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Fast Flux Test Facility Final Safety Analysis Report Amendment 78

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FFTF Transition Project Office

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FINAL SAFETY ANALYSIS REPORT, WHC-TI-75002,
AMENDMENT 78

Date: August 27, 1999

Attached is the Final Safety Analysis Report (FSAR) Amendment 78 for incorporation into the Fast Flux Test Facility (FFTF) FSAR set assigned to you. This page change incorporates Engineering Change Notices approved by the U.S. Department of Energy, Richland Operations Office. Significant changes are:

CHAPTER 7, INSTRUMENTATION AND CONTROL SYSTEMS

- Based on the requirement that the reactor vessel be defueled, the systems required for control rod movement that were previously disabled are now returned to service for control rod system testing.

CHAPTER 9, AUXILIARY SYSTEMS

- Due to a break in the cooling tower emergency makeup line, the line has been abandoned. Hose connections are provided for emergency makeup.
- Updates are made to the current communications systems in the FFTF and the associated technology and OSHA requirements.

CHAPTER 11, REACTOR REFUELING SYSTEM

- The loading limit for Core Component Containers (CCCs) is removed from Chapter 11 and loading restrictions in Appendix H are referenced.
- An evaluation of the storage of irradiated and unirradiated non-fuel hardware in the Test Assembly Conditioning Station is provided in preparation for turnover of the FFTF to the Environmental Restoration Contractor.
- The use of the grid structures in the Interim Examination and Maintenance (IEM) Cell is clarified with respect to the position of the Ident 35 cart.
- The purpose of the impact limiters for the Solid Waste Cask and Cask Loading Station is clarified.
- The fuel storage limits for CCC/Interim Storage Casks (ISCs) are changed based on new analyses.

CHAPTER 13, CONDUCT OF OPERATIONS

- The Emergency Response section of Chapter 13 is updated for the communications systems currently in place.

CHAPTER 15, SAFETY ANALYSIS

- The purpose of the impact limiter for Cask Loading Station drop accidents is clarified.

CHAPTER 17, FFTF TECHNICAL SPECIFICATIONS

- A new Limiting Condition for Operation is provided which prohibits fuel in the reactor vessel. Based on that, other changes are made to permit exercising the control rods and other head mounted components.
- The Administrative Controls section is updated, mostly to reflect organizational changes.

CHAPTER 18, FFTF ENVIRONMENTAL SPECIFICATIONS

- The 400 Area Process Sewer sampling requirements are changed to conform to State Waste Discharge Permit ST 4501.
- The requirements for continuous airborne emissions measurement and monitoring are deleted and replaced with periodic confirmatory measurements.

CHAPTER 20, FFTF CRITICALITY SPECIFICATIONS

- The IEM Cell Criticality Prevention Specification CPS 405-2 is revised to change the loading limits for CCCs and allow operation of the Sodium Removal System Ion Exchanger with the Containment Isolation Valves open when the floor grids are in place.

APPENDIX G, TRANSITION OF THE FFTF FROM OPERATING TO DEFUELED (SHUTDOWN) STATUS

- A new section is added describing Technical Specification changes pertaining to standby or potential restart.

APPENDIX H, SECTION H.3.3, FFTF SPENT FUEL STORAGE SYSTEM

- A list of additional fuel assemblies authorized for fuel storage is provided.
- The requirement that fuel pin loading during drop conditions be limited to <440 g's is removed.
- An editorial correction is made to clarify that the post-accident radiation limit of 1R/hr at 1 meter applies after design basis accidents, not hypothetical accidents.
- The fuel storage limits for CCC/ISCs are changed, consistent with the change to Chapter 11.
- The allowable stress values for the ISC closure bolts are revised.

Replace, remove or add applicable pages per the instructions on the enclosed Release and Change Control Record (RCCR) pages, RCCR-95 through RCCR-98. The RCCR pages should then be filed in the front of Volume I. FSAR Table of Contents pages i and ii provide a suggested assignment of chapters, appendices and supplements among the 11 volumes of the FFTF FSAR.

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		20.4-17 thru -18	634244, 642522
		20.4-26 thru -28	634244, 642522
		20.5-27 thru -28	634244, 642522
		20.5-30	634244, 642522
		20.5-32	634244, 642522
		20.5-35 thru -36	634244, 642522
		20.5-37	Text Rotation
		20.5-38 thru -39	634244, 642522
		20.5-45 thru -46	634244, 642522
		20.5-48	634244, 642522
		20.7-2	634244, 642522
		G-i thru -ii	639840
		G.1-1	639840
		G.2-1	639840
		G.3-1	639840
		G.4-1	639840
		H-ii	Update
		H-v	Update
		H.2-3	637817, 642522
		H.3.3-3	637817, 642522
		H.3.3-9	631801, 642522
		H.3.3-14	631801, 642522
		H.3.3-35	638337, 642522
		H.3.3-43	638337, 642522
		H.3.3-48 thru -51	631801, 642522
		H.3.3-77	631801, 642522
		H.3.3-96	637817, 642522
		H.3.3-98	631801, 637817, 642522
		H.3.3-99 thru -100.1	631801, 642522

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7.1 PLANT PROTECTION SYSTEM

7.1.0 TRANSITION TO SHUTDOWN

The reactor vessel has been defueled as a part of the transition to shutdown. Neither the Reactor Shutdown System nor the Containment Isolation System performs any safety functions. The description in this section is for an operating plant. As long as these systems are not required by the Technical Specifications, their actual configuration may be different from this FSAR section.

7.1.1 DESIGN BASES

7.1.1.1 Functions

The PPS comprises active devices, with the associated equipment that initiates their action, whose function (in conjunction with passive structures of the plant) is necessary to prevent unacceptable release or spread of radioactive materials by:

- 1) Protection of the fuel cladding (first barrier).
- 2) Protection of the primary coolant boundary (second barrier).
- 3) Prevention of release of radioactivity from the containment (third barrier).
- 4) Protection of Control Room occupants from radioactivity or sodium.

These three barriers are protected by preventing plant variables or conditions from exceeding their respective safety limits or by mitigating the consequences of exceeding safety limits.

The PPS also prevents unacceptable plant damage in accordance with the limits specified and initiates Heat Transport System (HTS) prime mover shutdown to prevent unacceptable thermal stress for specified components. The PPS includes the Reactor Shutdown Systems (RSS) and the Engineered Safety Features.

The functions of the PPS are to ensure that the guideline values on the release of radioactivity specified in 10 CFR 100 (or DOE Order 5480.1A, Ch. XI, as appropriate) are not exceeded for releases from the Containment

Building and to ensure that the reactor plant is protected as specified in NE Standard C 16-1T, December 1969, Paragraph 3.2.3. To accomplish these functions, the PPS senses the need for, initiates, and carries to completion:

- A. Reactor trip and shutdown of HTS prime movers (primary and secondary pumps and DHX fan motors) when the following abnormal plant conditions occur:
 - 1) Power excursion.
 - 2) Insufficient heat removal in either the reactor plant system or a CLS.
 - 3) Significant imbalance between the heat removal capabilities of the primary and secondary systems of an HTS loop.

Following reactor trip, the PPS also provides a signal used by the DHX airflow and Closed Loop Control Systems to initiate airflow shutdown and closed loop flow reduction (SDD-61).

- B. Containment isolation of the Heating and Ventilating (H&V) system when the following abnormal plant conditions occur:
 - 1) Detection of high radiation in the Containment Building exhaust.
 - 2) Detection of high radiation in the head compartment exhaust.
- C. Principal containment isolation when the following abnormal plant condition occurs:
 - 1) Detection of high radiation (higher than for H&V) in the Containment Building as monitored by detectors in either the Containment Building exhaust or the head compartment exhaust.
- D. Closure of pressure control valves (PV) in certain air and gas supply lines to prevent backflow that could result in contamination of the supply lines or possible loss of radioactivity to unrestricted areas.
- E. The PPS includes specified critical instruments that are required during post-accident conditions for maintaining the safe shutdown of the reactor/plant. These instruments are identified on Table 7.3-1.

7.2 PLANT CONTROL SYSTEMS

7.2.1 REACTOR PLANT CONTROL SYSTEM

7.2.1.1 Design Bases

7.2.1.1.1 Functions

The Reactor Plant Control System (RPCS) shall provide the following functions:

- A. The plant control configuration for the reactor and heat transport systems
- B. Control Room requirements, layout and arrangement of panels and consoles
- C. Coordination and mounting of operational information display and control for both manual and automatic control of the reactor plant
- D. Coordination and mounting of operational information display and control for test positions (SDD-61), electrical systems (SDD-11 and -12), and auxiliary systems (SDD-15, -91, -92, -93, -94, -95, -96, and -99)
- E. Manual and automatic control of the reactor plant utilizing flux, flow, and temperature signals (SDD-93 and -95)
- F. Operational control, logic and permissive circuits required for selecting and controlling rod movement, operating coolant pumps, dump heat exchanger actuators and fans, and sodium valves
- G. Position readout and display of all primary and secondary rods
- H. Convenient access to the manual control devices for the PPS (SDD-99)
- I. Stable control system response to operational disturbances to minimize thermal transients.

7.2.1.1.2 Interfacing System Requirements

A. Reactor System

The Reactor Plant Control System shall provide the following:

- 1. An integrated process instrumentation and control design to monitor and control the reactor system over the full power range

under normal and casualty operations. The reactor system maximum design conditions and operating conditions (e.g., reactor outlet temperature, reactor inlet temperature, reactor inlet temperature mismatch) are specified in SDD-31 and included in Sections 3.1 and 4.2.2.4 of SDD-90 as applied to the RPCS.

2. Capability for control and instrumentation for up to 6 control and 3 safety rods initially with capability for expansion to a total of 15 rods. Refer to Section 1.3.3.1 of SDD-90 and Section 1.2.7 and 1.3.2 of SDD-31 for the detailed requirements for control and instrumentation of all rods. The CRDM controllers shall provide minimum incremental rod motion of 0.025 in. on command and shall hold the rod at any selected position within the stroke of the travel during all modes of reactor operation except scram.
3. Capability for annunciation of low argon pressure inside the CRDM and of CRDM stator overheating in the control room. (The sensors for detection of those conditions are described in SDD-31.)
4. The CRDM controller and RPI designs, in conjunction with the CRDM shall accommodate an integrated program of surveillance and in-service inspection/testing to assure continuous reliable operation of the CRDM as follows:
 - a. Measured time to initiate a scram
 - b. Measured time to scram from full-out position
 - c. Minimum latch current
 - d. Minimum hold current for scram
 - e. Stator winding resistance
 - f. Argon pressure switch actuation
 - g. Position indication accuracy.

The equipment for this program need not be located in the Control Room. Portable test equipment may also be employed; however, provisions for connection of test instruments shall not jeopardize

Group control is also provided to operate any or all of the primary and secondary sodium pumps selected together as a group. Any or all of the DHX modules in an HTS loop may be operated with a common control setpoint. The 4 DHX fans in each loop are always controlled from a common controller.

The operator can operate the control system either in manual or automatic control. In manual control the operator varies primary and secondary sodium coolant flow by changing pump speed, flux by raising or lowering secondary rods, and dump heat exchanger air flow by changing the speed of the fans and/or the positions of the outlet dampers. In automatic control the setpoints of the controlled variables are set manually.

The group control consists essentially of a group control setpoint which is set by the operator. The group output signal is a command signal to simultaneously control the selected subloops to the desired setpoint.

A. Reactivity control

The reactivity control system converts the 480 VAC, 3 phase power supplied by SDD-12 to the pulsed DC necessary to operate the control rod drive mechanism, provides the necessary controls to permit the operator to maintain or vary the reactor power, measures the position of each rod, maintains the reactor power automatically at selected levels if desired, and prevents inappropriate sequences of operation through permissive and interlock circuitry. The reactivity control system includes the following hardware for the primary and secondary control rod system: rod drive motor-generator sets and generator output breakers; 3 phase to 6 phase transformers; CRDM controllers and associated power supplies; control rod selection logic and switches; automatic controller to operate a rod based on a flux feedback signal; provision for controlling a secondary rod from an external rod

position control signal in lieu of the normal automatic flux control signal; absolute and relative RPI instrumentation; and interlock and permissive circuitry.

The reactivity control system is designed to operate 3 primary control rods and 9 (6 active, 3 future) secondary rods. Powertrain capability and control panel space is provided for up to 15 secondary rods.

1. Control rod drive mechanism controller and power train - The CRDM controller and power train transforms the 3 phase, 480 Vac power supplied by SDD-12 into the pulsed DC necessary to operate the CRDM's in response to input commands from the control rod selection logic. The equipment monitors the power pulses to the CRDM to ensure that they correspond to the input command signals and blocks rod motion if any anomalous condition exists.
2. Control rod selection logic - The control rod selection logic is designed to provide operator access to movement of the control rods; to assure that only one control rod in the primary or one in the secondary can be manually withdrawn at one time, to assure that the automatic control can operate only one rod at a time, and to provide interlocks or alarms to prevent inappropriate operations.
3. Flux controller - The flux controller is designed to provide stable automatic control of the reactor power by operating control rods using flux as the feedback signal. This control also provides rod withdrawal stops based on high flux-flow ratio. The control also limits the reactivity insertion capability of the automatic controller.
4. Rod position indication - There are two independent systems (Relative and Absolute Rod Position Indication) which indicate and display the position of each rod. Both systems also supply signals of rod position indication to the Digital Data Handling and Display System.

which demonstrates the plant's ability to withstand a hypothetical accident much more severe than any credible failure without endangering public safety. These three levels of safety are treated in the following sections.

9.6.4.1 System Design Evaluation

A. Compressed Air System

The Service Piping System provides air to all other plant systems for instrumentation and utility usage. The compressed air system provides the plant requirement of 805 SCFM through use of three two-stage compressors rated at 540 SCFM each. The plant air requirement was determined based on the condition when the greatest probable number of demand points are in use simultaneously. The summation of users is listed in SDD-23, Table 2-2.

Optimization of supply to demand points is assured by two separate piping systems. Separate branches for instrument, breathing, and utility air assure that a failure in one system will not disable the other air system.

The instrument air branch, which provides the most essential of the compressed air services (valve operators, etc.), is given additional reliability by an interconnect with the utility air branch, allowing a standby air supply in the event of multiple instrument air component failures.

The containment isolation valves of the compressed air system, located internal and external to each piping penetration, are designed to close automatically on low system pressure. Since the closing pressure of the valves (nominally 30 psi) is well above the design pressure of containment, the self-actuating containment isolation valves ensure that containment integrity will be maintained for all system conditions without any operator action or Containment Isolation System input.

Although the containment atmosphere sampling connections do not have containment isolation valves both inside and outside the containment structure, adequate containment isolation provisions for these connec-

tions can be demonstrated by the arrangement of locked valves and pipe caps between, first, the containment atmosphere and the air lock and, second, between the air lock and the outside environment. Consequently, even with either door open, there is a closed valve and a pipe cap which provide a double barrier in the containment atmosphere sampling lines. The fact that these lines will never be opened to both the inside and the outside atmosphere simultaneously ensures containment integrity.

B. Sanitary Water System

The Sanitary Water System satisfies the plant requirements to supply water to the Fire Protection System (SDD-26), the cooling tower sumps, the makeup water inlets for the excontainment chilled water, incontainment chilled water and cooling water closed loops, the demineralizer, and all plant domestic water use points. Under normal operating conditions (a 204 gpm demand from FFTF, MASF and auxiliary support buildings), the system will provide up to 250% of rated demand for up to 11 hours before the minimum water reserve of 150,000 gal. in T-58 and 250,000 gal. in T-330 is reached. An increased degree of reliability is provided for the cooling tower sump supply through availability of alternate supplies in the area to provide water through hoses. The remaining functions are given increased reliability by a double backup arrangement: three fire pumps (SDD-26) and the sanitary pumps (P-467 and P-468) discharge to the fire/sanitary water yard loop, from which all pressurized water demands are taken. In the event of sanitary water pumps or hydro pneumatic tank failure, all functions of the pressurized portion of the sanitary water system will be assumed automatically by the Fire Protection System (SDD-26).

C. Demineralized Water System

Under normal conditions, the demineralizer and its associated components will provide the plant's needs for demineralized water. Through redundancy, capability is provided for the performance of maintenance on the circulation pumps without shutting down the system. Any other maintenance requires system shutdown. The infrequent nature of demineralized water demand and the nonessential nature of the demand points make further safeguards unnecessary.

SECTION 9.7

CONTENTS

	<u>Page</u>
9.7 COMMUNICATION SYSTEM	9.7-1
9.7.1 DESIGN BASES	9.7-1
9.7.1.1 Functional and Performance Requirements	9.7-1
9.7.1.2 Design Parameters	9.7-2
9.7.1.3 Service and Transient Conditions	9.7-2
9.7.1.4 Documentary Criteria	9.7-2
9.7.2 DESIGN DESCRIPTION	9.7-2
9.7.3 DESIGN VERIFICATION AND ACCEPTANCE	9.7-3
9.7.3.1 Design Analysis	9.7-3
9.7.3.2 Acceptance Test	9.7-3
9.7.4 EVALUATION	9.7-4
9.7.4.1 System Design Evaluation	9.7-4
9.7.4.2 Protective Features Evaluation	9.7-4
9.7.4.3 Hypothetical Accident Evaluation	9.7-6

9.7 COMMUNICATION SYSTEM

9.7.1 DESIGN BASES

9.7.1.1 Functional and Performance Requirements

The Communications System provides the following functions:

1. Communication of verbal instructions and warnings concerning high radiation or fire from the control room and emergency shutdown station (Primary CRDM Room) to all plant personnel to supplement the automatic alarms of the Radiation Monitoring System (SDD-96) and the Fire Protection System (SDD-26).
2. Communication of emergency conditions from the control room and the 400 Area Incident Command Post to off-site patrol and emergency organizations.
3. Internal voice communications connecting selected key plant operating locations for the exclusive use of plant operations personnel to control and coordinate the separate system operations which directly affect operation of the entire plant and to immediately respond and recover from emergencies.
4. The communication system shall be designed so that a failure or malfunction of an individual system will not impair the operation of the other systems.
5. The control room communications for the plant operations, maintenance and emergency warning functions shall be duplicated in the emergency shutdown station (Primary CRDM Room) for use in the event that access to the control room is lost.
6. Multiple means of communication shall be provided for control and coordination of normal and emergency operation.
7. The communication system shall be designed so that a failure resulting from loss of the control room will not impair the communications at the emergency shutdown station (Primary CRDM Room) with the rest of the plant.

8. Each Communications System which has all of its stations in containment connected to one common circuit shall have redundant penetrations into containment at separate locations.
9. Communications signals that shall not interfere with other plant instrument and control signals.

9.7.1.2 Design Parameters

The Communications System shall be composed of standard telephone, radio and audio equipment. Repair, replacement, and maintenance of components will allow the Communications Systems to be operable for as long as they are required. Radiation tolerant wire and telephone jacks used in cells shall be designed for a 20-year life.

The Plant Radio System shall have battery-supported power supplies sized to remain operable for at least four hours following a loss of off-site and on-site generated power.

9.7.1.3 Service and Transient Conditions

All Communications Systems shall be seismic Category III.

9.7.1.4 Documentary Criteria

The Communications System shall be designed and constructed in accordance with the 1996 Edition of the National Electrical Code, Article 800; the State of Washington Department of Labor and Industries, WAC-173-303 (951019) Section 340 Parts (1)(a)(b) and (2); HWS-8210-S, "Attack Alarms"; 29 CFR 1910.165, 1995 edition; and applicable portions of IEEE, NEMA, IPCEA and ANSI Standards.

9.7.2 DESIGN DESCRIPTION

The Communication System provides the equipment to perform the emergency warning, operation, maintenance and business communication functions described in Section 9.7.1.1 as they apply to all the interfacing systems throughout the FFTF. The equipment includes wall and desk-type telephones, telephone jacks, permanent and portable headsets with microphones, an automatic central exchange, battery supported power supplies, amplifiers, loudspeakers and the interconnecting wiring.

The following five separate communications systems are provided:

1. Private Automatic Exchange (PAX) for maintenance and plant inter-communication
2. General Plant Telephone (GPT)
3. Public Address (PA)
4. Plant Radio System
5. Interim Examination and Maintenance Cell System (IEM)

9.7.3 DESIGN VERIFICATION AND ACCEPTANCE

9.7.3.1 Design Analysis

To ensure that communication signals will not be introduced into plant instrumentation lines and also that instrument or other voltages will not interfere with communication signals, all of the communication cables will be installed in a separate conduit system throughout the plant. The conduit system will be routed separately from power cables. The shielding and separation provided by the conduit will eliminate signal interference. This precaution will not apply to fiberoptic cables, which are immune to EMI.

9.7.3.2 Acceptance Test

The Communications System operation was originally verified during the plant acceptance test program by test TS-15-2D011, "Communications System Telephone, PAX and PA System Pre-Operation and Startup Test." The acceptance test also confirmed that the system performance requirements of Section 9.7.1.1 were met.

