Storage and Disposition of Weapons Usable Fissile Materials (FMD) PEIS

**Blending of U-233 to <12% or <5% Enrichment
at the Idaho National Engineering Laboratory**

Data Report

(Draft: Version 1)

**Nuclear Operations
Spent Fuel Programs Directorate**

August 1995

**Prepared for the
Department of Energy
Headquarters Office
Under Contract DE-AC07-94ID13223**

MASTER

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ACKNOWLEDGEMENTS

Generation of this report required a team effort on the part of several dedicated individuals. The quality of the work is a direct result of the team efforts to ensure the completeness and accuracy of the information provided.

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ACRONYMS

ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASSC	American Society of Steel construction
AWS	American Welding Society
CAM	Continuous Air Monitor
CAS	Criticality Alarm System
CCA	Criticality Control Area
CEDE	Committed Effective Dose Equivalent
CFR	Code of Federal Regulations
COG	Cell Off Gas
D&D	Decontamination and Decommissioning
DAC	Derived Air Concentration
DBA	Design Basis Accident
DOE	Department of Energy
DOE-ID	Department of Energy-Idaho Operations Office
DOG	Dissolver Off Gas
DOT	Department of Transportation
EDE	Effective Dose Equivalent
EIS	Environmental Impact Statement
EPA	Environmental Protection Agency
ES&H	Environmental Safety and Health
FDP	Fluorinel Dissolution Process
FHU	Fuel Handling Unit
FSA	Fuel Storage Area
HEU	Highly Enriched Uranium
ICPP	Idaho Chemical Processing Plant (Building Designator)
IFSF	Irradiated Fuel Storage Facility
INEL	Idaho National Engineering Laboratory
LCO	Limiting Conditions for Operation
LWBR	Light Water Breeder Reactor
M&O	Management and Operating (Contractor)
mppcf	million particles per cubic foot of air
NFPA	National Fire Protection Association
NWCF	New Waste Calcining Facility
OSHA	Occupational Safety and Health Administration
ORNL	Oak Ridge National Laboratory
ORR	Operational Readiness Review
PEW	Process Equipment Waste
RAM	Remote Area Monitor
RCRA	Resource Conservation and Recovery Act
RWMC	Radioactive Waste Management Complex
SAR	Safety Analysis Report
SL	Safety Limit
SNF	Spent Nuclear Fuel
SNM	Special Nuclear Material

SSC	Safety, Standards, and Compliance
TLV	Threshold Limit Value
TRA	Test Reactor Area
TS/S	Technical Specification/Standard
TSR	Technical Safety Requirement
UBC	Uniform Building Code
USF	Underground Storage (Facility) Area
WINCO	Westinghouse Idaho Nuclear Company, Inc

1.0 BLENDING OF U-233 TO <12% OR <5% ENRICHMENT

1.1 Missions

Uranium-233 (U-233), a uranium isotope, is a fissionable material capable of fueling nuclear reactors or being utilized in the manufacturing of nuclear weapons. As such, it is controlled as a special nuclear material. The Idaho National Engineering Laboratory (INEL) and Oak Ridge National Laboratory (ORNL) currently store the Department of Energy's (DOE's) supply of unirradiated U-233 fuel materials. Irradiated U-233 is covered by the national spent nuclear fuel (SNF) program and is not in the scope of this report.

The U-233 stored at ORNL is relatively pure uranium oxide in the form of powder or monolithic solids. This material is currently stored in stainless steel canisters of variable lengths measuring about 3 inches in diameter. The ORNL material enrichment varies with some material containing considerable amounts of U-235. The INEL material is fuel from the Light Water Breeder Reactor (LWBR) Program and consists of enriched uranium and thorium oxides in zircaloy cladding.

The DOE inventory of U-233 contains trace quantities of U-232, and daughter products from the decay of U-232 and U-233, resulting in increased radioactivity over time. These increased levels of radioactivity generally result in the need for special handling considerations.

The elimination of U-233 as a source special nuclear material (SNM) is desired. This data report discusses blending existing inventories with U-238 to reduce the concentration of U-233 so that the final blended material is below the SNM concentration. The blending of uranium isotopes needs to be performed in a manner which prevents any later separation attempts. Blending, as used in this document, refers to the mixing of liquid phase uranium compounds to achieve a homogenous product. Although blending could also be performed using pyrometallurgical processes, such processing is not discussed here. Reducing the concentration of U-233 to 12% would eliminate its attractiveness as a weapons material. Further reducing the concentration of U-233 to below 5% could make it a viable commercial nuclear fuel material. Removal of thorium contained in the LWBR material would be necessary if the uranium oxide is to be saleable as a commercial fuel material.

The ORNL materials will be loaded into fuel casks and transferred to CPP-666 at the INEL for processing. They will then be unloaded and moved to a hot cell for removal of the stainless steel housings. The uranium oxide will be dissolved in acid and transferred to the separations cells located in CPP-601 where depleted uranyl nitrate will be added to reduce the concentration of U-233 to the desired level (12% or <5%). Highly radioactive daughter products will be removed from the uranium solution in the separations area, thus allowing manual denitration and packaging of the uranium material in CPP-602 for storage.

INEL LWBR fuel materials are currently stored at the CPP-749 facility in canisters and at the Radioactive Waste Management Complex (RWMC) in type 6M waste drums. Both materials will be transferred to CPP-666 for blending. The outer 6M drum casings will be removed and the drum cores moved to the hot cell for disassembly and fuel removal. The canisters from CPP-749 will go directly to the hot cell for fuel removal. Fuel pins within the containers will be remotely cut and flared and the fuel pellets removed and dissolved in acid at the Fluorinel Dissolution Process (FDP) Cell at CPP-666. Following dissolution the materials will be transferred to the separations area at CPP-601 for blending with depleted uranyl nitrate and separation of highly radioactive daughter products. Removal of thorium

from the uranium is planned at this stage in the process. However, because calcining of the thorium as waste may be problematical, an alternative for leaving the thorium in the final product is also discussed. The remaining material will be packaged and returned to storage at CPP-749.

A small quantity of material at RWMC has a composition of uranium and zirconium oxide. This material, if processed, will require a variant of the CPP-666 dissolution process.

1.2 Assumptions

Blending will be performed in existing INEL spent nuclear fuel (SNF) reprocessing facilities using chemical dissolution, blending in the liquid phase, separation of radioactive daughter products, and denitration of the uranium to provide a stable oxide product suitable for storage. The aqueous dissolution and separations reprocessing approach is the same basic processing scheme used for over 40 years to reprocess fuel at the INEL and other DOE facilities.

The Idaho Chemical Processing Plant (ICPP) facilities to be used in this processing effort were designed and built to the DOE Orders in effect at the time of construction. Most of these facilities have been upgraded over the years but they do not meet all of the current requirements of DOE Order 6430.1A.

The following general assumptions were used in the generation of this data report. Any change to the basic assumptions could significantly change the results presented in this report.

- ▶ Only unirradiated fuel materials with reportable levels of U-233 are being considered for blending.
- ▶ Blending will be restricted to the chemical processing of liquid phase uranium compounds to provide the desired isotope distribution in the end product.

1.2.1 Facility Capacity/Capability Assumptions

The following facility capacity/capability assumptions were used in the generation of this data report. Any change to these basic assumptions could significantly change the results presented in this report.

- ▶ Processing of U-233 materials involves a radiological safety hazard from alpha particles about 5×10^3 times that of U-235. This high alpha hazard may be a major problem in the CPP-601 facility, which requires hands-on maintenance activities. The lack of corresponding beta and gamma radiation makes the alpha more difficult to detect and, therefore, more dangerous. It is assumed that CPP-601 will be determined usable for U-233 processing only after an extensive radiological safety evaluation to determine if the cells can be used safely in the presence of the alpha hazard.
- ▶ Facility capacity is confined by the input material enrichment and the quantity of liquid which must be denitrated to provide a stable oxide product. If thorium in the U-233 fuel is removed, the DOE inventory could be blended in under two years of facility operation. If thorium remains in the product oxide about four years of operation would be required to process the DOE inventory of U-233.

- ▶ Existing ICPP facilities can be successfully upgraded to meet the intent of former DOE Order 6430.1A or successor DOE Life Cycle Asset Management Orders for the facilities involved. Such upgrading would involve seismic qualification of CPP-601 and CPP-602, which may be only a documentation exercise but could potentially require structural modifications to the facility. The State of Idaho has previously allowed operation of these facilities without the analysis, but additional variances are unlikely.
- ▶ Hot cell equipment for high volume remote de-cladding and de-canning of U-233 fuel materials can be installed in existing facilities.
- ▶ Existing INEL facilities will be used for this effort with safety analysis documentation being upgraded where required. Construction of new processing facilities was not considered.
- ▶ The LWBR fuel was placed in CPP-749 using only the “Super Tiger” transporter. IT is assumed that the fuel containers can be modified to transport the fuel using the Peach Bottom cask.
- ▶ CPP-749 dry wells will be used for storage of uranium oxide materials after the completion of processing. The dry wells will require a revised criticality analysis for the modified fuel enrichment and fuel form for a fully water moderated environment.

1.2.2 Facility Operating Assumptions

- ▶ It is assumed that chemical processing operation sequences will readily accommodate the requirements of U-233 blending. Facilities which would be utilized in this blending operation have been in operation for many years but were shut down when SNF reprocessing was discontinued in 1992. Facility operations to perform U-233 blending should be planned to occur approximately 3 years after the commitment of funding to perform the work. This time would allow the design and construction of fuel pin removal tooling, criticality calculations and updates to the Safety Analysis Report (SAR), facility re-start maintenance, etc.
- ▶ The fuel process system raffinates are currently hard piped to the High Level Waste Tank Farm. Although only high activity, low-level waste is to be generated by the U-233 processing, it would logically be sent to the Tank Farm for calcination and disposal. The Tank Farm is currently restricted by a Court Order to receive no additional waste and to treat the waste currently in storage. An adjustment to this Court Order would be required to allow the area to receive the U-233 liquid waste. Diversion of the waste streams to a low level waste collection area is an alternative which will need to be considered in detail prior to performing any U-233 work. However, such a collection area is not currently available. The use of the existing Process Equipment Waste (PEW) evaporator system, with only the evaporator bottoms being sent to the tank farm, may be a viable alternative.

1.2.3 Compliance Assumptions

It is assumed that this process will be in full compliance with all applicable rules, regulations, codes, etc. The following sections discuss in detail all identified compliance issues.

1.2.3.1 Rules, Regulations, Codes, and Guidelines. General health and safety criteria for the design of safety-significant structures, systems and components at the ICPP are identified in Table 1-1. Structural

design criteria specify the basic design requirements of a building or facility. The criteria include code-of-record criteria determined at the start of design and life safety criteria that must be met when the building is ready to operate. Most of the design codes are code-of-record codes which means that the code specified during design is the required code when the building is built. Design requirements do not apply to existing facilities awaiting decontamination and decommissioning (D&D) or to sites being remediated. These structural design requirements apply only to new facilities or to extensive modification to existing facilities.

Table 1-1. Principal Health and Safety Design Criteria Use for Facility Construction		
Safety Significant Design Features	DOE Orders	Codes and Standards
General Design of Structures, Systems, Components	DOE Order 6430.1A ¹ DOE Order 5480.28 ² DOE Order 5480.11 ³	Uniform Building Code (UBC) ⁴ DOE-ID A/E Standards ⁵ SEN-35-91 ⁶ UCRL-15910 ⁷ UCRL-53526 ⁸ ASME/ANSI B31.3 ⁹ INEL Welding Manual ¹⁰ American Welding Society Stds. (AWS) ¹¹ American Society of Steel Construction (ASSC) ¹² American Concrete Institute (ACI) ¹³ 29 CFR 1910.119 ¹⁴
Pressure Vessels, Unfired	DOE Order 6430.1A	ASME B&PV Code, Section 8 ¹⁵ ANSI Hazardous Material Safety Requirements
Radiological Control Design Features	DOE Order 5400.1 ¹⁶ DOE Order 5480.11	10 CFR 20 ¹⁷ 10 CFR 835 40 CFR 190 ¹⁸ 40 CFR 191 ¹⁹ ICPP Radiological Control Manual ²⁰
HEPA Filtration DOP Testing	DOE Order 5400.1 DOE Order 5480.11	10 CFR 20 ICPP Radiological Control Manual
Gaseous Release Detection		Idaho Air Quality Standard ²¹
Fire Design Features	DOE Order 5480.7A ²² DOE-ID Order 5480.1B, Chapter VII ²⁶	Uniform Fire Code ²³ National Fire Protection Association ²⁴ Uniform Building Code Section 505(e) DOE-ID 12044 ²⁵ FM Approval Guide ²⁶ FM Data Sheets ²⁷
Seismic Design Features	DOE Order 6430.1A DOE Order 5480.28	General Design Features
Flood Design Features	DOE Order 6430.1A DOE Order 5480.28	Niccum, "Flooding Potential at the ICPP" General Design Features ²⁸
Wind Design Features	DOE Order 6430.1A DOE Order 5480.28	General Design Features

1.2.3.2 Safeguards and Security. Security requirements for Special Nuclear Material and storage facilities are controlled by DOE Orders (5630 series). Because of the enrichment of U-233 and combination with enriched U-235, the input material to the blending process will require implementation of safeguards and security requirements for category 1 attractiveness level D material.

U-233 is currently used as a baseline spike material to calibrate equipment for analysis of uranium for accountability purposes in the CPP facility. Introduction of U-233 into these facilities will force a major adjustment in the approach used for uranium accountability. The adjusted approach is expected to be a Davies-Gray titration to obtain total uranium combined with the spectrographic results to show the relative percentage of isotopes. The Davies-Gray method would need to be performed remotely and a method devised to dispose of the contaminated chromium solution resulting from the test method.

1.2.3.3 Environmental, Safety, and Health (ES&H). It is assumed that the required ICPP facilities can be successfully permitted under the Clean Air Act and the Resource Conservation and Recovery Act (RCRA) to allow full operation of the facilities to process the U-233.

ICPP facilities have not performed reprocessing operations since 1992. During the years following shutdown the application of RCRA under the Federal Facilities Compliance Act has changed the operability status of the facilities. Chemical processing systems in the ICPP facilities are designed with hard piping of chemicals between processing vessels, makeup systems, and liquid waste systems. As currently applied, all chemical lines from the processing vessel downstream to and including the entire waste system are considered and controlled as a RCRA mixed waste storage area, licensed under an interim Part A, RCRA storage permit. The regulations concerning double containment of liquid wastes are not currently being met at the CPP-601 D Cell and the WG/WH Tanks because of the single wall piping from the processing systems. A revised definition of the boundaries of the RCRA storage area, agreed upon by the State of Idaho would be required to allow U-233 reprocessing.

It is the policy of the INEL to maintain the highest standards for occupational safety and health, including implementing industrial safety in compliance with DOE Order 5483.1A, Occupational Safety and Health Program for DOE Contractor Employees at Government-Owned Contractor-Operated Facilities. This is achieved at ICPP through the application of the requirements presented in the ICPP Industrial Safety Manual and implementation by the ICPP safety organization.

An integral part of the ICPP safety organization is the industrial hygiene program which is derived from DOE Order 5480.10, Contractor Industrial Hygiene Program. This program is implemented in accordance with the ICPP Industrial Safety Manual to reduce or eliminate injury or illness to personnel due to exposure to hazardous chemicals or physical agents in the work place, prevent damage to property by industrial hygiene factors, and provide assistance as needed to construction safety personnel in the protection of their employees. The program is achieved through the anticipation, identification, evaluation and control of environmental factors and stresses found in the occupational environment by experienced industrial hygiene personnel.

The exposure criteria for INEL co-located workers and the general public are based on the Emergency Response Planning Guide Level 1 (ERPG-1) concentration for chemicals. This is the maximum airborne concentration below which nearly all individuals for any time up to one hour could be exposed to a plume from a credible accident without experiencing other than mild transient adverse health effects. This and lesser concentrations may cause minor, reversible, and nuisance effects, such as lacrimation (eye tearing) or objectionable odor. If chemical airborne concentrations exceed ERPG-1 concentrations,

evacuation of INEL co-located workers and possibly the general public should be considered. Evacuation is required if levels exceed ERPG-2.

1.2.3.4 Buffer Zones. In 1949, the United States Atomic Energy Commission established the INEL site as the National Reactor Testing Station (NRTS) for testing various types of nuclear reactors and associated equipment. In 1974, the NRTS was made a national laboratory by congressional action and renamed the Idaho National Engineering Laboratory. The INEL is near the western edge of the Upper Snake River Plain in southeastern Idaho. It contains 890 square miles or 570,000 acres, with an average elevation of 4850 feet above sea level.

The ICPP is one of the complexes located at the INEL. It receives, stores, and prepares for storage many types of government-owned SNF. The ICPP is located in the south-central portion of the INEL, about 42 miles west of Idaho Falls, Idaho. The restricted area of ICPP encloses 0.19 square miles (120 acres) of the total ICPP area, which is 0.23 square miles (147 acres). Figure 1-1 shows the location of the ICPP relative to the INEL outer boundary. This location allows the INEL to act as a buffer zone for the ICPP.

1.2.3.5 Decontamination and Decommissioning/Conversion. DOE must ensure that facilities are constructed and operated so that they may be decontaminated and decommissioned in a cost-effective manner. The D&D consideration includes limiting contamination of the facility, equipment, and site to the minimum extent consistent with accomplishing the purposes of the facility; providing design features that ease decontamination; incorporating design features that promote reuse of the contaminated facilities; considering D&D as part of the life cycle cost; and selecting a tentative D&D method during facility design²⁹.

Design of areas with the potential to become contaminated must incorporate measures to simplify future decontamination. Items such as service piping, conduits, and ductwork must be kept to a minimum in these areas and must be arranged to facilitate decontamination. Filters must be positioned in ventilation systems in locations that minimize contamination of ductwork. Walls, ceilings, and floors must be finished with washable or strippable covering. In some areas, metal liners must be used. All cracks, crevices, and joints must be caulked or sealed and finished smooth to prevent contaminated material accumulation in inaccessible areas³⁰.

1.2.3.5.1 Decontamination and Decommissioning Process—Conceptual plans for D&D are developed during the design stage of a facility and embody specific design commitments required later in the process. D&D plans are further developed and completed prior to the D&D activities.

Decontamination and Decommissioning Management Process: Decommissioning activities for a facility include: (1) characterization and development of baseline data, (2) review of the environmental requirements and regulations applicable, (3) technical engineering planning, (4) decommissioning operations, and (5) post decommissioning activities. Prior to decommissioning all bulk radioactive material, hazardous materials, and material which can be removed using the decontamination capabilities of the system or facility must be removed as part of the last operational activities. During the transition stage to decommissioning, the facility must receive surveillance and maintenance to maintain physical safety and security, and to reduce the potential public and environmental hazards according to Projects Departmental Procedure F.47, "Surplus Facility Surveillance and Maintenance Program³¹." The potential nuclear hazards must be controlled as discussed in DOE Order 5480.24³².

Figure 1-1. INEL Boundary and Facilities Locations.

- A. Transition Plan: At ICPP the transition phase of the process is governed by WSOP WE-33, "Transition, Turnover, Decontamination, and Decommissioning of ICPP Surplus Contaminated Facilities³³." Applications, agreements, and requests for acceptance of the facility for environmental restoration are processed, and a transition plan is prepared and implemented to serve the following purposes:
- 1) collects facility, process, and operation information relevant to decommissioning, including site characteristics and facility design descriptions and drawings;
 - 2) confirms the removal of nonprocess equipment and hazardous materials, including flushing of hazardous material from vessels and piping, and decontamination to normal operational levels;
 - 3) confirms the containment and monitoring of potential hazardous material releases and the control of unauthorized entry;
 - 4) reports on facility compliance with environmental, safety, and health regulations, surveys of hazardous materials remaining, and conditions of structures and barriers; and
 - 5) establishes budgets and fiscal responsibility for the transition process.
- B. Characterization/Decision Analysis: The development of the D&D Plan is governed by Projects Department Procedure F.52³⁴. The first step is the completion of an environmental checklist (Form ID 5440.1A) to identify if the activity is a categorical exclusion or requires an action description memorandum to determine if an environmental assessment or an environmental impact statement is needed.
- Preparation of a management plan for selected decontamination alternatives is next. For each of the major project elements in the management plan, the following must be reviewed:
- 1) environmental compliance issues,
 - 2) estimates of personnel needed,
 - 3) estimates of possible exposures,
 - 4) estimates of equipment needed,
 - 5) required technical development research,
 - 6) estimates of types and quantities of waste,
 - 7) discussion of removal of waste from the ICPP and INEL sites, and
 - 8) estimates of the time necessary for completion.
- C. Decontamination and Decommissioning Plan: The project schedule is developed with costs, reporting, and performance measures and integrated into the project management system. The schedule includes the coordination of independent verification,

requirements for safety analysis, and departmental and DOE reviews. The D&D SAR is developed to address the normal, abnormal, and accident events and the technical and institutional controls necessary for the decommissioning plan. The Operational Readiness Review culminates and verifies the completeness of the plan prior to the plan implementation.

- D. Decontamination and Post Decommissioning: The decommissioning is completed as specified in the decommissioning plan. Post D&D requires consideration of future uses and long range site plans; preparation of a final report, according to Projects Department procedure F.53, "Preparation of Decontamination and Decommissioning Report"³⁵; and preparation and implementation of a post decommissioning surveillance and maintenance plan, according to Projects Department Procedure F.47, "Surplus Facility Surveillance and Maintenance Program"¹⁰.

1.2.3.6 Toxicological/Radiological Exposure. The radiation protection criteria are derived from the orders listed in Table 1-2:

Table 1-2. Regulations Pertaining to Radiation Protection Criteria	
DOE Order	Radiation Protection Criteria
DOE Order 5400.5	Radiation Protection of the Public and the Environment
DOE Order 5480.4 ⁴⁰	Environmental Protection, Safety and Health Protection Standards
DOE Order 5480.24 ³²	Nuclear Criticality Safety
DOE Order 5480.10 ⁴¹	Contractor Industrial Hygiene Program
DOE Order 5480.11	Radiation Protection for Occupational Workers
DOE Order 5480.15	DOE Laboratory Accreditation Program for Personnel Dosimetry
DOE Order 5483.1A	Occupational Safety and Health Program for DOE Contractor Employees at Government-Owned Contractor-Operated Facilities.
10 CFR 835	

Radiation protection criteria are drawn from DOE Directives including DOE Order s5480.11 and 5400.5, and DOE Notice 5480.6, which directs the use of the Radiological Control Manual (DOE-ID, 1992).

Radiation protection criteria include normal operating criteria, limiting operational criteria, and siting and design criteria. Normal operating criteria are intended to maintain radiation dose below operational limits and to assist in optimizing radiation dose at levels ALARA. Operational radiation protection criteria are presented in Table 1-3 and establish administrative control levels, action levels requiring monitoring and assessment of normal operational radiation hazards, use of protective equipment, and notification of workers.

Table 1-3. Operational Radiation Protection Criteria			
Criterion	Quantity	Safety Basis	Reference
Radiological Performance Indicator Goals	Operation-specific measures occupational dose, personnel contaminations, waste volume effluent discharges	Indicator of effectiveness of radiation protection program - may be established by M&O contractor at site level or operation level.	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
DOE Administrative Control Dose Level	2,000 mrem/year total effective dose equivalent (TEDE) to personnel at DOE facilities (see Note)	Assure control of occupational dose, evaluation and approval of DOE Program Secretarial Official required to exceed control level.	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Administrative Dose Control Level	$\leq 1,500$ mrem/year TEDE to individual DOE-ID personnel 200 mrem/yr whole body	Assures control of occupational dose, evaluation, and approval of M&O senior site executive required to exceed control level.	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Lifetime Individual Dose Control Level	N rem TEDE, where N is person's age in years	Assures chronic occupational doses approach 1 rem/year for prevention of non-stochastic effects.	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Monthly Occupational Dose Guideline - Fetus of Declared Pregnant Worker	50 mrem/month dose equivalent	Prevention of fetal stochastic radiation effects.	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Special Dose Control Level for Workers Exceeding Lifetime Control Level	≤ 1 rem/year TEDE	Assures chronic occupational doses approach 1 rem/year for prevention of non-stochastic effects.	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Guideline for Occupational Dose Exclusively in Controlled Areas	≤ 0.1 rem/year TEDE	Ensures work in Controlled Areas is adequately isolated from Radiological Areas.	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Posted "Hot Spot"	> 0.1 rem/hr and five times general area dose rate	Alerts workers to localized external radiation hazard	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Radiological Planning Review of Nonroutine or Complex Activity	Nonroutine or Complex Activity in High Contamination Area or area exceeding 1 rem/hr	Provides for ALARA review, worker indoctrination, and assessment of work to assure worker protection	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Radiation Work Permit Required	Entry to High and Very High Radiation Areas, High Contamination Areas, Airborne Radioactivity Areas	Provides for radiation protection review, worker information to assure worker protection.	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Protective Clothing Required	Entry to Surface Contamination and Airborne Radioactive Materials Areas	Protection against personnel contamination	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Doubled Protective Clothing Required	Entry to High Contamination Areas	Protection against personnel contamination	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Guidelines for Reuse of Protective Clothing	$< 10,000$ dpm/ 100 cm ² beta gamma, $< 1,000$ dpm/100 cm ² alpha	Protection against personnel contamination.	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)

Table 1-3. Operational Radiation Protection Criteria			
Criterion	Quantity	Safety Basis	Reference
Guideline for Use of Supplemental Dosimeters	Entry to High Radiation Area; potential to exceed 10% of a dose control level in one day	Self-monitoring to prevent exceeding control levels and to meet dose budgeted for work.	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Dose Evaluation Due to Personnel Contamination or Inadvertent Airborne Exposure	100 mrem based on site specific instrumentation factors	Assures monitoring of occupational exposure.	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Reproducibility of Radiation Protection Monitoring Instrument Checks	$\pm 20\%$ initial mean response checked daily before use	Assures statistical control of monitoring instrument accuracy.	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Calibration of Radiation Protection Monitoring Instruments	Within one year	Assures monitoring instrument accuracy.	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)

Note: Total Effective Dose Equivalent (TEDE) is the sum of the external deep dose equivalent and the internal 50-year committed effective dose equivalent.

