receives the highest radiation dose from the facility or activity in question. |
| maximally exposed off-site individual (MOI) | A theoretical individual located at the point on the DOE site boundary nearest to the facility or activity in question. |
| maximum individual | An individual who could consume items or occupy areas at rate |

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| maximum organ | which would be at a maximum for the population of interest. The organ which receives or could receive the largest amount exposure to radiation. |
| metric ton | [Abbreviation MT] A unit of mass which is equal to 1000 kilo or approximately 2205 pounds. |
| microcurie | [Abbreviation -Ci] A unit of activity which is equal to one- (1×10^{-6}) of a curie. |
| mil | A unit of length which is equal to one-thousandth (1×10^{-3}) inch. |
| millicurie | [Abbreviation mCi] A unit of activity which is equal to one-thousandth (1×10^{-3}) of a curie. |
| millirem | [Abbreviation mrem] A special unit for measuring dose equiva which is equal to one-thousandth (1×10^{-3}) of a rem. |
| monitoring, environmental | The periodic or continuous determination of the amount of radioactivity or radioactive contamination present in a regio |
| natural background radiation exposure | The total amount of radiation from cosmic radiation emitted b sun and the radiation emitted by natural minerals in the eart Typically, an average annual exposure of 100 mrem to the tota occurs from background radiation. |
| Naval Nuclear Propulsion Program | A joint program of the Department of Energy and the Departmen the Navy which has as its objective the design and developmen improved naval nuclear propulsion plants having high reliabil maximum simplicity, and optimum fuel life for installation in ranging in size from small submarines to large combatant surf ships. The program is frequently referred to as the Naval Re Program. |
| neutron | An uncharged particle with a mass slightly greater than that proton, found in the nucleus of every atom heavier than hydro Neutrons sustain the fission chain reaction in a nuclear reac |
| nuclear disin- tegration | A spontaneous nuclear transformation which is characterized b emission of particles and/or energy from the nucleus of an at |
| nuclear fuel | See fuel. |
| nuclear reactor | A device in which nuclear fission is initiated and controlled heat which is then used to generate power. |
| nuclear reactor accident | An accident which results in release of fission products from nuclear fuel. |
| nuclide | An atomic form of an element which is distinguished by its at number, atomic weight, and the energy state of its nucleus. factors determine the other properties of the element, includ radioactivity. |
| organ | A group of tissues which together perform one or more definit functions in a living body. |
| organism | Any living plant or animal. |
| overburden | Material overlying a deposit of useful geological materials. |
| particulate | Pertaining to a very small piece or part of a material. |
| pathway | The route or course along which radionuclides from defueled n powered ships could reach man. |
| percolate | To drain or seep through a material. |
| permeability | The quality or state of being able to diffuse or pass through material. |
| pH | A measure of the relative acidity or alkalinity of a solution solution has a pH of 7, acids have pH's less than 7, and base pH's greater than 7. |
| picocurie | [Abbreviation pCi] A unit of activity which is equal to one- (1×10^{-12}) of a curie. |
| prototype plants | Land-based naval nuclear reactor plants that are typical of a design for a naval warship and are used to test equipment and nuclear fuel prior to use on a shipboard nuclear plant. The plants are also used to train naval officers and enlisted per propulsion plant operators with extensive watchstanding exper and a thorough knowledge of all propulsion plant systems and |