Functional testing is performed on a periodic basis per an operating procedure. A retest of repaired equipment is performed by the FFTF Work Control procedure.

9.7.4 EVALUATION

9.7.4.1 System Design Evaluation

The function of the Communication System is to provide the means by which information can be transmitted throughout the FFTF for the purpose of coordinating facility functions. The system provides the means for communication with and from the control room, and the 400 Area Incident Command Post.

To ensure that these functions are provided as required under all anticipated normal and off-normal conditions, separate subsystems are provided. Portions of these systems, consistent with their importance in maintaining safe operation or shutdown of the plant, are provided with redundant power supplies and backup systems. The redundancy and flexibility of these systems provides sufficient communications to ensure safe and prudent operations during all anticipated modes of operation.

Emergency communications between the FFTF and offsite emergency organizations are provided by radio and telephone connections. These communications will be used to request assistance from offsite locations and to establish recovery action. Radio Communications with off-site locations are provided as discussed in Section 13.3.7.2.5. Adequate communication will be provided to ensure implementation of the emergency plan.

9.7.4.2 Protective Features Evaluation

To ensure that communications are maintained for off-normal events, portions of the system are provided with redundant backup power supplies.

A. Loss of Power

All of the Communications Systems except the GPT system receive electrical power from 120 VAC buses which are supported through Emergency Power System.

At least one telephone in the Control Room and 400 Area Incident Command Post is powered from the Hanford Site Telephone Exchange and is not dependent on the availability of power at the FFTF site.

B. Natural Phenomena

Operation of the Communication System is not necessary to assure control of safe shutdown. Reactor decay heat removal in the post-earthquake or post-tornado period as described in Section 7.2.1.2.4 and 7.2.1.4.2 does not require communications. The system meets the requirements of seismic category III. The system is tornado resistant to the degree of protection afforded by structures that house the various communication components.

C. Failure Evaluation

All communications systems have separate power supplies and amplifiers and are electrically separate from each other. A failure or malfunction in one system will not affect the operation of the others.

All of the communications amplifiers, switchboards and power supplies are located outside containment in the control building. Local sodium fires, spills or a reactor accident should result only in the loss of the local telephone stations without affecting the entire system. If the system operation is affected, the faulty components can be disconnected at local junction boxes to restore operation of the system.

The communications lines to the PAX and PA equipment located inside containment enter containment at two separate containment penetrations 180° apart. Therefore, each of these systems is redundant and separated inside containment. Approximately half of the containment communications are connected through each penetration. In the event of a sodium spill, fire or reactor accident which disables one penetration, cable group or junction box, the remaining separate communications systems will continue to operate using the other wiring network connected through the other penetration.

The Plant Radio System is a backup to the PAX and PA Systems.

9.7.4.3 Hypothetical Accident Evaluation

No requirements are imposed on the Communication System for Hypothetical Core Disruptive Accident (HCDA) conditions.

6. 1.5 KW limit for metal fuel assemblies.
7. 250 watts maximum per component loaded into a Core Component Container (CCC), but the total watts for all of the components within a CCC must not exceed 1500 watts.
8. *A minimum of 4 years post irradiation (to permit decay of short life radioisotopes) before loading into a CCC.
*NOTE: Shorter periods than four years may be allowed on a case basis provided the radionuclides inventory is bounded by current analysis.

C. The following limits will be applicable for the center fuel pin cladding temperature during all handling operations:

1. 800°F, maximum allowed for long-term storage of assemblies to preserve material properties test data when required by the experimenter.
Assemblies tested in excess of 800°F may be maintained at not more than 50°F above the test temperature at any point along the axial length of the fueled section, but in no case greater than 1000°F.
2. 1000°F, maximum allowed for short-term (1-hr) transfer, or long-term storage where data preservation is not required.
3. 1200°F, maximum allowed for fueled assemblies or pins being removed from the Containment Building.
4. 1500°F, maximum allowed to preclude pin cladding rupture.

The fuel to be transferred for cleaning shall not exceed a cladding temperature of 900°F. (See Appendix H for temperature value established by analysis to minimize fuel cladding degradation via stress rupture failure.)

- D. The System (machines and facilities) shall prevent the association of fueled assemblies in a nuclear critical configuration. Storage locations will utilize safe geometry, administrative control, or both to maintain subcriticality with a maximum $k_{eff} < 0.9$. FSAR, Chapter 20, Criticality Specifications, contains the detailed limits and controls.
- E. When the temperature difference between a new or reconstituted fueled core component being transferred and the immersion fluid potentially exceeds 200°F, preheating or cooling of the core component will be provided to reduce the difference to less than 200°F. Some special core components, such as 12-ft fueled test components, may require that the temperature

difference be less than 200°F. In these cases, the test originator will specify the maximum temperature difference allowed. In all cases, the temperature difference identified is between the core component structure (i.e., duct, handling socket, and nosepiece) and the immersion fluid.

When the temperature difference between a test assembly (above the 12-ft core section) and the immersion fluid (except CLIRAs) potentially exceeds 120°F, preheating of the test assembly will be provided to reduce the difference to less than 120°F. When the temperature difference between a CLIRA and the immersion fluid potentially exceeds 50°F, preheating and/or cooling of the CLIRA will be provided to reduce the difference to less than 50°F.

- F. Inerting will be provided for all assemblies and test assembly hardware prior to insertion into sodium-containing facilities in order to reduce the carryover of contaminants to the argon cover gas and to minimize the oxygen contamination of the sodium.
- G. The following maximum decay heat limit exists for the movement of a single Ident 69-type pin container:
 - 1. 7.0 KW per pin container within the IEM Cell.
 - 2. 1.4 KW per pin container without containment integrity in an inert gas, argon or nitrogen.
 - 3. 250 watts maximum per pin container in a CCC.
- H. The following maximum decay heat limit exists for the movement of spent fuel in a CCC loaded within the limits of FSAR, Appendix H:
 - 1. 1.5 KW in the SWC.
 - 2. 1.5 KW in the ISC.
 - 3. 1.5 KW in the IEM Cell.

11.1.1.3 Service and Transient Conditions

- A. Provisions will be made for the on-site receipt of new core components, test assemblies, and their sub-assembly replacement parts within the Reactor Containment Building (RCB) at specific locations which will contain the

CONTENTS (Cont'd)

	<u>Page</u>
11.6.4.1 System Design Evaluation	11.6-20
11.6.4.2 Protective Features Evaluation	11.6-22
11.6.4.2.1 Conditioning Stations	11.6-22
11.6.4.2.2 Temporary Inerting Systems	11.6-23
11.6.4.3 Evaluation of TACS Configuration at Turnover To the ERC	11.6-24
11.6.4.3.1 Reason for New Configurations	11.6-24
11.6.4.3.2 Different Types of Components	11.6-24.1
11.6.4.3.3 Irradiated Components in the New Storage Vault	11.6-24.1
11.6.5 REFERENCES	11.6-25

TABLES

<u>Table</u>	<u>Page</u>
11.6-1 Conditioning Station Operating Characteristics	T-11.6-1

FIGURES

<u>Figure</u>	<u>Page</u>
11.6-1 Test Assembly Conditioning Station	F-11.6-1
11.6-2 Core Component Conditioning Stations	F-11.6-2
11.6-3 Shield Plug Storage Stations	F-11.6-3
11.6-4 Temporary DFA Inerting Cell	F-11.6-4
11.6-5 Temporary Test Assembly Inerting Cell	F-11.6-5
11.6-6 SERF Cask Loaded in New Storage Vault	F-11.6-6

11.6.2.1.3.1 TACS Inerting and Preheating (IP) Vault

The TACS inerting and preheating vault consists of three IP cells and a heater cell. The IP cells are used for conditioning with argon and holding at temperature, new reactor test assemblies up to 42 ft in length (including grapple adapter) for transfer by the CLEM to the reactor. The heater cell is similar to the CCCS heater cell and houses the Heater Assembly that provides 15 kW maximum power for heating of the inert gas (argon) that is used to condition the new test assemblies.

Cooling nitrogen from System 25 flows up through the IP vault to maintain 100 °F average wall temperature.

11.6.2.1.3.2 TACS New Storage Vault

The TACS New Storage Vault, located immediately east of the IP vault, is designed to store up to 12 new test assemblies. However, administrative controls limit the number of fueled assemblies that may be stored in the vault (see Section 20.4.1, Chapter 20). The 46-ft deep vault is argon filled and supports new test assemblies from their grapple adapters in an open vault. The top shield plate is removable to provide space for lowering a CLEM cold-wall container into the vault during cold-wall exchange.

The top shield plate can be removed and replaced with a SERF Cask support that will provide the necessary structural support for the SERF Cask. This will provide a temporary facility that can be made available whenever it is necessary to load the SERF Cask in containment.

The facility is made available by removing an existing top shield plate from the New Storage Vault and replacing it with a SERF Cask support (see Figure 11.6.6). This support is rectangular (to fit cell opening),

45.0 in. x 64.12 x 15.0 in. high with a 4.0-in.-thick top plate. The top plate has a 43.63-in.-diam hole to accommodate the SERF Cask adapter flange. The SERF Cask, through the adapter flange, interfaces with the FFTF refueling floor valve. The dead load of the SERF Cask, floor valve, and CLEM movable closure valve is carried by the SERF Cask support to the main structure of the New Storage Vault.

Although the TACS New Storage Vault was designed for new assemblies, non-fuel irradiated assemblies may also be stored there. Analyses of radiation levels must be made before insertion. See Section 11.6.4.3 for inventory for FFTF turnover to the ERC.

11.6.2.1.3.3 TACS Radioactive Storage Vault

The Radioactive Storage Vault is designed to provide eight shield storage locations for radioactive and contaminated test assemblies and test assembly components. However, administrative controls limit the number of fueled assemblies that may be stored in the vault (see Section 20.4.1, Chapter 20). This storage vault is also 46 ft deep, argon filled, and supports the assemblies from the grapple adapter in an open vault. The design is for 90-day decayed tests.⁽⁵⁾

11.6.2.1.3.4 TACS Recycled CLIRA Storage Vault

The Recycled CLIRA Storage Vault is located immediately east of the New Storage Vault and immediately south of the Radioactive Storage Vault. Recycled CLIRA fuel assemblies will be stored in three, 43-ft-deep cells located within this vault. Both heating and cooling, and separate isolation argon purge capability are provided for recycled sodium-filled CLIRA units to maintain the test assembly duct temperature between 250 and 600 °F while in this storage position. The design is for a 90-day decayed test, generating a maximum

requirements of Seismic Category III⁽²⁴⁾. Furthermore, the CLIRA pressure boundary will not incur gross failure under Design Basis Earthquake (DBE) loading and produce a release to the cell. During the DBE the maximum stress in the TACS cell wall is 25,800 psi (allowable is 30,400 psi). Under the same input the CCCS wall would be 20,300 psi with an allowable of 36,000 psi. With the cell plugs in place the cell boundaries will remain intact.^(17,18) Loads under interfacing conditions with the fuel handling machines and equipment are discussed in Section 11.1.3.1.4. These facilities will survive the DBE intact and protect the stored fuel pins and assemblies for all expected equipment interfacing conditions.

11.6.4.2.2 Temporary Inerting Systems

In the event of a failure of the CLEM movable valve limit switches, the CLEM valve could continue traveling down after first contacting the top plate of the inerting cell. The inerting cells are not designed to support the full weight of the CLEM movable valve. To prevent the full weight of the CLEM valve from being applied to the cell, several features have been incorporated into the design of the cells as discussed below.

Based on elevation marks on the CLEM itself, the Movable Valve could move down 2.35 in. below the normal elevation of the top of the Floor Valve at TACS before bottoming out on the mechanical stop pins on CLEM. This would occur only if there were no interlocks/switches or manual stopping of the valve. To accommodate part of this downward movement of the CLEM valve, the top of the inerting cell top plate is located 1-1/2 in. lower than the top of the Floor Valve when located at TACS; thus, the CLEM valve must travel down further before contacting the top of the inerting cell. The remaining downward travel of the CLEM valve after it first contacts the top plate (maximum remaining travel is 0.85 in. max) is compensated for by 8 springs, which can compress more than this distance before reaching their maximum safe load as specified by the spring vendor. Further downward movement of the CLEM Valve beyond this point is prevented by the mechanical stop pins on CLEM. Thus, the full weight of the valve cannot be applied to the cell even if the CLEM limit switches do not actuate. In case of a dropped load, such as a core component pot or driver fuel assembly into a cell, the operation would be stopped until an assessment had been made of any damage to the components. The consequences of

such a drop by CLEM, while highly unlikely, would be constrained by established pressure boundaries. Since the temporary inerting cells are used only for handling of nonirradiated fuel, there is no potential for creating a situation that is hazardous to plant personnel.

When mating with the temporary test assembly inerting cell the CLEM MCV lowers to approximately 0.25 in. from the CLEM mating plate, mounted on the 10-in. vacuum valve, and contacts four posts located at four quadrants around the valve. Limit switches on the CLEM MCV are set to stop the motion of the MCV at this point and unload its weight on the posts (~30,000 lb). The posts are designed so that any two posts will support in excess of 800,000 lb without yielding. The operating procedure requires that the location of the four posts and the shimming of the four posts (match-marked) be visually verified before lowering the CLEM MCV and that an observer be in communication with the CLEM operator during the CLEM MCV lowering operation. If the limit switches failed the CLEM MCV would be stopped by the four posts and the CLEM operator without damage. If the posts were not in place or if the shims were not installed, the CLEM MCV would contact the mating plate located on the 10-in. vacuum valve. This could damage the 10-in. valve with loss of the argon atmosphere around the test assembly but would not damage the inerting cell or the test assembly.

11.6.4.3 Evaluation of TACS Configuration at Turnover to the ERC

11.6.4.3.1 Reason for New Configurations

As FFTF approaches turnover of the plant complex to the Environmental Restoration Contractor (ERC), some configurations for component storage in TACS are being considered based on viability, cost, and impact to the overall Transition Project schedule. None of these configurations include fuel or other fissile material. Reference 25 provides the details of the currently considered configurations and the engineering/safety evaluations. These details are briefly described in the following sections.

11.6.4.3.2 Different types of components

Although most of the components to be stored in TACS are standard assemblies (e.g. MOTAs, PIOTAs, etc.), other components are also listed in Reference 25. An example is the drill fixture to be used for the reactor core basket. Any non-fissile components are acceptable, provided appropriate evaluations such as radiation levels are performed.

11.6.4.3.3 Irradiated Components in the New Storage Vault

Reference 25 specifies storage of irradiated components in the new storage vault. An analysis was performed of the radiation levels outside of the TACS cells. It was shown that they would either be acceptable as is or shielding could easily be added to make it acceptable. While reference 25 analyses were for specific components in the new storage vault, the inventory may be changed provided appropriate radiation analyses are performed.

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SECTION 11.10

CONTENTS

	<u>Page</u>
11.10 INTERIM EXAMINATION AND MAINTENANCE CELL	11.10-1
11.10.1 DESIGN BASES	11.10-1
11.10.1.1 Functional and Performance Requirements	11.10-1
11.10.1.2 Design Parameters	11.10-2
11.10.1.2.1 Design Life	11.10-2
11.10.1.2.2 Cooling and Ventilation	11.10-2
11.10.1.2.3 Radiation Control	11.10-5
11.10.1.2.4 Sodium Removal Equipment	11.10-6
11.10.1.3 Service and Transient Conditions	11.10-7
11.10.1.4 Design Safety Margins for Hypothetical Events	11.10-8
11.10.1.5 Instrumentation and Control	11.10-8
11.10.2 IEM CELL DESIGN DESCRIPTION	11.10-9
11.10.2.1 Main Cell	11.10-10
11.10.2.2 Cell Annex	11.10-10
11.10.2.3 Operating Galleries	11.10-10.1
11.10.2.4 Cell Structure	11.10-10.1
11.10.2.4.1 Shielding Walls	11.10-11
11.10.2.4.2 Equipment Mounting and Cell Liner	11.10-11
11.10.2.5 Atmosphere	11.10-11
11.10.2.6 Support Services	11.10-11
11.10.2.6.1 Electrical Power	11.10-12
11.10.2.6.2 Communications	11.10-12
11.10.2.6.3 Radioactive Waste Disposal	11.10-12
11.10.2.6.4 Ventilation	11.10-12
11.10.2.6.5 Fire Protection	11.10-12
11.10.2.6.6 Radiation Monitoring	11.10-13
11.10.2.6.7 Lighting System	11.10-13.2
11.10.2.7 Special Equipment	11.10-13.2

CONTENTS (Cont'd)

	<u>Page</u>
11.10.2.7.1 Handling Equipment	11.10-13.2
11.10.2.7.2 Viewing Equipment	11.10-13.2
11.10.2.7.3 Electrical Power and Service Equipment	11.10-14
11.10.2.7.4 Through-Wall Transfer Equipment	11.10-14
11.10.2.7.5 Cooling and Ventilation Equipment	11.10-14
11.10.2.7.6 Sodium Removal System	11.10-14
11.10.2.7.7 Disassembly/Reassembly Equipment	11.10-15
11.10.2.7.8 Measuring Equipment	11.10-15
11.10.2.7.9 Receiving and Storage Equipment	11.10-15
11.10.2.7.10 Maintenance Equipment	11.10-15
11.10.3 DESIGN VERIFICATION AND ACCEPTANCE	11.10-16
11.10.3.1 Design Analysis	11.10-16
11.10.3.1.1 Sodium Removal System	11.10-16
11.10.3.1.1.1 Deleted	11.10-16
11.10.3.1.1.2 Thermal	11.10-16
11.10.3.1.1.3 Seismic	11.10-17
11.10.3.1.2 Maintaining Cell Integrity	11.10-17
11.10.3.1.2.1 Stress	11.10-18
11.10.3.1.2.2 Thermal	11.10-18
11.10.3.1.2.3 Seismic	11.10-18
11.10.3.1.3 Handling Irradiated Fueled Core Components	11.10-18
11.10.3.1.3.1 Stress	11.10-18
11.10.3.1.3.2 Seismic	11.10-19
11.10.3.1.3.3 Thermal	11.10-19
11.10.3.2 System and Feature Tests	11.10-19
11.10.3.2.1 Sodium Removal System Tests	11.10-19
11.10.3.2.2 Core Component Handling Tests	11.10-20

- C. The cooling gas flow for fuel subassembly cooling shall be controlled to limit the maximum exit gas temperature to 450°F when passed through or over a fuel subassembly.
- D. All controls for the IEM Cell Sodium Removal System shall be manually controlled and remotely actuated from the control console except for those functions which, if not automatically actuated, could result in system failure because of parts failure or operator error.
- E. Positive indication of key valve status during sodium removal operations shall be provided at the system operating control console.
- F. Interlocks shall be provided in the Sodium Removal System design which will preclude the possibility of water entering the system until such time as the system is sealed from the IEM Cell atmosphere.
- G. Controls for disassembly/reassembly equipment shall be provided for remote operation of the equipment using direct electrical connections or if practical, one of the manipulators.
- H. A welding control console shall be provided with the capability for programming and control of all welding operations in the IEM Cell.
- I. Primary control of measuring equipment for component positioning, data acquisition, reduction, storage, and readout shall be accomplished with a computer system and its associated peripheral equipment. Manual control shall also be provided for selected operations, from a console at the operating station.
- J. A means shall be provided for local calibration of the measuring equipment to ensure accuracy of data being measured.

11.10.2 IEM CELL DESIGN DESCRIPTION

The IEM Cell is a shielded, hot-cell complex which houses the remotely operated equipment necessary for the performance of nondestructive examination of core components, packaging irradiated and non-irradiated fueled/non-fueled components for shipment or dry storage, and limited maintenance of reactor plant equipment as shown in Figure 11.10-1. The IEM Cell consists of a main

cell area, cell annex and four operating galleries. Service cells containing process equipment for support of in-cell sodium removal equipment are located, adjacent to Containment (and the IEM Cell) in the Reactor Service Building (RSB). This process equipment consists of tankage, valves, heat exchangers, blowers, piping, etc.

11.10.2.1 Main Cell

The main cell consists of two regions: one 20 ft x 14 ft x 48 ft and the second 10 ft x 14 ft x 55 ft. Cell ceiling valves over the main cell are provided for transfer of all core components and maintenance equipment in and out of the IEM Cell.

There are five grid structures distributed on the lower level of the main cell and annex floor depending on the position of the Ident 35 cart. These grid structures reduce the possibility of the fuel assemblies and pin containers fuel region entering water in the unlikely event that water would accumulate on the cell floor. See Chapter 20 for specific details regarding when the grid structures are required to be installed.

11.10.2.2 Cell Annex

The cell annex making up the remainder of the cell is 10 ft x 14 ft x 34 ft 11 in. An access plug is provided over the cell annex for equipment insertion and removal and for change-over to air during a major cell shutdown. The configuration of the cell annex affords the capability of unobstructed right-angle viewing of the annex operations.