Limiting operational criteria establish limits for dose to workers and the public during normal operations, abnormal occurrences, and emergencies, including dose limits from radioactive material effluent which may be discharged to the environment and for emergency response personnel. Table 1-4 presents the limiting operational criteria for radiation protection of workers, the public, and the environment.

Table 1-4. Limiting Operational Radiation Protection Criteria			
Criterion	Quantity	Safety Basis	Reference
Occupational Whole Body Dose	5 rem TEDE	Prevention of stochastic radiation effects.	DOE 5480.11 (DOE, 1988); DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Occupational Dose to the Lens of the Eye	15 rem TEDE	Prevention of non-stochastic radiation effects to the eye, principally cataracts	DOE 5480.11 (DOE, 1988)
Occupational Dose to Any Individual Organ, Tissue, or Extremity Except the Lens of the Eye and Skin Exposures Localized to $< 10 \text{ cm}^2$	50 rem TEDE	Prevention of non-stochastic radiation effects to individual organs, tissues, and extremities.	DOE 5480.11 (DOE, 1989); DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Occupational Dose to the Fetus of a Declared Pregnant Worker	0.5 rem TEDE in 9 months (conception to birth)	Prevention of fetal non-stochastic and stochastic radiation effects.	DOE 5480.11 (DOE, 1988); DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Occupational Dose to Minors and Students	0.1 rem/year TEDE	Prevention of stochastic radiation effects; dose minimization	DOE 5480.11 (DOE, 1988)
Visitors And The Public Within Radiologically Controlled Areas	0.1 rem/year TEDE	Prevention of stochastic radiation effects; dose minimization	DOE 5480.11 (DOE, 1988)

Table 1-4. Limiting Operational Radiation Protection Criteria			
Criterion	Quantity	Safety Basis	Reference
Primary Dosimeter Required	>100 mrem Annual Dose Equivalent or 10% of occupational limits	Assures monitoring of significant occupational exposures.	DOE 5480.11 (DOE, 1988)
Internal Dose Evaluation Including Bioassay Monitoring	>100 mrem/year CEDE	Assures monitoring of significant occupational exposures.	DOE 5480.11 (DOE, 1988)
Air Monitoring Required	Air Sampling: 0.1 Derived Air Concentration (DAC) Continuous Air Monitor: 1 DAC averaged over 8 hours	Assures monitoring of potential internal exposures.	DOE 5480.11 (DOE, 1988); DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Posted "Caution Surface Contamination Area, RWP Required for Entry" and Required Detection Capabilities for Surface Contamination Survey	Uranium and Decay Products: 1,000/5,000 removable/total alpha dpm/100 cm ² ; Transuranics: 20/500 removable/total dpm/ 100 cm ² ; Th-natural: 200/1,000 removable/total dpm/ 100 cm ²	Ensures detection, worker awareness, surveillance, and control of contaminated areas to prevent contamination spread, personnel contamination, and occupational dose.	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Posted Surface Contamination and Required Detection Capabilities for Surface Contamination Survey (Cont.)	Beta-Gamma Emitters: 1,000/5,000 removable/ total beta/gamma dpm/ 100 cm ² ; Initiated Compounds: 10,000 dmp/100 cm ²		
Posted "Danger High Surface contamination Area. RWP Required for Entry"	100 times surface contamination area levels from above	Ensures worker awareness, surveillance; control of highly contaminated areas to prevent contamination spread, personal contamination, occupational dose.	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Airborne Radioactive Material Concentration (μCi/cc) for Posted "Caution Airborne Radioactive Materials Area, RWP Required for Entry"	10 % Derived Air Concentration (DAC in μCi/cc)	Alerts workers to potential internal dose hazard.	DOE 5480.11 (DOE 1988) DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Posted "Caution, Radiation Area, TLD, and RWP Required for Entry"	5 - 100 mrem/hr	Alerts workers to external dose hazard.	DOE 5480.11 (DOE 1988)
Posted "Danger, High Radiation Area, TLD, Supplemental Dosimeter, and RWP Required"	100 mrem/hr - 500 rad/hr	Alerts workers to external dose hazard.	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Posted "Grave Danger, Very High Radiation Area, Special Controls Required for Entry"	> 500 rad/hr	Alerts workers to external dose hazard and impose added access controls.	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Posted "Caution, Soil Contamination Area"	Contaminated soil not releasable in accordance with DOE 5400.5 (DOE, 1990)	Ensures worker awareness, surveillance and control of contaminated soils to prevent contamination spread, personnel contamination, and occupational dose.	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992); DOE 5400.5 (DOE, 1990)

Table 1-4. Limiting Operational Radiation Protection Criteria

Criterion	Quantity	Safety Basis	Reference
Labeled Radioactive Material, Posted Radioactive Material Area When Not in Contamination Area or Airborne Area	Internal or external contamination $\geq$ Posted Contamination Area levels; Sealed Sources	Ensures worker awareness, surveillance, and control of radioactive materials to prevent contamination spread, personnel contamination, and occupational dose.	DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Planned Special Exposure Limit	10 rem Annual Effective Dose Equivalent	Prevention of stochastic effects to exposed worker where lower-dose alternatives are unavailable or impractical. DOE-ID review and approval required.	DOE 5480.11 (DOE, 1988)
Emergency Exposure Guideline for Prevention of Major Property Damage, Reduction of Potential for Public Health Hazard, and Recovery of Deceased Victims	10 rem EDE	Prevention of stochastic effects to exposed worker during vital emergency response actions. Approved by emergency response official.	DOE 5480.11 (DOE, 1988); DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Voluntary Emergency Exposure Guideline For Prevention of Major Property Damage or Reduce Potential for Public Health Hazard	25 rem EDE	Minimization of non-stochastic effects to volunteer worker during vital emergency response actions. Approved by emergency response official.	DOE 5480.11 (DOE, 1988); DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Voluntary Emergency Exposure Guideline For Lifesaving and Prevention of Public Health Hazard	> 25 rem EDE	Minimization of non-stochastic effects to volunteer worker during vital emergency response actions. Approved by emergency response official.	DOE 5480.11 (DOE, 1988); DOE-ID 10399 Radiological Control Manual (DOE-ID, 1992)
Unrestricted Release of Items From Controlled Areas, Excluding Hazardous Materials	No hazardous material, contamination is ALARA and meets the following: uranium and decay products, alpha emitters, <1,000/15,000 removable/maximum alpha dpm/100 cm ² ; transuranics, 200/3,000 removable/maximum dpm/100 cm ² ; beta and gamma emitters, 1,000/ 15,000 removable/ maximum beta-gamma dpm/100 cm ²	Ensures evaluation of potentially contaminated items prior to release to prevent contamination spread and protect the public.	DOE 5400.5 (DOE, 1990)
Contamination Limits on Packages Offered For Offsite Transport	beta-gamma emitters, uranium, and thorium; <22,000 dpm/100cm removable: alpha emitters, <2,200 dpm/ 100cm ² removable	Prevents contamination spread; in public areas.	49 CFR 173.443 (DOE, 1993)

Table 1-4. Limiting Operational Radiation Protection Criteria			
Criterion	Quantity	Safety Basis	Reference
External Radiation Limits For Closed Exclusive-Use Vehicles Transporting Radioactive Materials	200 mrem/hr at any exterior contact point, 10 mrem/hr at 2 meters, 2 mrem/hr in any space occupied by person not under radiation protection program	Minimizes public exposure during transportation.	49 CFR 173.441 (DOE, 1993)
DOE Public Dose Limit-Total	100 mrem/yr TEDE, all pathways	Minimizes public exposure to radiation due to DOE activities.	DOE 5400.5 (DOE, 1990)
DOE Public Dose Limit - Air Emissions - Total	10 mrem/yr TEDE	Minimizes public exposure to radiation due to DOE activities.	DOE 5400.5 (DOE, 1990)
DOE Public Dose Limits Disposal of Spent Nuclear Fuel, High Level Waste, Transuranic Wastes	25 mrem/year dose equivalent 75 mrem committed dose equivalent to any organ	Minimizes public exposure to radiation due to DOE activities.	DOE 5400.5 (DOE, 1990)
DOE Public Dose Limit - Drinking Water	4 mrem/yr TEDE	Minimizes public exposure to radiation due to DOE activities.	DOE 5400.5 (DOE, 1990)

1.2.3.6.1 Protection from Toxicological Exposure—Hazardous material protection criteria for workers are based on Threshold Limit Value (TLV), or Permissible Exposure Limit (PEL). PELs are those issued by the OSHA in 1989.

Actual or anticipated exposures above TLVs or PELs necessitate the use of personal protective equipment (PPE) beyond normal work clothing. Protective level criteria are classified as Levels 1, 2, 3, and 4. The levels are based on whether a worker may be exposed to chemical concentrations above the PEL or TLV, or to immediately dangerous to life and health (IDLH) concentrations. Table 1-5 lists these levels.

Table 1-5. Recommended Levels of Personal Protective Equipment	
Level 4	Normal attire if the exposure $\leq$ PEL or TLV.
Level 3	PPE beyond what is normally required on a construction site is not required, but may be worn to keep clean, i.e., coveralls & gloves; Exposure $\leq$ PEL or TLV.
Level 2	PPE to include gloves, coveralls, and respirators required; Exposure $\geq$ PEL or TLV See Table 6-7.
Level 1	PPE to include encapsulating suit and Self-Contained Breathing Apparatus (SCBA); Exposure $\geq$ IDLH.

For estimated or monitored airborne concentrations of chemicals, consideration shall be given to chemicals with similar effects on body systems such as the respiratory tract, liver, nervous

system, kidneys, and reproductive systems. A chemical by itself may be below its TLV or PEL, but when added to airborne levels of other chemicals with similar effects on the body could result in exposure above the PEL or TLV.

1.2.3.7 Waste Management. The need to minimize solid and liquid discharges is recognized at the INEL. This section describes generation, handling, and disposal of nonradiological solid, liquid, and gaseous waste.

The waste management and control philosophy for the ICPP is that all solid, liquid, and gaseous discharges are safely managed according to applicable policies stated in DOE Order 5820.2A⁴², DOE Order 5480.11³, DOE Order 5484.1⁴³, DOE-ID Order 5480.2⁴⁴, DOE-ID Order 5484.1A⁴⁵, 40 CFR 61⁴⁶, 40 CFR 241⁴⁷, and 40 CFR 260-270⁴⁸.

Solid and liquid hazardous waste will be disposed of in accordance with the Transportation of Hazardous Materials Requirements Manual. Hazardous waste will be shipped off-site through the INEL waste disposal contractor.

Nonradioactive, nonhazardous, solid waste will be disposed of at the INEL's landfill. The nonradioactive, nonhazardous, liquid waste will be disposed of to the service waste/steam condensate return, or sanitary waste systems as appropriate.

1.2.3.7.1 Gaseous Waste—

1. **Source:** Exhaust air from each area will be appropriately processed in their off-gas system. Processing may include such items as HEPA filtration knockout pots, demisters, superheaters, and scrubbers. The exhaust gases will then be drawn through the off-gas blowers to a stack (if required) for discharge to the environment. The ICPP main stack and other exhausts are monitored for hazardous radioactive effluents.
2. **Management:** A monitoring program to measure nonradioactive and radioactive materials resulting from routine and accidental releases from the INEL is carried out by LITCO for all locations on the INEL site and by the Idaho environmental Research Foundation, the United States Geologic Society (USGS)^{51,52}, and the National Oceanic and Atmospheric Administration for all locations beyond the site boundary. The on-site⁵⁰ and off-site⁴⁹ monitoring program is summarized in Table 1-6.

Individual INEL facilities are monitored by their respective operating contractors. LITCO reports ICPP effluent and environmental monitoring results in quarterly reports⁵³.

Table 1-6. INEL On-Site and Off-Site Nonradiological Environmental Monitoring Program Summary

Medium Sampled	Type of Analysis	Analysis Method	Monitoring Frequency	Approximate MDC ^a	MDC ^a as % Appropriate EPA Regulation
Air	Particulate matter	Gravimetric	Quarterly	15 µg/mL	25
	Nitrogen dioxide	Chemiluminescent	Continuously	6 mg/m ³	4
	Sulfur dioxide	Fluorescent	Continuously	3 µg/m ³	4
Surface Water					
TRA pond	Ph	Electrode	Monthly	N/A	N/A
	Sodium	Specific ion electrode	Monthly	2 mg/L	N/A
	Chlorides	Specific ion electrode	Quarterly	2 mg/L	N/A
	Chromium (total)	Atomic absorption	Quarterly	0.05 mg/L	1
USGS samples ^b	Specific Conductance	Conductometric	Semiannually	N/A	N/A
	Chlorides	Specific ion electrode	Semiannually	2 mg/L	0.8 ^c
Groundwater					
USGS wells ^d	Specific conductance - 28 wells	Conductometric	Quarterly	N/A	N/A
	Specific conductance - 68 wells	Conductometric	Semiannually	N/A	N/A
	Sodium - 4 wells	Specific ion electrode	Quarterly	2 mg/L	N/A
	Sodium - 80 wells	Specific ion electrode	Annually	2 mg/L	N/A
	Chromium (total) - 12 wells	Atomic absorption	Quarterly	0.05 mg/L	100
	Chromium (total) - 25 wells	Atomic absorption	Semiannually	0.05 mg/L	100
	Chlorides - 28 wells	Specific ion electrode	Quarterly	2 mg/L	0.8 ^c
	Chlorides - 68 wells	Specific ion electrode	Semiannually	2 mg/L	0.8
	Nitrates - 48 wells	Ion chromatography	Annually	0.5 mg/L	1
	Major inorganic water quality constituents	Various	5 years	Varies	Varies

Table 1-6. INEL On-Site and Off-Site Nonradiological Environmental Monitoring Program Summary

Medium Sampled	Type of Analysis	Analysis Method	Monitoring Frequency	Approximate MDC ^a	MDC ^a as % Appropriate EPA Regulation
USGS wells ^d (cont.)	Organic contaminants				
	Dissolved organic carbon	Wet-oxidation	Once ^e	0.1 mg/L	N/A
	Herbicides	Gas chromatography	Once	0.01 µg/L	Varies ^f
	Insecticides	Gas chromatography	Once	0.01 to 0.1 µg/L	Varies
	Total PCBs	Gas chromatography	Once	0.1 µg/L	N/A

- a. MDC = minimum detectable concentration.
- b. Samples taken from creeks and rivers inside and outside INEL boundaries.
- c. Based on National Secondary Drinking Water Standard (aesthetic qualities relating to public acceptance).
- d. USGS observation and production wells.
- e. USGS Water Resources investigation 82-14, performed in 1980. Selected wells will be analyzed for organic contaminants periodically in the future.
- f. Regulations are being promulgated for individual compounds.

Ambient pollutant concentrations currently measured or calculated at the INEL are for sulfur dioxide (SO₂), oxides of nitrogen (NO_x), and total suspended particulates (TSPs). NO_x and SO₂ concentrations through 1989 were calculated from emissions data reported by the INEL Industrial Waste Management Information System⁵⁴. Calculated annual average ground-level concentrations of SO₂ and NO_x at the southern INEL boundary (near Atomic City) are shown in Table 1-7. The calculations use the MESODIF model⁵⁵. At the INEL boundary, calculated concentrations of SO₂ and NO_x are well below the national primary ambient air quality standards⁵⁶.

In 1983, a monitoring station containing NO_x and SO₂ analyzers was installed on U.S. Highway 20/26, west of Central Facilities Area (CFA). This monitor and others provided the data shown in Table 1-7. For example, NO_x concentrations from April 1 to December 31, 1983, averaged 0.7 µg/m³. The average SO₂ concentration from September 12 to December 31, 1983, was 0.4 µg/m³. These concentrations are only 4 and 1%, respectively, of air quality standards. From this example, a correlation can be drawn that concentrations are normally well below the air quality standards.

The air samplers at locations both on and off of the INEL are used primarily to detect radioactive particulates. The sampler filters are weighed quarterly to estimate the TSPs. Results are shown in Table 1-7. The average particulate concentrations at groups of stations occasionally exceed the national secondary ambient air quality standard of 60 µg/m³ due primarily to non-ICPP sources as described below.

Major sources of particulates are believed to be dust from the dry desert floor; off-site farming activities; and, in some locations, vehicle exhaust, industrial activity, and combustion byproducts of various fuels for heating. Comparisons of average concentrations of particulates at boundary and distant locations demonstrates no significant contribution from INEL activities to off-site

concentrations of particulates. A monitoring station specifically for particulates was started at the CFA in 1984 and currently provides data which is included in Table 1-7.

Table 1-7. Nonradiological Air Contaminant Concentrations On or Near the INEL Site (Concentrations in $\mu\text{g}/\text{m}^3$)					
Annual Average Ground-Level Concentration at the Southern INEL Boundary			Annual Average Concentrations* of Total Suspended Particulates		
Year	Sulfur Dioxide	Nitrogen Oxides	On-Site (11 Stations)	Boundary (5 Stations)	Distant (3 Stations)
1980	0.4	0.4	----	38 ± 11	53 ± 19
1981	0.7	0.7	32 ± 11	42 ± 12	39 ± 38
1982	0.3	0.3	22 ± 6	30 ± 9	34 ± 18
1983	0.4	0.7	10 ± 9	8 ± 9	18 ± 20
1984	0.4	0.6	----	20 ± 2	41 ± 31
1985	0.7	0.4	47 ± 3	43 ± 3	40 ± 3
1986	0.5	0.3	23 ± 6	31 ± 9	39 ± 9
1987	0.5	1.4	45 ± 16	34 ± 8	28 ± 8
1988	0.5	2	32 ± 13	35 ± 9	50 ± 20
1989	0.6	0.5	17 ± 2	30 ± 7	40 ± 14
Std.	80 ^b	100 ^b	60 ^c	----	----

a. The approximate minimum detectable concentration (MDC) is about $15 \mu\text{g}/\text{m}^3$ (mean $\pm$ 95% confidence interval about the mean).

b. National Primary Ambient Air Quality Standards (40 CFR 50).

c. National Secondary Ambient Air Quality Standard (40 CFR 50).

1.2.3.7.2 Liquid Waste—

1. **Source:** ICPP activities, by their very nature, generate radioactive and nonradioactive liquid waste. These may include various flush solutions and water used to decontaminate structures or machinery. Nonradiological liquid waste may include sanitary waste.
2. **Management:** Nonradioactive hazardous chemical waste is sent to the PEW system or off-site for disposal. Nonhazardous liquid waste is transferred to the sanitary sewer system.

Nonradiological characteristics of the aquifer are monitored at the same group of wells that are used for radiological monitoring. Some of the constituents monitored are

chlorides, chromium, and sodium. Specific conductance is also monitored. These waste constituents originate from percolation ponds at the Test Reactor Area (TRA), the ICPP, and some from waste disposal at the Naval Reactor Facility.

Water samples are collected from off-site drinking water production wells and from the Snake River. About 100 observation wells, including all on-site production wells in use, are also sampled. Locations and the frequency of sampling at various wells can be found in the ICPP SAR Chapter 1, "Site Characteristics"⁵⁸.

3. **Treatment:** Treatment of liquid waste, such as concrete stabilization, is discussed in specific addenda.

1.2.3.7.3 Solid Waste—

1. **Source:** Solid waste is either equipment that has served its useful life or failed equipment that cannot be economically repaired for reuse. Other solid waste may consist of disposable protective clothing, plastic, tape, and such materials. The waste also may consist of any construction materials such as concrete, steel, and soils. Small quantities of hazardous nonradiological solid waste may be generated during cleanup.

Nonhazardous, nonradioactive solid waste consists of trash from normally occupied areas. This waste may include empty shipping containers not contaminated with hazardous materials, used cleaning materials, broken furniture, tools, etc.

2. **Management:** All hazardous nonradioactive waste material is placed in transparent polyethylene color-coded bags with the words "Hazardous Waste" printed on the side. Additional labeling for specific waste materials may be required.
3. **Treatment:** There are no facilities currently available at the INEL for disposing of nonradioactive hazardous waste. Accumulated solid hazardous waste is shipped off-site to a commercially operated hazardous waste disposal site. A commercially operated transport company is contacted to transport this nonradioactive hazardous material on a periodic basis for off-site disposal. In some cases, solid waste will be packaged for off-site shipment and stored on-site until it can be transported to the appropriate off-site facility. Nonradioactive, nonhazardous solid waste items are collected and disposed of in waste dumpsters. The dumpsters are periodically emptied into refuse trucks, and the solid nonradiological, nonhazardous waste is transported to and disposed of at the INEL sanitary landfill.

1.2.3.7.4 Waste Minimization Program—The following practices are being used at ICPP facilities to minimize waste generation: closed-loop systems are used to minimize the waste generated and decontamination facilities are available for recyclable materials that can be decontaminated, repaired, and reused.

2.0 FACILITY DESCRIPTION

2.1 General Facility Description

2.1.1 Functional Description

LWBR fuel rods stored in CPP-749 dry silos will be transferred from the underground silos to the FDP remote handling cave hot process cell in CPP-666, decanned, and declad. Dissolution will take place in an acid thorex dissolvent using one of the three available dissolver vessels. Aqueous product will be diluted with a depleted uranium solution, and adjustment of the dissolver product will provide the solution for transfer to CPP-601. Depleted uranium blending to achieve isotopic dilution will reduce the U-233 enrichment to 12% or less than 5%. The adjusted and diluted aqueous solution will be processed in an existing CPP-601 first cycle extraction system, using an Acid Thorex process. The preferred option is for thorium to be separated from the uranium and remain in the first cycle extraction waste stream, which will drain into stainless steel tanks in an existing high level waste tank farm. Radioactive daughter products will remain with the waste stream. This aqueous waste will be mixed with existing aqueous HLW, and will be calcined to a granular product in a New Waste Calcining Facility (NWCF).

The first cycle product will be processed a second time through an acid-deficient hexone extraction system where the remaining thorium will be rejected. Here also, the aqueous waste stream will be drained to the existing Tank Farm. The purified uranium product will be denitrated and converted to granular uranium oxide (UO_3), and placed in 5.5 inch diameter steel cans. These product cans may be returned to the underground silos for interim storage.

2.1.1.1 RWMC U-233 Rods. U-233 bearing LWBR rods from the RWMC will follow the same path as the fuel above. An exception will be a small number of rods containing $\text{ZrO}_2\text{-UO}_2$ compounds which will be dissolved with an HF dissolvent similar to that used for zirconium clad fuels. Unirradiated LWBR rods are being stored in 55/110 gal drums at the RWMC.

2.1.1.2 U-233 Oxides from ORNL. Over 400 kg of U-233 in 1379 kg of total uranium is stored in stainless steel and aluminum cans in building 3019 at ORNL. The uranium exists as various uranium oxide compounds, both as powders and monoliths. Also, some containers have a mixture of enriched U-233 and enriched U-235.

For processing, the ORNL cans would be transferred to the ICPP FDP cell, where the cans could be opened, and the contents dissolved. The resulting product, uranyl nitrate in nitric acid, would be transferred to CPP-601 where the product would be diluted with aqueous depleted uranium as described for the LWBR material above. The isotopically diluted product would be processed only in the hexone extraction system, as there would be no thorium material to remove. Only the U-232 radioactive daughter products have to be removed before denitration. Waste streams and uranium product streams would be treated the same as with the LWBR process.

2.1.2 Plot Plan

The ICPP is owned and administrated by the DOE to receive, store, and prepare for storage many types of government-owned spent nuclear fuels. The overall ICPP mission is to cost-effectively manage all activities in a manner that protects the safety of INEL employees, the public, and the environment by the following:

1. Developing advanced technologies to condition SNF for permanent off-site disposal and to achieve waste minimization.
2. Receiving, storing, and preparing Navy and other DOE assigned SNF for storage.
3. Managing all wastes in compliance with applicable laws and regulations.
4. Identifying and conducting site remediation consistent with facility transition activities.
5. Seeking out and implementing private sector technology transfer and cooperative development agreements.