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| | operating requirements. |
| radiation | The emission and propagation of energy through matter or space means of electromagnetic disturbances which display both wave and particle-like behavior. In this context, the "particles" as photons. The term has been extended to include streams of moving particles such as alpha and beta particles, free neutrons, cosmic radiations. Nuclear radiation is that which is emitted by atomic nuclei in various nuclear reactions and includes alpha and gamma radiation and neutrons. |
| radiation field | A region where radiation is present. |
| radiation level | The measured amount of radiation in a region. |
| radiation survey | The evaluation of an area or object with instruments to detect, identify, and quantify radioactive materials and radiation fields that may be present. |
| radiation worker | A person specially trained and tested in basic information regarding radiation, its effects, and radiological control techniques and procedures. |
| radioactive contamination | The deposition of radioactive material in any place where it is not intended, persons, invalidate experiments, or make products or equipment unsuitable or unsafe for some specific use. The presence of radioactive matter. |
| radioactive decay | The process of spontaneous transformation of a radioactive nuclide into a different nuclide or different energy state of the same nuclide. Radioactive decay involves the emission of alpha particles, beta particles, or gamma rays from the nuclei of the atoms. If a radioactive nuclide is transformed to a stable nuclide, the process results in a decrease of the number of original radioactive atoms. Radioactive decay is also referred to as radioactive disintegration. |
| radioactive waste | Equipment and materials which are radioactive and for which there is no further use. Radioactive wastes are generally classified into high-level waste (those resulting from reprocessing reactor fuel or reactor fuel itself), as low-level waste, or as low-level waste containing transuranic elements or uranium-233. |
| radioactivity | The process of spontaneous decay or disintegration of an unstable nucleus of an atom; usually accompanied by the emission of ionizing radiation. |
| radioisotope | An unstable isotope of an element that decays or disintegrates spontaneously and emits radiation. |
| radiological consequences | The changes to the environment or the health of a person(s) as a result of the effects of radiation exposure or radioactive materials. |
| radionuclides | Atoms that exhibit radioactive properties. Standard practice in naming radionuclides is to use the name or atomic symbol of an element followed by its atomic weight (e.g., cobalt-60 or Co-60 radionuclide of cobalt). |
| reactor vessel (or pressure vessel) | A very strong, thick-walled steel structure which contains the fuel and cooling water under high pressure during reactor operation. |
| rem | A unit of measure used to indicate the amount of radiation exposure a person receives (an acronym for roentgen equivalent man). |
| risk | The product of the consequences of an event multiplied by the probability of that event. |
| river stage | The level of the surface of a river in relation to some reference elevation. |
| sediment | Particles of organic or inorganic origin that accumulate in a body of water. |
| seismicity | The quality or state of shaking or vibrating caused by an earthquake. |
| shipping container | A specially designed large, stainless steel or lead-lined, steel cask that is transported in the vertical position on a well-tensioned depressed center railcar. The container is certified by the Department of Energy and the Department of Transportation for the shipment of spent nuclear fuel. |
| short-lived radioactivity | Radioactive nuclides which decay rapidly, therefore having relatively short half-lives. |

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| socioeconomics | The welfare of human beings as related to the production, distribution and consumption of goods and services. |
| special nuclear material | Materials containing nuclides such as plutonium-239, uranium-uranium enriched to a higher percentage than normal in the uranium-235 isotope. |
| specific activity | The ratio between the amount of radioactive isotope present a total amount of all other isotopes of that same element, both radioactive and stable. It is usually expressed in microcuri radioisotope per gram of total element. |
| specimen | A small sample of material (fuel or non-fuel) inserted into a for testing to characterize the material's performance. Test may be constructed of plant materials, reactor structural materials, fuel materials. |
| steam generator | The portion of the nuclear power plant where the heat from the primary system is transferred to the secondary system without contact between the water in the two systems. |
| survey meter | Any portable instrument which is used to detect radiation and especially adapted for surveying or inspecting an area to estimate existence and amount of radioactive material present. |
| tectonic | Pertaining to or designating the rock structures which result deformation of the earth's crust. |
| threatened species | Any species or subspecies which is likely to become an endangered species within the foreseeable future throughout all or a significant portion of its range. |
| topography | The detailed physical description of the surface of a region, the relative elevations of features. The graphical representation of physical configuration of a region on a map. |
| toxic | Relating to or caused by a toxin which is a poisonous substance, a specific product of the metabolic activities of a living organism or is usually very unstable when introduced into human tissues. |
| tritium | A radioactive isotope of hydrogen with atoms that are three times the mass of ordinary light hydrogen atoms. Tritium is present in reactor coolant as the result of neutron interaction with naturally occurring deuterium present in the water. |
| uranium | [Symbol U] A natural radioactive element with the atomic number 92 and, as found in natural ores, an average weight of approximately 238. The two principal natural isotopes are uranium-235 (0.7 percent of natural uranium) and uranium-238 (99.3 percent of natural uranium). Natural uranium also includes a minute amount of uranium-234. |
| vadose zone | The unsaturated region of soil located between the ground surface and the water table. |
| water pools | Deep pools of water that are used to inspect and hold spent nuclear fuel modules. Storage racks are located below the water surface support and position the fuel modules in place for handling and prevent the formation of a critical mass. |
| water table | The upper surface boundary of an uncontrolled aquifer, below which groundwater occurs. It is usually defined by the levels at which water stands in wells that barely penetrate the aquifer. |
| watershed | The region which drains into a river, river system, or body of water. |
| wetlands | Those areas which are covered by water with a frequency sufficient to support a prevalence of vegetative or aquatic life that requires saturated or seasonally saturated soil conditions for growth and reproduction. Wetlands generally include swamps, marshes, bogs and similar areas such as sloughs, potholes, wet meadows, river overflow, mudflats, and natural ponds. |
| x-rays | Penetrating electromagnetic radiations with wavelengths shorter than those of visible light. They are usually produced (as in medical diagnostic x-ray machines) by irradiating a metallic target with numbers of high-energy electrons. In nuclear reactions, it is customary to refer to photons originating outside the nucleus and those originating in the nucleus as gamma rays, even though |

are the same.