There are five grid structures distributed on the lower level of the main cell and annex floor depending upon the position of the Ident 35 cart. This grid structure reduces the possibility of the fuel assemblies and pin containers fuel region entering water in the unlikely event that water would accumulate on the cell floor. See Chapter 20 for specific details regarding when the grid structures are required to be installed.

11.10.2.3 Operating Galleries

Four levels of operating galleries, which encompass the cell operating face along the east wall of the main cell and the north and east wall of the annex, provide visual access for remotely operating the in-cell equipment.

The operating galleries are provided with shielded viewing windows, master-slaves manipulators, communication equipment, in-cell viewing aids, and the consoles and controls necessary for the performance of the cell functions. The operating galleries also include the transfer locks, entrance plugs, inter-connection plugs and associated equipment, and other support equipment and systems necessary to perform the in-cell operations.

11.10.2.4 Cell Structure

The cell structure is designed to protect operating personnel from radiation by providing adequate radiation shielding, by containing the argon atmosphere to the cell by adequate sealing, and by maintaining a lower pressure in the cell than in the operating gallery.

- C. All pressure boundary components of the sodium removal system have been designed and fabricated to meet the specifications of ASME Section VIII or ANSI B31.1. The associated containment penetrations and valving meet Section III, Class 2 and MC of the ASME Code.⁽³¹⁾ The actual working pressure in most components is so low that structural rigidity requirements and the use of commonly available weights of piping and materials will result in a very conservative system design, when analyzed strictly from a pressure standpoint. Relief valves will be provided and vented to other parts of the closed system at a sufficient number of locations so that no part of the system can be blocked off and subjected to a dangerous overpressure⁽³⁴⁾. Those portions of the system where pressure surges due to rapid chemical reaction may occur, such as the sodium removal chamber, have the design pressure set (150 psig) to accommodate the highest calculated surge with relief valve (90 psig) venting provided to the water return. The effects of inadvertently or intentionally flooding a component coated with metallic sodium has also been taken into account in establishing design pressures⁽³⁴⁾.
- D. Temperatures in the sodium removal system will all be under 500°F during normal operation. The number of temperature cycles to design lifetime was considered in the detailed design phase. The system is designed with flexible expansion joints in all pipe runs connecting rigidly mounted units to prevent thermally induced strains. The use of flexible joints rather than expansion loops is dictated by the limited space available. The basis for selection of the actual design temperature of each major component will be given in the final design report. The system will be capable of withstanding temperatures well above the expected normal values without damage. The lowest temperature at which damage could occur will be given in the final design report.
- E. The control system consists of a full-graphic control console panel, signal conditioning equipment, a water vapor monitor, a water conductivity monitor, a hydrogen monitor, a temperature recorder, and a rack to mount all the components. The controls are manually operated and remotely actuated in accordance with a developed process procedure.

Functions which could result in system failure because of parts failure or operator error are interlocked or automatically sequenced. However, it is a manually operated system.

11.10.4.1.2 Fuel Handling Equipment

The equipment inside the IEM Cell used to handle irradiated core components has been designed following state-of-the-art principles and guidelines for remotely operated equipment. Consideration is also given to the need to preserve data inherent in the irradiated fuel and materials.

Prior to actual IEM Cell operation, the equipment will be installed in the IEM HCTF so that personnel can be trained in the operation of the equipment. As part of the training program, the man/machine relationship will be observed to assure the equipment has been properly designed for remote operation. Any deficiencies will be corrected prior to installation of the equipment into the IEM Cell at FFTF.

Procedures will instruct the operator(s) on the proper and safe performance of each operation to be performed in the IEM Cell.

The operating procedures will include instructions as to the course of action to follow during off-normal events.

11.10.4.1.2.1 Core Component Cooling. A fuel subassembly gas cooling manifold is provided as an arrangement of pipes, filters, valves, flow devices, hoses, and couplings which can be connected to the various items of handling equipment for the fuel components that require heat removal. The manifold applies a negative pressure to these components, either through their grapples, shrouds, or containers, to cause the cell argon atmosphere to be drawn across the components for cooling. The manifold is designed for a maximum cooling gas inlet temperature of 450°F. The gases are then passed through connecting hoses, control valves, flow devices, primary filters, and then out through a cell penetration to the Plant H&V System (SDD-25) for cooling the cell argon. Up to 400 SCFM of cooling gas flow is provided by the cell equipment for up to 18 kWt decay heat.

The manifold flexible hoses attach to the following equipment items for cooling:

11.10.5 REFERENCES

1. Nuclear Systems Materials Handbook, HEDL TID-26666.
2. Seismic Design Criteria for FFTF Site, JABE-WADCO-02, June 1973.
3. FFTF Structure and Response Curve for Seismic Design of FFTF Equipment, HWS-1386, Rev. 7, July 1973.
4. Crane and Hoist Safety Manual, HEDL-TI-76003, February 15, 1973.
5. 600 V Power and Control Wire - Adverse Environment, HWS-1353.-
6. 600 V Power and Control Wire, HWS-1352.
7. Radiological Design Criteria, WHAN-M-7.
8. Radiation Protection Procedures, WHAN-M-8.
9. Deleted.
10. Masterslave Manipulator Seismic Report, AMCO 02-M-175.
11. Shielding Windows Seismic Report, AMCO 02-M-164.
12. Periscope Seismic Report, AMCO 02-M-168.
13. Electrical Power and Control Plug Seismic Report, AMCO 02-M-144.
14. Special Service Plug Seismic Report, AMCO 02-M-147.
15. Deleted.
16. Cell Ceiling Valve Seismic Report, AMCO 02-M-159.
17. Small Tool Transfer Lock Seismic Report, AMCO 02-M-143.
18. Gamma Scan Collimator Seismic Report, AMCO 02-M-178.
19. Deleted.
20. Deleted.
21. Deleted.
22. Deleted.
23. Pipe Static and Dynamic Analysis Software System, (PIPESD Computer Code), Control Data Corporation.

24. Static and Dynamic Structure Analysis System, STARDYNE, Control Data Cybernet Service, Control Data Corporation.
25. W. N. Ramsos et al., CLIRA Decay Heat Removal Test, HEDL-TME-72-66, May 1972.
26. Deleted.
27. IEM Cell Pressure Study, VITRO-R-402, January 1976.
28. Military Standardization Handbook, MIL-Handbook 5B.
29. Seismic Design Criteria for the Fast Flux Test Facility, JABE-WADCO-02, Rev. 2, Richland, WA, July 1975.
30. Nuclear Energy Standard, Hoisting and Rigging of Critical Components and Related Equipment, NE F 8-6T.
31. ASME Boiler and Pressure Vessel Code, Section III, 1971.
32. Deleted.
33. Hanford Site Hoisting and Rigging Manual, DOE-RL-92-36.
34. IEM Cell Sodium Removal System Overpressure Protection Analysis, D-41-7025, Revision 1, June, 1996.

- D. Personnel Five-Minute Air Packs shall be available at the CLS lowest floor level and on the elevator platform during periods of personnel occupancy.
- E. Standard industrial safety design practices shall be used to assure personnel safety and equipment protection.
- F. Manual override features shall be incorporated that will permit return to a safe state should any equipment failure occur. Simultaneous actions which could result in major operational interference, except during override operations, shall be prevented.
- G. Redundant and/or backup systems shall be provided such that in event of a failure:
 - 1. Interrupted operations can be completed in a safe manner after occurrence of the failure.
 - 2. Accident sensing devices shall remain functional.
 - 3. Key status indicating sensors shall remain functional following all credible accidents less than a faulted condition.
- H. Non-combustible and fire resistant material shall be used where practical. Sodium contact with nonmetallic items (insulation, and electrical equipment) shall be prevented by proper design.
- I. Interlocks shall be provided to preclude unsafe operations.
- J. Impact limiters shall be provided to maintain acceptable structural integrity for the CCC and ISC in the event of a drop accident and to minimize potential damage to the fuel.
- K. The system (machines and facilities) shall ensure the association of fueled assemblies and pin containers in a safe configuration using K_{eff} or batch limit controls as specified in FSAR Chapter 20, FFTF Criticality Specifications.

11.20.1.2.6 Instrumentation and Control and Auxiliaries

- A. A Video Identification Unit (VIU) consisting of a TV camera, zoom lens, and lights mounted inside a sealed and shielded housing shall be available for remote visual identification of the payload in the cask. The VIU will be installed on the DSWC or Refueling Floor Valve at the 550-foot level. Identification is observed on a TV monitor screen.

- B. Argon control consoles shall be provided to regulate the cask cavity pressure during loading/unloading operations, to provide an argon cask cavity purge, and to provide argon buffer purge or leak test capability to seals.
- C. A bag sealer is installed in the CLS at the personnel platform level to provide capability for sealing and cutting plastic transfer sleeves for the T-3 cask loading/unloading operation.

11.20.2 DESIGN DESCRIPTION

The Cask Loading Station (CLS) is located in the Reactor Service Building (RSB). The center line of the CLS is in line with the east track and 2-ft 6-in. north of building column 1.2. The CLS extends down from the RSB 550-ft floor level to the 518-ft 0-in. elevation. See Figure 11.20-1.

The CLS has two CLS Interface Cover Plates: the CLS/BLTC Interface Cover Plate and the CLS/SWC Interface Cover Plate. The CLS/BLTC Interface Cover Plate is a removable cover that allows normal rail, vehicle, and personnel travel over the CLS when located in the 14-ft square pit opening at the 550-ft floor level. This cover plate is installed when the CLS is required to interface with the BLTC, or when rail access is required across the CLS. The cover plate has a removable center section for placement of small-diameter items into the CLS and is removed for installation of large-diameter casks or cask auxiliary equipment.

The CLS/SWC Interface Cover Plate is a removable cover with a limited deckplate loading capacity, allowing only personnel travel over the CLS when located in the 14-ft square pit opening at the 550-ft floor level. This cover plate is installed to configure the CLS for DSWC or ISC loading operations using the SWC. The CLS/SWC Interface Cover Plate has a removable center section for placement of small-diameter items into the CLS and is removed for installation of large-diameter casks or cask auxiliary equipment.

A 12-ft square stationary personnel and equipment work platform, having a center penetration, is bolted to the pit walls at the 540-ft elevation.

A motor-driven elevator platform, which provides the capability for supporting shipping casks up to 24-ft long and 90-in. diameter and weighing up to 75 tons, is located within the 12-ft square pit. The elevator platform is

11.21 SOLID WASTE CASK (SWC)

11.21.1 DESIGN BASES

11.21.1.1 Functional and Performance Requirements

NOTE: Throughout this section, "cleaned" means that either a component has never been in contact with sodium or that the sodium residuals have been removed in the IEM Cell.

The function of the Solid Waste Cask (SWC) is to provide means for accomplishing the following operations:

- 1) Removing solid waste containers, which contain cleaned non-fuel solid radioactive waste, from the IEM Cell via the 28-in. diameter ceiling valve when irradiated fuel is present in the Cell. The SWC is then used in transporting this waste, weighing up to 3,720 lb, to the RSB Cask Loading Station (CLS) for transfer to a disposal cask.
- 2) Removing a sealed Core Component Container (CCC) loaded within the limits of FSAR, Appendix H, Section H.3.3.10, Limit #10 from the IEM Cell via the 9-ft 6-in. ceiling valve and transporting it to the RSB CLS for transfer into an Interim Storage Cask (ISC). The fuel in the Ident 69 pin containers and the fuel assemblies may be either irradiated or unirradiated.

NOTE: The maximum and minimum weight of a loaded CCC is controlled by the impact limiter installed in the ISC. The cases presently covered in the system analysis are as listed above. Other CCC loadings may be approved on a case-by-case basis.

- 3) Transfer empty disposable waste containers into the IEM Cell.
- 4) Transfer empty CCCs between the RSB Solid Waste Transfer Pit (SWTP) and the IEM Cell.

11.21.1.2 Design Parameters

The SWC shall meet the following design requirements:

- 1) Consist of a shielded container capable of transporting either a disposable waste container (containing six non-fuel core components, 27 ft³ of radioactive waste, or any combination thereof) or a CCC (with contents as listed in FSAR Section 11.21.1.1, Item 2).
- 2) Provide an integral pressure boundary capable of maintaining a pressurized inert gas atmosphere for the SWC contents and the IEM Cell, SWTP, and CLS interfaces.
- 3) Have the capability of being transported in a horizontal attitude on a flat bed transporter or vertically with an overhead crane in any mode.
- 4) Provide an integral closure valve to allow the insertion of a waste container or CCC into the cask and retain the cask pressure boundary.
- 5) Contain a remotely controlled hoist for loading and unloading a disposable solid waste container or CCC with the following capabilities:
 - Minimum dead weight hoist load of 2.5 tons
 - Grapple velocity variable over a range of 14 to 32 in./min
 - Independent grapple position readout within ± 0.25 in.
- 6) Transport a sealed CCC containing:
 - Six or seven fuel assemblies which:
 - are sodium cleaned
 - each have a maximum decay heat of 250 watts
 - have a total decay heat ≤ 1500 watts
 - may be either irradiated or unirradiated.
 - Six Ident 69 pin containers containing fuel pins which:
 - are sodium cleaned
 - each have a maximum decay heat of 250 watts
 - may be either irradiated or unirradiated.
 - A combination of fuel assemblies and Ident 69 pin containers loaded within the limits of FSAR, Appendix H, Section H.3.3.10, Limit #10.
- 7) Transport a disposable waste container containing six cleaned non-fuel core components (250 watts maximum each), 27 ft³ of radioactive waste, or any combination thereof.

- 8) The SWC shall be capable of transport in a vertical or horizontal attitude with natural convection and radiative cooling only. Maximum fuel core component cladding temperature shall not exceed $900^{\circ}\text{F}^{(10)}$ during any transportation or transfer mode.
- 9) Limit the steady-state external dose rate (combined gamma/neutron) to 10 mrem/h at 3 ft from any cask surface except for the bottom of the cask closure valve, which shall not exceed 20 mrem/h at the valve surface when the cask is in the vertical attitude. During CCC or waste transfer operations, the transient dose rate due to streaming through mating valves and interfaces at the cask or valve surface shall not exceed 200 mrem/h.
- 10) Design SWC for the following minimum service life. Unit is easily disassembled to perform the third and fifth of these maintenance tasks and to minimize exposure to working personnel.
 - Static structural components - 20 years.
 - Control, electrical equipment, and mechanical drive line components other than those specified below - 10 years.
 - Bearings, motors, electro-mechanical clutches, and circuit breakers - 5 years.
 - Static seals - 5 years.
 - Dynamic seals - 5 years.
- 11) Provide a control console for controlling and monitoring the SWC functions by semi-automatic or manual means.

11.21.1.3 Service and Transient Conditions

The SWC shall be designed in accordance with the following requirements:

- 1) The SWC shall withstand a vertical drop of 5 ft to an unyielding surface on any bottom corner or on the flat bottom of the closure valve without sustaining shielding damage or loss of containment to the extent that would present an unacceptable radiation hazard to operating personnel, or unacceptable radiation doses at 1.5 mile or at the 4.5-mile site boundary, or damage to the cask sufficient to cause the contents to be ejected.

- 2) Seismic design criteria shall be as defined by JABE-WADCO-02.⁽¹⁾

The SWC shall be classified as a Seismic Category I item in accordance with JABE-WADCO-02⁽¹⁾ (except as modified by NRC Regulatory Guide 1.92⁽²⁾) and the FFTF structural response curves for seismic design of equipment.

- 3) The attaching and lifting structures shall be adequate to keep the SWC attached to the Multi-Purpose Rail Transporter (MPRT), the Maintenance Equipment Transport System (METS), or the crane during a seismic event. Should the MPRT or METS turn over during transport of the SWC, it is not expected that the SWC will sustain any unacceptable damage because the fall would be less than 5 ft.
- 4) The SWC pressure boundary (including cell interface) shall not be violated while the cask is attached to the IEM Cell.
- 5) The SWC shall maintain an acceptable confinement barrier to limit the potential release of radioactive material within off-site and on-site guideline values.

11.21.1.4 Instrumentation and Control

Monitoring and indication instrumentation shall be provided as follows:

- 1) Displays of grapple drive mechanisms:
- Pull force.
 - Position, continuous.
 - Position, full up, full down.
- 2) Displays of pressure:
- Internal SWC.
 - Interface cavity.
 - Argon supply.

11.21.2 DESIGN DESCRIPTION

11.21.2.1 General

The SWC is an enclosed, sealed, and shielded cylindrical cask (as shown in Figure 11.21-1) used to transport radioactive waste containers and CCCs from the IEM Cell to the RSB CLS for transfer of the container to the DSWC or the ISC, respectively. The SWC is capable of transporting a waste container

(containing six non-fuel core components or ~27 ft³ of radioactive waste) or a CCC (loaded within the limits of FSAR Appendix H) from the IEM Cell to the CLS. SWC components include the body, the cask closure valve, the hoist assembly with grapple, the lift fixture, and the control console.

The overall size of the SWC is 23 ft in length (with the lift fixture in the raised position) and ~5 ft in diameter. The assembly weights ~90,000 lb and consists mainly of a thick-walled steel cylinder with an internal hoist assembly at the upper end and a closure valve at the lower end. A separate console is used to control the internal hoist and cask argon system.

In a typical waste container transfer operation, the SWC is transported horizontally into the Reactor Containment Building (RCB) via the Multi-Purpose Rail Transporter (MPRT) or Maintenance Equipment Transport System (METS), raised to the vertical attitude with the crane, removed from the MPRT or METS and positioned/bolted onto the IEM Cell 28-in. valve. Next the SWC is inerted using the SWC argon system. The cask and IEM Cell 28-in. valves are opened, and the SWC grapple is lowered to engage the radioactive waste container. The waste container is then hoisted into the SWC. The valves are closed, and the SWC is returned to the MPRT or METS for transport through the airlock into the Reactor Service Building. Once inside the RSB, the SWC (with non-sodium-wetted radioactive waste) is again raised to the vertical attitude, removed from the MPRT or METS, and positioned/bolted onto the Disposable Solid Waste Cask (DSWC) floor valve at the Cask Loading Station (CLS). When so positioned, the cask closure valve and DSWC floor valve are opened. The waste container is lowered into the DSWC, and the grapple is disengaged and returned to the SWC. The valves are then closed, and the SWC is returned to its storage location.

During the fuel offload process, a clean, empty, and inerted CCC is loaded into the SWC at the SWTP and the SWC is inerted. The SWC is transported horizontally into the RCB via the MPRT. The SWC is then raised to the vertical attitude with the polar crane, removed from the MPRT, and positioned/bolted onto the IEM Cell on the 40-in. offset adapter plate on the 9 ft 6 in. valve. The SWC valve and ceiling valve are opened and the SWC lowers the empty CCC

into one of the CCC Storage Fixture locations on the maintenance turntable. After the SWC grapple is raised, the 9 ft 6 in. valve and the SWC valve are closed. The SWC could remain in place or be removed for further operations.

After a CCC is loaded with irradiated, cleaned fuel assemblies or Ident 69 pin containers and the CCC cover has been sealed and leak-tested, the SWC grapples the CCC in the CCC Storage Fixture and fully raises it into the SWC. The 9 ft 6 in. valve and the SWC valve are then closed. The SWC is transported horizontally out of the RCB via the MPRT and positioned at the CLS where the CCC is lowered into the ISC.

Although normal access of CCCs to the cell will be through the 9 ft 6 in. valve as described above, the SWC can also transfer CCCs through the 28-in. valve.

11.21.2.2 Cask Body

The cask body provides the radiation shielding and pressure boundary for the waste container (containing core components/radioactive waste) or the CCC (containing either irradiated, cleaned fuel assemblies and/or Ident 69 pin containers). The outside surface is 52.5 in. in diameter at the lower end and tapers to 39.25 in. in diameter at the upper end. A 22-in. diameter bore extends the full 12-ft length of the 65,000-lb cask body. Flanges at each end of the body interface with the hoist/chain locker assembly at the upper end and the valve assembly at the lower end.

The inside surface of the cask body is smooth to facilitate cleaning and to protect the cask body from attack by corrosive-cleaning agents, sodium, etc. Longitudinal stainless steel guides on the inside surface position the containers and prevent them from rubbing against the cask inside diameter.

11.21.2.3 Cask Valve Assembly

The SWC valve assembly is a gate valve that can only be operated with the valve in the horizontal position because of the massive weight of the gate. Actuation of the valve gate is accomplished manually by rotating a hand-wheel that, in turn, rotates a system of spur-gears, which rotate a parallel set of ball-screw shafts to drive the gate-mounted ball-screw nut to open or close the valve. The spur-gear train and the ball-screw assembly provide the mechanical advantage required to translate the 3000-lb gate.

The assumption was made that the RSB walls are gone and that the wind and missiles directly strike the SWC without any mitigation. The analysis shows that neither the 175 mph wind load or tornado missiles are sufficient to topple the SWC when it is in the vertical position. The analysis also shows that the effects of the tornado missiles on the SWC are some shallow penetration of the body of the cask but with negligible structural response. The containment is not breached; therefore, there are no problems associated with this event.