The original ICPP was constructed in 1951. Prior to 1992 the primary mission of the ICPP was the recovery of uranium from U.S. government SNF. A limited amount of research and development work is done, generally in support of ICPP processes. Isolating and solidifying the waste fission products that resulted from past reprocessing and facility clean-up are ongoing activities. In addition, nonprocessable fuel is stored at the ICPP in special long-term storage facilities. The current mission involves fuel storage and waste management activities with no fuel reprocessing.

The high-level radioactive liquid waste from past fuel reprocessing operations has been collected in stainless steel waste storage tanks inside underground concrete vaults. HLLW is no longer being generated from fuel reprocessing activities since reprocessing is no longer being done. The remaining stored liquid waste will be calcined to solid granules in the fluidized-bed calcination process in the NWCF. In future years it may be handled in other appropriate processes not yet in place. The resultant high-level radioactive solid waste, called calcine, is presently stored in air-cooled stainless steel bins contained in partially buried concrete vaults. Other liquid and gaseous waste streams are processed to comply with the appropriate limits established by DOE, the EPA, and the State of Idaho. These are controlled to levels of radioactivity ALARA. Major effluent streams are monitored to detect any deviations from expected and allowable contamination levels. Ultimately, all radioactive wastes not suitable for release to the environment are converted to solid wastes and are stored or disposed of in accordance with established procedures.

The original ICPP design was based on a direct contact maintenance philosophy: maintenance was performed by direct physical contact during periodic shutdowns and after extensive decontamination of cells and equipment. The direct contact maintenance approach may be inadequate to handle the alpha radiation problems associated with processing U-233 materials without major modification. Newer facilities, for example, NWCF and FAST, are designed for remote maintenance to reduce both radiation exposure and the extent of decontamination required.

2.1.3 Building Descriptions

2.1.3.1 Underground Storage Facility (CPP-749). The Underground Storage Facility (USF), CPP-749, is a facility for storing irradiated and unirradiated nuclear fuel elements and components in underground dry wells. The facility consists of vertical dry storage wells designed to keep the fuel storage units dry, to provide isolation to prevent neutron interaction between fuel units, and to provide a heat sink to dissipate the fission product decay heat from irradiated fuels.

The USF has two distinct dry well types. The original dry wells contain Peach Bottom fuel elements and Fermi blanket assemblies. The Peach Bottom elements are canned in aluminum, with 18 cans contained in an aluminum basket. This basket is the storage unit contained in one dry well. Fermi blanket assemblies are stored in long stainless steel canisters, one canister per dry well. The original dry wells consist of a cylindrical steel liner with a grout plug bottom and a welded-on cover. A shield plug is placed in the top part of each dry-well under the cover. Ports are provided in the covers for sampling the dry-well atmosphere.

LWBR fuels are stored in second generation dry wells. Unirradiated LWBR fuel rods are stored in stainless steel canisters. There are two sizes of second generation dry wells. Irradiated LWBR fuels, and the single larger unirradiated fuel canister, are stored in 30 inch dry wells. Unirradiated fuels are stored in 12 inch diameter dry wells. Two unirradiated LWBR canisters are stored in each dry well, stacked end to end. Storage of the smaller unirradiated LWBR canisters in a larger dry well is prohibited to prevent possible over batching^a.

The second generation dry wells have double-steel wall construction. The inner cylinder has a welded-in bottom and is grouted in place inside the outer cylinder. The top cover is bolted on, with a rubber gasket for sealing. Inlet and outlet lines are provided for purging, sampling, and water removal, if necessary. A shield plug is placed in the top part of each dry well containing irradiated fuel, then the top cover is bolted on. Crush pads are placed in the bottom of the dry wells to protect the canisters during installation and removal. A crush pad is also placed between the two unirradiated LWBR canisters in a dry well. Both original and second generation dry wells are equipped with cathodic protection systems for corrosion protection.

The entire USF is enclosed by a chain-link fence that is kept locked except when access is required for operational purposes. Electrical power is provided for powering small power tools and radiation monitors. No other utilities are provided. Portable radiation monitors are brought in as needed for a particular job. No permanent radiation monitors are provided.

Dry well atmospheres are sampled every two years to detect any condition that would indicate corrosion or compromised fuel containment. If any of these indicators are present, a corrective action plan is developed and appropriate steps are taken to correct or recover from the condition.

2.1.3.2 FDP (CPP-666). The FDP is located in the FAST Facility and was last operated in 1988 after two campaigns. The process was in a maintenance turnaround status until the announcement that reprocessing would be discontinued. Since April 28, 1992, the FDP has been in a transition status.

All reagent makeup and feed vessels have been emptied of their contents except for the cadmium nitrate and cadmium sulfate storage tanks. These two tanks have been physically isolated by blind flanges, and the appropriate valves and pipes from the chemical makeup system have been cut and capped. The fluoboric acid storage tank has been emptied, and the fluoboric acid has been returned to the vendor. Off-site vendors have been identified to accept the contents of the remaining vessels.

^a A "safe batch" is that quantity of nuclear material which can be safely handled or stored without danger of criticality. It is normally a quantity equal to 45% of the critical mass for a fully moderated and reflected system. Over-batching or double batching refers to two safe batch quantities located in the same container.

All three FDP dissolution trains have undergone chemical flushing followed by water rinses. Remaining radioactive solids have been sampled and determined to consist mainly of aluminum and zirconium (typical of the FDP). Sample analysis has found no hazardous constituents inside the dissolution trains and there is less than 0.03 lb (15 g) of uranium total in the three trains. All fuel transfers into the FDP Cell have ceased. All valves transferring product from the FDP to CPP-601 have been closed. As a result, the FDP cell is classified and maintained as a mass Criticality Control Area (CCA).

The dissolver off-gas (DOG) and cell off-gas (COG) prefilters and HEPA filters contain both radioactive materials and cadmium and are considered mixed waste. The FDP cell has RCRA interim status for storage of spent filters. There are 105 spent DOG prefilters, 40 final DOG HEPA filters, and 2 COG HEPA filters currently stored in the FDP cell. The spent HEPA filters are expected to be treated by the Filter Leach Process located in the NWCF.

A minimum number of systems remain in service indefinitely to provide adequate H&V, contamination control, fire protection, and basic utility services. All other process-related systems have been taken out-of-service. Most of the process instrumentation, the Data Processing System Enhancement (DPSE), and the Plant Protection System (PPS) have been taken out-of-service.

Most of the technical specification/standard (TS/S) for the FDP have been inactivated and are being canceled. The operating and administrative procedures have been reviewed and have either been rewritten or canceled to reflect the current configuration. Many of the remote operating procedures have been retained for use as needed. Their use will be evaluated during annual reviews.

No staffing requirements remain for the FDP facility because the FDP is in shutdown status. Personnel are required to maintain the remaining instruments and perform routine surveillance.

Any further uses of FDP would require a new SAR and new technical safety requirements (TSRs) prior to startup.

2.1.3.3 Fuel Processing Facility (CPP-601). Most processes that were used for fuel reprocessing are housed in one processing complex (CPP-601). Only the FDP, which is housed in the FAST building (CPP-666), is not located in the processing complex. This main processing complex is a rectangular structure about 240 ft (73.2 m) long, 100 ft (30.5 m) wide, and 90 ft (27.4 m) high from lower level to the top of the roof. In one section, the building extends 60 ft (18.3 m) below ground level. Nearly 30 percent of the building consists of concrete walls for shielding and structural purposes.

Process equipment in CPP-601 is contained within 29 cells, most of which are about 20 ft (6.1 m) square and 28 ft (8.5 m) deep. The bottom of each cell is lined with stainless steel, and most of the equipment is stainless steel, except those pieces made of more exotic alloys for resistance to sulfuric or hydrofluoric acids or to electrolytic currents.

With the exception of four special cells provided with viewing windows and manipulators, the in-cell process equipment was controlled from an operating corridor. The corridor runs the length of the building between cells and contains graphic control panels, control and monitoring instrumentation, and pneumatic-valve-control stations for regulating the flow of liquids through the process. Below the operating corridor are service piping and cell access corridors. Sampling and cell ventilation corridors are located outside the row of cells. The top story of the building is an unpartitioned area that was used for reagent storage, makeup, and charging of fuel elements into the cells.

CPP-601 has had all remaining product and waste solutions removed from its systems. All systems have undergone flushing to remove any residual material. Equipment that was required to recover uranium from dissolved fuels remains within the various cells and is awaiting D&D. Systems remaining in CPP-601 include aluminum fuel dissolution, zirconium clad alloyed fuel dissolution, electrolytic fuel dissolution, graphite fuel dissolution, uranium rework and salvage, feed preparation, feed clarification, TBP extractions, TBP purification, hexone extraction, hexone purification, intercycle storage, aqueous uranyl nitrate storage, and various off-gas systems.

2.1.3.4 Uranium Denitration Process and Product Storage (CPP-602). The Uranium Denitration Process is used to purify and concentrate uranyl nitrate solution from the Hexone extraction processes to a granular uranium trioxide (UO_3) in a heated fluidized bed. The granular product was packaged and stored in a vault pending shipment to off-site users. The waste condensate stream, composed of water, nitric acid, and trace amounts of uranyl nitrate, was sent to J-Cell for salvage. Off-gases, after preliminary treatment within the denitration system, discharge to the plant vessel off-gas system. The denitrator systems have undergone system flushing and a limited amount of decontamination after their last process run.

Figure 2-1. U-233 Fuel Chemical Blending Process

Figure 2-2. Plot Plan of the ICPP

Figure 2-3. CPP-666 Flourinel Dissolution Process Cell Area

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Figure 2-5. CPP-602 Laboratory, including the Denitrator Area.

Table 2-1. Facilities Data

Existing Facility Description	Building Name	Footprint or Floorspace	Number of Levels	Special Nuclear Materials	Construction/Modification Type
Current U-233 Storage Facilities	CPP-749 LWBR Dry Storage	22,296 m ²	1	yes	Underground Storage
Proposed U-233 Blending Facilities	CPP-666, CPP-602 & CPP-601	13,361 m ²	3	yes	Concrete Below Grade and Transite Above Grade
Facilities to Store 12% or 5% Long-Term or Short-Term Before Disposal to Waste Facility	CPP-749 LWBR Dry Storage	22,296 m ²	1	yes	Underground Storage
Declading Facility	Within CPP-666		2	yes	Concrete Below Grade and Steel Above Grade

2.2 Design Safety

The following parameters are considered for facility design safety: earthquakes, high winds, floods, snow, lightening, fire protection, safety class instrumentation and control, nuclear criticality, and ventilation. These parameter are described in greater detail below.

2.2.1 Earthquake

The INEL is classified as a Seismic Zone 2B region as defined by the UBC (1991)⁵⁷. This region is classified as an area where destructive earthquakes occur. The probability or frequency per year of a UBC Seismic Zone 2B or above (0.2 g surface horizontal movement) is 2×10^{-3} .²

2.2.2 High Winds

The INEL is defined as an extreme wind area according to DOE Order 5480.28. Winds that exceed 38 m/s (84 mph) have a probability or frequency per year of 1×10^{-3} .⁷

2.2.3 Floods

The postulated maximum flood at the INEL is expected to have a 10,000-year recurrence interval. This flood is estimated to crest at 1498.6 m (4916.6 ft) above mean sea level⁵⁸.

2.2.4 Snow

Snow fall has occurred in sufficient amounts to cause flooding but not to cause roof loading concerns in the 40 years of climatic records. The maximum snow depth recorded of 0.6 m (24 in.) has a frequency of occurrence of $2 \times 10^{-2}/y$.⁸

2.2.5 Lightning

Lightning does occur over the INEL during the few thunderstorms occurring each year. Lightning protection is provided for this facility.

2.2.4 Fire Protection

Fire protection is provided by an automatic wet-pipe sprinkler system. The improved risk criteria, as defined in DOE Order 5480.7A⁵⁹ establishes the minimum level of fire protection for the facility. The design also complies with the requirements of DOE-ID Architectural Engineering Standards and the National Fire Protection Association (NFPA) codes.

2.2.5 Safety Class Instrumentation and Control

The identification of individual safety systems, structures, and components (SSCs) will be addressed in the facility specific safety analysis. Five DOE Orders, shown in Table 2-2, have established the framework that sorts structures, systems or components (SSCs) into grades of hazard and safety.

Table 2-2. DOE Order Grading Process	
DOE Order	Grading Process
DOE Order 5480.1B ³⁶ , Environmental, Safety, and Health Protection for Department of Energy Operations	Grades from a physical and chemical hazard approach.
DOE Order 5480.22 ³⁷ , Technical Safety Requirements	Grades on a programmatic system based on operation requirements.
DOE Order 5480.23 ³⁸ , Nuclear Safety Analysis Reports	Dominated by radiological hazards.
DOE Order 5480.28 ² , Natural Phenomena Hazards Mitigation	Addresses radiological hazards but largely defines structural and life safety requirements.
DOE Order 6430.1A ³⁹ , General Design Criteria	Grades SSCs on radiological safety with regard to design and quality.

Each of these orders develops the grading process from a different perspective; however, all of them can be roughly broken down into offsite, onsite, and industrial risks. Table 2-3 shows a comparison of DOE Orders broken down by this basis.

Table 2-3 CLASSIFICATIONS CORRELATION

SA	HAZARD CLASS (.1B)	HAZARD CATEGORY (.23)	SA	PC	SAFETY CLASS	TAR
SAR .1B	HIGH OFF-SITE MODERATE OFF-SITE	CAT 1 OFF-SITE NUCLEAR REACTOR EQUIVALENT	SAR .23 (21)	PC-4 $1 \times 10^{-5}/Y$	SAFETY CLASS OFF-SITE EXPOSURE CONSEQUENCES. G2 MAINTAIN BELOW OFF-SITE SAFETY LIMIT. G2 MONITOR RELEASES DBA. G1 CONTROL OF SAFETY CLASS ITEMS. G1 SAFE SHUTDOWN OF FACILITY THAT IS NOT FAILSAFE AND WITH OFF-SITE CONSEQUENCE EVENTS. G1	OFF-SITE SL+ NPH < 25 REM CHEM < ERPG-2 OR 0.1xIDLH
	HIGH ON-SITE	CAT 2 CRITICALITY CRITICALITY			NUCLEAR CRITICALITY. G2	ADMIN. CONTROLS
SAR .1B	MODERATE ON-SITE	CAT 2 ON-SITE > CAT 2 THRESHOLDS DOE-STD- 1027	SAR 3009 (17)	PC-3 $1 \times 10^{-4}/Y$	SAFETY SIGNIFICANT SUBJECT TO CONVENTIONAL DESIGN STANDARDS, CODES, AND QUALITY STANDARDS. G2	ON-SITE LCO+ CHEM < ERPG-3 OR IDLH
SAR .1B	LOW ON-SITE LOW OFF-SITE	CAT 3 LOCAL > CAT 3 THRESHOLDS DOE-STD- 1027	SAR 3009 (17)	PC-2 $5 \times 10^{-4}/Y$	ESSENTIAL LIFE SAFETY G1	GENERAL ADMIN CONTROLS
ASA	<u>CHEMICALS MAIN HAZARD</u> > OSHA 1910.119 THRESHOLDS - CHEMICALS > 40 CFR 355 EP LEVELS > 40 CFR 302.4 RQ THRESHOLDS - CHEMICALS	<u>RADIOLOGICAL MAIN HAZARD</u> < OSHA 1910.119 THRESHOLDS - CHEMICALS < 40 CFR 355 EP LEVELS > 40 CFR 302.4 RQ THRESHOLDS - RADIOLOGICAL	ASA	PC-1 $1 \times 10^{-3}/Y$	GENERAL SERVICE SUBJECT TO CONVENTIONAL DESIGN STANDARDS, CODES, AND QUALITY STANDARDS G3	(ROUTINE INDUSTRIAL PROGRAMS)
NA	ROUTINE PUBLIC < 40 CFR 302.4 RQ THRESHOLDS - CHEMICALS	NA	NA		(LIFE SAFETY ITEMS) G3	(ROUTINE PUBLIC SAFETY PROGRAMS)
NOT OF SAFETY CONCERN				PC-0	NOT COVERED BY QUALITY PROGRAM	

.1B CORRESPONDS TO DOE ORDER 5481.1B SAFETY ANALYSIS
.23 CORRESPONDS TO DOE ORDER 5480.23 SAFETY ANALYSIS

G1 FUNCTION DURING AND AFTER (ACTIVE SYSTEMS).
G2 MAINTAIN STRUCTURAL AND PRESSURE BOUNDARY.
G3 MAINTAIN STRUCTURE FOR LIFE SAFETY.

Nuclear criticality is a prominent boundary issue that is treated in the top-most grade by some orders and in the mid-grade by others. Another major difference is the use of the hazardous materials inventory versus accident analysis results as the measure of consequence. The inventory represents knowledge in the early stage of design and analysis results represent knowledge in the later stage of design. These two stages of consequence can be referred to as hazard and safety, respectively. The most flexible grading system is the performance categories of DOE Order 5480.28. They start out as DOE Order 5480.23 hazard categories and evolve by the end of the design period closer to DOE Order 6430.1A safety class. This shift in consequence definition is a result of the design process.

Hazard classifications are most appropriate for use by safety analysts and regulators because they define the level of scrutiny an SSC should receive. Safety classifications are most appropriate for use by design engineers and quality assurance because they define the level of qualification and robustness a SSC should have

Hazard class or hazard category and “preliminary” performance categories can be determined at the earliest stages of design. As the safety analysis and design process proceeds, the performance categories will quickly shift from hazard criteria to safety criteria.

An integrated design and safety analysis approach provides assurance through the design and construction phases that a facility will comply with the DOE safety policy with a minimum of retrofit.

During the design and preliminary safety analysis phase, the steps involved in the systematic engineering and safety process of applying hazard and safety classification and design criteria are as follows.

1. Early hazards analysis to determine major hazards and inventories of potential sources that could release hazardous materials. Guidance such as DOE-STD-1027 can be applied to establish hazard and “preliminary” performance categories.
2. Identification of design basis accidents including natural phenomena events. DOE Order 5480.28 and its daughter standards can provide guidance in the selection process.
3. Determination of required safety functions along with the SSCs and TSRs necessary to provide those functions. The design process should provide continued feedback between the safety analysts, and the designers. DOE Order 5480.22 and DOE Order 6430.1A provide guidance at this stage.

As the safety analysis and design progresses toward construction, the integrated design and safety analysis approach provides classification of both facility hazard and the safety levels of both SSCs and TSRs.

The starting point for grading of SSCs is the establishment of DOE Order 5480.28 preliminary performance categories based on Order 5480.23 hazard categorization. Table 2-4 correlates these two orders.

Using the hazard categorization as a starting point, the designer must limit consequences to at least the off-site dose criteria for siting and designing nuclear facilities of DOE Order 6430.1A. Site specific criteria are then assigned for each PC level for seismic and wind while a single criteria for all PC categories is assigned for flood and snow load.

Table 2-4. Correlation of Hazard Categorization and Natural Phenomena Classification	
DOE Order 5480.23 Hazard Categorization	DOE Order 5480.28 Preliminary Performance Categories
Category 1: INEL off-site exposure to the public	Preliminary Performance Category 4 (PC-4): Seismic provisions consistent with civilian nuclear power plants
Category 2: INEL on-site exposure to the worker	Preliminary-PC-3: Natural phenomena provisions consistent with those used for reevaluation of commercial plutonium facilities
Category 3: Local exposure to the worker	Preliminary-PC-2: Natural phenomena provisions consistent with model building codes for continuity of essential operations
General Use: Industrial types of hazards controlled by General Industry regulations, codes, and standards	Preliminary-PC-1: Natural phenomena provisions consistent with model building codes for life safety for on-site personnel
N/A	Preliminary-PC-0: Natural phenomena is not an issue

Using the Hazard Categorization as a starting point, the designer must limit consequences to at least the criteria for siting and designing nuclear facilities and nuclear operations provided in Table 2-5.

Table 2-5. Siting and design Radiation Protection Criteria		
Criterion	Quantity	Safety Basis
Design Estimate Occupational Annual Effective Dose Equivalent Limit	500 mrem/year	Maintains occupational dose below operational limits and assists in maintaining dose ALARA.
Design Value for Neutron Quality Factor	20 (dimensionless)	Introduces conservatism into design calculations.
Siting Guideline Dose Limits to Maximally Exposed Individual at Site Boundary Following a Credible Unmitigated Accident (1E-6/yr probability)	25 rem whole body 300 rem thyroid 300 rem bone 75 rem lung 150 rem any other organ	Reference values for evaluation of Design Basis Accidents for given site.

The natural phenomena criteria used at the INEL for new structures is presented in Table 2-6. Criteria is assigned for each PC level for seismic and wind while a single criteria for all PC categories is assigned for flood and snow load.

Table 2-6. INEL Natural Phenomena Design Criteria

DOE Order 5480.28 PC	Site Phenomena	Performance Goal Probability/Y	Magnitude Range or Upper Limits	Probability or Frequency/Y	Design Requirements
PC-4 PC-3 PC-2 PC-1	Seismic	1×10^{-5} 1×10^{-4} 5×10^{-4} 1×10^{-3}	0.24 g (bedrock) 0.18 g (bedrock) 0.18 g (bedrock) 0.13 g (bedrock)	2×10^{-4} 1×10^{-3} 1×10^{-3} 2×10^{-3}	0.24 g (bedrock) 0.18 g (bedrock) 0.20 g (surface) UBC I=1.25 0.20 g (surface) UBC I=1.0
PC-4 PC-3 PC-2 PC-1	Wind	1×10^{-5} 1×10^{-4} 5×10^{-4} 1×10^{-3}	42 m/s (95 mph) 38 m/s (84 mph) 31 m/s (70 mph) 31 m/s (70 mph)	1×10^{-4} 1×10^{-3} 1×10^{-3} 2×10^{-3}	42 m/s (95 mph) @ 10 m 38 m/s (84 mph) @ 10 m 31 m/s (70 mph) @ 10 m I=1.07 31 m/s (70 mph) @ 10 m I=1.0
PC-4 PC-3 PC-2 PC-1	Flood	1×10^{-5} 1×10^{-4} 5×10^{-4} 1×10^{-3}	1498.5 m (4916.6 ft) DBF	1×10^{-4}	1498.6 m (4917 ft) above DBF
PC-4 PC-3 PC-2 PC-1	Snow	1×10^{-5} 1×10^{-4} 5×10^{-4} 1×10^{-3}	1.14 m (45 in) maximum annual snow fall (120 to 146 kg/m ²)	1/y	150 kg/m ² (30 lb/ft ²) roof

Table 2-7 shows the assignment of life, safety, and security equipment to natural phenomena classes. These designs correspond to those required by state and local governments.

Table 2-7. Classification of Life Safety Systems and Security	
DOE Order 5480.28 PC	Life Safety & Security
PC-4	
PC-3	
PC-2	Essential System-Life Safety: fire fighting response, medical response, police response.
	Life Safety Systems: emergency power distribution system, fire protection alarm networks, evacuation alarm system, fire protection water distribution systems, criticality alarm network, > 300 people in one room egress.
PC-1	Building Life Safety: building emergency power, voice paging network, egress lights, exit lights, fire protection, evacuation alarms, criticality alarm
	Security: special nuclear material safeguards, security alarm networks, building security alarms, and property protection

Classification of radiological and toxic hazards are given in DOE Orders and guidance. Facility equipment and systems which support contained process functions typically include: confinement barriers, effluent systems, ventilation and off-gas systems, and equipment and instrumentation for safety function, as shown in Table 2-8. Criticality prevention, waste handling and treatment, and industrial/chemical systems in support of contained process functions are shown in Table 2-9.

1. Confinement Barriers: Confinement barriers are the last design feature that prevents the transport of radiological and toxic materials to the on-site personnel and the public.
2. Effluent Systems: (including reentry monitor systems): Environmental Protection Agency (EPA) required monitoring of the main stack is established in the Code of Federal Regulations, Title 40, Part 61 (40 CFR 61) and DOE/EH-0173T for monitoring release points that are capable of producing radiation exposure of 0.1 mrem/year at the site boundary.
3. Ventilation and Off-gas Systems: (includes contamination control that is not required by Orders or codes but is an INEL policy): The building blowers are used to maintain air zones to prevent contamination spread during an emergency. Contamination that would escape from the building and exceed the DOE on-site radiological or toxic material limit or that could migrate to the building process control room are noted as category 2. Contamination only within the building is a category 3 hazard.