ABBREVIATIONS AND ACRONYMS

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| AEA | Atomic Energy Act |
| AEC | Atomic Energy Commission |
| ANL-E | Argonne National Laboratory - East |
| ANL-W | Argonne National Laboratory - West |
| ATR | Advanced Test Reactor |
| Btu | British thermal unit |
| BWR | boiling water reactor |
| CAA | Clean Air Act |
| CDE | committed dose equivalent |
| CEDE | committed effective dose equivalent |
| CERCLA | Comprehensive Environmental Response, Compensation, and Liability Act |
| CFA | central facilities area |
| CFR | Code of Federal Regulations |
| cfs | cubic feet per second |
| Ci | curies |
| cms | cubic meters per second |
| CNS | Charleston Naval Shipyard |
| CWRM | Commission on Water and Resource Management |
| DEP | Department of Environmental Protection |
| DOD | Department of Defense |
| DOE | Department of Energy |
| EB | Electric Boat Division of General Dynamics |
| ECF | Expended Core Facility |
| EDE | effective dose equivalent |
| EIS | Environmental Impact Statement |
| EPA | Environmental Protection Agency |
| ERPG | Emergency Response Planning Guideline |
| FAA | Federal Aviation Administration |
| FMEF | Fuels and Materials Examination Facility |
| FWPCA | Federal Water Pollution Control Act |
| HEPA | high-efficiency particulate air |
| ICPP | Idaho Chemical Processing Plant |
| ICRP | International Commission on Radiological Protection |
| IDLH | immediately dangerous to life and health |
| INEL | Idaho National Engineering Laboratory |
| INEL-ECF | Idaho National Engineering Laboratory Expended Core Facility |
| INGL | Ingalls Shipbuilding |
| KAPL | Knolls Atomic Power Laboratory |
| KSO | Kesselring Site Operation |
| kv | kilovolts |
| kw | kilowatts |
| kwh | kilowatt hours |
| LET | linear energy transfer |
| MCW | maximally exposed collocated worker |
| MEI | maximally (or maximum) exposed individual |
| mg | milligram |
| mgd | million gallons of water per day |
| MINS | Mare Island Naval Shipyard |
| MMI | Modified Mercalli Index |
| MOI | maximally exposed off-site individual |
| mph | miles per hour |
| MVA | megavolt amperes |
| MW | megawatts |
| MWh | megawatt hours |
| NAAQS | National Ambient Air Quality Standards |
| NEA | Nuclear Energy Agency |
| NEPA | National Environmental Policy Act |
| NESHAP | National Emission Standards for Hazardous Air Pollutants |
| NNPP | Naval Nuclear Propulsion Program |

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| NNS | Newport News Shipbuilding |
| NOAA | National Oceanic and Atmospheric Administration |
| NOR | Norfolk Naval Shipyard |
| NPA | nearest public access |
| NPDES | National Pollutant Discharge Elimination System |
| NRC | Nuclear Regulatory Commission |
| NRF | Naval Reactors Facility |
| NTS | Nevada Test Site |
| NYSDEC | New York State Department of Environmental Conservation |
| OECD | Organization for Economic Co-operation and Development |
| ORNL | Oak Ridge National Laboratory |
| ORR | Oak Ridge Reservation |
| PAH | polycyclic (or polynuclear) aromatic hydrocarbons |
| PCB | polychlorinated biphenyl |
| pCi | picocuries |
| PHNS | Pearl Harbor Naval Shipyard |
| PHWMA | Pearl Harbor Water Management Area |
| PNS | Portsmouth Naval Shipyard |
| PSNS | Puget Sound Naval Shipyard |
| PWR | pressurized water reactor |
| RCRA | Resource Conservation and Recovery Act |
| RWMC | Radioactive Waste Management Complex |
| SAPS | Shippingport Atomic Power Station |
| SARA | Superfund Amendments and Reauthorization Act |
| SNF | spent nuclear fuel |
| SRS | Savannah River Site |
| SRS-ECF | Savannah River Site Expanded Core Facility |
| TEDE | total effective dose equivalent |
| TI | transport index |
| TLV-TWA | threshold limit value, time-weighted average |
| TRA | test reactor area |
| USFWS | United States Fish and Wildlife Service |
| VOC | volatile organic compound |
| WIPP | waste isolation pilot plant |
| WSO | Windsor Site Operation |

This draft may be exempt from disclosure
pursuant to the
Freedom of Information Act at 5 USC 552 (b.5)