11.21.3.4 Seismic Analysis

The SWC is Category I equipment when handling fuel and was therefore analyzed accordingly. The following are the criteria used in the analysis.

SWC in the RCB: The building and crane structures do not fail.

SWC in the RSB: The building and crane structure stands but the crane can fail. Therefore, the SWC could be dropped if in the process of being moved.

In the RCB where the SWC cannot be struck by the building or crane but it can be dropped, the seismic accident (faulted) loadings, in general, induce rather modest stresses within the SWC. The most significant seismic-induced stresses occur when the SWC is hanging from the overhead crane. This event represents the most significant stress condition for the lifting fixture and the upper trunnions, but this stress remains within the design criteria limits. Dynamic analysis was also performed to ensure that the SWC does not impact Seismic Category I equipment in FFTF, including the equipment airlock or adjacent buildings or the liquid waste tank car.

The only vulnerable area of the SWC during any of the specified accident conditions is the hoist enclosure. Therefore, the design of the SWC includes a mechanism to lock and seal the grapple in the raised position, thus isolating the cask contents from the hoist enclosure except during the loading and unloading operations.

The analysis shows that with spent fuel in the SWC, and the SWC closure valve closed and the grapple raised and locked, there will be no release of any particulates under any of the specified accident conditions.

The other condition that was addressed is during the loading and unloading when the grapple is not locked and sealed and the lower valve is open. During this condition the SWC is always bolted to the facility. The analysis shows that under any of the seismic conditions the SWC will not disengage from the facility nor will the main cask body be penetrated by a projectile. The hoist, however, can be sheared off, allowing the container of six or seven spent fuel core components to fall. There are impact limiters in all the facilities except the IEM Cell with which the SWC interfaces, when handling fuel, that will prevent any container failure and minimize potential damage to the fuel.

The SWC was analyzed for seismic loadings when it is being transported on the MPRT and found to be acceptable.

When the SWC is on the test stand or if it is transported on METS, it will not contain fuel. Therefore, a Seismic Category I evaluation was not performed.

11.21.3.5 Radiation Analysis

The SWC shielding was designed to limit the steady-state external dose rate to 10 mrem/h at 3 ft from any cask surface except for the bottom of the cask closure valve, which is not to exceed 20 mrem/h at the valve surface when the cask is in the vertical position. During waste transfer operations, including radiation from irradiated fuel in the IEM Cell the transient dose rate due to streaming through mating valves and interfaces at the cask or valve surface is not to exceed 200 mrem/h.

Analysis, using seven spent fuel assemblies (fuel at maximum 250 watts and decayed for a minimum of 4 years) as a worst case, shows that the design will meet the 10 mrem/h limit for the cask and the 20 mrem/h limit at the valve surface. Additional shielding was added to the chain locker, grapple, and cask valve to satisfy the 200 mrem/h limit during waste transfer operations. Analysis indicates that the 200 mrem/h limit may be exceeded by the cask valve just above the installed shield blocks, possibly requiring that area to be administratively controlled during a transfer operation. When actual IEM Cell transfer operations begin, administrative procedures will be used to perform radiation measurements during the waste container lifting activities to confirm the adequacy of the valve shield design.

Administrative procedures used to determine actual dose rates each time the IEM Cell 10-ft valve or the 28-in. valve is opened in conjunction with SWC operations will provide a barrier to exclude personnel, as required, to minimize radiation exposure. Operation of the SWC over the IEM Cell will not impact the 18-kW cell cooling limit.

11.21.3.6 Thermal Analysis

The SWC is designed to transport up to seven fuel assemblies or six Ident 69 pin containers which meet the following limits:

- a) the maximum decay heat of any fuel assembly or pin container is 250 watts and
- b) the total decay heat in a CCC shall be limited to 1500 watts (maximum SWC capacity).

The SWC is designed to transport the fuel in a vertical or horizontal attitude with natural convection and radiative cooling. With loads up to 1500 watts, analysis indicates that the maximum non-fuel component temperature will be well below the 800°F limit and that the maximum surface temperature of the cask will be below 180°F when exposed to direct sunlight in still air at 115°F. Maximum temperature for a fueled core component (center fuel assembly cladding temperature) temperature will not exceed 900°F if decay heat is 250 watts or less. The CCC loadings specified above were for the original analyses. FSAR, Appendix H may provide different loadings provided they are bounded by the above loading.

11.21.3.7 Confinement Boundary

The SWC forms a part of the IEM Cell liner boundary, or the boundary of the ISC at the CLS, when the interfacing facility valve is open to the cask. The cask meets Seismic Category I requirements when at these locations. The SWC will be leak-tested when attached to the IEM Cell or to the ISC at the RSB CLS before the interfacing facility valve is opened.

11.21.3.8 Acceptance Test

The SWC acceptance test shall be performed to confirm the design adequacy of the hoist/grapple system, the argon system which is used at FFTF and to perform interface fit checks between the SWC and the SWTP, CCC, RSB CLS, ISC,

Multi-Purpose Rail Transporter (MPRT), and the IEM Cell. The SWC hoist/grapple and the inerting capabilities are to be tested at each interface as applicable. The SWC/IEM Cell interface will be leak-tested before the connecting valves are opened. Testing includes verification of interlock features pre-programmed in the logic controller, hoist grapple remote operation with the CLEM auxiliary control box from the IEM Cell, and the SWC hoist/grapple. The test procedure documents the radiation surveys performed with a radiation source to verify shield design adequacy.

11.21.3.9 Fire Analysis

The SWC is constructed of non-combustible metal which, except for electrical equipment and seals, does not constitute a fire hazard. Fire fighting equipment for electrical and sodium fires is available at the stations where the SWC is operated. The chance for a fire is highest when actual transfers are in progress. During this time the building inerting gas supply, argon at the FFTF, is connected to the SWC, thus maintaining a continuous inert atmosphere within the SWC containment barrier.

Following placement of components in the SWC, the gas supply is removed from the SWC. The SWC design ensures that in-leakage of air would be minimal. The SWC is then transported to the next station without an inerting gas makeup supply.

11.21.3.10 Loss of Electrical Power

The loss of electrical power during SWC operation does not constitute a hazard. If power is lost the operation safely stops and the cask remains inert. When power is restored the operation continues.

11.21.3.11 Loss of Facility Argon

The loss of facility argon would not constitute a hazard. The cask would remain inert. Operations may be interrupted depending on the task being performed but would safely resume when facility argon was reestablished.

11.21.3.12 Loss of Radioactive Argon Vent System

The loss of the Radioactive Argon Vent System would not constitute a hazard. The cask would remain inert. If a pressure buildup occurred, the excess pressure would safely vent through a HEPA filter. Operations may be interrupted depending on the task being performed but would safely resume when the vent system was returned to operation.

11.21.3.13 SWC Inert Gas System Failure

Failure of the SWC inert gas system would not constitute a hazard. If the inert gas system pressure supply fails and pressure inadvertently builds in the cask, the pressure relief valve will open automatically before the cask design pressure is reached and safely vent the excess gas through a HEPA filter. If the inert gas system fails to initially pressurize the cask, spent fuel would not be placed in the cask until the system was repaired.

11.21.3.14 Hoist System Failure During CCC Transfer to/from SWC

Failure of the hoist system causing the CCC with fuel to stop part way into the SWC would be of little consequence because the SWC is open to the connecting interface and has argon supply and exhaust. Repair work could proceed on the hoist, using proper contamination control techniques.

11.21.4 EVALUATION

11.21.4.1 System Design Evaluation

The SWC is designed to transfer non-fuel sodium-free radioactive waste from the IEM Cell to the RSB CLS for deposit into a DSWC. In addition, in support of the Fuel Offload Project, the SWC is designed to transfer:

1. Clean, empty CCCs from the SWTP in the RSB to the IEM Cell. While it is expected that the SWC will normally be mounted to the 40-in. offset adapter plate on the 10-ft valve, it can also mate with the 28-in. valve on the IEM Cell.

2. CCCs loaded within the limits of FSAR, Appendix H, from the IEM Cell to the RSB CLS.

SWC transfers through the Equipment Airlock will use the MPRT. At the RSB CLS, the CCC will be transferred to an ISC.

Analysis shows the cask meets the Seismic I requirements of JABE-WADCO-02 (except as modified by Regulatory Guide 1.92⁽²⁾) and locally developed spectrum response curves.

The cask also has been shown by analysis to:

- Withstand a vertical 5-ft drop on any bottom corner or on the flat bottom of the closure valve.
- Withstand rotational falls following corner drops.
- Withstand normal operating load conditions when the cask is transported by the MPRT or METS, lifted by a crane, or subjected to internal pressure.
- Withstand a tornado and tornado missiles.
- Satisfy spent fuel and waste temperature, and cask surface and cask body temperature limits.
- Limit the radiation dose rate to 10 mrem/h at 3 ft from any cask surface except the bottom surface of the cask closure valve, which is limited to 20 mrem/h when the cask is loaded.

The design of the CCC ensures that the cask is safely subcritical ($\leq 0.95 k_{eff}$ even with optimum interspersed water moderation) when it is loaded within the limits of FSAR, Ch. 20 and FSAR, Appendix H.

All load-bearing members of the hoist and grapple assemblies, lift fixture, and upper trunnion meet the requirements of DOE-RL-92-36.⁽³⁾ The SWC pressure boundary that includes the SWC body, valve, and hoist assembly cover was designed in accordance with ASME Code, Section VIII, Division 1 guidelines,⁽⁴⁾ but does not contain an ASME code stamp.

CONTENTS (Cont'd)

	<u>Page</u>
13.3.5 ORGANIZATIONAL CONTROL OF EMERGENCIES	13.3-31
13.3.5.1 Normal Plant Organization	13.3-31
13.3.5.2 Facility Emergency Organization	13.3-32
13.3.5.2.1 Direction/Coordination	13.3-32
13.3.5.2.2 Plant (Facility) Staff Emergency Assignments	13.3-34
13.3.5.3 Augmentation of Facility Emergency Organization	13.3-36
13.3.5.3.1 Supplementary Staff	13.3-36
13.3.5.3.2 Local Services Support	13.3-40
13.3.5.4 Coordination with Participating Agencies	13.3-40
13.3.6 EMERGENCY MEASURES	13.3-44
13.3.6.1 Activation of Emergency Organization	13.3-44
13.3.6.2 Assessment Actions	13.3-44
13.3.6.2.1 Surveillance of Instrumentation	13.3-44
13.3.6.2.2 Surveillance of Radioactive Systems Integrity	13.3-46
13.3.6.2.3 In-Plant Radiological Surveys	13.3-46
13.3.6.2.4 Fast Flux Test Facility Site and 400 Area Boundary Surveys	13.3-46
13.3.6.2.5 Environs Surveys and Monitoring	13.3-46
13.3.6.2.6 Data Reporting, Reduction, and Analysis	13.3-47
13.3.6.2.7 Interviewing Witnesses or Evacuees	13.3-47
13.3.6.2.8 Assessment Result Communications	13.3-47
13.3.6.3 Corrective Actions	13.3-48
13.3.6.3.1 Fire Control	13.3-48
13.3.6.3.2 Repair and Damage Control	13.3-50
13.3.6.3.3 System Control	13.3-50
13.3.6.4 Protective Actions	13.3-50
13.3.6.4.1 Protective Cover, Evacuation, and Personnel Accountability	13.3-50
13.3.6.4.2 Use of Protection Equipment and Supplies	13.3-56
13.3.6.4.3 Offsite Contamination Control Measures	13.3-58

CONTENTS (Cont'd)

	<u>Page</u>
13.3.6.5 Aid to Affected Personnel	13.3-59
13.3.6.5.1 Emergency Personnel Exposure Criteria	13.3-59
13.3.6.5.2 Decontamination and First Aid	13.3-62
13.3.6.5.3 Medical Transportation	13.3-63
13.3.6.5.4 Medical Treatment	13.3-63
13.3.7 EMERGENCY FACILITIES AND EQUIPMENT	13.3-65
13.3.7.1 Emergency Control Centers	13.3-65
13.3.7.1.1 Fast Flux Test Facility Incident Command Post (ICP)	13.3-65
13.3.7.1.2 Emergency Operating Center	13.3-66
13.3.7.2 Communications Systems	13.3-66
13.3.7.3 Assessment Facilities	13.3-72
13.3.7.3.1 Onsite Fast Flux Test Facility Systems and Equipment	13.3-72
13.3.7.3.2 Facilities and Equipment for Monitoring Outside the Fast Flux Test Facility Site	13.3-73
13.3.7.4 Protective Facilities	13.3-73
13.3.7.5 First Aid and Medical Facilities	13.3-74
13.3.8 MAINTAINING EMERGENCY PREPAREDNESS	13.3-74
13.3.8.1 Training	13.3-74
13.3.8.2 Drills	13.3-74
13.3.8.3 Review and Updating of the Plan and Procedures	13.3-75
13.3.8.4 Emergency Equipment and Supplies	13.3-75
13.3.9 RECOVERY	13.3-75
13.3.10 APPENDIX	13.3-79
13.3.10.1 Agency Agreement Letters and Interfacing Emergency Plans	13.3-79

D. Kennewick General Hospital

This facility, located in Kennewick, Washington, serves as a backup to Kadlec Hospital; its services shall be limited to activities normally performed at Kennewick General.

E. Our Lady of Lourdes Hospital

This facility, located in Pasco, Washington, serves the same backup function as does Kennewick General Hospital.

13.3.7 EMERGENCY FACILITIES AND EQUIPMENT

13.3.7.1 Emergency Control Centers

13.3.7.1.1 FFTF Incident Command Post (ICP)

Rooms 138, 139, and 140 in the FFTF Control Building are habitable under most unusual occurrence conditions and are the designated FFTF ICP, with a backup designated by the Building Emergency Plan for the 400 Area.

13.3.7.1.2 Emergency Operating Center

The RL-EOC is headquarters for the DOE-RL and contractors management staff for civil defense purposes. Equipment descriptions and basic operations are delineated in the "Hanford Emergency Response Plan," DOE/RL-94-02 published in May 1995.

13.3.7.2 Communications Systems

Communications systems are used to transmit the information that an emergency condition has been detected by sounding an alarm. Standard signals are used to alert personnel to an emergency. The use of standard signals throughout the Hanford Site helps to avoid confusion, particularly for personnel who enter numerous areas and facilities. The signals are a howler, gong, steady siren, and steady ringing telephone to signal high radiation, fire, evacuation, and crash alarm, respectively. The steady ringing telephone (crash alarm) is a special telephone circuit between key emergency stations in the 200, 300, and 400 Areas.

There are several communications systems that can be used for reporting an emergency regardless of the point of origin or the nature of the emergency but that are not necessarily restricted to emergency use. Within the 400 Area these include the following:

- The FFTF PAX-telephone system, which has stations within the reactor facility

- Crash-Alarm telephone system, which rings directly at the Patrol Operations Center (POC)
- General Plant telephone, which is connected throughout the Hanford Site and offsite through the Richland exchange
- Radio, which is in the FFTF ICP and Control Room which can access the same frequencies as Patrol.

These systems are supplemented by vehicle radio units, walkie-talkie equipment, and messengers.

For situations that arise at times other than normal working hours, a residential crash alarm telephone system can be used to report an emergency condition. This system operates in the same way as the Hanford Site crash alarm telephones, but it is connected to residences of selected DOE-RL contractor personnel in Richland and is activated by the PEO in the POC.

A. Fire Detection and Alarm

Fire protection systems are located throughout the FFTF complex. These systems detect and/or extinguish fires while alerting building occupants by sounding a gong. The systems also provide a coded signal to the 300 Area Fire Station.

In addition to the preceding functions, an annunciator in the Control Room is activated. Fire detectors in areas that are vital to FFTF operating continuity will activate specific area annunciators in the Control Room. Instructions to evacuate areas and zones (or take other actions), in addition to evacuating the immediate vicinity of the local alarm, can then be transmitted over the public address system and/or the FFTF Plant telephone system. On receipt of the fire alarm signal by the Hanford Fire Department, the PEO is notified by emergency telephone or by radio. The PEO notifies selected representatives of Westinghouse Hanford management, DOE management, and management of other

Hanford Site contractors. During normal working hours, notification is by Plant telephone system using emergency call lists. At other hours, the notification may be by using call lists or by the residential crash alarm system via the commercial telephone system in Richland. The PEO at the Federal Building activates the system, which simultaneously rings residential phones of selected contractor personnel (including Westinghouse Hanford representatives) in Richland. During regular work hours, the first available manager on the Westinghouse Hanford call list will be notified by the PEO by Plant telephone. The manager contacted will provide appropriate additional notifications within Westinghouse Hanford. This can be accomplished by directing the 300 Area Patrol Radio Room Operator to activate the Crash Alarm System and pass along a specific message.

B. High Radiation Monitor and Alarm

Work locations where significant quantities of fissionable materials are stored or handled can present a radiation hazard. Such locations are equipped with radiation-sensitive detectors that provide alarms in the work location if high radiation is detected and annunciated in the Control Room. The RCB and the RSB operating floors (550-ft level) are provided with radiation-sensitive detectors that will initiate an automatic evacuation of these areas by activating the High Gamma Alarm System. A complete description of radiation monitors at the FFTF is included in SDD.96.⁽¹⁴⁾

C. Evacuation

An automatic personnel evacuation of the RCB and/or RSB is initiated whenever the High Gamma System alarms are tripped in either or both of these areas.

A signal is transmitted to FFTF personnel by a steady siren for a plant evacuation over the public address system when manually actuated from the Control Room as authorized by the FFTF ED.

The signal will result in an evacuation of the FFTF and may evolve from local conditions or from information received from sources external to the plant via communications networks previously described. The Public Address (PA) System, PAX-telephone system, and the Plant Radio System also provide for continuity of communication with personnel who take cover and remain at the FFTF during a general evacuation. If the cause of the evacuation is of FFTF origin, the POC officers are notified that evacuation is in progress. The other communication systems previously mentioned would be available and used to alert personnel to the emergency conditions.

Conditions arising elsewhere, which may indicate that evacuation of the FFTF is needed, will be communicated to the FFTF ED via the networks previously mentioned. Most likely, the initial signal would be via the Crash Alarm Telephone System. The other systems are available as alternates. The FFTF ED would proceed using the siren, the intercom, and possible other alternates (e.g., Radio, PAX-telephone and messengers) to evacuate portions or all of the FFTF complex.

In an evacuation, Operations crew personnel will place the Plant in a safe condition. Personnel will evacuate the building via the closest exit route and proceed to the parking lot Staging Area, avoiding the immediate area of the emergency as much as possible, and report to their Staging Area Manager. The Staging Area Managers will then report any missing personnel to the Staging Area Director who will notify the FFTF ED.

The FFTF ED will determine alternate evacuation actions and alternate routes according to the emergency situation. For Plant evacuation under adverse weather conditions, coordination with the DOE-RL will be necessary with respect to information about routes that are open and areas that are to be avoided, since this information is received by the DOE-RL.

D. Other Emergencies

Emergencies may arise from other causes. In addition to the signals that automatically alert personnel to emergency conditions, there are the following systems.

- (1) Monitoring of the exhaust air effluent provides continuous indication of release of radioactivity. The system alarms in the Control Room at a predetermined level specified in the operating procedures. Action to be taken is also specified in the emergency procedures.
- (2) Gamma radiation monitoring equipment in work locations provides an alarm in the Control Room and at the location of the detector. The alarm level is preset for Control Room annunciation in accordance with operating procedures.
- (3) In case an emergency occurs for which no site alarm exists, the public address (PA) system will be used to alert personnel. The PA may be used, for example, to alert personnel in office areas to remain inside the building and close all doors and windows, if an accidental airborne release from a neighboring Hanford Site facility threatens the 400 Area.

After personnel have been alerted that there is an emergency condition, communication needs change to the vital role of stabilizing the condition(s) giving rise to the emergency, controlling the movement of personnel, and generally aiding as a tool in the control and recovery stages.

The Federal Building, in which the DOE ECC is located, has telephone access to other State and Federal agencies for use in an emergency. There is also the ability to communicate with Washington State Highway Patrol by radio. A teletype provides direct private communication with

- The Interim Storage Cask (ISC) is impacted
- The CLS elevator fails
- The CLS elevator, ISC, CLS cover, DSWC FV, and SWC fall and impact the CLS cell floor
- The CLS floor absorbs the impact

The Impact limiter on the CLS elevator limits the loading on all components to acceptable limits and no release of radioactive material or unacceptable radiation field occurs.^{(9) (10)}

15.2.1.7.2 Drop of DSWC FV on the CLS Cover

A. Identification of Cause and Accident Description

If the DSWC FV were to be dropped from a height up to 24 ft for any reason, it would be bounded by the SWC drop case (SWC wt. 90,000 lb vs DSWC FV wt. 25,000 lb).