Table 2-8. Categorization Criteria for Confinement, Effluent, Ventilation and Off-gas, and Support Systems

DOE Order 5480.28 PC	Confinement Barriers	Effluent Systems-Post Accident Monitoring for Public and Worker Reentry	Ventilation and Off-gas Systems-Property Damage Protection and Process Operations Continuity	Equipment for Support of Safety Functions
PC-4	Radiological or Toxic Confinement Controls: ≥ 0.5 rem at site boundary, $\geq$ ERPG-3 at site boundary	Site boundary monitoring equipment and controls		
PC-3	Radiological or Toxic Confinement Controls: ≥ 1 rem at 300 m or approximately ≥ 10 rem at 100 m or $\geq$ category 2 TQs in DOE-STD-1027 for inventories, $\geq$ ERPG-3 at 300 m Direct Radiation Exposure: high hazard radiation area	Monitoring Equipment and Controls: used for mitigation	Radiological or Toxic Contamination Controls: VOG ventilation fans/exhaust fans for confinement cells, instrumentation POG cleanup system and blowers	Standby Power: contamination control systems instrumentation, lighting in process operation control rooms Criticality Controls Standby Power: instrumentation for triple contingency above double contingency
PC-2	Radiological or Toxic Confinement Controls: ≥ 1 rem at 100 m dispersion or $\geq$ category 3 TQs in DOE-STD-1027 for inventories, $\geq$ ERPG-3 around building or 100 m Direct Radiation Exposure: radiation area	Monitoring Equipment and Controls: $>$ EPA emission for 24 hour at site boundary	Radiological or Toxic Contamination Controls: VOG ventilation fans/exhaust fans for confinement cells	Standby Power: instrumentation (RAM and CAM alarms and network systems), contamination control systems instrumentation, safe shutdown equipment) ^b Essential System Emergency Power-Life Safety: power systems for fire fighting, medical, and police response structures. Life Safety Systems Emergency Power: emergency power and fire protection water distribution systems, fire protection alarms and voice paging networks, radiation monitoring, and criticality alarm systems as an evacuation alarm networks
PC-1	General Use Controls: < 1 rem worker, (sealed sources, sources in commercial products, and sources in DOT Type B shipping container with over packs as specified by DOE-STD-1027), $<$ ERPG-1 for public, $<$ ERPG-3 for worker,	Monitoring: < 10 mrem at site boundary, $<$ EPA emission for 24 hour at site boundary		Standby Power: (equipment damage $>$ \$1M control, plant shutdown $>$ 6 months) plant steam generation systems instrumentation, raw water deep well pumps for cooling, freeze control program Building Life Safety Emergency Power: building emergency power use, egress lights, exit lights, fire protection, evacuation alarms, radiation alarms as evacuation alarms, and criticality alarm system as an evacuation alarm, fire protection water pumps

ERPG-1: Emergency response planning guide for prevention of mild transient adverse health effects or objectionable odor

ERPG-2: Emergency response planning guide to prevention impairment of an individual's ability to take protective action

ERPG-3: Emergency response planning guide for prevention of life threatening effects

PC: Performance category

TQs: Threshold quantities

Table 2-9. Classification Criteria for Criticality Prevention, Waste Handling and Treatment, and Industrial/Chemical Systems

DOE Order 5480.28 PC	Criticality Prevention	Waste Handling and Treatment	Industrial/chemical Systems
PC-4			Radiological or Toxic Release Controls: ≥ 0.5 rem at site boundary, $\geq$ ERPG-3 at site boundary
PC-3	Criticality alarms in the immediate area of a potential unshielded criticality area		Radiological or Toxic Release Controls: ≥ 1 rem at 300 m (approximately ≥ 10 rem at 100 m or $\geq$ category 2 TQs in DOE-STD-1027 for inventories), $\geq$ ERPG-3 at 300 m
PC-2	Criticality alarms not in the immediate area of a criticality	Waste controls	Radiological or Toxic Release Controls: ≥ 1 rem at 100 m dispersion ($\geq$ category 3 TQs in DOE-STD-1027 for inventories), $\geq$ ERPG-3 around BLDG or at 100 m
PC-1			General Use Controls: < 1 rem worker, (sealed sources, sources in commercial products, and sources in DOT Type B shipping container with over packs as specified by DOE-STD-1027), $< \text{ERPG-1}$ for public, $< \text{ERPG-3}$ for worker

CAS: Criticality alarm system

ERPG-1: Emergency response planning guide for prevention of mild transient adverse health effects or objectionable odor

ERPG-2: Emergency response planning guide to prevention impairment of an individual's ability to take protective action

ERPG-3: Emergency response planning guide for prevention of life threatening effects

PC: Performance category

Tqs: Threshold quantities

Once the performance categories are established, the transition to establishing safety class designations based on the consequence of failure and in turn the requirements of each SSC can proceed. DOE Order 6430.1A Section 1300-2 and DOE-STD-3009 provide the basis for this stage of classification. Table 2-10 summarizes the graded safety classes and the consequence thresholds that define them; expanded definitions of threshold limits outlined in Table 2-10 are described below.

Table 2-10. Safety Class Grading System	
Category of Safety	Consequence Threshold Description
Safety Class ^a	Public Exposure to Radiation: offsite criticality > 500 mrem for public protection, safe shutdown, or post-accident monitoring, if used for mitigation, ESFs.
Safety Significant Class, (below 6430.1A limits)	Co-Located Worker Exposure to Radiation or Toxic Hazard: on-site > 5 rem, ^b or > ERPG3, environmental equipment control. Off-site toxic exposure > EPA emission limits for 24 hrs, or > ERPG2
General Service Class	Facility Worker/Life Safety Codes & Industry Standards: low risk release, OSHA limits, engineered chemical contamination & fire protection, ALARA, property protection, common industrial hazards, SNM safeguards

a. Safety Class corresponds to the Safety Class Items as outlined in DOE Order 6430.1A Section 1300-2.

b. See DOE Order 5480.11 for complete threshold limits.

2.2.5.1 Safety Class Criteria (based on DOE Order 6430.1A). Safety Class Criteria are based on DOE Order 6430.1A and include: Public Exposure, Safety Limits in Technical Safety Requirements (TSRs), Nuclear Criticality Safety, Monitoring Releases to the Environment, Safe Shutdown Mode, and Control of Safety Class Systems and Equipment. These are described in detail below.

1. **Public Exposure:** Safety Class systems and equipment include equipment whose failure could produce exposure to radioactive material that would exceed the guidelines at the INEL site boundary listed below. Releases postulated to occur as a result of accidents shall be limited by designing facilities such that at least one confinement system remains fully functional following any credible single failure.
 - a) No unfiltered releases of radioactive material are permitted following credible accidents that could produce doses in excess of DOE Order 5400.5 limits for public exposure (see Table 2-11). The Clean Air Act portions of DOE Order 5400.5 are for normal operation and are not applicable to accident releases.
 - b) Attenuation features must be provided when off-site accident releases could cause doses in excess of the DOE Order 5400.5 limits for public exposure.
 - c) As-low-as-reasonably-achievable (ALARA) concepts shall be applied when designing facilities to mitigate post accident releases.
 - d) No radiological on-site exposure criteria are associated with Safety Class.

Table 2-11. DOE Order 5400.5 Limits for Public Exposure		
Accident Frequency Per Year	Criteria Category	Criteria (REM)
1 to 0.1	Normal Operations	DOE Order 5400.5 limits on normal emissions.
0.1 to 0.01	Anticipated Events	0.5 (DOE Order 5400.5)
1 E-2 to 1 E-4	Unlikely Events	5 (10 CFR 72)
1 E-4 to 1 E-6	Extremely Unlikely Events	25 (10 CFR 100)

2. Safety Limits in Technical Safety Requirements (TSR): Safety Class systems and equipment include those required to maintain operating parameters within "safety limits" during normal and abnormal occurrences. Early in design, the accident analysis in the Safety Analysis Report (SAR) will be used to determine safety limits until TSRs are produced. The safety class items (SCIs) are the final product of the safety analysis. Any equipment required to maintain operating parameters within safety limits specified in TSRs during normal and anticipated events will be SCIs. These may include some Group 1 instruments. This could also include pieces of equipment such as mist eliminators, filters, and others.
3. Nuclear Criticality Safety: SCIs include those barriers required to prevent an accidental nuclear criticality. These will be defined as any piece of equipment for which credit is taken in the SAR criticality accident analysis.
4. Monitoring Releases to the Environment: Safety Class systems and equipment include those required to monitor the release of radioactive materials to the environment during and after an accident that are used for accident mitigation. These may include a post accident stack monitor.
5. Safe Shutdown Mode: Safety Class systems and equipment include those required to achieve and maintain the facility in a safe shutdown mode. The term safe shutdown refers to prevention of a criticality or off-site release during shutdown conditions. These systems and equipment must be identified on a case-by-case basis, but will likely consist of some instruments, power supplies, and ventilation blowers.
6. Control of Safety Class Systems and Equipment: Safety Class systems and equipment are those required for control of the SCIs described above. Those systems and equipment that control SCIs are also SCIs, e.g., uninterruptible power supply (UPS), emergency control panel, and any other systems and equipment for which safety credit has been taken.

The criteria outlined above are derived from DOE Order 6430.1A. Safety Class systems and equipment are those that meet any of the above criteria. This order primarily addresses off-site public exposure. Safety Class is intended to cover the requirements to protect the public that resides off-site. It is not intended to apply to the transient on-site public.

Many structural and system design requirements are mandated by DOE Order 6430.1A as a result of the facility Natural Phenomena Performance Category as defined in DOE Order 5480.28. The Natural Phenomena Performance Category is based on both on-site and off-site consequences and is, therefore, different from safety class which is based mostly on off-site consequences. Just because a component

must be designed to tolerate natural phenomena is not an indication that it must be designated as safety class. Indeed, most of the features considered as important to safety fall into this non-safety class classification.

2.2.5.2 Safety Significant Class. Safety Significant Class regards exposure limits for Radiation Exposure to Co-located Workers and the Transient Public; Toxic Chemical Exposure to Co-located Workers, Transient Public and Off-site Public; and Environmental Equipment, as described below.

1. Radiation Exposure to Co-Located Workers and the Transient Public: Safety Significant Class systems and equipment include those whose failure could produce exposure that would exceed the guidelines listed below, at or beyond a 0.4-mile exclusion zone around the facility boundary.
 - a) No unfiltered releases of radioactive material are permitted following credible accidents that could produce doses in excess of DOE Order 5480.11 limits for exposure.
 - b) Attenuation features must be provided when on-site accident releases could cause doses in excess of the DOE Order 5480.11 limits for exposure.
 - c) Criticality alarm systems are required as safety significant class worker protection if, under the criteria of DOE Order 5480.24, an on-site dose of 12 rad in free air would occur as a result of a criticality.
 - d) ALARA concepts shall be applied when designing facilities to mitigate post accident releases.
2. Toxic Chemical Exposure to Co-Located Workers, Transient Public, or Off-site Public: Safety Significant Class systems and equipment include those whose failure could produce exposure that would exceed the guidelines of Emergency Response Planning Guideline (ERPG) 3 at or beyond a 0.4-mile exclusion zone around the facility boundary, off-site exposure exceeding Environmental Protection Agency (EPA) emission limits for 24 hours or off-site toxic exposure exceeding ERPG 2.
3. Environmental Equipment: This is equipment that is required to be operational by a federal or state permit for environmental controls. This includes equipment whose failure could result in exceeding EPA emission limits for 24 hours.

2.2.5.3 General Service Class. General Service Class regulates (1) Protection of Facility Workers; (2) Special Nuclear Materials Safeguards; Property Protection ; and (3) Environmental Monitoring, as described below.

1. Protection of Facility Worker: General Service Class systems and equipment include those whose failure could expose facility workers to conditions that would exceed state and federal Occupational Safety and Health Administration (OSHA) requirements, and requirements in DOE Order 5480.11, ALARA, or National Life Safety codes. Included in this class are engineered chemical contamination controls and barriers for low risk releases of radioactive and toxic substances. Monitoring or surveillance to verify compliance with regulations is within the scope of this class.

2. Special Nuclear Materials Safeguards: Equipment and systems used for the safeguarding of special nuclear materials are included in the General Service Class.
3. Property Protection: Security systems, standby and emergency power, fire protection, flood barriers, and freeze protection are examples of protective systems/equipment.
4. Environmental Monitoring: A requirement is established in 40 CFR 61 and DOE/EH-0173T for monitoring release points that are capable of producing radiation exposure of 0.1 mrem/year at the site boundary.

The last stage of the classification process is to relate the procurement requirements (quality level) of an SSC to the various grading systems. Dividing points do not match any individual grading scheme but have boundaries found in at least one of the schemes. Seismic and other structural requirements can be related to performance categories independent of Safety Class or Hazard Class.

2.2.5.4 Process Equipment and Engineered Systems That Are Safety Significant. Process equipment and engineered systems that could be safety significant include the following:

1. Criticality Prevention: General information on ICPP inadvertent criticality protection is contained in Chapter 8. The current DOE Order governing nuclear criticality safety is 5480.24⁶⁰. CAS (criticality alarm systems) are designed in accordance with the specifications and requirements specified in the ANSI/ANS Standards^{61,62} and DOE Order 5480.24; the CAS in the CPP-601 facility does not meet current standards and will require replacement. The procedures are given in the Criticality Safety Requirements Manual⁶³.
2. Waste Handling and Treatment: The waste management and control philosophy for the ICPP is that all solid, liquid, and gaseous discharges are safely managed according to applicable policies stated in DOE Order 5820.2A⁶⁴, DOE Order 5480.11⁶⁵, DOE Order 5484.1⁶⁶, DOE-ID Order 5480.2⁴⁴, DOE-ID Order 5484.1A⁴⁵, 40 CFR 61⁶⁷, 40 CFR 241⁶⁸, and 40 CFR 260-270⁶⁹. The need to minimize solid and liquid discharges is recognized at the INEL.

Solid and liquid hazardous waste will be disposed of in accordance with the ICPP Transportation of Hazardous Materials Requirements Manual. Hazardous waste will be shipped to licensed hazardous waste sites, as contracted by the INEL waste disposal contractor.

3. Industrial/chemical Safety Systems: A facility or project has airborne chemical contaminant monitors, where required. Ambient air monitors are located to monitor the working environment for chemical vapor releases. Ensuring proper and timely calibration of all instruments at the ICPP is directed through WSOP WE-3, "Calibration Program"⁷⁰. This operating procedure addresses, among others, the following requirements: accuracy, calibration agency, calibration interval, instrument calibration data base, set points and ranges, standards and measuring devices, and tagging. These criteria are discussed in detail in Section 10.

2.2.5.5 Safety Grade System.

1. Allocation of safety functions:
 - a. Passive systems - are preferred to other controls. No specific TSRs are required for passive controls. These are covered under a generic configuration control specification. Changing of passive systems are reviewed through USQ evaluation process.
 - b. Active systems - require varying levels of support from other systems or personnel and thus require procedural controls for personnel interfaces and TSR specifications for system controls. Active systems require corresponding TSRs with safety limits and LCOs for Category 1 hazards, TSRs and safety limits for Category 2 hazards, and TSRs (procedural controls) for category 3 hazards.
 - c. Procedural controls - For category 3 hazards the procedural control is the TSR with no limiting conditions for operation (LCO) or Safety Limit (SL). In cases where TSR procedural controls are used for Category 1 or 2 the TSR contains the corresponding LCOs or LS as required.

2.2.6 Nuclear Criticality

The criticality safety of fuel handling and storage operations at the ICPP is based on comparing the calculated effective multiplication factor (k_{eff}) values for fissionable material configurations (normal and accident cases) to a set of criticality safety criteria. The current DOE Order governing nuclear criticality safety is 5480.24⁷¹ with additional requirements applicable to this facility given in the ANSI/ANS Standards^{72,73}, DOE Order 5480.24³², and DOE-ID 5480.5A⁷⁴. The procedures are given in the Criticality Safety Requirements Manual⁷⁵.

The safety parameters used for the prevention of criticalities at ICPP are as follows:

1. A maximum k_{eff} value of 0.95 is allowed for criticality safety parameters developed from calculations. This acceptance criterion value must include the effects of credible accidents, equipment tolerances, and the uncertainties of the calculational method.
2. If criticality safety is by mass control, then a system must be limited to 45% of the minimum critical mass (for worst-case geometry) unless double batching (doubling the allowable amount of fissile material) is not credible, in which case a higher percentage may be used. The higher value is usually set at 75% of the minimum critical mass. This principle is also applied to controls based on the minimum critical number of elements or pieces.
3. The double contingency principle is applied to well-shielded areas to determine limits of operation⁶. The term "well-shielded" means that sufficient shielding is provided such that external whole body radiation exposures will not exceed 25 rem to personnel outside of the shielding under accident conditions. Double contingency means that sufficient factors must be incorporated in the design or the operational controls to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

The approach used to apply the double contingency principle is described in subsection 6.2 above. In addition, operating limits are set such that no single accidental event results in exceeding a k_{eff} of 0.95 for well-shielded facilities⁶. It is also policy to implement physical barriers whenever possible, rather than to rely solely on administrative controls to prevent criticality accidents.

Criticality safety features, control methods, and design limits are based on the following assumptions:

1. Complete and accurate descriptions of the fissionable material.
2. Accurate criticality safety evaluations (CSEs) to demonstrate adequate safety margin.
3. Adequate controls to ensure the designated material is properly accounted for.
4. Adequate controls to ensure fissionable material integrity or containment is maintained.
5. Adequate controls to ensure approved administrative safety controls for limits and configuration are not violated.
6. An adequate equipment surveillance, maintenance, and inspection plan and replacement, including appropriate and timely corrective action.
7. Operating personnel trained and certified to comply with the applicable administrative controls.

General criticality control limits are developed from calculations based on worst case fissile material concentration or a generic fuel unit, although material-specific calculations are done as necessary.

Administrative and physical controls are imposed as necessary to meet the double contingency criterion for criticality accident prevention. Complete dependence on physical barriers for these contingencies is desirable but not practical due to the large number of manual operations performed.

Complete dependence on passive physical barriers, such as geometrically safe containers, to prevent criticality is not possible. Generally, criticality safety must be accompanied by administrative controls, which either require the use of a physical feature (such as fuel racks) or limit the scope of its applicability (such as fissile material loading limits). Thus, administrative controls are used where necessary to implement physical controls.

Criticality calculations are performed for various fissile material configurations, normal and abnormal, to determine the margin of safety either inherent in the system or imposed by the administrative requirements. Criticality calculations also determine the sensitivity of k_{eff} to equipment design and operating parameters and identify fissile material configurations that pose a potential criticality hazard. Selected criticality safety calculations are provided in each specific facility Safety Analysis Report.

The criteria used for performance of criticality calculations, including treatment of errors and verification and validation of results, are given in DOE-ID 5480.5A and the Criticality Safety Requirements Manual as implemented by ICPP Operating Contractor procedures.

2.2.6.1 Scoping Calculations. Solvent for the uranium/thorium fuel would be mostly nitric acid. Therefore, undissolved solids will contain at least the same ratio of thorium to uranium as the original pellets. The most reactive time will be immediately after fuel charging. The heavy pellets would probable be distributed around the bottom of the dissolver torus. However, for simplicity they were

conservatively modeled as a slurry sphere in the larger spaces higher in the dissolver, fully reflected by poisoned water.

In another more conservative model it was assumed that the uranium oxide remained undissolved, while all the thorium oxide dissolved except for a protective shell of 4 thorium oxide molecules around each uranium oxide molecule. These clusters were then modeled as a slurry sphere.

The uranium/zirconium fuel would be dissolved in hydrofluoric acid. The uranium would dissolve more slowly than the zirconium and would form solid UF_4 . Therefore it was modeled as a slurry sphere of undissolved UO_2 , with no other solids.

Fuel compositions vary. The composition of the uranium/thorium fuel with the highest fissile content is given in Table 2-12.

Table 2-12. Composition of Most Reactive Thorium-Containing Fuel		
Material	Weight Percent	Density (g/cc)
$^{233}UO_2$	4.05	10.76
$^{234}UO_2$	0.08	10.8
ThO_2	95.87	9.69

From the data in Table 2-12, the volume fraction available for solvent (ignoring the $^{234}UO_2$) in the fissile region is calculated as shown in Formula (1).

$$1 - 0.00288 \left(g \cdot \frac{U^{233}}{L} \right) \quad (1)$$

The volume fraction of water in the borated reflector was calculated as based on a dissolved boric acid density of 1.44 g/cc and is calculated as shown in Formula (2).

$$1 - 0.00397 \left(g \cdot \frac{B}{L} \right) \quad (2)$$

In the case of cadmium poisoning, water volume fractions in the reflector were based on handbook densities of cadmium nitrate solutions. Water and poison concentrations in the fissile sphere were obtained by multiplying the concentrations on the reflector by the "volume fraction available for solvent" as given by Formula (1).

For the slurry sphere with only UO_2 , the "volume fraction available for solvent" is calculated as shown in Formula (3).

$$1 - 1.057E-4 \left(g \cdot \frac{U^{233}}{L} \right) \quad (3)$$

Poison and water atomic densities were calculated by the same methods as used for the uranium/ thorium oxide slurries.

Most of the calculations were performed with TWODANT, using 27-group, ENDF/B-IV cross sections processed by SCALE. A few calculations were checked by MCNP4a using continuous energy ENDF/B-V cross sections. Validation calculations against experiments were not performed, since these are scoping calculations.

2.2.6.1.1 Results—The calculations for poison requirements in the dissolver solution for a uranium/thorium oxide fuel charge containing 17.5 kg U-233 are given in Tables 2-13 and 2-14.

Table 2-13. Boron Poison Requirements for 17.5 kg of U-233 as a UO ₂ /ThO ₂ Slurry Sphere					
	Case 1	Case 2	Case 3	Case 4	Case 5
g B/L in reflector	5.39	5.39	5.39	4.94	4.49
g ²³³ U/L	150	175	200	175	175
radius, cm	30.313	28.795	27.541	28.795	28.795
k _{eff}	0.9116	0.9253	0.9145	0.9375	0.9492 MCNP 0.946±0.002

Table 2-14. Cadmium Poison Requirements for 17.5 kg of U-233 as a UO ₂ /ThO ₂ Slurry Sphere				
	Case 1	Case 2	Case 3	Case 4
g Cd/L in reflector	8.4	8.4	8.4	7.1
g ²³³ U/L	110	150	200	150
radius, cm	33.614	30.313	27.541	30.313
k _{eff}	0.8841	0.9302	0.8914	0.9535 MCNP 0.938±0.002
By interpolation, 7.4 g Cd/L required for k _{eff} = 0.95				

Calculations for a slurry sphere containing only 4 atoms of thorium per uranium atom are shown in Table 2-15. Calculations of minimum critical U-233 masses in slurry spheres with only uranium oxide, which pertain to the uranium/zirconium oxide fuels, are presented in Tables 2-16 and 2-17.

Table 2-15. Minimum Critical Mass in Oxide Slurry Sphere Th/ ²³³ U=4,@9.88 g B/L				
	Case 1	Case 2	Case 3	Case 4
kg ²³³ U	7.07	7.07	7.07	8.58
g ²³³ U/L	400	500	650	500
radius, cm	16.158	15	13.744	16
k _{eff}	0.9131	0.9205	0.8838	0.9641
By interpolation, 7.95 kg ²³³ U at k _{eff} = 0.95				

Table 2-16. Minimum Critical Mass of ²³³ U in ²³³ UO ₂ Slurry Sphere with 9.88 g B/L				
	Case 1	Case 2	Case 3	Case 4
kg ²³³ U	4.25	4.25	4.25	6.74
g ²³³ U/L	500	650	800	650
radius, cm	12.66	11.6	10.824	13.524
k _{eff}	0.9178	0.9265 MCNP 0.904±0.002	0.9209	1.0422
By interpolation, 4.7 kg ²³³ U at k _{eff} = 0.95				

Table 2-17. Minimum Critical Mass of ²³³ U in ²³³ UO ₂ Slurry Sphere with 0.5 M Cd				
	Case 1	Case 2	Case 3	Case 4
kg ²³³ U	5.98	5.98	5.98	5.43
g ²³³ U/L	800	1000	1200	1000
radius, cm	12.131	11.261	10.597	10.9
k _{eff}	0.9524	0.9603 MCNP 0.926±0.002	0.9556	0.9342
By interpolation, 4.7 kg ²³³ U at k _{eff} = 0.95				

The results are summarized in Table 2-18. The check calculations with MCNP usually yielded lower k_{eff}'s, indicating that the conditions in Table 2-18 are based on the more conservative of the two codes.

Table 2-18. Summary of Limiting Safe Conditions ($k = 0.95$)		
System	Poison Concentration	kg ^{233}U
$^{233}\text{UO}_2/\text{ThO}_2$ slurry sphere, $\text{Th}/^{233}\text{U}=23.8$	4.46 g B/L	17.5
$^{233}\text{UO}_2/\text{ThO}_2$ slurry sphere, $\text{Th}/^{233}\text{U}=23.8$	7.4 g Cd/L	17.5
$^{233}\text{UO}_2/\text{ThO}_2$ slurry sphere, $\text{Th}/^{233}\text{U}=4$	9.88 g B/L	7.9
$^{233}\text{UO}_2$ slurry sphere	9.88 g B/L	4.7
$^{233}\text{UO}_2$ slurry sphere	0.5 <u>M</u> Cd	5.7

It should be noted that the criticality safety control manual at this facility requires absorber concentrations be 125% of those required for a k_{eff} of 0.95 at the accident or upset condition of maximum reactivity. Therefore, higher poison concentrations or lower fissile concentrations would be required than those listed in the tables above.

2.2.7 Ventilation

The heating and ventilation system is designed in accordance with the DOE-ID Architectural Engineering Standards and DOE Order 6430.1A. The H&V system is a once-through system with no air recirculation. The supply system heats the entire facility using steam. The steam condensate is recycled to the ICPP condensate return system.

Ventilation for ICPP processing facilities is provided using input blowers and exhaust fans dampered to maintain a negative pressure in all processing areas at all times. Exhaust air is scrubbed and filtered before being released to the atmosphere. Separate vessel offgas systems are also maintained in the process facilities to handle higher concentrations of airborne effluent from specific processing operations. The vessel offgas systems are individually designed with appropriate offgas purification, handling, scrubbing, and HEPA filtering to minimize the potential release of airborne effluents⁷⁷.

3.0 SITE MAP AND LAND USE REQUIREMENTS

3.1 Site Map

Figure 3-1 presents a pictorial map of the flow of material through the ICPP

Figure 3-1. Site Map with Process Flows

3.1.1 Fuel Receipt and Storage

The CPP-603, -666, -651, and -749 facilities provide fuel receipt, storage, and management capabilities for the ICPP. CPP-651, the Unirradiated Fuel Storage Facility, provides interim storage for recovered uranium product and a variety of unirradiated nuclear materials from other government facilities. CPP-749, the Underground Dry Fuel Storage Facility, uses below grade vaults to provide dry, underground storage for nuclear fuel materials. The LWBR fuel to be processed as part of the project is stored in the CPP-749 facility.

3.1.2 Fuel Reprocessing

Although several fuel element dissolvers were available for specialized fuels in CPP-601, the principal facility for fuel dissolution and the location of the dissolver to be used for this project was the FDP units in CPP-666. Dissolved fuel was piped to CPP-601 for three stages of organic separation. Aqueous waste from the extraction system was piped to the ICPP tank farm, and uranium product solution was piped to CPP-602 for denitration leaving a solid granular product of UO_8 .

3.1.3 High Level Liquid Waste Management

HLLW from the extraction systems is transferred directly to the tank farm for storage until it can be transferred to the New Waste Calcining Facility (NWCF) for processing. The tank farm consists of eleven stainless steel 300,000 gallon underground tanks, housed in concrete vaults. These tanks provide interim HLLW storage space before calcination. Three 18,200 gallon tanks housed at the north end of the Waste Treatment Building, CPP-604, provide temporary storage of special waste solutions. After segregation and sampling, waste from these tanks is transferred to the tank farm.

HLLW from the tank farm is calcined to a granular solid in the NWCF. Conversion of HLLW to an inert solid reduces the volume of waste by a ratio of about seven to one and results in a less mobile waste form. Calcining liquid waste reduces the potential mobility of the HLLW fission products, provides greater assurance of isolation from the environment, improves safety, lowers overall costs for waste management, and reduces storage volume requirements. An average of 3,000 gallons of HLLW per day can be processed through NWCF.

Calcine is pneumatically transferred to the Calcine Solids Storage Facilities (CSSFs) for storage. CSSFs provide safe and economical containment of radioactive solids produced by the NWCF process. There are seven CSSFs at the ICPP each consisting of several stainless steel bins enclosed in a reinforced concrete vault. The stainless steel bins are the primary containment of the calcined solids. The concrete vaults provide secondary containment and radiation shielding for plant personnel. The CSSFs are located east and south of the NWCF. They are designed conservatively for a nominal 500-year lifetime. The short half-life radioisotopes will be essentially gone in 300 years or less, but the transuranic and hazardous components will remain indefinitely.

3.2 Land Area Requirements During Operations

The restricted area of the ICPP encloses 0.49 km^2 (0.19 mi^2 or 120 acres) of the total ICPP area, which is 0.60 km^2 (0.23 mi^2 or 147 acres).

4.0 PROCESS DESCRIPTION

Two alternative processes are presented to blend existing U-233 materials with depleted uranium to obtain a final product having an enrichment of either 12% U-233 or <5% U-233. In one process, uranium feed materials are dissolved and blended, and thorium and fission products are removed via two extraction systems. For LWBR fuels stored both at the Idaho ICPP and the RWMC, the bulk of the thorium would be removed in an Acid Thorex extraction, and the residual thorium removed in a two-cycle hexone extraction system. When using this process for ORNL U-233 materials, which contain no thorium, the Acid Thorex process is bypassed. In the alternative process, uranium and thorium would be coextracted in a single extraction cycle, then concentrated and denitrated, resulting in a combined uranium/thorium oxide product. Coextracting the thorium with the uranium would remove any doubt regarding the ability to calcine the extraction waste in the NWCF as planned. Processing of the ORNL material is the same for both alternatives.

4.1 Thorium Removal Process

The thorium removal process was conceived to dilute fissile U-233 material presently stored at the INEL, including unirradiated LWBR fuel at the ICPP and the RWMC, and various uranium containing materials stored at ORNL. This process, described below, is outlined in Figure 4-1.

In the proposed process, unirradiated LWBR fuel is transferred from underground silos in CPP-749 to the FDP remote handling cell in CPP-666. The LWBR material that is contained in 55 and 110 gallon drums at the RWMC may first need to be transferred to the Irradiated Fuel Storage Facility (IFSF) dry cave for reloading into cans and then transferred to the FDP cave. ORNL materials will be brought to the INEL and placed in the FDP cave.

In the FDP cell, the cans are opened, fuel rods removed, rod ends cut off, and fuel pellets removed; a critically safe batch of pellets will be charged to the FDP dissolvers. Cans containing ORNL material are opened and, depending on the form of the material in the can, the material is either removed or placed in the dissolver in the can. Fuel pellets and ORNL uranium compounds are dissolved in a boiling solution of 12 M HNO₃, 0.15 M HF and 0.1 M Al(NO₃)₃. HF is needed to catalyze ThO₂ oxide dissolution in nitric acid. Aluminum nitrate complexes the fluoride sufficiently to avoid corrosion of the dissolver. Cadmium nitrate is added at a concentration of 0.066 M as a neutron poison to assure criticality safety. Cadmium is the preferred poison because it would have the least impact on increasing calcine HLW volume and because its dissolution chemistry is straightforward. However, should environmental factors dictate against using the RCRA-hazardous cadmium, either fluoroboric acid or boric acid could be substituted.

Dissolution occurs by the following reactions:

Figure 4-1. Block Flow Diagram for U-233 Fissile Dilution

A dissolution batch of the ICPP LWBR fuel would consist of 411.5 kg of clad fuel containing 402.5 kg ThO₂ and 9.0 kg of UO₂ (7.79 kg of U-233), and 1550 L of dissolvent. The initial boiling temperature is estimated to be 117° C. The steady state dissolver product composition (after two batches) is 8 M HNO₃, 0.983 M Th(NO₃)₄, 0.022 M UO₂(NO₃)₂, 0.066 M Cd(NO₃)₂, 0.15 M HF, and 0.1 M Al(NO₃)₃. Complexation of the fluoride by dissolved Th⁴⁺ will result in negligible free HF.

Dissolution of the LWBR fuels from the RWMC will be similar to dissolution of ICPP LWBR fuels. However, for the ZrO₂/UO₂ fuel, dissolution will involve a two-step process in which the fuel is first exposed to 9.6 M HF at 80° C, primarily to attack ZrO₂, followed by complexation of excess HF with Al(NO₃)₃ and addition of nitric acid. This complexed solution is then heated to 80° C to oxidize and dissolve the UO₂ and any UF₄ formed during exposure to HF. Preliminary criticality safety modeling indicates that 0.5 M Cd(NO₃)₂ will provide the necessary neutron poisoning. The total quantity of ZrO₂/UO₂ fuel can be processed in one batch.

Dissolution of the ORNL material is performed with 7.72 M HNO₃. At this concentration, the dissolution reaction is:

A 1550 L batch in a FDP dissolver would process 748 kg of UO₂. Thus the total quantity of ORNL material would be processed in less than 3 batches. A batch dissolution is completed in approximately three hours. Gadolinium nitrate would be used as the poison, at a concentration estimated to be about 1.8 M.

Steam jet transfers and dissolver heel rinses result in dilution by the time the dissolver product reaches the G-cell accountability tank in CPP-601, where it will be isotopically diluted and purified by extraction.

Depleted uranium, as oxide, is dissolved using nitric acid to obtain a solution containing 450 g/L uranium. The depleted uranium solution is then blended with the U-233 enriched solution. The final U-233 concentration is easily controlled by the feed rate of depleted uranium. The diluted solution, if containing INEL U-233 material, is first processed through an Acid Thorex process and then a two-cycle hexone system. The diluted ORNL uranium solution, which contains no thorium, bypasses the Thorex system and is purified only by the hexone system.

The Acid Thorex process uses a 5 vol% tributylphosphate/95 vol% normal paraffin hydrocarbon (TBP-NPH) as the extractant. Uranium and thorium are extracted into the organic phase in the first column. The uranium and thorium containing organic phase is scrubbed to remove thorium, and then stripped of uranium. The organic solvent is recycled, while the aqueous product is evaporated to concentrate uranium to about 250 g/L, and then stored prior to further purification. The aqueous waste drains into stainless steel tanks in the existing high level waste tank farm.

The concentrated product from the acid Thorex process, or the isotopically diluted ORNL uranium-containing solution, is then processed through an acid-deficient hexone extraction system. Uranium and any remaining thorium and radioactive decay products are extracted into hexone (methyl isobutyl ketone). The radionuclide-rich organic phase is scrubbed to remove thorium, if any, and decay products; it is then stripped to transfer uranium to the aqueous phase, and the organic solvent is recycled. Following two cycles of extraction, the uranium product is concentrated by evaporation to about 300 g/L. The aqueous wastes drain to the tank farm.

Following extraction, the purified uranium is converted from nitrate to oxide in a fluidized bed denitrator, and the UO_3 product is put in 2.85 liter (5.5 inch x 7.5 inch) cans. These cans are then returned to CPP-749 for storage. Aqueous wastes will be mixed with other high level wastes generated at the ICPP in the tank farm, and then calcined in the NWCF. Calcium nitrate is added in the NWCF to react with fluoride ions in the waste.

Equipment exists at the ICPP for performing all of the unit operations except for decladding fuel from rods. It is expected that decladding could be accomplished using an automatic pipe cutter to remove the rod ends, then flaring the open ends and sliding out the pellets. Should this prove infeasible, a hydrofluoric acid chemical decladding process could be used. After chemical decladding, the decladding solution would be transferred to one of the three FDP dissolver train vessels for reuse on a subsequent assembly batch. The fuel pellets would be rinsed and dissolved as described above. At the end of the useful life of the decladding solution, excess hydrofluoric acid would then be complexed with aluminum nitrate, the solution would be adjusted with nitric acid for extraction and put through the solvent extraction process to recover any small amount of uranium that may have dissolved during the decladding.

Chemical decladding would leave nichrome springs largely undissolved. These could be removed from the dissolvers using a vacuum device that has been developed for the FDP dissolvers. The springs would then be sent to the RWMC for disposal.

Some testing would be needed to demonstrate the high thorium U-233 blending process in ICPP equipment. The calciner has not been operated with thorium wastes, and the formation of thorium dioxide particulate may cause problems in the NWCF offgas scrubbing system. A second concern over processing the thorium wastes in the calciner is that in the event of the calciner going down, removal of the resultant solidified thorium-containing solid from the calciner may be difficult. Because of these concerns over processing thorium waste in the NWCF, an alternative process is presented in Section 4.2 which avoids calcination of thorium waste.

While the calcination issues are recognized and must be addressed, including providing a contingency, it is expected that testing will show that they are of no concern. The small particle sizes exposed to the scrub solution will likely dissolve rapidly, based on known dissolution kinetic models. Similarly, the small concentration of thorium blended in the calcine most likely would not present problems in bed dissolution. The presence of other "insoluble" materials, such as molybdenum and ruthenium, have not resulted in difficulties.

4.1.1 Function

The function of the process is to dilute LWBR fuel and other U-233 containing materials to less than 5 or 12% U-233 in a dissolution process. A second function of the thorium removal process is to purify the resultant diluted uranium product from thorium and radioactive decay products in order to reduce the volume and activity of the product.

4.1.2 Feeds

The feeds to the thorium removal process include unirradiated LWBR fuel rods presently stored in CPP-749 dry silos, unirradiated LWBR rods stored in drums at the RWMC, and U-233 materials stored in cans in Building 3019 at ORNL. The LWBR fuels consists of thorium oxide/uranium oxide pellets clad in Zircaloy. The fuel rods contain nichrome springs in the plenum. The LWBR fuel in

CPP-749 contains a total of 319.8 kg of uranium, of which 98.1% is U-233. It also contains 14,255 kg of Thorium and 5,280 kg of Zircaloy-4 cladding.

The U-233 materials at the RWMC include sixty-eight (68) 55-gal drums of fuel rod sections and fuel pellets, seventy-seven (77) 110-gal drums of fuel rod sections and fuel pellets, and one hundred fifty-five (155) 55-gal drums of metallographic mounts, fuel assembly ends, and other materials. Most fuel pellets are uranium and thorium oxides, but 22 drums contain uranium/zirconium oxide pellets. The total uranium contained in the RWMC fuel rods amounts to 35.1 kg, of which 97.6% is U-233. These rods also contain 1291 kg of thorium and 1168 kg of Zircaloy and zirconium.

U-233 materials are stored in aluminum and stainless steel cans at ORNL in a variety of forms. The largest amount is present as U_3O_8 monolithic material, while other forms include oxide powders, uranium metal, and uranium tetrafluoride. The uranium in much of the oxide monolith is 75% U-235 and 10% U-233, while most other forms and types of ORNL materials have a U-233 enrichment of greater than 90%. The combined ORNL material contains a total of 1379 kg of uranium, of which 30% is U-233 and 57% is U-235.

4.1.3 Products

The product from the thorium removal process is UO_3 in granular form, with an enrichment of U-233 of less than 12%, and containing only trace quantities of thorium and radioactive decay products. A 5% U-233 enrichment can easily be produced by the process by using an appropriate rate of depleted uranium. The product is packaged in 2.85 liter (5.5 inch x 7.5 inch) cans. The product quantities shown in Table 4-1 would be produced from the different feeds, based on 12.5 kg U per can:

Table 4-1. Product Quantities—Thorium Removal		
Product	12% U-233 Quantity	5% U-233 Quantity
ICPP LWBR U-233 fuel	220 cans	510 cans
RWMC LWBR U-233 fuel	25 cans	60 cans
ORNL U-233 materials	820 cans	2020 cans
Total	1065 cans	2590 cans

4.1.4 Utilities Required

The required utilities include electricity, low pressure steam, raw water, cooling water and kerosene, used as fuel for the calciner.

4.1.5 Chemicals Required

Chemicals used in the process include 12 M HNO_3 , HF, aluminum nitrate nonohydrate (ANN), cadmium nitrate, gadolinium nitrate, calcium nitrate, TBP/NPH solvent, and hexone.

4.1.6 Special Requirements

No special requirements have been identified.

4.1.7 Wastes Generated

The solid waste generated by the process includes the thorium-containing calcined waste from the extraction systems, metallic waste including rod ends, springs and fuel rods void of fuel particles, and the empty drums and cans which contained the feed materials. Liquid waste from offgas treatment is also calcined. Gaseous effluents are treated by the offgas treatment systems.

4.2 Coextraction Process

The coextraction process is identical to the thorium extraction process in the decanning, decladding, dissolution, and blending steps. However, in contrast to the thorium removal process, only one cycle of extraction is used for the INEL feeds, and thorium is not separated from uranium. Only radioactive decay products are extracted from the blended solution, using a 30% TBP/70% NPH solvent. The product from extraction thus contains both uranium and thorium, is concentrated by evaporation and then denitrated. The TBP solvent, stripped of radionuclides, is recycled, and the aqueous waste, containing very little thorium, is concentrated in the process equipment evaporator, and the concentrated waste sent to the tank farm. For ORNL feed, this process is identical to the thorium extraction process described above. A block diagram of the coextraction process is shown in Figure 4-2.

For the coextraction process, denitration of the thorium/uranium product would need to be demonstrated in the ICPP denitrator. Cleanout of the denitrator after shutdown may require dissolution of thorium oxides and may be difficult. Also, because of the increased volume of product material produced in the coextraction process compared to the thorium removal process, the denitrator is the bottleneck of the process, requiring 457 operating days, or about 3.8 years to process the INEL U-233 material.

4.2.1 Function

The function of the coextraction process is to dilute LWBR fuel and other U-233 containing materials to less than 12% U-233 in a dissolution process. A secondary function of the process is to purify the resultant diluted uranium product from radioactive decay products in order to reduce activity of the product.

4.2.2 Feeds

The feeds to the coextraction process are the same as for the thorium removal process, described in Section 4.1.2.

4.2.3 Products

The product from the coextraction process is a mixed uranium and thorium oxide in granular form. The enrichment of U-233 in the product can be set in the process at 12%, 5% or any other desirable level. The product contains only trace quantities of radioactive decay products. The product is packaged in 2.85 liter (5.5 inch x 7.5 inch) cans. The product quantities shown in Table 4-2 would be produced from the different feeds, based on 12.5 kg uranium per can.

Table 4-2. Product Quantities—Coextraction		
Product	12% U-233 Quantity	5% U-233 Quantity
ICPP LWBR U-233 fuel	1350 cans	1640 cans
RWMC LWBR U-233 fuel	125 cans	160 cans
ORNL U-233 materials	820 cans	2020 cans
Total	2295 cans	3820 cans

4.2.4 Utilities Required

The required utilities include electricity, low pressure steam, raw water, cooling water, and kerosene, used as fuel for the calciner.

4.2.5 Chemicals Required

Chemicals used in the process include 12 M HNO₃, HF, ANN, cadmium nitrate, gadolinium nitrate, calcium nitrate, and TBP/NPH solvent. Hexone is required only for processing ORNL materials.

4.2.6 Special Requirements

No special requirements have been identified.

4.2.7 Wastes Generated

The wastes generated by the coextraction process are the same as those generated by the thorium removal process, except that the calcined waste from the extraction systems does not contain thorium.

4.3 Waste Management

The handling, processing (decladding), dissolution, and extraction of unirradiated U-233 fuel will result in several different sources of solid and liquid wastes. Although the process is only moderately complex the variability inherent in virtually all chemical processes provide attractive opportunities for waste prevention and minimization. The objectives of this section are (1) to identify the waste types and categories which are likely to result from the processes and (2) to identify possible alternatives for their prevention and/or minimization.

The method of presentation follows essentially the same steps as the processing of the fuel, beginning with the receipt of the materials at the ICPP and following the process through to ultimate disposition.

Although every process step is addressed, there are several resulting in no waste or only one waste type or category, and that fact is noted accordingly. The results of the waste estimates are summarized in sections addressed to individual categories, including brief definitions of the waste classes themselves.

4.3.1 The Waste Management Process

4.3.1.1 Fuel Receipt. The U-233 fuel and fuel assemblies are unirradiated; therefore, no fission products are expected to be present above natural background. The fuel receipt operations will result in the generation of some waste, consisting primarily of the containers the U-233 materials were stored and shipped in and coincidental wastes attributable to their handling during receiving (boots, coveralls, tape, gloves, etc.). The fuel is packaged in a number of different containers ranging from 55 gallon capacity 17H drums, or equivalent, to custom-made stainless steel vessels of various lengths and diameters. Unless the storage/shipping containers are expected to be reused, some of the custom made stainless vessels, most notably the CPP-749 containers will have to be reduced in size to 6 feet or less for packaging into B-25 type waste disposal boxes. The use of cold roller or diamond wire cutters, as opposed to friction wheel cutters or plasma torches, is recommended for volume reduction. The 55 gallon drums should be compacted for packaging, also into B-25 type disposal boxes. There is a possibility that the stainless steel could be reprocessed through commercial smelting.

There appears to be a possible volume reduction benefit in cutting the stainless steel containers into longitudinal sections and nesting the cut components inside each other. As noted above, cold cutting using diamond wire (similar to that used in concrete cutting), as opposed to friction wheels or torches, would result in the least amount of coincidental waste generated. The equipment is commercially available and relatively inexpensive to use. Most importantly, the operation does not require extensive atmospheric control, and the only waste generated (LLW) would be the diamond coated wire used by the cutter.

The predominant contaminants resulting from the receiving operations are the natural U-233 decay daughters which, by regulatory and departmental definitions, are low-level wastes. A cursory examination indicates that decontamination of the containers to certifiably clean levels is unlikely to be successful. Additionally, even the attempt to do so would result in the generation of large quantities of decontamination fluids requiring additional processing prior to disposal. Also, there appears to be no net gain by a reduction in the waste classification (from Class C to A) as all of the containers and associated wastes, as generated, are expected to be Class A LLW.

Due to the fact that the contact radiation levels are expected to exceed 200 mR/hr, the fuel shipments are anticipated to be made in shipping casks or Type A transport shields. The handling and unloading operations are expected to generate some additional amounts of LLW waste, possibly including some quantities of cleaning fluids used in decontamination.

Regardless of the method of disposition, all of the fuel containers must be tested for removable contamination by taking gross swipes of the internal and external surfaces and counting for alpha and gamma emitters. The information is required by both the processors of the waste (compactors) and by the disposal site. The information is also necessary for the generation of shipping papers for the transportation of the containers to their ultimate disposition.

As all of the operations are expected to take place inside a Radioactive Materials Management Area (RMMA), or a Radiologically Controlled Area (RCA), all of the wastes resulting from fuel receiving are expected to be LLW. As a consequence, none of the wastes will be categorized as hazardous, nonhazardous, or sanitary wastes. Additionally, attempts to reclassify or decontaminate the materials

from LLW to any other "lesser" category is unlikely to be successful and is judged to be economically or volumetrically counterproductive. Standard operating practices at the ICPP prevent the use of chemicals which would be deemed hazardous (listed or characteristic) by either the EPA and/or its equivalent, the State of Idaho Department of Environmental Quality (DEQ). Therefore, mixed waste generation is not expected.

4.3.1.2 Fuel Decladding. Most of the unirradiated U-233 fuels subject to this report are clad in stainless steel or zirconium tubes. A certain number of the fuel pins are also assembled into fuel assemblies with appropriate numbers of end-fittings, spacers, and other related hardware. Due to their close mechanical similarities and associations, as well as to the fact that both the fuel decladding and separation of the non-fuel components are going to be performed in the same location (hot cell), the two are discussed together under one heading without extensive distinctions. Except for the obvious dimensional differences, the resulting wastes are also expected to be virtually identical and subject to essentially the same concerns, waste categories, and disposal scenarios.

The mechanical removal of the fuel assembly components as well as the physical means of separating the cladding from the fuel versus chemical removal, such as dissolution, provides a significant opportunity for the avoidance of waste generation. Although the waste volume generated by chemical dissolution is unlikely to be significantly greater than that resulting from mechanical removal, the currently available method of waste handling at the ICPP would result in the dissolved cladding being combined with other HLW, thus adding to an already complex management and disposal problem. All reasonable attempts will be made to achieve mechanical separation of fuel from non-fuel components allowing the handling and management of empty cladding and fuel hardware as LLW which can be disposed of at the RWMC.

Aside from the radiation levels virtually dictating the use of remote tooling, the removal of fuel hardware and decladding operations are relatively simple in theory. As fuel pins and assemblies are manufactured to highly refined and rigidly enforced tolerances, the tooling requirements are relatively simple and unlikely to vary during the entire process. The ends of the fuel assemblies are normally bolted in place and, as the fuel is unirradiated, should separate easily. The same argument applies to the associated fuel pin spacers and/or spacer grids, flowguides, instrumentation tubes, and poison rods, if any are present. It should again be noted that, as the fuel is unused, the dimensions should be as manufactured and not distorted as would occurred in the hostile environment of a reactor.

The separation of the fuel pellets from the cladding requires that the end plugs be removed by cutting, chamfering the ends of the tubes and expelling the pellets. The use of circular cutting wheels, as opposed to shears, is recommended for the removal of the end plugs from the fuel pins. The reason is targeted toward minimizing end distortion of the cladding to allow the pellets to be expelled from the cladding with greater ease. If necessary, the pins may be sectioned into shorter lengths that may be heated or cooled.

The foregoing discussion is included here to underscore the obvious advantages of separating all of the non-fuel components prior to dissolution of the fuel. The resulting wastes, such as the fuel assembly components and cladding, would be LLW and relatively easy to disposition. Similar to the storage and shipping containers, the only contamination would be the natural decay daughters in concentrations well within low-level waste disposal criteria and limits. Also similar to the treatment and handling of the shipping and storage containers, there is no advantage in attempting to decontaminate the resulting components to certifiably clean levels, however; the stainless steel may be reprocessed through smelting if there is available capacity.

The alternative of the foregoing, chemical dissolution will result in large quantities of LLW liquids. However due to the peculiarities of the systems at the ICPP, the only convenient route for the liquids is to drain them directly into the HLW storage tanks. The treatment of HLW is a costly operation, and the storage and eventual disposal are subject to considerable uncertainties, thereby adding complexities best avoided. Decladding via chemical means should only be attempted as a last resort after all reasonable mechanical means have failed to yield acceptable results.

4.3.1.3 Fuel Dissolution. The dissolution and extraction of the unirradiated U-233 fuel is essentially identical to the reprocessing of any irradiated fuel assemblies (SNF). The chemicals used in these processes are generally either non-hazardous (TBP) or exhibit characteristic (corrosive) hazards. The resulting characteristic mixed wastes are drained to the HLW tanks as raffinate from the extraction columns and eventually transferred to the waste calciner for solidification. The systems for handling the extraction wastes are essentially hard-piped and double contained. However, it should be noted that the waste resulting from U-233 fuel dissolution and extraction process, by definition, is not HLW. The U-233 extraction wastes are, in fact, LLW, and the resulting solutions could be handled and processed as ordinary LLW liquids. Depending on the degrees of cross-contamination by fission products already in the system, the resulting solids may be disposable as a Class A or higher LLW.