15.2.1.7.3 Drop of ISC on the CLS Elevator

A. Identification of Cause and Accident Description

The ISC is administratively controlled to a lift height of 4 ft over the CLS 550-ft elevation. When the ISC contains fuel and is lifted, the closure plug has been installed, sealed, and tested. If the ISC were to be dropped from this height for any reason the following is the sequence of events that would occur:

- The CLS elevator is contacted
- The CLS elevator fails (by shearing of the four support pins)
- The CLS elevator and ISC fall and contact the CLS floor at the 518 ft elevation
- The CLS floor absorbs the impact

The impact limiter on the elevator limits the loading on the ISC and CCC to acceptable levels, minimizes potential damage to the fuel, and no release of radioactive material or unacceptable radiation field occurs.⁽⁹⁾⁽¹⁰⁾

15.2.1.7.4 Drop of CCC from the SWC

A. Identification of Cause and Accident Description

If the CCC were to be dropped from the SWC, at its maximum height, into the ISC for any reason the CLS elevator that supports the ISC could fail. The ISC and the CCC could then fall to the CLS floor. The CLS floor would absorb the impact and the impact limiter on the CLS elevator would limit the loads such that no release of radioactive material would occur.⁽⁹⁾⁽¹¹⁾ However, because the ISC closure plug would not have been installed when this drop could occur, an intense radiation field would result. The radiation field is restricted to a beam by the bore of the ISC and results in about 1,100 rem/hr at the 550-ft level of the RSB located some 18 ft above the top of the ISC. The closest individual is assumed to be standing at the edge of the CLS in the 1,100 rem/hr radiation field. Following an initial delay of 5 seconds, the individuals are assumed to walk rapidly (4 miles per hour) away from the CLS. It is assumed that the radiation field decreases with the square of the distance from the edge of the CLS. This is conservative because the radiation beam would be restricted by the bore of the ISC and the walls of the CLS and radiation fields away from the edge of the CLS would result only from scattered radiation. The resultant maximum dose is less than 2 rems, which is much less than the 25 rem allowable for an unlikely event per WHC-CM-4-46.⁽¹⁶⁾

15.2.2 EX-REACTOR FUEL STORAGE (INTERIM DECAY STORAGE)

15.2.2.1 Loss of Cover Gas Pressure Boundary During Normal Operation

A. Identification of Causes and Accident Description

A failure of the IDS cover gas pressure boundary could potentially release the design basis activity of 500 Ci of ¹³³Xe and other less important radioactive noble gases into the containment atmosphere.

PREFACE

The Fast Flux Test Facility (FFTF) Final Safety Analysis Report (FSAR) and Technical Specifications were originally developed for FFTF operating conditions.

The FSAR and Technical Specifications have been revised to accommodate facility operations subsequent to receipt of the Shutdown Order. This revision was developed based on the following FFTF conditions:

1. There are no fuel assemblies in the reactor vessel.
2. The 2.4 kV power to the Primary Heat Transport System pump main motors is electrically disabled.
3. The decay heat of each assembly or pin container is less than 1.4 kW.

CONTENTS (Cont'd)

<u>Limits</u>	<u>Page</u>
17.3/4.7.2.4 Test Assembly Conditioning Station (TACS)	17-46
17.3/4.7.2.5 Core Component Conditioning Station (CCCS)	17-47
17.3/4.7.2.6 Bottom Loading Transfer Cask (BLTC)	17-48
17.3/4.7.2.7 (Deleted)	
17.3/4.7.3 Fuel Storage Facility	17-50
17.3/4.8 Earthquake (Deleted)	
17.3/4.9 Experiments (Deleted)	
17.3/4.9.1 (Deleted)	
17.3/4.9.2 (Deleted)	
17.3/4.9.3 (Deleted)	
17.3/4.9.4 (Deleted)	
17.3/4.10 Fire (Deleted)	
17.5 DESIGN FEATURES	17-53
17.5.1 Site	17-55
17.5.1.1 Site Boundary	17-55
17.5.1.2 Flood Control	17-55
17.5.2 Containment	17-56
17.5.2.1 Configuration	17-56
17.5.2.2 (Deleted)	
17.5.3 Reactor Core	17-57
17.5.3.1 Fuel Assemblies	17-57
17.5.3.2 Control Rod Assemblies and Drive Mechanisms	17-59
17.5.4 Reactor Heat Transport System	17-61
17.5.4.1 Design Pressure and Temperature	17-61
17.5.4.2 Decay Heat Removal	17-61

CONTENTS (Cont'd)

<u>Limits</u>	<u>Page</u>
17.5.5 (Deleted)	
17.5.6 (Deleted)	
17.5.7 Interim Decay Storage	17-63
17.5.8 (Deleted)	
17.5.9 Fluid Volumes for In-Containment Systems	17-64
17.6 ADMINISTRATIVE CONTROLS	17-65
17.6.1 Responsibility	17-67
17.6.2 Purpose	17-67
17.6.3 Implementation	17-67
17.6.4 Technical Specification VIOLATIONS and NONCONFORMANCES	17-68
17.6.4.1 Technical Specification VIOLATIONS	17-68
17.6.4.2 Technical Specification NONCONFORMANCES	17-69
17.6.5 Changes to the Technical Specifications	17-71
17.6.6 Waiver to the Technical Specifications	17-71
17.6.7 Organization	17-71
17.6.7.1 Plant Operations	17-71
17.6.7.1.1 Watchstanding	17-72
17.6.7.1.2 Operator Training and Qualification	17-73
17.6.7.1.3 Procedures and Documentation	17-74
17.6.7.1.4 Criticality Control	17-75
17.6.7.1.5 Plant Change Control	17-75
17.6.7.2 Safety Analysis and Reviews	17-75
17.6.7.2.1 FFTF Safety	17-76
17.6.7.2.2 Experiment Reviews	17-76
17.6.7.2.3 BWHC Safety Review Board	17-76
17.6.7.2.4 (Deleted)	

CONTENTS (Cont'd)

<u>Limits</u>	<u>Page</u>
17.6.7.2.5 FFTF Quality Assurance	17-77
17.6.8 Radiation Protection	17-77
17.6.9 Surveillance and In-Service Inspection	17-77

CONTENTS (Cont'd)

<u>BASES</u>	<u>Page</u>
17.2 SAFETY LIMITS AND SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS	17-79
17.2.1 SAFETY LIMITS	17-81
17.2.1.1 (Deleted)	
17.2.1.2 Reactor Heat Transport System	17-81
17.2.1.3 (Deleted)	
17.2.1.4 Fuel Storage Facility (FSF)	17-82
17.2.2 SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS	17-83
17.2.2.1 (Deleted)	
17.2.2.2 Reactor Heat Transport System Actuation Limits and Time Constants	17-83
17.2.2.3 (Deleted)	
17.3/4 LIMITING CONDITIONS FOR OPERATION, ACTION, AND SURVEILLANCE REQUIREMENTS	17-85
17.3/4.0 GENERAL	17-87
17.3.0.1 Limiting Conditions for Operation and Action Statements	17-87
17.3.0.2 Declaring Equipment INOPERABLE	17-87
17.3.0.3 Equipment Removal from Service	17-88
17.3.0.4 (Deleted)	
17.3.0.5 LCO Format	17-89
17.4.0 Surveillance Requirements	17-90
17.4.0.1 Performance of Surveillance Requirements	17-90
17.4.0.2 Surveillance Prior to an LCO Becoming Applicable	17-92
17.4.0.3 (Deleted)	

17.3/4.7 REFUELING AND FUEL HANDLING

17.3/4.7.1 IN-VESSEL HANDLING

17.3/4.7.1.4 ASSEMBLY SERIAL NUMBER IDENTIFICATION

LIMITS

17.3.7.1.4 LIMITING CONDITIONS

In-vessel handling operations shall be terminated unless the serial number of each assembly transferred into or out of core by the IVHM is measured and found to be consistent with the refueling plan.

APPLICABILITY: Whenever fueled assemblies are in the core.

ACTION

- a. If an assembly has been transferred into or out of the core and not identified, IVHM fuel handling operations in that sector shall be suspended, pending a SAFETY REVIEW.
- b. If an error in serial number identification is found; movement of core components and test assemblies (other than those necessary to identify the cause) shall be suspended until the cause is found and corrected.

17.4.7.1.4 SURVEILLANCE REQUIREMENTS

For each assembly transferred into or out of core, the identification via the IVHM shall VERIFY that the assembly serial number agrees with that specified in the loading plan.

17.3/4.7 REFUELING AND FUEL HANDLING

17.3/4.7.1 IN-VESSEL HANDLING

17.3/4.7.1.5 DEGREE OF SUBCRITICALITY

LIMITS

17.3.7.1.5 LIMITING CONDITIONS

There shall be no fuel assemblies in the reactor vessel.

APPLICABILITY

At all times.

ACTION

With the specification not met, in-vessel handling shall be suspended pending a SAFETY REVIEW.

17.4.7.1.5 SURVEILLANCE REQUIREMENTS

The Detailed Refueling Plan shall ensure that this specification is met at all times.

17.3/4/7 REFUELING AND FUEL HANDLING

17.3/4.7.2 EX-VESSEL HANDLING

17.3/4.7.2.1 CLEM TRANSFER

LIMITS

17.3.7.2.1 LIMITING CONDITIONS

- a. Core components and test assemblies shall have predicted decay power <10 kW at the planned time of transfer from the reactor vessel.
- b. Any transfer that involves irradiated fuel pins in CLEM where the pins are not immersed in sodium shall be limited to assemblies with less than 0.60-kW decay heat, unless the limit is confirmed by a SAFETY REVIEW to be higher for the particular configuration.

APPLICABILITY: At all times.

ACTION

If a fuel assembly or component in the CLEM exceeds these limits, the assembly or component shall be placed in the nearest location capable of accepting it.

17.5 DESIGN FEATURES

17.5.3 REACTOR CORE

17.5.3.2 CONTROL ROD ASSEMBLIES AND DRIVE MECHANISMS

The reference safety and control rods contain boron carbide. The following design features are applicable to the reference absorber assemblies and to all drive mechanisms. Any modification to the reference control rod subsystem, which includes the absorber assemblies and associated driving mechanisms, must not change these design features:

- a. Deleted.
- b. Series 1.5 and Series 2 Absorber Assemblies have the following lifetimes:
 - Row 3 Primary Rods -- 15.0×10^{22} n/cm² ($E > 0.1$ MeV) axial average fast neutron fluence delivered to the peak duct corner.* This limit applies for Row 3 primary rods operated in the shimmed configuration (≥ 30 inches withdrawn), and conservatively applies to full-out (36 inches withdrawn) conditions as well.
 - Row 5 Secondary Rods -- 15.0×10^{22} n/cm² ($E > 0.1$ MeV) axial average fast neutron fluence delivered to the peak duct corner.*

* Requires rotation per following prescribed schedule. Without rotation, the fluence limit is 11.8×10^{22} n/cm².

If First Rotation Performed at _____:	Then Second Rotation to be Performed at _____:
$5.0 \leq \phi t < 7.0$	$8.8 \leq \phi t \leq 10.8$
$7.0 \leq \phi t < 9.3$	$10.8 \leq \phi t \leq 12.8$
$9.3 \leq \phi t < 11.8$	Not Required

17.5 DESIGN FEATURES

17.5.3 REACTOR CORE

17.5.3.2 CONTROL ROD ASSEMBLIES AND DRIVE MECHANISMS (Cont'd)

- c. Series 1.5 and Series 2 absorber pins have the following design parameters:

	<u>Series 1.5</u>	<u>Series 2</u>
Cladding outside diameter	0.474 in.	0.474 in.
Cladding thickness	0.051 in.	0.046 in.
Pellet diameter	0.352 in.	0.362 in.
BOL diametrical gap	0.020 in.	0.020 in.
Pellet density	92% TD	92% TD
Fill gas composition and pressure	helium, 1 atmos at room temp	

17.6 ADMINISTRATIVE CONTROLS

17.6.1 RESPONSIBILITY

The Fast Flux Test Facility (FFTF) is operated by B&W Hanford Company (BWHC) pursuant to a sub-contract between Fluor Daniel Hanford, Inc. (FDH) and BWHC. FDH is the prime contractor for the U.S. Department of Energy Contract Number DE-AC06-96RL13200. Under the provisions of the BWHC sub-contract, B&W Hanford Company is responsible for the safe operation of the FFTF.

17.6.2 PURPOSE

The purpose of the Technical Specifications is to define the limits/operating envelope within which the FFTF must be maintained.

17.6.3 IMPLEMENTATION

The BWHC President delegates authority and assigns specific responsibility for meeting contract requirements to his staff. The FFTF Project Director has been delegated this authority and responsibility for the FFTF and its associated facilities, i.e. the Fuel Storage Facility (FSF), the Sodium Storage Facility (SSF), the Maintenance and Storage Facility (MASF), and the 400 Area Interim Storage Area (ISA).

Note: MASF and SSF have separate safety documentation from the FFTF.

MASF is a radiological facility with an Auditable Safety Analysis.

SSF is in standby; when operational, it will also be a radiological facility. FSF and the ISA are part of FFTF and covered by the FFTF FSAR.

The FFTF Project Director is responsible for compliance with the Technical Specifications, including implementation of all of the actions, surveillances, etc., necessary to ensure that these limits are not exceeded.

17.6 ADMINISTRATIVE CONTROLS

17.6.4 TECHNICAL SPECIFICATION VIOLATIONS AND NONCONFORMANCES

17.6.4.1 TECHNICAL SPECIFICATION VIOLATIONS

DEFINITION

A VIOLATION of the requirements of these specifications will be deemed to have occurred when:

1. The requirements of a "Safety Limit," Section 17.2.1, are not met,
2. The "Safety System Actuation Limits and Time Constants," Section 17.2.2, are not met and the action is not invoked, or
3. An LCO of an "a" type (see Section 17.3.0.5.a) is not met.

ACTION

In the event that a VIOLATION of the requirements of these specifications occurs, proceed as follows:

- a. Return the Plant to compliance as soon as possible and terminate the affected operation, for example:
 - VIOLATION of LCO 17.3.7.2.5 requires a termination of all fuel handling,
 - VIOLATION of Safety Limit 17.2.1.2 requires termination of all plant activities (including IEM Cell), or
 - If an Actuation Limit is exceeded, only the affected operations require termination;
- b. Declare a Technical Specification VIOLATION;
- c. Immediately notify the Occurrence Notification Center (ONC) of the VIOLATION; and

17.6 ADMINISTRATIVE CONTROLS

17.6.4.1 TECHNICAL SPECIFICATION VIOLATIONS (Cont'd)

ACTION (Cont'd)

- d. Written reports shall be made as required in DOE Order 232.1.
- e. Deleted.
- f. When the cause and effect of the VIOLATION have been determined and corrective action taken, the President, B&W Hanford Company will recertify to Fluor Daniel Hanford, Inc. and to DOE-RL, readiness to resume operation. This certification shall document the bases and justification for resuming operation. Release to resume operation will be at the direction of DOE-RL and Fluor Daniel Hanford, Inc.

17.6.4.2 TECHNICAL SPECIFICATION NONCONFORMANCES

DEFINITION

A NONCONFORMANCE of the requirements of these specifications will be deemed to have occurred when:

- 1. The Design Feature Requirements (see Section 17.5) are not met,
- 2. The Administrative Control Requirements (see Section 17.6) are not met,
- 3. The ACTION statement is not invoked when it is discovered that the plant is outside the conditions specified by a "b" type LCO (see Section 17.3.0.5.b), or
- 4. The Surveillance Requirements are not met (except due to instrument INOPERABILITY when the Action has been invoked per Section 17.4.0.1).

17.6 ADMINISTRATIVE CONTROLS

17.6.4.2 TECHNICAL SPECIFICATION NONCONFORMANCES (Cont'd)

ACTION

In the event that a NONCONFORMANCE of the requirements of these specifications occurs, proceed as follows:

- a. The Action statement shall be invoked (unless compliance with the LCO has already been restored);
- b. If the NONCONFORMANCE was a Surveillance Requirement not met, immediately perform the Surveillance or implement the Action;
- c. Immediately notify DOE of the NONCONFORMANCE; and
- d. Written reports shall be made as required in DOE Order 232.1.

17.6 ADMINISTRATIVE CONTROLS

17.6.5 CHANGES TO THE TECHNICAL SPECIFICATIONS

All changes to the Technical Specifications shall be approved by DOE prior to implementation.

17.6.6 WAIVER TO THE TECHNICAL SPECIFICATIONS

Waivers to suspend various portions of the Technical Specifications when necessary for performance of special activities such as acceptance testing or irradiation testing may be granted. Waivers shall be approved by DOE. A waiver shall be for an explicit activity or defined time duration.

17.6.7 ORGANIZATION

The President of BWHC shall be supported in meeting his responsibility under Section 17.6.1 by organizations and functions as follows:

17.6.7.1 PLANT OPERATIONS

The FFTF Project Director shall be responsible to the President of BWHC for safe operation of the facility. He shall be responsible for ensuring that the following functions are adequately implemented.

17.6 ADMINISTRATIVE CONTROLS

17.6.7.1.1 WATCHSTANDING

The FFTF Project Director shall ensure that the number of qualified watchstanders on watch is adequate to operate the plant safely. He shall consider plant conditions in determining watchstation assignments.

The following watchstations shall be manned by qualified watchstanders:

Watchstation (Shutdown)	Number of Watchstanders
Shift Operations Manager (SOM)	1
Control Room Shutdown Watch	1
Ex-Control Room Qualified Watchstanders ^(a)	4
Total	6
Refueling - Fuel Handler ^(a)	1
- IEM Cell Operator ^(b)	1

(a) One qualified watchstander may be the Fuel Handler. Watchstation assignments will be specified in FFTF administrative procedures.

(b) IEM Cell Operator required when handling fuel in the IEM Cell or when forced cooling of fuel in the IEM Cell is required.

17.6 ADMINISTRATIVE CONTROLS

17.6.7.1.1 WATCHSTANDING (Cont'd)

Exceptions to the above list of minimum qualified watchstanders may be made as provided in the following requirements:

1. The Shift Operations Manager is authorized, on an emergency basis (personal or plant related), to allow temporary deviations from the stated minimum levels. The Control Room Shutdown Watchstation shall be manned, even if the SOM is required to take over in this position. This authorization shall not permit the SOM to excuse watchstanders when their reliefs have not yet checked in, but rather to excuse current watchstanders due to valid, serious emergencies.
2. Affected evolutions shall be limited as much as reasonable and practical to account for the reduced manning.
3. A replacement operator shall be called immediately and shall be in the plant within 2 hours.
4. The Operations Manager, FFTF Project Director, facility Safety Manager and the DOE operations representative shall be notified.
5. All above actions, and the arrival of the replacement watchstander, shall be logged in the Control Room log.
6. The event shall be reviewed at the next meeting of the Plant Review Committee. No other reports are necessary.

17.6.7.1.2 OPERATOR TRAINING AND QUALIFICATION

FFTF Operations personnel shall be trained and qualified in accordance with applicable DOE directives. FFTF equipment shall be operated by qualified operators or under the supervision of qualified operators.

17.6 ADMINISTRATIVE CONTROLS

17.6.7 ORGANIZATION

17.6.7.1.3 PROCEDURES AND DOCUMENTATION

Operation and maintenance activities shall be performed in accordance with approved written procedures. Plant design, operating and safety documentation shall be maintained as necessary to facilitate safe operation of FFTF. The procedures and other documentation shall be subjected to appropriate review, approval, and verification prior to use. The FFTF Project Director shall ensure that up-to-date copies of documents are available for operator use as necessary. Included shall be procedures and documentation which:

- a. direct plant operation in normal, infrequent, and casualty situations,
- b. provide administrative control of critical plant components (e.g., locked valves and key control switches),
- c. provide administrative controls to alert the operators to abnormal or potentially hazardous situations (e.g., danger and caution tags, jumper control),
- d. direct periodic preventive maintenance and calibration,
- e. direct maintenance and modification of the plant,
- f. provide for systematic testing of newly installed or modified equipment prior to normal operation,
- g. direct emergency response to accident conditions,
- h. direct conduct of EXPERIMENTS, and
- i. direct handling of radioactive materials and radiation protection.

17.6 ADMINISTRATIVE CONTROLS

17.6.7 ORGANIZATION

17.6.7.1.4 CRITICALITY CONTROL

Nuclear material shall be controlled in accordance with applicable DOE directives and FSAR Chapter 20, FFTF Criticality Specifications.

17.6.7.1.5 PLANT CHANGE CONTROL

Changes or waivers made to FFTF Equipment, design features, procedures and documentation shall be controlled so that they do not adversely affect plant safety. This control incorporates:

- a. Review and appropriate levels of approval for proposed changes.
- b. Control of the plant while the change is being made to avoid personnel and equipment hazards.
- c. Testing to assure that modified equipment or components perform as designed.
- d. Revision of FFTF documentation to reflect the plant modification.

17.6.7.2 SAFETY ANALYSIS AND REVIEWS

The following functions shall assist the President, BWHC, and the FFTF Project Director in carrying out their responsibility for safe operation of the plant and for performing or obtaining adequate safety analysis and reviews associated with plant operation.

17.6 ADMINISTRATIVE CONTROLS

17.6.7.2.1 FFTF SAFETY

An organization, reporting directly to the FFTF Project Director, shall be maintained to assist the FFTF Project in implementing Industrial Safety, Industrial Hygiene, Fire Protection and Nuclear Safety programs meeting DOE and Project Hanford requirements. The FFTF line organizations are responsible for safe operation of the FFTF.

17.6.7.2.2 EXPERIMENT REVIEWS

BWRC shall conduct an independent review of all EXPERIMENTS and advise the FFTF Project Director as to their safety. This review shall be conducted prior to EXPERIMENT insertion into the reactor and shall be a multi-disciplinary review with participation by organizations representing at least the following disciplines: engineering, EXPERIMENT assembly, quality assurance, safety analysis, and reactor operations.

17.6.7.2.3 BWRC SAFETY REVIEW BOARD

A board shall be maintained within BWRC which provides reviews of Safety Basis Documents that require DOE approval. The board shall assess the technical adequacy and quality of the documents prior to submittal to Fluor Daniel Hanford and DOE-RL. The board shall be composed of members from the major BWRC projects who are knowledgeable of the safety documentation system.

17.6 ADMINISTRATIVE CONTROLS

17.6.7.2.5 FFTF QUALITY ASSURANCE

An organization shall be maintained to assist the FFTF line organizations in the implementation of a problem prevention-oriented Quality Assurance (QA) program and to verify its implementation. The FFTF line organizations are responsible for the quality operation of the FFTF.

17.6.8 RADIATION PROTECTION

Radiation protection shall be provided consistent with the requirements of 10 CFR 835, "Occupational Radiation Protection."

17.6.9 SURVEILLANCE AND IN-SERVICE INSPECTION

A Surveillance and In-Service Inspection program shall be established in accordance with DOE-approved WHC-SD-FF-SISI-006⁽¹⁰⁴⁾ (Formerly HEDL-MG-89), FFTF SISI Requirements.

17.3/4.7 REFUELING AND FUEL HANDLING

17.3/4.7.1 IN-VESSEL HANDLING

17.3/4.7.1.5 DEGREE OF SUBCRITICALITY

BASES

17.3.7.1.5 BACKGROUND/DERIVATION OF LIMITS

The limit of no fuel in the reactor vessel eliminates the need for neutronic considerations and limits relating to fuel integrity. In addition, DOE approved reducing the FFTF Hazard Category from 1 to 2 based on maintaining the reactor vessel void of fuel.

APPLICABILITY

Because the limit of no fuel in the reactor vessel is one of the conditions upon which the Technical Specifications are based (see Preface), this limit is applicable at all times.

ACTION

The standard action whenever criticality related limits are exceeded is to totally stop all operations (i.e. no recovery or backing out is permitted until a technical review is performed). That is the action specified in this LCO and the SAFETY REVIEW would provide that technical review and provide a safe recovery plan.

17.4.7.1.5 SURVEILLANCE REQUIREMENTS

All refueling at FFTF is controlled and sequenced by the Detailed Refueling Plan. VERIFICATION will be made before the approval of the Detailed Refueling Plan that this specification is met.

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TABLE OF CONTENTS

	<u>Page</u>
PREFACE	18.0-iii
18.1 DEFINITIONS	18.1-1
18.1.1 FFTF	18.1-1
18.1.2 400 AREA	18.1-1
18.1.3 SITE BOUNDARY	18.1-1
18.1.4 1-1/2 MILE CONTROL DISTANCE	18.1-1
18.1.5 CHANNEL CALIBRATION	18.1-1
18.1.6 CHANNEL CHECK	18.1-2
18.1.7 CHANNEL FUNCTIONAL TEST	18.1-2
18.1.8 CHECK SOURCE TEST	18.1-2
18.1.9 MONITOR, MONITORING, MONITORED	18.1-2
18.1.10 NONCONFORMANCE	18.1-3
18.1.11 OPERABLE, OPERABILITY, INOPERABLE	18.1-3
18.1.12 OPERATING	18.1-3
18.1.13 MODE, OPERATIONAL MODE	18.1-3
18.1.14 SURVEILLANCE FREQUENCIES	18.1-4
18.1.15 UNRESTRICTED AREA	18.1-5
18.1.16 VERIFY, VERIFICATION	18.1-5

TABLE OF CONTENTS (Cont')

	<u>Page</u>
18.2 ENVIRONMENTAL PROGRAM	18.2-1
18.2.0 Environmental Monitoring	18.2-1
18.2.1 Environmental Surveillance	18.2-1
18.2.2 Effluent Monitoring	18.2-2
18.3 LIMITING CONDITIONS, ACTION, AND SURVEILLANCE REQUIREMENTS	18.3-1
18.3.0 GENERAL	18.3-1
18.3.0.1 Limiting Conditions and Action Statements	18.3-1
18.3.0.2 Declaring Equipment INOPERABLE	18.3-1
18.3.0.3 Removing Equipment from Service	18.3-1
18.3.0.4 Limiting Condition NONCONFORMANCE	18.3-2
18.3.0.5 Performance of Surveillance Requirements	18.3-3
18.3.1 Liquid Effluents	18.3-4
18.3.1.1 Nonradioactive Liquids	18.3-4
18.3.1.2 Radioactive Liquids	18.3-6
18.3.2 Airborne Effluents	18.3-7
18.3.2.1 Radioactive Gases	18.3-7
18.3.2.2 (Deleted)	
18.3.2.3 Nonradioactive Gases	18.3-10
18.3.3 Solid Waste	18.3-15
18.3.3.1 Radioactive Solid Waste	18.3-15
18.4 Design Features	18.4-1
18.4.1 Site	18.4-1
18.4.2 Thermal Impacts	18.4-2
18.4.3 Hydraulic Impacts	18.4-3
18.4.4 Meteorology	18.4-4

TABLE OF CONTENTS (Cont'd)

	<u>Page</u>
18.5 ADMINISTRATIVE CONTROLS	18.5-1
18.5.1 Responsibility	18.5-1
18.5.2 Organization	18.5-1
18.5.3 Review and Audit	18.5-1
18.5.4 Action to be Taken if a Limiting Condition is Exceeded	18.5-2
18.5.5 Procedures	18.5-2
18.5.6 Plant Requirements	18.5-3
18.5.6.1 Routine Reports	18.5-3
18.5.6.2 Reportable Occurrences	18.5-4
18.5.6.3 Requests for Design, Operational or Procedural Changes	18.5-4
18.5.7 Environmental Specifications Waivers	18.5-5
18.5.8 Records Retention	18.5-5
18.6 BASES	18.6-1
18.7 REFERENCES	18.7-1
<u>APPENDIX A - WAIVERS</u>	18.A-i

TABLES

<u>Table</u>	<u>Page</u>
18.3.2-1 FFTF Effluent Exhaust Paths	18.3-9

18.3 LIMITING CONDITIONS, ACTION, AND SURVEILLANCE REQUIREMENTS

18.3.1 LIQUID EFFLUENTS

LIMITS

SURVEILLANCE REQUIREMENTS

1. Samples of the 400 Area process sewer shall be collected and analyzed in accordance with requirements of the State Waste Discharge Permit, ST 4501.
2. ANNUALLY the chemical inventory and releases associated with the plant activities will be reported per the requirements of the Emergency Planning & Community Right to Know Act (EPCRA)³, Sections 312 and 313.

18.3 LIMITING CONDITIONS, ACTION, AND SURVEILLANCE REQUIREMENTS

18.3.1 LIQUID EFFLUENTS

18.3.1.2 RADIOACTIVE LIQUIDS

LIMITS

LIMITING CONDITIONS

1. Radioactive liquid wastes resulting from FFTF operations shall be collected and transported to the appropriate processing facility.
2. No direct radioactive liquid shall be released to the soil in the vicinity of the Hanford 400 Area.

APPLICABILITY: All MODES.

ACTION

If a liquid release to the ground occurs,

- a. Stop the release
- b. Report the release per the requirements specified in the Environmental Compliance Manual, WHC-CM-7-5.
- c. Evaluate the reportability of the release per 18.5.6.2a, Reportable Occurrences.
- d. Restore the system(s) to, as near as possible, prerelease conditions.
- e. Clean up the spill site as directed by the Environmental Protection group.

SURVEILLANCE REQUIREMENTS

1. Record the volume, curie content, and principle radionuclides of each shipment of liquid waste transported to the Hanford Site Waste Storage Area.

18.3 LIMITING CONDITIONS, ACTION, AND SURVEILLANCE REQUIREMENTS

18.3.2 AIRBORNE EFFLUENTS

FFTF Airborne effluents include both radioactive and nonradioactive pollutants. In accordance with the Clean Air Act requirements, airborne radionuclide emission rates and associated offsite Effective Dose Equivalents received by the Maximally Exposed Individual are required to be estimated, based on periodic confirmatory measurements.

18.3.2.1 RADIOACTIVE GASES

LIMITS

APPLICABILITY: All MODES.

SURVEILLANCE REQUIREMENTS

1. Collect four - seven day record particulate samples per year from each of the exhaust paths listed in table 18.3.2-1. Record samples to be analyzed for gross beta and alpha activity.
2. Collect four - seven day record effluent tritium samples per year from FFTF-CB-EX (C-350) exhaust path. Record samples to be analyzed for tritium activity.
3. Gross beta, gross alpha and tritium activity analysis data will be reviewed by the Hanford Site effluent monitoring/oversight group to ensure compliance with regulatory requirements.
4. The particulate sampler and tritium sampler flow instrumentation shall be calibrated at least annually.
5. Each of the exhaust paths listed in table 18.3.2-1, shall have its flow rate determined at following times, as a minimum:
 - a. Annually.
 - b. After any equipment or process modification that has the potential to significantly alter stack flow rates.

18.3 LIMITING CONDITIONS, ACTION, AND SURVEILLANCE REQUIREMENTS

18.3.2 AIRBORNE EFFLUENTS

18.3.2.1 RADIOACTIVE GASES

LIMITS

SURVEILLANCE REQUIREMENTS (Cont'd)

6. The flowrate data from the exhaust paths listed in table 18.3.2-1, measured in accordance with the previous requirement shall:
 - a. Be provided within 30 days of measurement to Hanford Site effluent monitoring/oversight group.
 - b. Be provided to FFTF Regulatory Compliance, for inclusion in the regulatory files.

18.3.2.1 RADIOACTIVE GASES

TABLE 18.3.2-1
FFTF EFFLUENT EXHAUST PATHS

<u>Exhaust Path I.D.</u>	<u>Associated Plant Monitoring Panel</u>	<u>Description</u>
FFTF-CB-EX	C-350(RE-96010)	Containment and Auxiliary Equipment Building East Combined Exhaust
FFTF-HT-TR	C-348(RE-96060)	Heat Transport System South Exhaust
FFTF-RE-SB	C-346(RE-96030)	Reactor Service Building Lower Exhaust

18.3 LIMITING CONDITIONS, ACTION, AND SURVEILLANCE REQUIREMENTS

18.3.2 AIRBORNE EFFLUENTS

18.3.2.3 NONRADIOACTIVE GASES

LIMITS

LIMITING CONDITIONS

Not applicable.

18.6 LIMITING CONDITIONS, ACTION, AND SURVEILLANCE REQUIREMENTS

18.6.2 AIRBORNE EFFLUENTS

18.6.2.1 RADIOACTIVE GASES

BASES

BACKGROUND/DERIVATION OF LIMITS

The annual potential Effective Dose Equivalent (EDE) received by the off-site nearest residence from any of the FFTF exhaust points listed in Table 18.3.2-1, has been determined to be less than 0.1 mrem/yr, as documented in EP-0894, Hanford Site Radionuclide National Emission Standards for Hazardous Air Pollutants Registered and Unregistered Stack (Powered Exhaust) Source Assessment.

The Surveillance Requirements are based on the requirements for exhaust points with annual potential EDE less than 0.1 mrem/yr, as contained in the Radioactive Emission Standards in CM-7-5, Environmental Compliance Manual.

Table 20.4-1 IEM Cell Locations may contain one of the listed Fuel Units or Containers										
Ident		Fuel Assy	Open 69	Closed 69	Open 1578	Closed 1578	Open 1578 Pin Basket	Closed 1578 Pin Basket	CCC With up to a MAX OF 7 Fuel Ass'ys OR 6-69s OR 5-69s & 2 Fuel Ass'ys ⁽¹⁾	Pins
15	Core Component Rec. Cont.	1		1		1				
89	Na Removal Station (T-100)	1		1				1		
17-1	Core Component Storage	1		1						
17-2	Core Component Storage	1		1						
39-1	Pin Container Storage	1		1						
39-2	Pin Container Storage	1		1						
41	Component Measuring	1								
42	Pin Weighing									50
116-B	Pin cont/Component Xfer	1		1		1		1		
116-D1	Pin Cont/Component Storage	1		1		1				
116-D2	Pin Cont/Component Storage	1		1		1				
116-D3	CCC Storage/Loading								1	
116-D4	CCC Storage/Loading								1	
116-D5	CCC Storage/Loading								1	
M-852	Pipe Lathe	1								
35-1	Disassembly	1								
35-2	Pin Basket Rack						1	1		
35-3	Pin Basket Rack						1	1		
116-A3	Pin Cont/Component Storage					1				
116-C	Pin Container Loading	1	1	1	1	1				

(1) When mixing pin containers and assemblies in the same CCC, a maximum of 5 pin containers may be in the CCC, and the balance of the 7 positions may be filled with fuel assemblies.

1.3 Fuel Assembly and Fuel Pin Handling

1.3.1 A total of no more than 2 (max) of the following:

- o Fuel assemblies
 - o Closed Ident 1578 pin containers containing fuel pins
 - o Closed Ident 69 pin containers containing fuel pins
 - o Ident 1578 pin baskets containing fuel pins
- shall be in locations other than those specified in Table 20.4-1, at any time.

- A. Only one loaded CCC, alone or in combination with one of the above fuel units, shall be in locations other than those specified in Table 20.4-1.

1.3.2 In addition to the limit of Section 20.4.2, Part 1.3.1 above, no more than five (max) "loose" fuel pins shall be in the cell at anytime. A "loose" fuel pin is a pin that does not meet one of the following criterion:

- A. In an assembly with the duct installed
- B. At the Disassembly Station and constrained at the bottom by a rail and at the top by either the assembly duct, the Pin Bundle Rotating Clamp, or the Upper Restraint Clamp.
- C. Within an Ident 1578 pin basket or Ident 69 pin container
- D. Attached to an end fitting (H-4-57320-23) and stored on one of the storage positions in the Pin Weighing System lower shroud.
- E. Attached to an end fitting (H-4-57320-23) and supported by the Pin Weighing System balance arm.

1.6 Limits for fueled Components handling at Station Ident 116-D and Ident 61.

Core Component Containers (CCCs) are the only approved Container for Assemblies, or Pin Containers, at the maintenance turn table.

1.6.1 CCCs, except as specified in 1.6.1.B, shall only be located in the designed storage of Ident 116-D3, D4, & D5.

A. Ident 116-D3, D4, and D5, are the only locations where a CCC with fuel is allowed to have it's Closure Lid off.

B. A single CCC may be placed into Ident 61 location, as an alternate transfer location, provided the following requirements are met:

- A Recovery Plan of Normal Cell Transfer Operations is developed and approved by Operations and Engineering.
- A Memo is signed by the Manager of The Independent Safety Organization and the Manager of FFTF Transition Project, agreeing to allow use of the Ident 61 location for transfer of CCCs containing fuel.
- The Ident 61 location will not be used for storage of any CCC containing fuel. Since the Ident 61 is not a fixed storage location, a CCC in this location is considered to be one of the two fuel units allowed outside of fixed storage locations

1.6.2 If the CCC's contain any fuel, they shall be moved only with the Closure Lid fully installed.

1.6.3 Each station at the Maintenance Turntable may contain up to the stated amount in TABLE 20.4-1.

A. Mixing of assemblies and up to five Ident 69 pin containers in the same CCC is permitted.

B. Ident 69 pin containers are not allowed in the center tube of the CCCs.

1.6.4 The 4 Ton Crane will be used in conjunction with the IEM Cell Pipe Lathe to remove the nozzle of an assembly so the assembly can be stored in the center tube of a CCC.

1.7 Moderator Control

1.7.1 No more than 20 gal (max) of free moderating liquid (exclusive of the SRS) shall be in the IEM Cell while fuel is present. A continuous inventory of moderating liquid is not required, provided the amount is maintained <5 gal.

1.7.2 There are no restrictions on containers that can be brought into the IEM Cell. Neither fuel pins nor fuel assemblies shall be placed in any container unless it has been approved for the handling or storage of fuel.

1.7.3 Fuel and moderating liquids shall not be combined in any container, with the exception of the Sodium Removal Station (T-100) and two (max) Water-Wash Fuel Pin Seal Tubes (H-4-62764). Any moderating liquid discharged from the Water-Wash Fuel Pin Seal Tubes shall be removed from the IEM Cell and shall not be transferred to any system in the Cell.

1.7.4 During transfers of water between the Demineralized Water System and T-101, at least one CIV (inner or outer) per SRS line CTMT penetration, must be closed.⁽¹⁷⁾

1.7.5 During recycle of water from T-102, through the Ion Exchange System, to T-101, at least one CIV (inner or outer) for each SRS line CTMT penetration, must be closed, if the floor grids are not all in place.

20.4.4 Criticality Prevention Specification 4718-1. Interim Storage Area.
Work Location: Interim Storage Area (ISA), 4718 Concrete Pad.

A. Fissionable Material Description

1. FFTF Fuel

1.1 Fuel Pins - - Only fuel pins contained in assemblies or pin containers specified in Section A.1.2.1 below shall be handled.

1.2 Assemblies

1.2.1 Type A assemblies

- A. Driver fuel assemblies, with fuel composition of ≤ 29.32 wt% Pu in the $\text{PuO}_2\text{-U}(\leq 0.72\% \text{U}^{235})\text{O}_2$, with fuel column dimensions of 0.200-in. dia X 36-in. length. The Pu^{240} content of Pu is ≥ 10.00 wt%. Each assembly contains 217 pins. This specification covers the fuel, either fresh or irradiated.
- B. Any and all fuel assemblies that are no more reactive than a Type A driver fuel assembly with the fuel composition as described in Section 20.4.4.A.1.2.1.A above.
- C. A list of Type A assemblies (meeting the requirements of section 20.2.2.A.1.2.1.A and/or B above), approved for loading into CCCs shall be maintained by the Criticality Safety Representative and with the Shift Operations Manager's copy of the FSAR, Chapter 20.

1.3. Closed Ident 69 Pin Containers Containing Fuel Pins

1.3.1 Closed Ident 69 pin containers containing no more than 217 Type A fuel pins.

1.3.2 Closed Ident 69 pin containers, containing any Type B fuel pins, must have an analysis performed, and be approved on a case by case basis.

B. Operations Involved

<u> </u>	Processing	<u> X </u>	Storage	<u> X </u>	Transporting within a location (ISA)	<u> X </u>	Transporting between buildings and locations (RSB and ISA)
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C. Brief Description of Operations and Equipment Involved

1. FFTF Fuel

1.1. The only FFTF fuel handling activities permitted by this specification are:

1.1.1 Transfer of fuel in the Interim Storage Cask (ISC) between the RSB and the ISA, and within the ISA.

1.1.2 The storage of fuel in the ISCs at the ISA.

1.2 The mixing of assemblies and up to five Ident 69 pin containers in the same Core Component Container (CCC) is allowed.

1.3. Ident 69 pin containers are not allowed in the center tube of CCCs.

D. Fuel Handling Limits and Controls

1. Limits for Criticality Safety

1.1 FFTF Fuel - - All FFTF fuel in transit and at the ISA shall be within sealed CCCs, sealed in ISCs, which are safe by their design features. Therefore, no batch limits or workstation limits are applicable at the ISA.

1.1.1 The only operations permitted in the ISA is the storage of fuel in approved casks and associated handling. No operations affecting the confinement boundaries are permitted.

1.1.2 Moderator Control - - No moderator is allowed within the CCC or the ISC.

2. Controls for Criticality Safety

2.1 General Spacing Limits

2.1.1 FFTF Fuel - - Because all required spacing is guaranteed by the ISC design, there are no applicable spacing limits.

2.2 Inventory

2.2.1 FFTF Fuel - - There is no limit on the number of ISCs allowed at the ISA.

A. A current inventory of ISCs and their loadings, shall be maintained.

2.3 Geometry

2.3.1 FFTF Fuel - - None.

2.4 Firefighting Restrictions

2.4.1 FFTF Fuel - - Firefighting at the ISA is rated as Category A.

E. Nonfuel Fissionable Material Limit

None Allowed.

F. Design Controls

1. Physical Design Restrictions

1.1 FFTF Fuel

1.1.1 Each storage location in the CCC will hold only 1 fuel assembly or one Ident 69 pin container.

A. Each CCC can hold up to:

1. 7 Type A fuel assemblies

or

2. 6 Ident 69 fuel pins containers.