As noted above, the entire HLW liquid train is hard-piped, and diversion of the liquids out of that system would require extensive prior evaluation and technical preparation. This is consistent with the design philosophy of the ICPP process in making every attempt to minimize the possibility of leakages of HLW liquids. Due to the closed nature of the processing system, liquid wastes will be combined with the already existing HLW and treated in an identical fashion. Processing the U-233 fuel will result in approximately 100,000 gallons of waste and a complement of LLW waste volume similar to that attributable to the normal operation of the system, such as during the previous processing of irradiated or spent fuel.

4.3.1.4 Fuel Dilution. The separated U-233 nitrate solution will be combined and diluted with depleted U-238 (depleted in U-235) nitrate solution to either 12% or <5% concentrations. The dilution operations do not require the addition of any chemicals external to the process and, hence, are not expected to produce any new uncharacterized wastes other than low-level wastes normally attributable to plant operations.

4.3.2 Waste Management Feeds

Waste management feeds resulting from the dissolution of U-233 are described below and include: LLW, mixed low-level wastes, hazardous wastes, and sanitary and other nonhazardous wastes.

4.3.2.1 Low-Level Wastes. LLW is defined in 10 CFR 61 and is further described in the appropriate chapter of DOE Order 5820.2A and its current or subsequent revisions. The management and/or disposal of such wastes are governed by site criteria which reflect the regulatory requirements.

LLW is the predominant and essentially only waste category which will result from the U-233 fuel dilution project. The LLW waste generation rate can be estimated to be approximately 0.1 ft³ per fuel pin, or fuel pin equivalent amount of fuel being dissolved. The exact value will vary due primarily to the divergence of fuels handled (i.e., clad, unclad, assembled, canned, and drummed) and to the number of operations the fuel is processed through (i.e., received, declad, dissolved, and diluted). The LLW is primarily miscellaneous waste (99% volume) with small quantities of laboratory wastes generated during the analyses of the various effluents. Similarly, the LLW liquids which may be generated coincident to the process will consist mostly of decontamination liquids not unlike those already in general use at the

ICPP. Low-level liquids wastes at the ICPP presently are evaporated in the PEW evaporator, and the bottoms are sent to the tank farm for eventual calcination.

4.3.2.2 Mixed Low-Level Wastes (MLLW). Mixed wastes are those which contain both radioactive materials and chemicals which the EPA, and/or its State equivalent, has determined to be hazardous and so identified in the appropriate Subparts of 40 CFR 261 or the equivalent State regulations. As described in the section titled Fuel Dissolution, the main source of MLLW is the fuel dissolution and extraction processes.

All of the MLLW generated during the process meets the regulatory definition for the category of Characteristic Mixed Wastes by being either corrosive or flammable. The other concurrent and/or coincident operations during the U-233 fuel dilution process are not expected to generate major sources of MLLW, but minor quantities will be generated and treated at various steps throughout the process.

4.3.2.3 Hazardous Wastes. A solid waste is a hazardous waste if it meets the regulatory criteria defined in 40 CFR by exhibiting certain characteristics and/or containing certain specific chemicals deemed hazardous by the EPA and/or its equivalent State agency.

As noted, all of the waste generated coincident to the U-233 fuel dilution process must be regarded as being contaminated with some amounts of radioactive material, primarily uranium decay daughters, and, therefore, by exclusion, none of it can be considered as strictly hazardous waste. As the ICPP is expected to receive periodic shipments of nitric and/or hydrofluoric acids, some amount of hazardous waste may result from the receiving handling of the shipments, however; under normal operating conditions, the total waste volume, consisting of a few cleaning rags, should be negligible.

4.3.2.4 Sanitary and Other Nonhazardous Wastes. Sanitary wastes will be generated by the process operations and will be sent to the existing system.

4.3.3 Waste Management Products

Excluding the liquids produced during dissolution, virtually all of the waste (99+% volume) resulting from the U-233 fuel dilution project will be low-level solids. The liquid wastes from the extraction system will be drained to the HLW tanks for eventual calcination, while the LLW solids will be packaged into appropriate containers for disposal at the RWMC.

4.3.4 Waste Management Special Requirements

None of the wastes resulting from the U-233 fuel dilution process are expected to be outside of the normal categories and quantities encountered during steady-state operations at the ICPP. The nominal characteristics of the waste and volumes also results in the absence of a need for new, unavailable, or special waste treatment systems, packaging, and/or handling requirements. As the majority of the waste volume is normal type LLW, existing systems will be used for their packaging and disposal.

5.0 RESOURCE NEEDS

Various resources are required during operation and facility construction/modification, as described in the following section.

5.1 Materials/Resources Consumed During Operation

Sections 5.1.1-5.1.5 present rough estimates of the utilities and chemicals that will be required to process the U-233 feed materials.

5.1.1 Utilities Consumed

Estimates of utilities are shown in Table 5-1 for dilution to 12% and in Table 5-2 for dilution to 5% U-233. Utilities are given for both the thorium removal and coextration processes. These estimates include only major utility users in the U-233 process and do not include contributions for common support utilities. Utilities are based on processing the entire volume of INEL and ORNL U-233 feed materials.

Table 5-1. Utilities Consumed in Dilution of U-233 Materials to 12% U-233 Enrichment		
	Thorium Removal Process	Coextration Process
Coal, 10 ³ kg	80	160
Kerosene, 10 ³ liters	80	120
Electricity, MW-hr	60	140
Raw water, 10 ⁶ liters	60	90

Table 5-2. Utilities Consumed in Dilution of U-233 Materials to 5% U-233 Enrichment		
	Thorium Removal Process	Coextration Process
Coal, 10 ³ kg	240	320
Kerosene, 10 ³ liters	260	280
Electricity, MW-hr	230	320
Raw water, 10 ⁶ liters	200	220

The present raw water requirement for the ICPP is approximately 2 million gallons per day⁷⁶.

5.1.2 Water Balance

The water balance is shown in Figure 5-1.

Figure 5-1. Water Balance

5.1.3 Chemicals Consumed

Chemicals are given in Table 5-3 for isotopic dilution of feed materials to 12% U-233 and in Table 5-4 for dilution to 5% U-233, for both the thorium removal process and the coextraction process. Volumes of chemicals are based on processing the entire quantities of INEL and ORNL U-233 feed materials.

Table 5-3. Chemicals Consumed in Dilution to 12% U-233		
	Thorium Removal Process	Coextraction Process
Depleted Uranium (as UO_2), kg	13,300	35,400
13 M Nitric Acid, liters	40,000	30,000
Hydrofluoric Acid (50%), liters	350	350
Aluminum Nitrate Nonohydrate, kg	100,000	70,000
TBP/NPH Solvent, liters	12,000 (5% TBP)	12,000 (30% TBP)
Hexone, liters	4,700	3,400
$\text{Cd}(\text{NO}_3)_2 \cdot 4\text{H}_2\text{O}$, kg	1,400	1,400
$\text{Ca}(\text{NO}_3)_2 \cdot 4\text{H}_2\text{O}$, kg	1,000	1,000
$\text{Gd}(\text{NO}_3)_3 \cdot 6\text{H}_2\text{O}$, kg	3,000	3,000

Table 5-4. Chemicals Consumed in Dilution to 5% U-233		
	Thorium Removal Process	Coextraction Process
Depleted Uranium (as UO_2), kg	13,300	35,400
13 M Nitric Acid, liters	80,000	60,000
Hydrofluoric Acid (50%), liters	350	350
Aluminum Nitrate Nonohydrate, kg	400,000	330,000
TBP/NPH Solvent, liters	13,000 (5% TBP)	13,000 (30% TBP)
Hexone, liters	19,000	16,000
$\text{Cd}(\text{NO}_3)_2 \cdot 4\text{H}_2\text{O}$, kg	1,400	1,400
$\text{Ca}(\text{NO}_3)_2 \cdot 4\text{H}_2\text{O}$, kg	1,000	1,000
$\text{Gd}(\text{NO}_3)_3 \cdot 6\text{H}_2\text{O}$, kg	3,000	3,000

5.1.4 Mass Balances

This section presents mass balances for the blending of U-233. It is important to note that this material balance is based on preliminary results and small discrepancies occur between feed and product tables due to rounding of values.

Case 1: 12% Dilution and Thorium Removal

A. LWBR Fuel at ICPP

DISSOLUTION			
Feeds (kg)		Product (kg)	
UO ₂	363.7	UO ₂ (NO ₃) ₂	533.9
ThO ₂	16,221.0	Th(NO ₃) ₄	29,493.0
HF	186.0	HF	186.0
Al(NO ₃) ₃	1,300.0	Al(NO ₃) ₃	1,300.0
Cd(NO ₃) ₂	7,000.0	Cd(NO ₃) ₂	7,000.0
HNO ₃ (100%)	47,100.0	HNO ₃	31,270.0
H ₂ O	36,400.0	H ₂ O	46,860.0
Excess H ₂ O	8,200.0	NO ₂	126.0
TOTALS:	116,770.7		116,769.0

DEPLETED URANIUM DISSOLUTION			
Feeds (kg)		Product (kg)	
UO ₂ (DU)	2,836.0	NO ₂	966.0
HNO ₃ (100%)	4,330.0	H ₂ O	4,690.0
H ₂ O	3,350.0	HNO ₃ (100%)	1,680.0
H ₂ O dilution	956.0	UO ₂ (NO ₃) ₂	4,138.0
TOTALS:	11,472.0		11,474.0

BLENDING				
	U-233 Fuel Solution	DU Solution	Feed Adjustment	Product
UO ₂ (NO ₃) ₂	534.0	4,138.0		4,672.0
Th(NO ₃) ₄	29,493.0			29,493.0
HF	186.0			186.0
Al(NO ₃) ₃	1,300.0			1,300.0
Cd(NO ₃) ₂	7,000.0			7,000.0
HNO ₃	31,270.0	1,680.0		32,950.0
H ₂ O	46,860.0	4,690.0	40,000.0	91,550.0
TOTAL (kg):	116,643.0	10,508.0	40,000.0	167,151.0
TOTAL (L):	63,300.0	5,555.0	40,000.0	108,855.0

TBP EXTRACTION							
	Feed Solution	Scrub Solution	Strip Solution	HL Waste	Evaporator		
					Feed	Product	Condensate
UO ₂ (NO ₃) ₂	4,672.0			47.0	4,625.0	4,625.0	
Th(NO ₃) ₄	29,493.0			29,410.0	83.0	83.0	
HF	186.0			186.0			
Al(NO ₃) ₃	1,300.0			1,300.0			
Cd(NO ₃) ₂	7,000.0			7,000.0			
HNO ₃	32,950.0	2,700.0	40.0	35,690.0			
H ₂ O	91,950.0	42,200.0	70,860.0	133,710.0	70,900.0	11,200.0	59,700.0
TOTAL:	167,151.0	44,900.0	70,900.0	207,343.0	75,608.0	15,908.0	59,700.0

HEXONE EXTRACTION							
Note: Two-cycle extraction will require doubling quantities of scrub, strip, and waste. Al(NO ₃) ₃ quantities, not shown.							
	Feed Solution	Scrub Solution	Strip Solution	Thorium Waste	Evaporator		
					Feed	Product	Condensate
UO ₂ (NO ₃) ₂	4,625.0			46.0	4,579.0	4,579.0	
Th(NO ₃) ₄	83.0			83.0			
HNO ₃		1,850.0	30.0	1,880.0			
H ₂ O	11,200.0	28,950.0	48,470.0	40,120.0	48,500.0	9,200.0	39,300.0
TOTAL (kg):	15,908.0	30,800.0	48,500.0	42,129.0	53,079.0	13,779.0	39,300.0
TOTAL (L):	11,200.0	29,900.0	48,500.0	41,100.0	48,500.0	9,200.0	39,300.0

DENITRATION			
	Feed	UO ₃ Product	Off-Gas
UO ₂ (NO ₃) ₂	4,579.0		* 3321.6 kg UO ₃ contains 2763.3 kg U. 2763.3 ÷ 12.6 = 221 cans.
H ₂ O	9,200.0		
NO ₂			
O ₂			
UO ₃		3,321.6*	
TOTALS (kg):	13,779.0	3,321.6	10,457.4

B. RWMC U-233 Material

DISSOLUTION			
Feeds (kg)		Product (kg)	
UO ₂	39.9	UO ₂ (NO ₃) ₂	58.6
ThO ₂	1,469.0	Th(NO ₃) ₄	2,669.0
HF	21.0	HF	21.0
Al(NO ₃) ₃	150.0	Al(NO ₃) ₃	150.0
Cd(NO ₃) ₂	800.0	Cd(NO ₃) ₂	800.0
HNO ₃ (100%)	4,890.0	HNO ₃	3,450.0
H ₂ O	3,780.0	H ₂ O	5,180.0
Excess H ₂ O	1,194.0	NO ₂	14.0
TOTALS (kg):	12,343.9		12,342.6

DEPLETED URANIUM DISSOLUTION			
Feeds (kg)		Product (kg)	
UO ₂ (DU)	311.0	UO ₂ (NO ₃) ₂	454.0
HNO ₃ (100%)	475.0	HNO ₃ (100%)	184.0
H ₂ O	368.0	H ₂ O	515.0
H ₂ O dilution	105.0	NO ₂	106.0
TOTALS (kg):	1,259.0		1,259.0

BLENDING				
	U-233 Fuel Solution	Feed Adjustment	DU Solution	Product
UO ₂ (NO ₃) ₂	58.6		454.0	512.6
Th(NO ₃) ₄	2,669.0			2,669.0
HF	21.0			21.0
Al(NO ₃) ₃	150.0			150.0
Cd(NO ₃) ₂	800.0			800.0
HNO ₃	3,450.0	4,400.0	184.0	3,634.0
H ₂ O	5,180.0		515.0	10,095.0
TOTAL (kg):	12,328.6	4,400.0	1,153.0	17,881.6
TOTAL (L):	11,270.0		610.0	12,000.0

TBP EXTRACTION							
	Feed Solution	Scrub Solution	Strip Solution	HL Waste	Evaporator		
					Feed	Product	Condensate
UO ₂ (NO ₃) ₂	512.6			5.2	507.4	507.4	
Th(NO ₃) ₄	2,669.0			2,664.0	5.0	5.0	
HF	21.0			21.0			
Al(NO ₃) ₃	150.0			150.0			
Cd(NO ₃) ₂	800.0			800.0			
HNO ₃	3,634.0	300.0	5.0	3,939.0			
H ₂ O	10,095.0	4,640.0	7,795.0	14,730.0	7,800.0	1,225.0	6,575.0
TOTAL (kg):	17,881.6	4,940.0	7,800.0	22,309.2	8,312.4	1,737.4	6,575.0
TOTAL (L):	12,000.0	4,800.0	7,800.0	16,800.0	7,800.0	1,225.0	6,575.0

HEXONE EXTRACTION							
Note: Two-cycle extraction require doubling quantities of scrub, strip, and waste. Al(NO ₃) ₃ quantities not shown.							
	Feed Solution	Scrub Solution	Strip Solution	Feed Adjust.	Thorium Waste	Evaporator	
						Feed	Product
UO ₂ (NO ₃) ₂	507.4				5.1	502.3	502.3
Th(NO ₃) ₄	5.0				5.0		
HNO ₃		200.0	3.0		203.0	203.0	
H ₂ O	1,225.0	3,165.0	5,305.0	103.0	4,490.0	5,308.0	4,298.0
TOTAL (kg):	1,737.4	3,365.0	5,308.0	103.0	4,703.1	5,810.3	4,298.0

DENITRATION			
	Feed	UO ₃ Product	Off-Gas
UO ₂ (NO ₃) ₂	502.3		* 364.4 kg UO ₃ contains 303.5 kg U. 303.5 ÷ 12.6 = 24.3 cans.
H ₂ O	1,010.0		
NO ₂			
O ₂			
UO ₃		364.4*	
TOTALS (kg):	1,512.3	364.4	1,147.9

C. ORNL U-233 Material

DISSOLUTION			
Feeds (kg)		Products (kg)	
U (metal)	14.0	Offgas: NO ₂	214.0
UF ₄	4.0	NO	3.6
UO ₄	15.0	O ₂	0.6
UO ₂ and UO ₃	280.0	Solution: HF	1.0
U ₃ O ₈	1,307.0	HNO ₃ (100%)	160.0
10 M HNO ₃	2,450.0	H ₂ O	2,508.0
H ₂ O	1,109.0	UO ₂ (NO ₃) ₂	2,295.0
TOTALS (kg):	5,179.0		5,182.2

DEPLETED URANIUM DISSOLUTION			
Feeds (kg)		Products (kg)	
UO ₂ (DU)	10,145.0	Offgas: NO ₂	3,457.0
HNO ₃ (100%)	15,479.0	Solution: UO ₂ (NO ₃) ₂	14,804.0
H ₂ O	7,331.0	H ₂ O	16,755.0
Excess H ₂ O	8,071.0	HNO ₃ (100%)	6,010.0
TOTALS (kg):	41,026.0		41,026.0

BLENDING

	U-233 Fuel Solution	DU Solution	Feed Adjustment	Product
UO ₂ (NO ₃) ₂	2,295.0	14,804.0		17,099.0
HNO ₃	160.0	6,010.0		6,170.0
HF	1.0			1.0
H ₂ O	150.0	16,755.0	12,067.0	31,330.0
TOTAL (kg):	4,964.0	37,569.0	12,067.0	54,600.0

HEXONE EXTRACTION

Note: Two-cycle extraction require doubling quantities of scrub, strip, and waste.
Al(NO₃)₃ quantities, not shown.

	Feed Solution	Scrub Solution	Strip Solution	Waste	Evaporator		
					Feed	Product	Condensate
UO ₂ (NO ₃) ₂	17,099.0			171.0	16,928.0	16,928.0	
HF	1.0			1.0			
HNO ₃	6,170.0	5,670.0	90.0	11,930.0			
H ₂ O	31,330.0	88,830.0	149,010.0	120,070.0	149,100.0	22,710.0	126,390.0
TOTAL (kg):	54,600.0	94,500.0	149,100.0	132,172.0	166,028.0	39,638.0	126,390.0
TOTAL (L):	34,406.0	91,750.0	149,100.0	126,155.0	149,100.0	22,710.0	126,390.0

DENITRATION

	Feed	UO ₃ Product	Off-Gas
UO ₂ (NO ₃) ₂	16,928.0		* 12,280 kg UO ₃ contains 10,219 kg U. 10,219 ÷ 12.5 = 818 cans.
H ₂ O	22,710.0		
NO ₂			
O ₂			
UO ₃		12,280.0*	
TOTALS (kg):	39,638.0	12,280.0	27,358.0

Case 2: 12% Dilution and Coextraction

A. LWBR Fuel at ICPP—Dissolution, DU Dissolution, and Blending Material Balances are the same as for Case 1.

TBP EXTRACTION

	Feed Solution	Scrub Solution	Strip Solution	Adjust	Waste	Feed Adjust	Evaporator		
							Feed	Product	Condensate
UO ₂ (NO ₃) ₂	4,672.0				14.0	4,658.0	4,658.0	4,658.0	
Th(NO ₃) ₄	29,493.0				310.0	29,183.0	29,183.0	29,183.0	
HNO ₃	32,950.0	8,830.0	140.0		41,920.0				
H ₂ O	91,550.0	138,250.0	232,060.0	248,500.0	478,160.0	715,040.0	232,200.0	56,300.0	658,740.0
Cd(NO ₃) ₂	7,000.0				7,000.0				
Al(NO ₃) ₃	1,300.0				1,300.0				
HF	186.0				186.0				
TOTAL (kg):	167,151.0	147,080.0	232,200.0	248,500.0	528,890.0	748,881.0	266,041.0	90,141.0	658,740.0
TOTAL (L):	109,000.0	142,800.0	232,200.0	248,500.0	500,300.0	751,040.0	232,200.0	56,300.0	658,740.0

DENITRATION

	Feed	Mixed Oxide Product	Off-Gas
UO ₂ (NO ₃) ₂	4,658.0		* 3,379 kg UO ₃ contains 2,811 kg U, and 16,051 kg ThO ₂ contains 14,105 kg Th. 2,811 kg U + 14,105 kg Th = 16,916 kg (total). 16,916 ÷ 12.5 = 1,353 cans.
Th(NO ₃) ₄	29,183.0		
UO ₃		3,379.0*	
ThO ₂		16,051.0	
NO ₂			12,276.0
O ₂			2,135.0
H ₂ O	56,300.0		56,300.0
TOTALS (kg):	90,141.0	19,430.0	70,711.0

- B. **RWMC U-233 Materials**—Dissolution, DU Dissolution, and Blending Material Balances are the same as for Case 1.

TBP EXTRACTION: Feed ~ 10% of ICPP Fuel									
	Feed Solution	Scrub Solution	Strip Solution	Adjust	Waste	Feed Adjust	Evaporator		
							Feed	Product	Condensate
TOTAL (kg):		14,700.0	23,200.0	24,800.0	53,000.0*				
TOTAL (L):		14,300.0	23,200.0	24,800.0	50,000.0*				

* 1.4 kg $\text{UO}_2(\text{NO}_3)_2$, 34 kg $\text{Th}(\text{NO}_3)_4$

DENITRATION			
	Feed	Mixed Oxide Product	Off-Gas
UO_3		371.0*	* 371.0 kg UO_3 contains 308.5 kg U, and 1,446.3 kg ThO_2 contains 1,273.6 kg Th. 308.5 kg U + 1,273.6 kg Th = 1,582.1 kg (total). 1,582.1 ÷ 12.5 = 126 cans.
$\text{UO}_2(\text{NO}_3)_2$	511.2		
$\text{Th}(\text{NO}_3)_4$	2,635.0		
ThO_2		1,449.3	
NO_2			1,129.5
O_2			196.4
TOTALS (kg):	3,146.2	1,820.3	1,325.9

- C. **ORNL U-233 Materials**—Since the feed contains no thorium, all material balances are the same as for Case 1.

Case 3: 5% Dilution and Thorium Removal

A. ICPP and RWMC U-233 Fuel

BLENDING					
	ICPP Fuel Solution	RWMC Materials Solution	Total Dissolved Fuel	Depleted Uranium Solution	Total
UO ₂ (NO ₃) ₂	534.0	59.0	593.0	11,336.0	11,929.0
Th(NO ₃) ₄	29,493.0	2,669.0	32,132.0		32,132.0
HF	186.0	21.0	207.0		207.0
Al(NO ₃) ₃	1,300.0	150.0	1,450.0		1,450.0
Cd(NO ₃) ₂	7,000.0	800.0	7,800.0		7,800.0
HNO ₃	31,270.0	3,450.0	34,720.0	4,600.0	39,320.0
H ₂ O	86,860.0	9,580.0	96,440.0	12,820.0	109,260.0
TOTAL (kg):	156,643.0	16,729.0	173,373.0	28,756.0	201,029.0
TOTAL (L):	103,300.0	11,300.0	114,600.0	15,200.0	130,000.0

DEPLETED URANIUM DISSOLUTION			
Feeds (kg)		Product (kg)	
UO ₂	7,769.0	UO ₂ (NO ₃) ₂	11,336.0
HNO ₃ (100%)	11,850.0	HNO ₃ (100%)	4,600.0
H ₂ O	9,160.0	H ₂ O	12,820.0
Excess H ₂ O	2,626.0	NO ₂	2,647.0
TOTALS:	31,405.0		31,403.0

TBP EXTRACTION							
	Feed Solution	Scrub Solution	Strip Solution	Waste	Evaporator		
					Feed	Product	Condensate
UO ₂ (NO ₃) ₂	11,929.0			119.0	11,810.0	11,810.0	
Th(NO ₃) ₄	32,132.0			32,042.0	90.0	90.0	
HF	207.0			207.0			
Al(NO ₃) ₃	1,450.0			1,450.0			
Cd(NO ₃) ₂	7,800.0			7,800.0			
HNO ₃	39,320.0	3,220.0	50.0	42,590.0			
H ₂ O	109,260.0	50,330.0	84,510.0	159,540.0	84,560.0	28,510.0	56,050.0
TOTAL:	201,029.0	53,550.0	84,560.0	243,748.0	96,460.0	40,410.0	56,050.0

HEXONE EXTRACTION

Note: Two-cycle extraction will require doubling quantities of scrub, strip, and waste.
 Al(NO₃)₃ quantities, not shown.