3. The center tube of a sealed CCC will not accommodate a fuel pin container.

4. Mixing Type A fuel assemblies and up to 5 (max) Ident 69 fuel pin containers is allowed within the same CCC.

a. There are no loading restrictions other than an Ident 69 will not be placed in the center tube.

1.1.2 Each ISC will hold only 1 CCC.

possibility of increasing the total number of fuel units allowed in the Cell, and a possibility of exceeding Criticality Limits for the Cell. This location is also difficult to access, and other equipment has the possibility of interference when going into or out of this location. Extra attention is required to use this location, since there is a possibility of damage to either a CCC or other Cell equipment. The ID-61 is for transfer only and not to be used for storage of a loaded CCC.

- 1.6.2 The CCC Closure Lid has a handling socket built into the center location. All fasteners must be installed per the engineering design, before the CCC with fuel can be moved. There are no other approved means of moving CCCs containing fuel without the closure lid being installed. One CCC, a prototype has the handling socket built-in to the center of the CCC, and has no closure lid. This CCC will not be loaded into a ISC with fuel.
- 1.6.3 The CCC may contain 7 Type A fuel assemblies, 6 in the outer tubes, and 1 in the center tube. The closure lid has the Grapple Socket in the center tube, and is recessed into the tube about 8 inches. For an assembly to be placed into the center tube, the nozzle assembly must be removed. With full reflection and moderation, this configuration has a K_{eff} of less than 0.95. 6 Ident 69 containers with fuel may be placed in the 6 outer tubes. No Ident 69 containers are allowed in the center tube. 6 containers in the outer tubes will have a K_{eff} of less than 0.95 under all conditions. With the three CCCs filled with 7 DFAs and a DFA in Id-116-D1 and -D2, and assuming full moderation of the CCCs with hydrogenous material between the CCCs, the K_{eff} of this array is less than 0.92. The normal condition for CCCs is to be completely dry. To fill a CCC containing 7 assemblies, with water requires 94 gallons. There is a limit of 20 gallons (max) of free moderator in the Cell. Multiple contingencies or a DBE would be required to make this amount of water available. Under normal conditions, the K_{eff} of this array would be well below the 0.95 limit.

- A. Mixing assemblies and up to five Ident 69 pin containers in the same CCC (7 components max) is permitted for storage and transportation in the ISC, and storage of the ISC at the Interim Storage Area (ISA).
- B. Although a pin container will fit in the center tube, the closure lid will not fit on the CCC because the handling socket will interfere with the Ident 69 container in the center tube. This is bounded by the 7 assembly case dry because each container will not have any more pins than an assembly, and will probably be much less. The K_{eff} will be well below the 0.95 limit dry.

Seven Ident 69 containers with optimally arranged fuel pins with full moderation and reflection, have a K_{eff} greater than 0.935, but less than 1.0. To achieve this condition in Ident 116-D, would require a water level of at least 23 feet. If the CCC were to fall to the lower level upside down, and retain all containers, a level of 9.9 feet would be required. If the CCC were to fall to the lower level on it's side and retain all 69 containers, a level of 3.24 feet would be required. Maximum moderator level available to the Cell is 3.5 inches. The moderator prevention gridwork is 6.79 inches above the floor of the lower level of the Cell. This puts this scenario beyond the second contingency of the Double Contingency principle.

- 1.6.4 The IEM Cell pipe lathe will hold the bottom of the assembly when an assembly is in this location. The 4 ton crane must remain attached to the handling socket during the cutting operation, as the pipe lathe is not designed to hold the full weight of the assembly.

1.7 Moderation Control

In an unmoderated condition, there is insufficient fuel in the IEM Cell to go critical. The purpose of this section is to provide

restrictions affecting moderating material to ensure that the unintentional moderation of fuel is sufficiently unlikely such that it can be the second contingency preventing criticality (see the introduction to the basis for CPS 405-2).

- 1.7.1 With respect to this limit, "free moderating liquid" is considered to be hydrogenous liquids not constrained by permanent cell equipment or systems.

The following are the potential sources of moderating liquid from plant systems for fuel in the IEM Cell.

- A. Sodium Removal Equipment--A breach of the T-100 chamber or in-cell piping would permit up to approximately 40 gal of water to be released to the cell before an interlock would prevent further leakage. Failure of the interlock would permit (under the worst conditions) approximately 580 gal of water to be released. (The system was designed for a nominal 500 gal. Maximum capacity as measured, without voids in the tanks, is 580 gal. The normal volume is 516 gal. It would be reasonable to expect the volume to be as large as 550 gal.) This is the entire contents of the system. Connections to other water sources are closed during processing.

Each of the two Water-Wash Fuel Pin Tubes (H-4-62764) is physically restricted to less than 1 quart of liquid. A breach of both tubes would result in <0.5 gal of free liquid in the cell.

Should water be released, it would accumulate on the floor of the cell. A sump location on the floor has a 2-ft by 2-ft cross-sectional area and is approximately 2.5 in. deep. The lower portion of the Cell is 28 x 10 ft in dimension and would fill to a depth of about 3.4 in. if flooded by 580 gal

of water. Although the upper portion of the Cell has a smaller cross section, water would not accumulate there.

In very few places in the IEM Cell can water accumulate in a geometry favorable for effective moderation. The central portion of the maintenance turntable provides a favorable geometry, but the potential for water accumulation there is remote. A metal plate covers the cavity and would prevent water entry. In addition, holes near the base of the turntable provide drain outlets. No fissionable material should ever be in this cavity, and it is not considered possible for a criticality to occur there.

- B. Ion Exchange System -- The Ion Exchange System (IXS) loop and resin bed unit makes an additional 415 gal. (nominal) available to the SRS. However, administrative controls (closing the SRS Containment Isolation Valves during resin loop operation when the floor grids are not in place) preclude this additional water from entering the IEM Cell. Even if the CIVs remained open during operation of the IXS, it is highly unlikely that the transfer of water contained in the IXS, to the IEM Cell would occur, even during a DBE. This is because T-101, SRS water supply tank, must be pressurized to transfer water to the IEM Cell. T-101 must be vented to CAPS to receive water from the IXS. During a loss of electrical and/or pneumatic power the IXS isolation valves fail closed, precluding additional moderator transfer to the IEM Cell. Since the CIVs are the only part of the SRS that is seismically qualified to Category I, they must be closed during operation of the IXS, when the floor grids are not in place and fuel is in the Cell.
- C. Shield Windows -- The shield windows contain oil that might conceivably drain into the Cell. Rupture of a window is very unlikely (the windows are Seismic Category I) and would be independent of failure of other sources of moderator, except as a result of a major earthquake.

D. Cell Cooling System -- The argon atmosphere in the Cell is cooled by being pumped through coolers (heat exchangers) located 5 to 10 ft below the lower Cell floor. The two main Cell coolers are each comprised of 2107 straight tubes immersed in a coolant solution of 45% glycol and water. Each tube has a 0.709-in. ID and a 0.083-in. wall thickness. The two smaller subassembly coolers have somewhat smaller diameter tubes. All four coolers are designed to Seismic Category I criteria and contain alarmed leak detectors. Argon gas from the Cell passes through at least 80 ft of 30-in. ID piping before reaching the tubes of the main coolers. If a leak were to occur in the main coolers, the piping would have to fill with at least 3,000 gal of coolant before any liquid would reach the Cell. The total amount of water-glycol solution available in the closed system is about 33,000 gal. A leak into the subassembly cooling system would not have to fill as much piping before entering the Cell, but its entry into the Cell would be considerably more impeded by the design of the piping.

Although it is hypothetically possible for the water-glycol coolant to flood the Cell to at least 10 ft, such an occurrence is considered to be precluded for the FFTF Design Basis Event (DBE). The high integrity of the cooler design (i.e., Seismic Category I) makes the probability of failure very small. Small leaks are considered possible, but they would be detected and the flow stopped before any flooding of the Cell would occur.

If one postulates the rupture of one of the 0.7-in. ID tubes in the cooler, the maximum flow rate would be <0.7 gal/s, and this would decrease as the water level rose in the piping. Well over an hour would be required for water to reach the Cell from such a failure. The cooler leak alarm would sound and a control room alarm would signal a low chilled-water

level. Even a small leak would be quickly detected and terminated by isolation at the leaking cooler. Even if one further postulates that the leak is allowed to continue, an additional failure (such as might be caused by a DBE) is required to deposit the pins in the water. In any case, the operator may at any time terminate the leak by closure of the Seismic Category I chilled water containment isolation valves.

Operating Deck -- Inventories of chilled water and of Mobiltherm exist above the 550-ft level and could theoretically enter the Cell through a breach of the Cell boundary. These sources are seismically qualified and isolable remotely with containment isolation valves. Entry of liquid from these sources into the Cell would require failure of Seismic Category I piping as well as the Cell boundary. Such a compounding of events would require conditions beyond the design bases.

In addition, some moderating liquids are required in the Cell for normal operation (e.g., humidification, lubrication, and cleaning). At least 4.2 in. of water is required in the bottom of the Cell for a criticality to occur. As stated above, leakage of the 580 gal (max) from the SRS could cause a level of 3.4 in. to accumulate in the lower part of the Cell. If 20 gal of free moderating liquid were also to spill, the total level would be 3.5 in., still well below the minimum required to permit a criticality. It is not expected that nearly this much moderating liquid will ever be needed in the Cell. However, picking a (safe) limit of 20 gal permits a lower value to be chosen below which inventories are not required. Maintaining inventories of liquids used for such things as humidification control would be impractical.

- 1.7.3 The only locations in the Cell where moderating liquids come in contact with fuel during normal operations are the Sodium Removal Station (T-100) and the two Water-Wash Fuel Pin Seal Tubes (H-4-62764).

In T-100, a single fuel assembly or closed 1578 pin basket is placed in the chamber, and the chamber is sealed. The assembly or pin basket is eventually flooded with water, which is collected and recirculated through the system. It is physically impossible to place more than one assembly or closed 1578 pin basket in this chamber. Closed Ident 69s may be put in the SRS.

Each of the two Water-Wash Fuel Pin Seal Tubes (H-4-62764) is physically restricted to a single fuel pin and less than one quart of liquid. Transfer of moderating liquid from Water-Wash Fuel Pin Seal Tubes to any system in the Cell is not allowed to prevent a possible accumulation of fissionable material in the systems.

- 1.7.4 The Demineralized Water System is connected to the sanitary water system. Failure to close the CIVs during a water transfer between the demineralized water system and T-101 could allow water to exceed the amount of moderator needed to exceed one of the contingencies of the double contingency principal for the IEM Cell. The CIVs are seismic Category I and will prevent the moderator from reaching the Cell, if closed.
- 1.7.5 During recycle of water between T-102 and T-101, through the Ion Exchange System, up to 415 more gallons of moderator could be added to the SRS by off normal conditions. Closing the CIVs during recycle will prevent this additional moderator from reaching the IEM Cell under all conditions including DBE. If the floor grid system were completely installed in the IEM Cell, and the CIVs were not closed during a DBE, and all of the moderator from the SRS and IXS

were released into the Cell, the level of moderator in the Cell would be less than 6 in. The floor grid system will keep the fuel region of assemblies, pins, pin containers, and CCCs at least 6.79 in. above the IEM Cell Annex floor. This will prevent moderation of any fuel in the cell, so operating the IXS with the CIVs open is allowed, when the floor grid system is in place. This series of events is not considered credible.

2. Controls for Criticality Safety

- 2.1 The basic premise of this Criticality Prevention Specification is that for the two events, fuel falling to a suitable configuration and moderating liquid accumulating to a sufficient extent must each be independent and unlikely. While the conditions specified in this section will not, by themselves, be considered "violations" of the specification, they each increase the potential for one of the two events occurring. Therefore, fuel handling should be suspended until conditions are restored.

The notifications are required to ensure that proper corrective action is taken. The basis for each of the conditions is as follows:

- 2.1.1 The argon atmosphere, per se, is not significant to criticality safety. However, it is significant to the extent that the Cell boundary integrity is a necessary condition of an inert atmosphere. Should a significant amount of water be present on the 550-ft level, the Cell boundary would be a significant barrier preventing the water from entering the Cell.
- 2.1.2 This specification only addresses fuel in assembly and pin configurations. Because cladding failures could lead to unanalyzed conditions, they should be thoroughly investigated before fuel handling proceeds.

2.1.3 Because Section 20.4.2, Part D.1.3.1, permits loose fuel pins and assemblies, a dropped fuel pin, assembly, or container with fuel will probably not result in a violation. However, the occurrence must negatively reflect on the ability to maintain fuel in desired configurations. Therefore, the occurrence shall be reviewed prior to continuing normal fuel handling.

2.1.4 A leak in one of the IEM Cell coolers is a potential source of a significant amount of moderating material. Upon any indication that one of the coolers may have an integrity failure, fuel handling shall be suspended, pending satisfactory resolution of the problem.

2.1.5 None of the fuel restraints in the IEM Cell are seismically qualified. Therefore, if any seismic event is confirmed to have occurred at FFTF, a review of potential damage to the Cell shall be made.

2.2 Maintaining a current inventory using the standard inventory control records meets the requirements of WHC-CM-4-29.⁽¹⁾

E. Nonfuel Fissionable Material Limit

A limit of 15 g (max) is placed on the quantity of miscellaneous fissionable material permitted. This limit is completely independent of the limits on fuel pins and assemblies. That is, some or all of it may be moved within the IEM Cell with no spacing restrictions with respect to fuel present. Since this material has a maximum fraction critical of 0.03 under conditions of optimum moderation, it will not significantly increase reactivity in the Cell. This limit is intended primarily to cover fission chambers and neutron sources (should they be needed).

F. Design Controls

Each of the listed requirements is assured by equipment design.

1. Each location specified in Table 20.4-1 is limited by design, to either one (max) CCC, pin container, assembly, or the stated number of pins.
2. There are 2 types of Ident 1578 pin baskets in use. One has a 19 pin capacity, and the other has a 40 pin capacity. These numbers are less than the number of pins in a fuel assembly.
3. Core Component Containers are each limited to 7 (max) fuel assemblies each, and have a K_{eff} of less than 0.95 including interaction with other fuel.
4. A Floor Grid System is installed on the lower Cell floor (495' elev.) to prevent the fuel region of any pins or assemblies from being submerged in moderator. During a DBE, the SRS could be damaged and discharge up to 580 gallons of moderator into the Cell. This would result in a depth of 3.4 inches in the lower Cell area. The floor grid system is 6.79 inches above the floor. This is sufficient to prevent the fuel region of any pin, assembly, or CCC, from coming into contact with moderator. With 25 fueled components allowed in the Cell, and the Floor Grid System installed, a Criticality is not possible with the moderator available to the Cell (including the SRS, IXS, and free moderator for a total of 1,015 gallons, or <6 in depth.)
5. The Sodium Removal System (SRS) was originally designed with a maximum volume of 580 gallons. An Ion Exchange System (IXS), has been added to the SRS to decrease the amount of waste water generated by fuel washing in the IEM Cell. The IXS consists of 2 sets of 3 beds each, of which 1 set is normally lined-up for water cleaning. Water is transferred from T-102 through the beds into T-101. When cleaning is complete, the water has been transferred into T-101. The IXS holds 415 gallons of water nominal. This

additional moderator is prevented from reaching the IEM Cell by Engineered and Administrative controls. During the water cleaning process, the SRS CIVs are closed if the floor grids are not in place. After the cleaned water is transferred to T-101, the valves for the IXS are closed. These valves fail closed on loss of pneumatics or electrical. After IXS isolation valves are closed, the SRS CIVs are opened (if they were closed). Due to the hydraulic profile of the SRS, T-101 must be pressurized to transfer water to the IEM Cell Sodium Removal System, T-100 (this is where the actual washing of fuel takes place). To get any transfer of water from the IXS to T-101 or T-100, T-102 must be pressurized. Flow would initiate until T-101 and T-102 pressures equalized. Under normal conditions, even if the IXS isolation valves were left open, no water transfer from or to the IXS would occur because of check valves that prevent reverse flow, and system pressure that would prevent additional water from transferring. Further water transfer would require T-101 to be vented to CAPS, and thus would not transfer to T-100, because of the poor hydraulic profile.

6. There are 2 Water-Wash Fuel Pin Seal Tubes in the IEM Cell, and each will only hold 1 fuel pin, along with less than one quart of moderation liquid. If the amount of water in both tubes were to be released, no measurable increase in the moderator level in the cell will occur.

20.5.3 Technical Bases for CPS 403-1, Fuel Storage Facility

Work Location: This specification permits handling greater than 1/3 Minimum Critical Mass in the FSF.

A. Fissionable Material Description

1. Fuel Pins -- The FSF has been designed to store fuel assemblies and fuel pins within closed pin containers. Only fuel pins contained in either fuel assemblies or closed pin containers are permitted in the facility. The descriptions of the fuel pins permitted in fuel assemblies and pin containers are contained in CPS 405-1 (Section 20.4.1, Part A.1).
2. Originally, the fuel description of this specification (FSF) was identical to that of CPS 405-1 for the 550-ft level. Since that time, CPS 405-1 has been changed to accept Type B fuel assemblies. Because the FSF has not been analyzed for the more reactive Type B assemblies, they are excluded from use at the FSF. The description of Type A assemblies is the same as in CPS 405-1 and within the FSF analyses.
3. This section allows for handling Ident 69 and Ident 1578 pin containers in a manner equivalent to Type A fuel assemblies.

Criticality Prevention Specification 405-2, Part A.3, assures that pin containers are loaded in the IEM Cell such that they are no more reactive than a Type A assembly when unmoderated. The only difference between this section and that for the IEM Cell is that the IEM Cell allows the pin containers to be open.

For further details, see Section 20.5.2, Part A.3.

4. Nonfuel Fissionable Material -- This specification places a limit on fissionable material in fission chambers and unmoderated fissionable sources.

B. Operations Involved

1. FFTF Fuels

This CPS covers transportation of ISCs between the RSB and the ISA, transportation of ISCs at the ISA, and storage of ISCs at the ISA.

C. Brief Description of Operations and Equipment Involved

1. FFTF Fuel

The mixing of assemblies and up to five Ident 69 pin containers, in the same CCC, is permitted by the CPS. Analysis was performed for mixed loading of the CCC, inside of the ISC. Ident 69 pin containers are not allowed in the center tube of CCCs. An analysis was performed with the center tube containing an ID-69, and 6 ID-69s in the outer ring, and the K_{eff} was greater than 0.95.

D. Fuel Handling Limits and Controls

1. Limits for Criticality Safety

1.1 FFTF Fuel

The ISC is the only approved storage container for FFTF fuel at the ISA. All fuel, whether in assemblies or pin containers, must be within sealed CCCs, and the CCCs sealed within ISCs. The ISA is designed for storage of fuel only. The only work that may be performed at the ISA is maintenance to the outside of the ISCs only (seal testing, environmental cover, etc). Any work that will violate the integrity of the Cask or has the probability of violating the integrity of the cask is not allowed at the ISA. Even though analysis shows an infinite number of ISCs, with their approved loading, with full flooding inside and out will remain subcritical, moderators are not allowed in the ISC or CCC, as they are designed for dry storage.

Although the ISA is for storage of fuel, this limit is not meant to prevent the storage of empty casks, such as Disposable Solid Waste Cask (DSWC), ISC, T-3 and their related equipment.

2. Controls for Criticality Safety

2.1 General Spacing Limits

2.1.1 FFTF Fuel

Since the ISC is 83.75 inches in diameter, the closest that any two CCCs could come to each other at the ISA, is 62.75 inches edge to edge. The Nuclear Safety Limits, Section 20.3.5 states: "The spacing between two batches of fissionable material shall be at least one foot edge-to-edge when maintained by mechanical means;", the mechanical spacing guaranteed by the ISC meets this requirement.

2.2 Inventory

2.2.1 FFTF Fuel

Analysis has shown that an infinite array of ISCs, even with flooding inside the ISC and CCC, has a K_{eff} of less than 0.930, which is less than the 0.95 limit. It is clear from this that any number of ISCs can be safely stored at the ISA.

2.3 Geometry

2.3.1 FFTF Fuel

All geometry is guaranteed by the engineered design of the ISC, and no other requirements are needed.

A. Standard inventory control records maintained by FFTF Refueling will satisfy the inventory requirements for the ISA.

2.4 Firefighting Restrictions

2.4.1 FFTF Fuel

The ISC weight is 110,000 pounds empty. No known pressurized fire fighting media will rearrange or tip over the ISC, or violate any CPS, so Fire Fighting Category A is appropriate.

E. Nonfuel Fissionable Material Limit

The ISA is intended for the storage of Fuel only, thus no other fissionable material is allowed.