	Feed Solution	Scrub Solution	Strip Solution	Waste	Evaporator		
					Feed	Product	Condensate
UO ₂ (NO ₃) ₂	11,810.0			118.0	11,692.0	11,692.0	
Th(NO ₃) ₄	90.0			90.0			
HNO ₃		4,710.0	80.0	4,790.0			
H ₂ O	28,510.0	73,690.0	123,380.0	102,120.0	123,460.0	23,530.0	99,930.0
TOTAL (kg):	40,410.0	78,400.0	123,460.0	107,118.0	135,152.0	35,222.0	99,930.0
TOTAL (L):	28,510.0			104,537.0			

DENITRATION

	Feed	UO ₃ Product	Off-Gas	
UO ₂ (NO ₃) ₂	11,692.0			* 8,485 kg UO ₃ contains 7060 kg U. 7060 ÷ 12.5 = 565 cans (508 for ICPP Fuel and 57 for RWMC materials).
H ₂ O	23,530.0		23,530.0	
UO ₃		8485.0*		
NO ₂			2,732.0	
O ₂			475.0	
TOTALS (kg):	35,222.0	8,485.0	26,737.0	

B. ORNL Feeds**DEPLETED URANIUM DISSOLUTION and BLENDING**

	DU Dissolution Feed	DU Dissolution Product	ORNL Solution	Blended Solution
UO ₂	27,597.0			
HNO ₃	42,105.0	16,348.0	160.0	16,508.0
H ₂ O	32,549.0			
Excess H ₂ O	114,480.0	150,710.0	2,508.0	153,218.0
UO ₂ (NO ₃) ₂		40,270.0	2,293.0	42,563.0
NO ₂		9,404.0		
TOTAL (kg):	216,731.0	216,732.0	4,961.0	212,289.0

HEXONE EXTRACTION

Note: Two-cycle extraction will require doubling quantities of scrub, strip, and waste.
 $\text{Al}(\text{NO}_3)_3$ quantities, not shown.

	Feed Solution	Scrub Solution	Strip Solution	Waste	Evaporator		
					Feed	Product	Condensate
$\text{UO}_2(\text{NO}_3)_2$	42,563.0			426.0	42,137.0	42,137.0	
HNO_3	16,508.0	26,640.0	420.0	27,060.0			
H_2O	153,218.0	417,360.0	700,040.0	570,158.0	700,460.0	106,700.0	593,760.0
TOTAL (kg):	212,289.0	444,010.0	700,460.0	597,644.0	742,597.0	148,837.0	593,760.0
TOTAL (L):	161,640.0	431,050.0	700,460.0	592,700.0	700,460.0	106,700.0	593,760.0

DENITRATION

	Feed	UO_3 Product	Off-Gas
$\text{UO}_2(\text{NO}_3)_2$	42,137.0		* 30,581 kg UO_3 contains 25,445 kg U. $25,445 \div 12.5 = 2036$ cans.
H_2O	106,700.0		106,700.0
NO_2			9,844.0
O_2			1,712.0
UO_3		30,581.0*	
TOTALS (kg):	148,837.0	30,581.0	118,256.0

Case 4: 5% Dilution and Coextraction

ICPP and RWMC U-233 Materials

EXTRACTION									
	Feed Solution	Adjust	Scrub Solution	Strip Solution	Waste	Evaporator			
						Feed	Adjust	Product	Condensate
UO ₂ (NO ₃) ₂	11,929.0				15.0	11,914.0	11,914.0	11,914.0	
Th(NO ₃) ₄	32,132.0				337.0	31,795.0	31,795.0	31,795.0	
HF	207.0				207.0				
Al(NO ₃) ₃	1,450.0				1,450.0				
Cd(NO ₃) ₂	7,800.0				7,800.0				
HNO ₃	39,320.0		9,600.0	150.0	49,070.0				
H ₂ O	109,260.0	258,000.0	150,260.0	252,050.0	517,370.0	252,200.0	784,300.0	76,000.0	708,300.0
TOTAL (kg):	201,029.0	258,000.0	159,860.0	252,200.0	576,249.0	295,909.0	828,009.0	119,709.0	708,300.0
TOTAL (L):	130,000.0	258,000.0	155,200.0	252,200.0	543,000.0	252,200.0	784,300.0	76,000.0	708,300.0

DENITRATION			
	Feed	UO ₃ Product	Off-Gas
UO ₂ (NO ₃) ₂	11,914.0		
Th(NO ₃) ₄	31,795.0		
H ₂ O	76,000.0		76,000.0
NO ₂			14,971.0
O ₂			2,603.0
UO ₃		8,647.0*	
ThO ₂		17,488.0	
TOTALS (kg):	119,709.0	26,135.0	93,735.0

- NOTES: 1) Dissolution assumed 12 M HNO₃; ICPP purchases HNO₃ at 13 M.
 2) 1 M HNO₃ and 0.01 N HNO₃ assumed for scrub and strip solutions. These are preliminary and may change.
 3) Al concentration in Hexone extraction is 2 M. Al(NO₃)₃ was not included in material balances as feed to Hexone extraction, but required the following calculated amounts: 100,000 kg ANN for Case 1; 70,000 kg ANN for Case 2; 400,000 kg ANN for Case 3; and 330,000 kg ANN for Case 4.

SUMMARY OF WASTES

CASE 1: 12% DILUTION, THORIUM REMOVAL		
ICPP Fuels	HL Waste from TBP Extraction	207,000.00
	HL Waste from Hexone Extraction	84,000.00
RWMC Materials	HL Waste from TBP Extraction	22,000.00
	HL Waste from Hexone Extraction	94,000.00
ORNL Materials	HL Waste from Hexone Extraction	264,000.00
TOTAL (kg):		586,400.00
TOTAL (L):		560,000.00

CASE 2: 12% DILUTION, COEXTRACTION		
ICPP Fuels	HL Waste from TBP Extraction	529,000.00
RWMC Materials	HL Waste from Hexone Extraction	53,000.00
ORNL Materials	HL Waste from Hexone Extraction	264,000.00
TOTAL (kg):		846,000.00
TOTAL (L):		810,000.00

CASE 3: 5% DILUTION, THORIUM REMOVAL		
ICPP Fuels	HL Waste from TBP Extraction	244,000.00
RWMC Materials	HL Waste from Hexone Extraction	214,000.00
ORNL Materials	HL Waste from Hexone Extraction	1,195,000.00
TOTAL (kg):		1,653,000.00
TOTAL (L):		1,600,000.00

CASE 4: 5% DILUTION, COEXTRACTION		
ICPP and RWMC Fuels	HL Waste from TBP Extraction	576,000.00
ORNL Materials	HL Waste from Hexone Extraction	1,195,000.00
TOTAL (kg):		1,771,000.00
TOTAL (L):		1,700,000.00

5.2 Materials/Resources Consumed During Construction/Modification

Because of the uncertainties related to re-permitting and re-start of existing facilities which will involve determinations by Federal and State regulatory agencies, and the Judicial system, the construction/modification emissions and waste scope cannot be determined at this time.

6.0 EMPLOYMENT NEEDS

Employment and task structure at the ICPP has fluctuated since the discontinuation of processing in 1992. The changes in employment involved with the U-233 processing are outlined below.

6.1 Employment Needs

Table 6-1 provides an estimate of the employment involved for operation of onsite transportation and operations for blending of U-233 Materials. The assumption is made that the denitrator product will be stored at CPP-749 until disposal and that no other transfer of the materials will be required prior to such disposal. Therefore, variations in employment for short vs. long term storage are not considered.

Employment levels are dictated by the operating crew size required for operation of the processing areas involved. The crew size is independent of final enrichment picked (12% or <5%) although the total length of the processing time will be greater for the 5% case.

The data in Table 6-1 reflects the best estimates of personnel required for operation of the processes based on historical information of similar processing performed prior to 1992. The Professionals and Technicians categories include additional personnel to manage the alpha radiological hazard.

Table 6-1. Employment During Operation				
Labor Category	Current (1995)	Reassigned/ Lay-offs	New Hires	Total
Officials & Managers	8	0	0	8
Professionals	10	5	10	45
Technicians	5	3	6	8
Office and Clerical	3	0	0	3
Craft Workers	9	9	9	9
Operatives	16	8	24	32
Laborers	8	4	12	16
Service Workers	5	2	2	5
Total Employees	64	31	63	126

6.2 Employees and Radiological Exposure

With the strict adherence to current DOE radiation exposure guidelines and the extensive use of ALARA for all ICPP operations activities, operating personnel will be at minimal risk from radiological exposure. The ICPP processing cells are designed for reprocessing of spent nuclear fuel with shielding walls designed accordingly. The facility radiation protection systems are over designed for direct radioation,

but may be underdesigned for alpha radiation from U-233. Approximately 5% of the employees working at the ICPP routinely work in radiologically controlled areas. Nonroutine maintenance involving cell entry during facility startup and manual uranium denitration are considered the operations tasks with potential for significant radiation exposure. These tasks involve an estimated 1% of the badged employees. Table 6-2 provides estimates of worker exposure during normal operation. As in Section 6.1, the blending level and the storage term are not relevant to the annual worker dose.

Table 6-2. Worker Exposure During Normal Operation (Direct Radioation)		
Worker Exposure	Current Operations	Blending to 12% or <5%
Number of radiation workers	55	80
Number of maintenance and denitrator workers	9	9
Predicted average annual dose/worker (mrem)	10	10
Predicted average annual dose/maintenance and denitrator worker (mrem)	150	150
Total Person-rem	1.9	2.2

6.3 Employment Needs During Construction/Modification

Because of the uncertainties related to re-permitting and re-start of existing facilities which will involve determinations by Federal and State regulatory agencies, and the Judicial system, the construction/modification emissions and waste scope cannot be determined at this time.

7.0 EMISSIONS AND WASTES FROM THE FACILITY

7.1 Wastes and Emissions During Operation

7.1.1 Chemical and Radiological Emissions

Air emissions from the ICPP for processing U-233 fuel would occur from existing ICPP sources. Chemical emissions are expected to be similar to the emissions from spent fuel reprocessing and are estimated in Table 7-1. Because the U-233 materials are unirradiated, however, the potential radiological emission increase over the ICPP baseline can only include the radioactive daughter products from the natural decay of isotopes in the fuel.

Table 7-1. Chemical Emissions Comparison Estimate			
Pollutant	ICPP Emission Baseline (Kg) ¹	Emissions from 12% Blending (Kg)	Emissions from <5% Blending (Kg)
Acetaldehyde	1.11e+01	1.11e+01	1.11e+01
Ammonia	6.34e+03	6.34e+03	6.34e+03
Arsenic	5.89e-02	5.89e-02	5.89e-02
Benzene	3.77e+01	3.77e+01	3.77e+01
Butadiene	2.97e+01	2.97e+01	2.97e+01
Chromium III	1.77e+01	1.77e+01	1.77e+01
Chromium VI	1.76e+01	1.76e+01	1.76e+01
Cyclopentane	2.03e-01	2.03e-01	2.03e-01
Formaldehyde	4.52e+01	4.52e+01	4.52e+01
Mercury	1.99e+02	1.99e+02	1.99e+02
Naphthalene	<IEL	<IEL	<IEL
Nickel	4.05e+00	4.05e+00	4.05e+00
Nitric Acid	9.58e+04	9.58e+04	9.58e+04
Phosphorus	1.48e+00	1.48e+00	1.48e+00
Propionaldehyde	6.74e+00	6.74e+00	6.74e+00
Styrene	2.16e-01	2.16e-01	2.16e-01

Table 7-1. Chemical Emissions Comparison Estimate			
Pollutant	ICPP Emission Baseline (Kg) ¹	Emissions from 12% Blending (Kg)	Emissions from <5% Blending (Kg)
Toluene	1.93e+01	1.93e+01	1.93e+01
Trimethylbenzene	<IEL	<IEL	<IEL
Hydrofluoric Acid	0.00e+00	1.60e+01	1.60e+01
Tributylphosphate	0.00e+00	5.50e+01	5.50e+01
N-Dodecane	0.00e+00	1.70e+02	1.70e+02
Hydrocarbons	0.00e+00	9.10e+02	9.10e+02
Hexone	0.00e+00	2.30e+04	2.30e+04
Ammonium Hydroxide	0.00e+00	1.60e+03	1.60e+03

1. Reflects emissions without fuel reprocessing operations. Taken from the site-wide Environmental Impact Statement (EIS) support data.

The isotopes which have a half life over one minute are listed in Table 7-2; however, only the gases would be expected to be released since the particulates would be removed in the off-gas flows.

Table 7-2. Radiological Decay Chain for U-233 Fuel		
Radiological Isotope	Decay Chain	Half Life
Tl-207	U-235	4.77 min
Tl-208	U-232	3.05 min
Tl-209	U-233	2.20 min
Pb-211	U-235	36.1 min
Bi-211	U-235	2.13 min
Po-211	U-235	36.1 min
Pb-212	U-232	10.64 hrs
Bi-212	U-232	60.55 min
Bi-213	U-233	45.65 min
Fr-2212	U-233	4.8 min

Table 7-2. Radiological Decay Chain for U-233 Fuel

Radiological Isotope	Decay Chain	Half Life
Fr-223	U-235	21.8 min
Ra-223	U-235	11.43 days
Ra-224	U-232	3.62 days
Ra-225	U-233	14.8 days
Ac-225	U-233	10 days
Ac-227	U-235	21.73 yrs
Th-227	U-235	18.7 days
Th-228	U-232	1.913 yrs
Th-229	U-233	7340 yrs
Th-231	U-235	25.5 hrs
Pa-231	U-235	3.276e+4 yrs
U-232	U-232	72 yrs
U-233	U-233	1.592e+5 yrs
U-235	U-235	7.038e+8 yrs

Table 7-3 provides a comparison of the radiological emissions baseline and the emissions during operation of a U-233 blending process based on 1993 maximum baseline emissions (no reprocessing) with addition of decay products released during spent fuel reprocessing and known to exist in the U-233 decay chain. This table provides a bounding emissions level for the facility during U-233 operations.

Table 7-3. Radiological Emissions Comparison (curies)

Radionuclide	ICPP Emissions Baseline ¹	Annual Emissions Blending to 12%	Annual Emissions Blending to <5%
C-14	3.40e-02	3.40e-02	3.40e-02
Co-57	2.89e-07	2.89e-07	2.89e-07
Co-60	1.71e-05	1.71e-05	1.71e-05
Cs-134	3.24e-04	3.24e-04	3.24e-04
Cs-137	2.78e-02	2.78e-02	2.78e-02

Table 7-3. Radiological Emissions Comparison (curies)			
Radionuclide	ICPP Emissions Baseline ¹	Annual Emissions Blending to 12%	Annual Emissions Blending to <5%
Eu-152	6.68e-06	6.68e-06	6.68e-06
Eu-154	3.66e-05	3.66e-05	3.66e-05
Eu-155	6.01e-07	6.01e-07	6.01e-07
H-3	4.30e+01	4.30e+01	4.30e+01
I-129	6.40e-02	6.40e-02	6.40e-02
Kr-85	1.00e+04	1.00e+04	1.00e+04
Pu-TOTAL	9.43e-09	9.43e-09	9.43e-09
Ru-106	3.75e-03	3.75e-03	3.75e-03
Sb-125	1.86e-04	1.86e-04	1.86e-04
Sr-90	3.60e-03	3.60e-03	3.60e-03
Tl-208	N/A	1.40e-07	1.40e-07
Pa-231	N/A	8.10e-06	8.10e-06

1. Reflects the maximum ICPP emissions without fuel reprocessing operations. Taken from the site-wide EIS support data.

7.1.2 Solid and Liquid Wastes

The wastes generated during operation of the blending process are discussed in detail in Chapter 4. Table 7-4 is provided to summarize the quantity estimates for the activity.

Table 7-4. Annual Waste Volumes Generated from U-233 Operations (base case - thorium removal)				
	Category	Quantity Generated		Post Treatment Quantity
		Solids (M ³)	Liquids (Liters)	Solids (M ³)
12% Blending	Low Level Waste	61.2	N/A	40
	Mixed Low Level Waste	0	0	0
	High Level Waste ²	N/A	396,101	39
	Nonhazardous Waste ³	N/A	N/A	N/A

**Table 7-4. Annual Waste Volumes Generated from U-233 Operations
(base case - thorium removal)**

	Category	Quantity Generated		Post Treatment Quantity
		Solids (M ³)	Liquids (Liters)	Solids (M ³)
12% Blending (cont.)	Recyclable Waste	N/A	169,988	8.5
	Sanitary Waste ³	N/A	N/A	N/A
<5% Blending	Low Level Waste	66.2	N/A	45
	Mixed Low Level Waste	0	0	0
	High Level Waste ²	N/A	396,101	39
	Nonhazardous Waste ³	N/A	N/A	N/A
	Recyclable Waste	N/A	169,988	8.5
	Sanitary Waste ³	N/A	N/A	N/A

1. Data in this table reflects waste generation over and above the baseline wastes from system maintenance and related activities.. Some waste categories would be reduced by performing the processing, because it would stop intensive waste generation maintenance and repair activities.
2. Although high level waste by definition is NOT possible from the U-233 inputs, the waste system which may be required to handle the waste is a high level waste system which will contaminate U-233 process wastes with other high level waste resulting in a high level waste product. Alternatives to bypass this waste mixing situation would be explored in detail prior to generation of any additional high level waste from U-233 processing. If such alternatives are successful, the high level waste quantities would be moved to the mixed low level waste column.
3. Sanitary waste and nonhazardous waste are not expected to change significantly as a result of processing.

7.2 Wastes and Emissions Generated During Construction/Modification

Because of the uncertainties related to re-permitting and re-start of existing facilities which will involve determinations by Federal and State regulatory agencies, and the Judicial system, the construction/modification emissions and waste scope cannot be determined at this time.

8.0 DESIGN PROCESS FOR ACCIDENT MITIGATION

8.1 Operational, Design, and Beyond Design Basis Bounding Accidents

A review of approved Safety Analyses for all ICPP facilities resulted in the following list of postulated radiological accidents. The accidents presented below are not all of the accidents presented in various SAR Sections, but they are the largest contributors to risk. The accidents described here are upper-limit cases whose consequences could be much lower when processing the U-233, since the fission product inventory would not be present.

- 1) NWCF explosion
- 2) Hazardous Chemical and Radioactive Waste Facility fire
- 3) CPP-601 TBP extraction fire
- 4) CPP-601 K-Cell solvent fire/explosion
- 5) APS filter failure (ICPP Main Stack)
- 6) FAST criticality
- 7) CPP-601 L-Cell criticality
- 8) Sixth and Seventh CSSFs release

8.1.1 NWCF Explosion

The NWCF processes high-level liquid waste into a solid form called calcine by burning a variety of fuels and injecting the high-level waste. The worst-case accident is postulated to be a calciner vessel explosion when it is filled with the optimum explosive mixture of kerosene and oxygen. The most likely scenario postulated is the violent failure of the calciner vessel causing a blast loading that results in a breach in the integrity of the maintenance area.

The release is postulated to occur at the point where the building seals are expected to fail, 15.2 m (50.0 ft) above the ground outside of the NWCF. The release, however, was modeled as a ground-level release. The explosion results in the release of 1 L (0.26 gal) or 0.2 kg (0.4 lb) of calcine. The source term was taken as that of "normal" waste, with an additional concentration factor of 7 applied. One hundred percent of the release is assumed to become airborne instantaneously. Individuals involved in the accident are assumed to remain in the plume for the duration of the accident and have a fast breathing rate of $3.33 \text{ E-4 m}^3/\text{s}$ ($1.2 \text{ E-2 ft}^3/\text{s}$). A windspeed of 0.5 m/s (1.6 ft/s) was used for distances less than 2 km (1.2 mi) and a windspeed of 2 m/s (6.6 ft/s) for greater distances. Class F meteorological conditions prevail during the accident. The building wake effects used were 15.2 m (50.0 ft) high and 44.2 m (145.0 ft) wide. Ingestion was included for off-site individuals only. Direct radiation from the plume passing was accounted for by using the finite plume cloud gamma model.

Effective Dose Equivalents (EDEs) were calculated for ICPP individuals at distances of 100 m (328 ft) and 200 m (656 ft), for INEL individuals at TRA and CFA, and for off-site individuals are about 3.7 E+1 , 3.3 E+1 , 9.0 E-1 , 4.5 E-1 , and 1.2 rem, respectively.

8.1.2 Hazardous Chemical and Radioactive Waste Facility Fire

The Hazardous Chemical and Radioactive Waste Facility was designed and constructed to satisfy RCRA requirements. This facility is intended to provide a centralized location for the management and handling of various incoming and outgoing radioactive and hazardous materials.

A fire is assumed to start in a 4 x 4 x 8-ft wooden radioactive waste box located in the radioactive mixed waste storage area. The release occurs as a result of a fire consuming the box. The box contains 3 Ci of material. One hundred percent of the activity is assumed to be released outside the facility. On-site workers are assumed to have a fast breathing rate. A windspeed of 0.5 m/s (1.6 ft/s) is used for distances of less than 2 km (1.2 mi), and a windspeed of 2 m/s (6.6 ft/s) is used for greater distances and under Class F meteorological conditions. Exposure times are assumed to be for the duration of the passing plume, and no credit for evacuation or building wake effects is taken. The release is also assumed to be unfiltered.

EDEs were calculated based on the above assumptions. EDEs for ICPP individuals at distances of 100 m (328.1 ft) and 200 m (656.2 ft), for INEL individuals at TRA and CFA, and for off-site individuals are about 2.2 E+1, 6.8, 2.9 E-2, 1.3 E-2 and 3.6 E-2 rem, respectively.

8.1.3 CPP-601 TBP Extraction Fire

The TBP Extraction Process is housed in G- and H-Cells in CPP-601. This system was used to separate fission products from uranium, and, for this process, would be used to remove the U-233 from other materials. A fire in the Preventive Maintenance (PM) area of CPP-601 is assumed to consume a spill of organic material, containing uranium and fission products, and is assumed to be a worst-case accident. For this project, only the decay products, rather than fission products, would be present, reducing the consequences of such an accident.

The spilled organic material is 100 L (26.4 gal) of 1.37 g/L (1.1 E-2 lb/gal) of product containing TBP in hydrocarbon diluent and is assumed to result in a ground-level release. Individuals involved in the accident are assumed to remain in the plume for the duration of the accident and have a fast breathing rate. A windspeed of 0.5 m/s (1.6 ft/s) is used for distances of less than 2 km (1.2 mi), and a windspeed of 2 m/s (6.6 ft/s) is used for greater distances. Both windspeeds assume Class F meteorological conditions. Exposure time for ICPP workers is assumed to be five minutes, with all material being released from the PM area during that period of time. No credit is taken for evacuation. Building wake effects were used with the building height of 8.38 m (27.5 ft), width of 25.3 m (83.0 ft), and no HEPA filtration.

EDEs were calculated based on the above assumptions. EDEs for ICPP individuals at distances of 100 m (328.1 ft) and 200 m (656.2 ft), for INEL individuals at TRA and CFA, and for off-site individuals are about 5.4, 4.1, 5.4 E-2, 2.6 E-2, and 1.0 E-2 rem, respectively.

8.1.4 CPP-601 K-Cell Solvent Fire or Explosion

The TBP recovery and waste solvent disposal process is housed in H-Cell and K-Cell in CPP-601. This system had been operated in conjunction with the first-cycle extraction system and is intended to recycle as much TBP as possible. However, the solvents used can create a fire or explosion hazard. A fire is considered more likely than an explosion.

An organic spill is assumed to occur on the floor of K-Cell, and the release occurs as a result of an explosion of the material in the cell. Material is then ejected into the PM area at ground level (ground-level release). The spill is 2185 L (577.3 gal) from the K-109 vessel with the radionuclide content derived from samples taken in past runs. Seventy-five percent of the material is released to the PM area through the cell hatches, and the remainder vents through the APS system where it is filtered prior to final release. Individuals involved in the accident remain in the plume for the accident duration and have a fast breathing rate. A windspeed of 0.5 m/s (1.6 ft/s) is used for distances less than 2 km (1.2 mi), and a windspeed of 2 m/s (6.6 ft/s) is used for greater distances. Both windspeeds are under Class F meteorological conditions. Exposure time is assumed to be five minutes. All material released to the PM area also is released to the environment during the exposure time. No credit was taken for plateout or evacuation. Building wake effects used were 8.38 m (27.5 ft) in height and 25.3 m (83.0 ft) in width. The PM release is unfiltered. Material released to the APS is negligible and results in negligible consequences.

EDEs were calculated based on the above assumptions. EDEs for ICPP individuals at distances of 100 m (328.1 ft) and 200 m (656.2 ft), for INEL individuals at TRA and CFA, and for off-site individuals are about 2.7, 2.1, 2.7 E-2, 1.3 E-2, and 5.0 E-3 rem, respectively.

8.1.5 Main Stack APS Filter Failure

The APS consists of 104 HEPA filters that are used to filter ventilation air from ICPP activities. Each filter has a TS/S-exposure-rate limit of 450 mR/h at 1 ft and is controlled to 20 mR/h as defined in procedures. All 104 filters are assumed to release all entrained materials (at the 20 mR/h level) in three to five minutes. Dispersion parameters were based on a six-year average of the calculated annual total integrated concentrations for the Main Stack.

EDEs were calculated based on the above assumptions. EDEs for ICPP individuals at distances of 100 m (328.1 ft) and 200 m (656.2 ft), for INEL individuals at TRA and CFA, and for off-site individuals are about 2.3, 1.1 E-1, 7.1 E-2 and 3.4 E-2 rem, respectively.

8.1.6 FAST and FSA Criticality

The FAST facility is a combination of fuel storage and a relatively new headend dissolution process that had processed a variety of irradiated, highly enriched uranium fuels. A criticality is postulated to occur after a cask is dropped during loading, unloading, or movement of fuel.

The solids criticality is assumed to consist of $3.0 \text{ E}+19$ fissions lasting for a period of 966 seconds. The exhaust fans are assumed to be running so that the fission products released from the two ruptured fuel handling units (FHUs) are swept from the Fuel Storage Area (FSA). Individuals involved in the accident are assumed to remain at the exposure point for the duration of the accident (i.e., two hours). The worst-case release is assumed to be with a failed stack and filters. Individuals were assumed to have a breathing rate of $3.47 \text{ E}-4 \text{ m}^3/\text{s}$ ($1.2 \text{ E}-2 \text{ ft}^3/\text{s}$). A windspeed of 2 m/s (6.6 ft/s) during Class F meteorological conditions is assumed. Individuals remaining at distances of 175 m (574 ft) and 400 m (1312 ft) are assumed to have the same χ/Q .

EDEs were calculated based on the above assumptions. EDEs for on-site individuals at distances of 175 m (574 ft) or 400 m (1312 ft) and individuals located at the site boundary $1.35 \text{ E}+4 \text{ m}$ ($4.4 \text{ E}+4 \text{ ft}$) are about 1.1 and 4.3 E-2 rem, respectively.

8.1.7 CPP-601 L-Cell Criticality

The uranium rework system is housed in L-Cell of CPP-601. This system had collected various streams containing uranium and chemically adjusted the solution for reintroduction into the appropriate portion of the extraction system. The worst-case accident is postulated to be a criticality in a rework tank.

The criticality is assumed to be in a liquid lasting 16 minutes with $1.0 \text{ E}+19$ fissions. The tank containment is breached by the nuclear excursions. All plant ventilation systems are functioning normally and remain running for the duration of the accident. The criticality is assumed to last long enough to allow for the buildup of some of the longer-lived particulate daughter nuclides that are biologically significant but are not direct short-lived fission products. A total of 415 kg of uranium is assumed to be involved. Individuals located downwind from the stack are exposed during Class F fumigation meteorological conditions with a windspeed of 2 m/s (6.6 ft/s) and are assumed to have a breathing rate of $3.47 \text{ E}-4 \text{ m}^3/\text{s}$ ($1.2 \text{ E}-2 \text{ ft}^3/\text{s}$).

EDEs were calculated based on the above assumptions. EDEs for an ICPP individual located 100 m (328 ft) downwind of the release and an off-site individual $1.35 \text{ E}+4 \text{ m}$ ($4.4 \text{ E}+4 \text{ ft}$) away are about $2.3 \text{ E}-1$ and $2.6 \text{ E}-3 \text{ rem}$, respectively.

8.1.8 Sixth and Seventh CSSFs

The Sixth and Seventh CSSFs store the calcined solid product from the NWCF. The first five CSSFs are full and are, therefore, bound by the accident scenario presented here. The worst-case accident is postulated to be a break or hole in a fill line in the CSSF.

The calcined solids are assumed to be released and undetected for 16 hours under natural convection cooling because of failure of the vault cooling air radioactivity monitor and failure of plant personnel to properly check the monitor. After 16 hours, the vault cooling air is manually diverted through the vault cooling air HEPA-filtered discharge system. NWCF then begins to stop the flow of calcine to the bin set. This shutdown requires 40 hours to complete.

The following conditions also are assumed for the above accident. The transport rate for solids in the transport line is 34.7 g/s (4.6 lb/min). The fraction released into the vault from the leak in the fill line is assumed to be 1% with 1% of that becoming airborne and exiting the vault into the vault cooling air. No filtration is used for the first 16 hours of the accident. The total stack height is 19.3 m (63.3 ft), and the plume injection velocity is 1.5 m/s (5 ft/s) from a 1.2 m (4 ft) inside diameter stack. The next 40 hours of release are HEPA-filtered and exhausted from a stack height of 25.4 m (83.3 ft), injection velocity of 6.6 m/s (21.7 ft/s), and through a stack diameter of 1.4 m (4.6 ft). Individuals involved with the accident are assumed to have a normal breathing rate of $2.7 \text{ E}-4 \text{ m}^3/\text{s}$ ($9.5 \text{ E}-3 \text{ ft}^3/\text{s}$).

Workers on-site are assumed to be exposed for eight of the first 16 hours of the accident and then 16 of the last 40 hours. The ingestion pathway was not used in the exposure calculations. The source term chosen is for a fluorinel calcine waste blend. Meteorological conditions during the accident were chosen such that they would represent 95% dispersion factors. The 95% dispersion factors were related to the atmospheric conditions which resulted in calculation of the 95% value of source-strength normalized concentration (χ/Q). Particles greater than 10 microns are assumed to settle out, but only for the individual off-site. Particle size and settling rate of released material were accounted for only during the unfiltered phase.

EDEs were calculated based on the above assumptions. EDEs for on-site individuals located at 100 m (328 ft), 200 m (656 ft), and at TRA and CFA and for off-site individuals are about 5.4 E-3 , 2.8 E-3 , 2.3 E-3 , 1.9 E-3 , and 3.4 E-3 rem , respectively.

9.0 TRANSPORTATION

Three sources of U-233 exist for processing, one at the Oak Ridge National Laboratory, another at the INEL ICPP, and a third at the INEL RWMC. It is anticipated that the material located at the INEL RWMC will be transferred to the ICPP prior to processing. Therefore, the main transportation issues are the movement of LWBR fuel at CPP-749 storage facility to CPP-666 processing facility, the movement of the ORNL material to the INEL, and the movement of processed material to storage.

9.1 Intraste Transportation

The unirradiated INEL LWBR presently is stored in underground silos at the CPP-749 storage facility. This material would be moved to the CPP-666 processing facility as described earlier where it would be dissolved and processed for blend down. Fuel stored at CPP-749 is generally transferred using the Peach Bottom Cask. However, the unirradiated LWBR fuel was originally placed in CPP-749 using a different container and is not specifically packaged for easy retrieval into the Peach Bottom cask. Modification to the cask, construction of a new handling tool, and new procedures will be required. With these modifications the Peach Bottom cask would be used to remove the fuel from the storage location and transfer it to CPP-666.

After processing is completed, the uranium oxide would be returned to this facility as described in Section 10. The Peach Bottom cask would again be used for this purpose. Scheduling of this cask will be required, depending on when processing occurs.

9.2 Intersite Transport

Material from ORNL would be shipped to the INEL if processing is done here. Approximately 423 kg of U-233 in 1380 kgs of total uranium are stored at ORNL. This material is typically stored in 3- or 4-in. diameter cans that are from 6 to 24-in. in length. Over 1000 cans must be shipped. All intersite shipping from ORNL to the INEL will require full security measures due to the quantity of weapons useable material involved. The material is present in various physical and chemical forms, including oxide powder, oxide monolithic form, metal, and miscellaneous forms. The exact cask to be used will depend on the timing of the project and the configuration of the ORNL material as well as cask availability at the time of transfer. Various casks might be used for these transfers, and they are listed in Table 9-1. The dimensions, weights, present availabilities, and certifications are shown in the table. Obviously, more detailed studies will be required before a cask and shipping configuration can be determined. Criticality calculations will be required to determine the number and configuration of canisters that can be used for each cask. The availability of any or all of these casks will require coordination with SNF shipments expected between various DOE sites throughout the country.

Shipments to the ICPP can be received from either highway or railroad systems. U. S. Highways 20 and 26 cross the southern portion of the site. The Union Pacific Railroad has a branch line that services the southern portion of the INEL. A DOE-owned railroad track passes north past the east side of the ICPP. To place the shipment and processing of the material to the ICPP in a better perspective, Table 9-2 shows shipments that would be required.

Table 9-1. Cask Types That May be Used For Material Transfers

Package Type	Certificate Expiration	Gross Weight (lbs)	External Length (in)	External Diameter (in)	Transport Mode	Payload Weight (lbs)	Internal Length (inches)	Internal Diameter (inches)	Internal Compartments	Maximum Heat Load (kW)	Availability	Number
GE T-2	5/31/98	18,400	113	36	Legal Weight Truck	160	99.69	6.06	1	0.29	N/A	N/A
GE-100	2/28/98	4,800	26.88	20.25	Legal Weight Truck	N/A	10	7.625	1	0.4	2	3
BMI-1	5/31/95*	23,660	73.37	33.37	Legal Weight Truck	1800	54	15.5	1	N/A	1	1
FSV-1	4/30/96	47,600	208	31	Legal Weight Truck	4430	187.6	17.7	1	4.1	2	2
NLI-1/2	3/31/96	49,250	178	75	Legal Weight Truck	3000	237	12.63	1	0.75	5	5
NLI-6502	10/31/93*	45,300	130	33.5	Legal Weight Truck	N/A	120	11.5	1	4.1	N/A	N/A
T-3	3/31/96	38,200	213	52	Legal Weight Truck	700	147	7.98	1	1.4	1	3
NAC-1	9/30/94*	49,000	214	50	Legal Weight Truck	3700	178	13.5	1	0.75	6	6
CNS 1-13G	12/31/97	25,500	81.8	68	Legal Weight Truck	Not Restricted	54	26.5	1	0.6	1	1
NRBK-41	1/31/96	9,000	40	27.16	Legal Weight Truck	N/A	16	5	1	0.264	N/A	N/A
NAC-LWT	12/31/94*	51,200	232	65	Legal Weight Truck	4000	178	13.4	1	2.5	4	4
GE Model 2000	10/31/94*	33,550	131.5	72	Legal Weight Truck	5450	54	26.5	1	0.6	2	3
TN-FSV	5/31/99	47,000	247	78	Legal Weight Truck	5000	199	18	1	0.36	2	2
TN-8	5/31/96	79,380	217.2	67.64	Overweight Truck	4850	168.5	9.06	3	35.5	2	2
TN-9	5/31/96	79,380	226.61	67.64	Overweight Truck	4652.5	177.95	5.91	7	24.4	2	2
M-130	5/31/97	228,000	158	84	Rail	N/A	130	29	1	9815.5	0	27
IF-300	5/31/95*	185,000	210	64	Rail	21000	180.25	37.5	1	11.72	2	4
NLI-10/24	3/31/97	194,000	204.5	96	Rail	34100	179.5	45	1	70	2	2
125-B	5/31/96	181,500	279.5	120	Rail	20580	183.25	14.5	7	0.7	3	3
TN-BRP	6/30/94*	215,000	244.5	131	Rail	41250	171	64	1	3.1	1	1
TN-REG	5/31/95*	225,000	234	131	Rail	50500	163.25	71.75	1	2.7	1	1
M-160	7/31/97	237,000	199	79	Rail	N/A	160	21	1	6.2409	0	1
M-140	10/31/96	375,000	194	126	Rail	N/A	146	70	1	18.25	0	24

* Best available information on Certification Expiration Date. Certification renewals are expected for all casks.

Table 9-2. Transportation Summary					
		Blend to 12% and Store Short-term Before Disposal		Blend to <5% and Store Short-term Before Disposal	
		Input Material	Output Material	Input Material	Output Material
Transported Material	Type	LWBR fuel from INEL and Miscellaneous U-233 sources from ORNL	UO ₃	LWBR fuel from INEL and Miscellaneous U-233 sources from ORNL	UO ₃
	Physical Form	fuel, powder, monolith, and metal	powder	fuel, powder, monolith, and metal	powder
	Chemical Composition	Oxide, Fluoride, and elemental	UO ₃	Oxide, Fluoride, and elemental	UO ₃
Packaging	Type	See Table 9-1	Peach Bottom or other	See Table 9-1	Peach Bottom or other
	Certified By		Local		Local
	Identifier		CA-SF-006		CA-SF-006
	Container Weight		58,000 lb		58,000 lb
	Material Weight	17,300 kg U and Th	16,000-33,000 kg	17,300 kg U and Th	38,000-56,000 kg
	Isotopic Content	U-233	U-233 and U-238	U-233	U-233 and U-238
Average Shipping Volume	Quantity per Year	Various	720 cans	Various	720 cans
	Average Number of Packages	Various	720 cans	Various	720 cans
	Estimated Number of Packages Shipped Over Life	1,050	1,230 (originally had 2,295, minus 1,065=1,230)	1,050	1,230 (originally had 3,820, minus 2,590=1,230)
	Average Number of Packages per Shipment	to be determined (TBD) based on criticality and other considerations	TBD	TBD	TBD
	Number of Shipments per Year	TBD	TBD	TBD	TBD
	Number of Shipments Over Life	TBD	TBD	TBD	TBD
Routing	Destination Facility Type	INEL, FAST	CPP-749	INEL, FAST	CPP-749

9.2.1 Input Material Streams

Table 9.2 shows the shipments of materials to be made and summarizes how the material will be stored following processing. Although the INEL storage of oxide products would be convenient, materials management and political questions are involved. The material is anticipated to be stored at the INEL following processing until a final disposal site or customer is selected.

Depleted uranium will also be required for blending with the U-233. This material is expected to be provided in the form of a UO_2 powder. Process calculations show that about 13,300 kg of UO_2 will be required to blend the U-233 to less than 12% enrichment and 35,400 kg will be required to blend to <5%. The UO_2 is assumed to be shipped to the ICPP in 6-M containers, each containing 16 kg of UO_2 . This would require receipt of 950 containers to blend to 12% and 2530 containers to blend to <5%.

9.2.2 Output Material Streams

Since the processed uranium is assumed to be stored at the ICPP for a period of time until the final disposal location is available, no output, other than from the processing plant to the storage site at the ICPP, is shown. The final product (see Section 10) will consist of UO_3 stored in small cans in the quantities listed in Tables 4.1 and 4.2.

10.0 FINAL FORM CHARACTERISTICS

The final process step in disposing and blend down of the U-233 is the denitrating step where the aqueous purified product is converted to a solid oxide form. This step has been completed routinely with recycled uranium at the ICPP in the past, and existing facilities will be used for this process. The solid product, UO_3 or $\text{UO}_3 + \text{ThO}_2$, is then packaged in cans, and for this process, will be returned to the CPP-749 underground storage area where the unirradiated LWBR fuel is currently being stored. The product will be stored here until a final disposal location is identified and the material is shipped to that site.

The aqueous uranyl nitrate solution is sprayed into a hot fluidized bed of UO_3 product particles. The water is evaporated, and the uranyl nitrate is converted to uranium oxide. The newly formed UO_3 either coats the existing particles, thus increasing their size, or becomes nuclei for new particles. As the bed volume becomes larger, it eventually overflows into the product overflow line. The size of the particles in the denitrator may be controlled by using a jet grinder mounted in the wall of the vessel, by adjusting the ratio of atomizing air rate to the liquid feed rate, or by adjusting the bed temperature.

The uranium oxide product is transferred to the glove box where samples are taken by passing the product through a sample splitter. The product then is poured into a 2.85 liter container having a diameter of 5.5 inches and a height of 7.75 inches. This inner container is then sealed with a press-fit lid and bagged out of the glove box. It is then transferred to a shipping container for return to its storage location. Since the purpose of this process is to simply blend down the existing uranium, the control of the purity of the product is not as important as it is for preparing product which will be used for fuel fabrication. However, it is anticipated that essentially all impurities of decay products will be removed during the extraction process, leaving a final product that is only uranium. If it is determined that the U-233 will not be separated from the thorium, the thorium will be present in the product stream from the extraction system and will be converted to an oxide along with the uranium and will be contained in the denitrator oxide product. If this occurs there will be approximately 3,316 kg of uranium oxide product mixed with 16,051 kg of thorium oxide. The isotopic content of the product will change with time as the U-232 decays, and the percent of U-233 will depend on the blend down being considered (less than 12% or less than 5%), with the majority of the remaining uranium being present as the U-238 isotope.

The denitration process generates little waste. Oxides of nitrogen are released during the denitration process. These are exhausted to the ICPP Vessel Off-Gas System where they are filtered and discharged through the main ICPP stack. The off gas also is filtered at the denitrator to remove uranium oxide fines, and the gas stream flows through a condenser before entering the Vessel Off-Gas System. Condensate and acid flush solutions are recycled, and solid wastes are leached of uranium in salvage operations. Other contaminated solid wastes are disposed of to the RWMC. The UO_3 product then is stored until ultimately disposed and would not be classified as a waste at this time.

The radioactivity in the uranium product will depend on the timing between extraction and denitration. To keep the radiation level in the product as low as possible during processing times, the denitration step will occur concurrently or as quickly as possible following the second cycle extraction step. This will minimize the exposure to personnel from the build up of the decay daughters that occurs over time.

The product cans from the denitration step will be placed inside baskets and canisters for movement to the storage location. Two types of vaults are available for storage, the 12 inch diameter vaults which presently hold the LWBR fuel and 30 inch diameter vaults which also are available. The 12 inch vaults will be used first, and if more space is necessary, the larger vaults will be used. The vaults are about 20 feet deep, and the product cans will be placed inside steel baskets which will be used to hold the material in the vaults.

For radiation protection reasons, the uranium product will be moved as quickly as possible to the underground storage vaults. They will be stored at that location until the final disposal site is available. The timing of this movement will depend on the schedule for the preparation of the final storage location.

REFERENCES

1. DOE Order 6430.1A, "General Design Criteria," April 1989. (Now Cancelled)
2. DOE Order 5480.28, "Natural Phenomena Hazards Mitigation," January 15.
3. DOE Order 5480.11, "Radiation Protection for Occupational Workers," June 1992.
4. International Conference of Building Officials, Uniform Building Code, 1991.
5. DOE-ID, "DOE-ID Architectural Engineering Standards," September 21, 1992.
6. Secretary of Energy Notices, SEN-35-91, "Nuclear Safety Policy," September 9, 1991.
7. LLNL, UCRL-15910, "Design and Evaluation Guidelines for DOE Facilities Subjected to Natural Phenomena Hazards," June 1990.
8. LLNL, UCRL-53526, "Natural Phenomena Hazards Modeling Project: Extreme Wind/Tornado Hazard Models for Department of Energy Sites," Rev. 1.
9. ANSI, ASME/ANSI B31.3, "Chemical Plant and Petroleum Refinery Piping."
10. DOE-ID, "Welding Procedure Specification Manual."
11. American Welding Society, "Welding Standards," 1969.
12. American Institute of Steel Construction, "Specifications for the Design, Fabrication, and Erection of Structural Steel for Buildings."
13. American Concrete Institute, ACI-318, "Building Code Requirements for Reinforced Concrete."
14. 29 CFR 1910, "Occupational Safety and Health Standards."
15. ASME, "Boiler and Pressure Vessel Code," Section 8, "Unfired Pressure Vessels."
16. DOE Order 5400.1, "General Environmental Protection Program," June 1990.
17. 10 CFR 20, "Standards for Protection Against Radiation."
18. 40 CFR 190, "Environmental Radiation Protection Standards for Nuclear Power Operations."
19. 40 CFR 191, "Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes."
20. INEL Radiological Control Manual (IRCM), December 1992.
21. Rules and Regulations for the Control of Air Pollution in Idaho, 1991.
22. DOE Order 5480.7A, "Fire Protection," February 1993.

23. Western Fire Chiefs Association and International Conference of Building Officials, "Uniform Fire Code," May 1991.
24. NFPA, "National Fire Codes, A Compilation of NFPA Codes, Standards, Recommended Practices and Guides," Volumes 1 to 11, 1992.
25. DOE-ID, ID-12044, "Operational Safety Design Criteria Manual," Chapter I, "Fire Protection," March 1985.
26. Factory Mutual, "Approved Guide," 1991.
27. Factory Mutual, "Loss Prevention Data Sheets," January 1992.
28. M. R. Niccum, "Flooding Potential at the ICPP," April 1973.
29. DOE Order 4700.1, "Project Management System," Chapter V, Part A, (2)(p), "Decontamination and Decommissioning," dated June 2, 1992.
30. DOE Order 6430.1A, "General Design Criteria" (Now Cancelled), 1300-11, "Decontamination and Decommissioning," dated April 6, 1989.
31. ICPP, Operations Department Procedure, F.47, "Contaminated Surplus Facility Surveillance and Maintenance Program," (draft will replace the existing procedure E.63, April 1992).
32. DOE Order 5480.24, "Nuclear Criticality Safety."
33. ICPP, WSOP WE-33, "Transition, Turnover, Decontamination, and Decommissioning of ICPP Surplus Contaminated Facilities," dated October 8, 1992.
34. ICPP, Operations Department Procedure F.52, "Preparation of Decontamination and Decommissioning Plan," (draft).
35. ICPP, Operations Department Procedure F.53, "Preparation of Decontamination and Decommissioning Report," (draft).
36. DOE Order 5480.1B, Change 004, "Environmental, Safety, and Health Protection for Department of Energy Operations," March 27, 1990. (Now Cancelled).
37. DOE Order 5480.22, "Technical Safety Requirements," February 25, 1992.
38. DOE Order 5480.23, "Nuclear Safety Analysis Reports," April 1992.
39. DOE Order 6430.1A, "General Design Criteria," April 1989.
40. DOE Order 5480.4, "Environmental Protection, Safety, & Health Protection Standards," January 1993.
41. DOE Order 5480.10, "Contractor Industrial Hygiene Program," June 1985.
42. DOE Order 5820.2A, "Radioactive Waste Management," dated September 26, 1988.

43. DOE Order 5484.1, "Environmental Protection, Safety, and Health Protection Information Reporting Requirements," Chapter III, "Effluent and Environmental Monitoring Program Requirements," dated November 6, 1987.
44. DOE-ID Order 5480.2A, "Hazardous and Radioactive Mixed Waste Management," dated January 3, 1984.
45. DOE-ID Order 5484.1, "Environmental Protection, Safety, and Health Protection Information Reporting Requirements," Chapter III, "Effluent and Environmental Monitoring Program Requirements," dated August 26, 1982.
46. "Code of Federal Regulations," Title 40, Environmental Protection Agency, Part 61, "Emission Standards for Hazardous Air Pollutants." (Hereafter referred to as 40 CFR)
47. 40 CFR, Environmental Protection Agency, Part 241, "Guidelines for Land Disposal of Solid Wastes."
48. 40 CFR 260–270, Resource Conservation and Recovery Act.
49. DOE, IDO 12081, "The Idaho National Engineering Laboratory Site Environmental Report for Calendar Year 198X (1980–1989)," 1980–1989.
50. DOE, "INEL Site Environmental Monitoring Data for the Fourth Quarter, 1983," 1984.
51. J. T. Barraclough and R. G. Jensen, IDO-22055, "Hydrology of the National Reactor Testing Station Idaho, 1971–1973," 1976.
52. J. R. Pittman, R. G. Jensen, and R. D. Fischer, DOE/ID-22078, "Hydrologic Conditions at the Idaho National Engineering Laboratory, 1982–1985," December 1988.
53. WINCO, "ICPP Effluent Monitoring Report 4th Quarter CY-1983," WINCO 1002, 1984.
54. DOE, IDO-10057, "Industrial Waste Management Information for 1989 and Record-to-Date," 1990.
55. G. E. Start and L. L. Wendell, "Regional Effluent Dispersion Calculations Considering Spatial and Temporal Meteorological Variations," Technical Memorandum ERL ARL-44, National Oceanic and Atmospheric Administration, 1974.
56. 40 CFR 50, "Protection of Environment," "National Primary and Secondary Ambient Air Quality Standards," U.S. Government Printing Office, Washington D.C., July 1980.
57. International Conference of Building Officials, "Uniform Building Code," 1991.
58. ICPP SAR, Part 1, "General Safety Analysis," Chapter 1, "Site Characteristics," Rev. 0, March 1995.
59. U.S. DOE Order 5480.7A, "Fire Protection."
60. DOE-ID Supplemental Directive 5480.24, "Nuclear Criticality Safety."

61. ANSI/ANS-8.1-1983; R1988, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."
62. ANSI N16.5/ANS-8.7-1975; R1987, "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials."
63. ICPP Criticality Safety Requirements Manual.
64. DOE Order 5820.2A, "Radioactive Waste Management," dated September 26, 1988.
65. DOE Order 5480.11, "Requirements for Radiation Protection," dated December 21, 1988.
66. DOE Order 5484.1, "Environmental Protection, Safety, and Health Protection Information Reporting Requirements," Chapter III, "Effluent and Environmental Monitoring Program Requirements," dated November 6, 1987.
67. 40 CFR 61, "Emission Standards for Hazardous Air Pollutants."
68. 40 CFR, Environmental Protection Agency, Part 241, "Guidelines for Land Disposal of Solid Wastes."
69. 40 CFR 260-270, Resource Conservation and Recovery Act.
70. ICPP, WSOP WE-3, "Calibration Program," October 14, 1991.
71. DOE-ID Supplemental Directive 5480.24, "Nuclear Criticality Safety."
72. ANSI/ANS-8.1-1983; R1988, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."
73. ANSI N16.5/ANS-8.7-1975; R1987, "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials."
74. DOE-ID Supplemental Directive 5480.5A, "Nuclear Criticality Safety." (Draft).
75. ICPP Criticality Safety Requirements Manual.
76. B.T. Richards, "ICPP Water Inventory Study Progress Report," WINCO-1184, May 1993.
77. "Radioactive Airborne Effluent Plant Safety Document & Technical Specifications/Standards/Requirements," WIN 107-8.6, Rev. 0, Final Draft, March 1995.