F. Design Controls

1. Physical Design Restrictions

1.1 FFTF Fuel

Each CCC has only 7 storage locations which will hold either one fuel assembly or pin container. Up to 7 assemblies are allowed in each CCC, but the assembly that goes into the center tube must have the nozzle modified to fit. The center storage location also has the handling socket recessed into it, so the assembly must be shortened in order for the closure lid to be installed and sealed. Six Ident 69 pin containers are allowed into a CCC. The center tube must remain empty. If a pin container were to be placed into the center tube, in the IEM Cell, the closure lid would not fit on, as the handling socket would interfere with the pin container. To allow a pin container into the center tube, the container must be shortened. This is prohibited administratively. Each ISC will only hold one CCC. The inside diameter of the ISC is 21 inches, and this precludes side by side double batching because the CCC has a diameter of 20 inches. The inside length of the

ISC is 150 inches. The total length of a CCC is 146 inches, so double batching by stacking one CCC on top of another is precluded.

Analysis has determined that mixing assemblies and Ident 69 pin containers in the same CCC is allowed, as long as the center tube contains an assembly or is empty, and a maximum of five pin containers are allowed in the CCC.

If the assumption is made that "hot cell rot" has occurred and the pin cladding and duct disintegrate at the ISA, a criticality would require the occurrence of the following contingencies:

- Flooding of the ISA must occur.
- The ISC outer seal must fail.
- The ISC inner seal must fail
- The CCC seal must fail.
- Optimization of the water/fuel geometry.

This combination of events far exceeds the double contingency principle used to assure criticality safety. The loss of any single one of these barriers would not result in loss of double contingency protection. If any one of these events does not transpire, the K_{eff} of the ISC and its contents will remain below 0.95 at the two sigma confidence level. This is not considered a credible scenario.

It is concluded from this evaluation that a Criticality Alarm System is not needed at the ISA.

20.7

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APPENDIX G
TRANSITION OF FFTF FROM OPERATING TO DEFUELED (SHUTDOWN) STATUS
CONTENTS

G.1	INTRODUCTION	G.1-1
G.2	TECHNICAL SPECIFICATION CHANGE JUSTIFICATION SUMMARY FOR SHUTDOWN	G.2-1
G.2.1	GENERAL SAFETY STRATEGY	G.2-1
G.2.2	PRESERVATION OF FUEL CLADDING	G.2-3
G.2.2.1	POTENTIAL MECHANISMS FOR BREACHING CLADDING INTEGRITY	G.2-3
G.2.2.2	JUSTIFICATION SUMMARY FOR PRESERVATION OF FUEL CLADDING TECHNICAL SPECIFICATION CHANGES	G.2-4
G.2.3	HEAT TRANSPORT SYSTEM	G.2-7
G.2.3.1	DISCUSSION OF CONTROL OF REACTOR COOLANT TEMPERATURES (HEAT REMOVAL REQUIREMENTS)	G.2-7
G.2.3.2	DISCUSSION OF SODIUM SPILL ANALYSES AND CONSEQUENCES	G.2-8
G.2.3.3	JUSTIFICATION SUMMARY FOR HEAT TRANSPORT SYSTEM TECHNICAL SPECIFICATION CHANGES	G.2-11
G.2.4	CONTAINMENT	G.2-23
G.2.4.1	DISCUSSION OF CONTAINMENT INTEGRITY REQUIREMENTS	G.2-23
G.2.4.2	JUSTIFICATION SUMMARY FOR CONTAINMENT INTEGRITY TECHNICAL SPECIFICATION CHANGES	G.2-26
G.2.5	INSTRUMENTATION	G.2-28
G.2.5.1	DISCUSSION OF POST ACCIDENT MONITOR REQUIREMENTS	G.2-28
G.2.5.2	DISCUSSION OF 1-E POWER REQUIREMENTS	G.2-29
G.2.5.3	DISCUSSION OF OTHER SAFETY RELATED INSTRUMENTATION	G.2-30
G.2.5.4	JUSTIFICATION SUMMARY FOR INSTRUMENTATION TECHNICAL SPECIFICATION CHANGES	G.2-31

APPENDIX G

TRANSITION OF FFTF FROM OPERATING TO DEFUELED (SHUTDOWN) STATUS

CONTENTS (CONTINUED)

G.3	TECHNICAL SPECIFICATION CHANGES DUE TO STANDBY OR RESTART PREPARATIONS	G.3-1
G.3.1	REACTOR VESSEL RELATED TESTING	G.3-1
G.4	REFERENCES	G.4-1

TABLES

G.2-1	1E ELECTRICAL SYSTEM LOADS NO LONGER REQUIRED	T-G.2-1
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APPENDIX G

TRANSITION OF FFTF FROM OPERATING TO DEFUELED (SHUTDOWN) STATUS

G.1 INTRODUCTION

FFTF received a permanent shutdown order on 12/15/93. To support an orderly transition to shutdown status, it was prudent to reduce the FFTF Technical Specifications to only those needed to protect health and safety. A large reduction of requirements was possible because the hazards presented by the FFTF reactor plant, which had been shutdown for in excess of 2 years and was partially defueled, were of a much more limited scope and consequence than those of an operating reactor.

The purpose of this Appendix is to summarize the justification for changes made to FFTF Technical Specifications, thereby demonstrating the reduction in safety issues due to plant shutdown. Exclusive of FSAR Chapter 17, the FSAR was not updated to reflect the changes outlined in this Appendix. Using this Appendix in conjunction with the FSAR, the reviewer can determine the current safety envelope and portions of the FSAR which are no longer applicable.

On 1/17/97 the Secretary of Energy approved a Memorandum of Decision directing that the Fast Flux Test Facility be maintained in a standby condition. Subsequent to that decision it is expected that some different configurations, evolutions, etc. may be required that would include changes to the Technical Specifications. Section G.3 of this appendix will be utilized for these standby/preparation for restart changes such that Appendix G continues to reflect the current safety envelope.

G.2 TECHNICAL SPECIFICATION CHANGE JUSTIFICATION SUMMARY FOR SHUTDOWN

G.2.1 GENERAL SAFETY STRATEGY

The FFTF is intended to conduct all operations in such a manner that the impact on the public and the environment is minimized. Normal activities shall not result in radiological consequences greater than those allowed in 10 CFR 20 and accident consequences shall be within those allowed in 10 CFR 100. Technical Specifications were put in place to ensure that the operation of FFTF at up to 400 Mwt thermal power would conform to these objectives. However, with the FFTF reactor having been shutdown for more than two years, the available radiological source term is dramatically reduced from that which existed during power operation. In addition, the current plant condition/configuration results in many of the events analyzed in the FSAR being impossible.

Since FFTF is to be defueled and not be operated critical again, it is no longer necessary to retain all of the Technical Specifications which were required to assure safe power operation. Rather, the stated objective of conducting all activities within the limits of 10 CFR 20 and 10 CFR 100 can be assured with a greater cost efficiency through a reduced set of Technical Specifications.

To implement this improved efficiency, the FFTF Technical Specifications are changed to delete areas where safety issues no longer exist and to modify or add specifications where needed to provide new safety margins. The following five general conditions represent the principle features (barriers) which provide the safety margins necessary to allow the Technical Specification changes:

1. Fuel has been removed from the reactor such that core k_{eff} is less than 0.95 with the three most reactive absorber assemblies removed from the core and will be subcritical with all absorber assemblies removed.
2. The control rod drive mechanism (CRDM) power supply system is electrically disabled.

3. The Primary Heat Transport System main motors 13.8 kV Power supply system is electrically disabled.
4. The total decay heat in the reactor vessel is less than 100 kW.
5. The decay heat of any assembly or pin container is less than 1.4 kW.

For ease of categorization, the resulting Technical Specification changes are divided into 4 general areas of applicability. Sections G.2.2 through G.2.5 below provide the justification summary for each general area. For additional information concerning the Technical Specifications which remain, see the respective Technical Specification Basis.

G.3 TECHNICAL SPECIFICATION CHANGES DUE TO STANDBY OR RESTART PREPARATIONS

G.3.1 REACTOR VESSEL RELATED TESTING

When the original Technical Specification reduction effort was put into effect, limits were put in place to ensure the safety of the fuel in the reactor vessel. Subsequently, all the fuel was removed from the Reactor Vessel. Upon receipt of the DOE direction for standby, there was a desire to exercise and test the reactor vessel head mounted components. One technical basis for the original reduction, stated in the Preface to the Technical Specifications, is "The control rod drive mechanism power supply system is electrically disabled."

Therefore, a new basis is being provided reflecting that there is no fuel in the reactor vessel. With that, the following prior preface conditions are removed:

- k_{eff} limit on the reactor. It is no longer applicable.
- Control Rod Drive Mechanism power supply disabled. With no fuel, the control rods have no neutronic safety function.
- The total decay heat in the reactor vessel less than 100 kw. This also is no longer applicable.

G.4 REFERENCES

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APPENDIX H

FAST FLUX TEST FACILITY FUEL OFFLOAD AND FUEL STORAGE IN THE INTERIM STORAGE AREA

CONTENTS

H.1	INTRODUCTION	H.1-1
H.1.1	FUEL PROCESSING EQUIPMENT AND FACILITIES	H.1-2
H.1.2	GLOSSARY: ABBREVIATIONS AND ACRONYMS	H.1-4
H.2	FAST FLUX TEST FACILITY (FFTF) FUEL OFFLOAD PROCESS DESCRIPTION	H.2-1
H.2.1	INTRODUCTION	H.2-1
H.2.2	PROCESS DESCRIPTION	H.2-1
H.2.2.1	Processing Fuel Components into the CCC	H.2-2
H.2.2.2	CCC Receipt, Loading, and Transfer to the ISC	H.2-4
H.2.2.3	ISC Receipt, Loading, and Transfer to the ISA	H.2-4
H.2.2.4	FFTF Fuel Offload Acceptance Testing	H.2-5
H.2.3	REFERENCES	H.2-5
H.3	400 AREA DRY CASK SPENT FUEL STORAGE	H.3.1-1
H.3.1	400 AREA INTERIM DRY CASK STORAGE AREA	H.3.1-1
H.3.1.1	Purpose of the Interim Storage Area (ISA)	H.3.1-1
H.3.1.2	General Site Description of the ISA	H.3.1-2
H.3.1.3	ISA Design	H.3.1-2
H.3.1.4	References	H.3.1-8
H.3.1.5	Bibliography	H.3.1-9
H.3.2	INTERIM STORAGE AREA (ISA) SPENT FUEL STORAGE SYSTEM REQUIREMENTS	H.3.2-1
H.3.2.1	GENERAL DESCRIPTION OF SPENT FUEL STORAGE SYSTEM	H.3.2-2
H.3.2.2	PRINCIPAL DESIGN CRITERIA	H.3.2-2
H.3.2.3	STRUCTURAL EVALUATION	H.3.2-10
H.3.2.4	THERMAL EVALUATION	H.3.2-13
H.3.2.5	CONFINEMENT BARRIERS AND SYSTEMS	H.3.2-14
H.3.2.6	SHIELDING EVALUATION	H.3.2-15
H.3.2.7	CRITICALITY EVALUATION	H.3.2-16
H.3.2.8	RADIOLOGICAL PROTECTION	H.3.2-16
H.3.2.9	OPERATIONS/TESTING/MAINTENANCE	H.3.2-18
H.3.2.10	FUNCTIONAL AND OPERATIONAL REQUIREMENTS	H.3.2-19
H.3.2.11	ACCIDENT ANALYSIS	H.3.2-20
H.3.2.12	RESPONSIBILITIES	H.3.2-22
H.3.2.13	REFERENCES	H.3.2-23
H.3.3	FAST FLUX TEST FACILITY (FFTF) SPENT FUEL STORAGE SYSTEM	H.3.3-1
H.3.3.1	FFTF STORAGE SYSTEM DESCRIPTION	H.3.3-1
H.3.3.2	PRINCIPAL DESIGN CRITERIA	H.3.3-9
H.3.3.3	STRUCTURAL EVALUATION	H.3.3-28
H.3.3.4	THERMAL EVALUATION	H.3.3-51
H.3.3.5	CONFINEMENT BARRIERS AND SYSTEMS	H.3.3-66
H.3.3.6	SHIELDING EVALUATION	H.3.3-76

CONTENTS (cont)

H.3.3.7	CRITICALITY EVALUATION	H.3.3-79
H.3.3.8	RADIOLOGICAL PROTECTION	H.3.3-80
H.3.3.9	OPERATIONS/TESTING/MAINTENANCE	H.3.3-83
H.3.3.10	FUNCTIONAL AND OPERATIONAL LIMITS	H.3.3-87
H.3.3.11	ACCIDENT ANALYSIS	H.3.3-100.1
H.3.3.12	RESPONSIBILITIES	H.3.3-118
H.3.3.13	REFERENCES	H.3.3-119
H.3.3.14	BIBLIOGRAPHY	H.3.3-124
H.3.4	TRIGA FUEL STORAGE SYSTEM	H.3.4-1
H.3.4.1	DESIGN DESCRIPTION	H.3.4-3
H.3.4.2	PRINCIPAL DESIGN CRITERIA	H.3.4-14
H.3.4.3	STRUCTURAL EVALUATION	H.3.4-26
H.3.4.4	THERMAL EVALUATION	H.3.4-40
H.3.4.5	CONFINEMENT BARRIERS AND SYSTEMS	H.3.4-43
H.3.4.6	SHIELDING EVALUATION	H.3.4-51
H.3.4.7	CRITICALITY EVALUATION	H.3.4-55
H.3.4.8	RADIOLOGICAL PROTECTION	H.3.4-56
H.3.4.9	OPERATIONS/TESTING/MAINTENANCE	H.3.4-60
H.3.4.10	FUNCTIONAL AND OPERATIONAL LIMITS	H.3.4-63
H.3.4.11	ACCIDENT ANALYSIS	H.3.4-72
H.3.4.12	RESPONSIBILITIES	H.3.4-94
H.3.4.13	REFERENCES	H.3.4-95
H.3.5	324 BUILDING SPENT FUEL STORAGE SYSTEM (This section will be provided at a later date.)	H.3.5-1

LIST OF TABLES

H.3.2-1	Design Bases Tornado Missiles	T-H.3.2-1
H.3.3-1	Natural Environmental Conditions	T-H.3.3-1
H.3.3-2	Maximum Solar Radiation Received from the Sun	T-H.3.3-2
H.3.3-3	Diurnal Ambient Temperature at the Hanford Site for the Peak Summer Month	T-H.3.3-3
H.3.3-4	Summary of Interim Storage Cask Normal Condition Load Combinations	T-H.3.3-3
H.3.3-5	Interim Storage Cask Confinement Boundary Normal Condition Allowable Stresses	T-H.3.3-4
H.3.3-6	Summary of Interim Storage Cask Load Combinations Hypothetical Accident Loads	T-H.3.3-4
H.3.3-7	Interim Storage Cask Confinement Boundary Accident Condition Allowable Stresses	T-H.3.3-5
H.3.3-8	Material Properties for Interim Storage Cask Thermal Analysis	T-H.3.3-6
H.3.3-9	Material Properties of Air at 1 atm	T-H.3.3-7
H.3.3-10	Material Properties of Argon at 1 atm	T-H.3.3-7
H.3.3-11	Material Properties of Helium at 1 atm	T-H.3.3-8
H.3.3-12	Material Properties of Type 1020 Carbon Steel	T-H.3.3-8
H.3.3-13	Material Properties of Type 304 Stainless Steel	T-H.3.3-9
H.3.3-14	Material Properties of Concrete	T-H.3.3-9
H.3.3-15	Material Properties of Fast Flux Test Facility Metal Oxide (MOX) Driver Fuel Assembly	T-H.3.3-10
H.3.3-16	Bounding Temperatures for Interim Storage Cask/Core Component Container Load Conditions	T-H.3.3-11
H.3.3-17	Material Data	T-H.3.3-13
H.3.3-18	Source Specification	T-H.3.3-13
H.3.3-19	Summary of Interim Storage Cask Dose Rates	T-H.3.3-14
H.3.3-20	Local Damage Prediction	T-H.3.3-14
H.3.3-21	Experimental Fuel Assemblies and Ident 69 Pin Containers Analyzed for Loading into a CCC/ISC	T-H.3.3-15

LIST OF TABLES (cont)

H.3.4-1	Principal Design Parameters for TRIGA Fuel	T-H.3.4-1
H.3.4-2	TRIGA Fuel	T-H.3.4-2
H.3.4-3	Maximum Irradiated Fuel Element Radiation Source Term . .	T-H.3.4-7
H.3.4-4	TRIGA Reactor Coolant Sampling	T-H.3.4-8
H.3.4-5	Principal Design Requirements	T-H.3.4-9
H.3.4-6	Drop Accident Summary	T-H.3.4-14
H.3.4-7	Allowable Maximum and Calculated Maximum Temperatures for TRIGA Fuel Storage System Components	T-H.3.4-15
H.3.4-8	Maximum Individual Receptor Doses for NRF TRIGA Fuel Release Accident	T-H.3.4-15
H.3.4-9	Unlikely Category (10^{-2} Frequency $>10^{-4}$) Dose Consequence Limits	T-H.3.4-15

described in the FFTF FSAR, Chapter 11, Section 11.10. Detailed discussion of criticality studies are contained in the FFTF FSAR, Chapter 20. All operations shall be conducted in accordance with approved plant procedures.

After a component has been placed in the IEM Cell, residual sodium will be removed using the SRS. Both assemblies and Ident-69 pin containers are authorized by the FFTF FSAR to be washed in the SRS. Following sodium removal, each component is loaded into a clean CCC on the MTT. Each CCC shall be loaded in accordance with Section H.3.3.10, Limits #10 and #11.

With the exception of criticality safety, all aspects of the CCC design are described in Section H.3.3.A of this appendix. The FFTF fuel storage system criticality safety analyses and corresponding limits are provided in the FFTF FSAR, Chapter 20.

When a CCC is loaded, its cover is bolted in place. Because the IEM Cell atmosphere has an inert argon atmosphere, the container also is inerted. The cover bolts are torqued and the CCC is leak tested. Following a successful leak test, the CCC is ready for removal from the IEM Cell by the SWC.

H.2.2.2 CCC Receipt, Loading, and Transfer to the ISC

Normally, an empty CCC is received in the Reactor Service Building (RSB) where it is uprighted and moved to the solid waste transfer pit (SWTP), using the RSB crane, where it is inspected prior to use. The SWC is placed onto the SWTP where it picks up the empty CCC. The SWC containing the CCC is placed on the multipurpose rail transporter (MPRT) for transfer into the RCB. Operation of the SWC is described in the FFTF FSAR, Chapter 11.21. Alternate locations for receipt, inspection, and inerting of empty CCCs may be used without changes to the FFTF FSAR.

Once in the RCB, the SWC is moved to the IEM Cell 10-ft port using the polar crane. The SWC interfaces with the IEM Cell at the 10-ft ceiling valve via the 40-in. offset adapter plate. After inerting the SWC and leak checking the interface, the 10-ft ceiling valve is opened and the SWC lowers the CCC into one of three CCC storage fixtures on the MTT. Loading of the CCCs occurs as described in Section H.2.2.1 of this appendix.

The SWC grapples the loaded CCC and withdraws it from the IEM Cell. After it is unmated from the IEM Cell, the SWC is transferred to the MPRT using the polar crane. The MPRT transports the SWC to the RSB where the RSB crane transfers the SWC to the CLS. At the CLS, the SWC is mated with the disposable solid waste cask (DSWC) floor valve, and the CCC is lowered into the ISC. The SWC is then either returned to the SWTP to pick up an empty CCC, returned to the IEM Cell to pick up a loaded CCC, or placed in storage.

H.2.2.3 ISC Receipt, Loading, and Transfer to the ISA

Because of the size and weight of the ISC (114,200 lb, 85-in. diameter, 180-in. height), it is transferred by the DSWC transporter to the RSB. The transporter is a towed flatbed trailer with provisions for tie-down attachments. The load-bearing section of the trailer is 26-in. (nominally) above-grade. In the RSB, the RSB crane is used to transfer the ISC to the CLS where the closure plug is removed. The RSB CLS center/seal plate and the DSWC floor

installed. The Ident-69 containers are supported on either the nozzle end or the transition section within the CCC.

2. CCC

The CCC (Figures H.3.3-1 and -2) is an unshielded, sealed fuel storage container with seven fuel storage positions. The CCC provides canning for the fuel to ensure cladding equivalency is maintained during the spent fuel dry storage design lifetime. The CCC also provides the geometry to ensure criticality control during handling and storage of the fuel. The CCC is designed such that it is fully retrievable from the storage configuration, although the capability to remove individual fueled components from a CCC is not guaranteed. The center storage location can accept a fuel assembly which has had the bottom 15.5 in. removed. However, due to the indented CCC handling socket, an Ident-69 pin container cannot be stored in the center location.

The CCC was designed to interface with the FFTF fuel handling system and only minor modifications to existing equipment were required. The CCC was sized to fit into the solid waste cask (SWC) and the ISC. It is fabricated from stainless steel and nickel alloy material to provide corrosion-resistant fuel storage with overall dimensions of 20.0 in. in diameter by 146 in. in height. The weight of the empty CCC is 1,100 lb. The maximum weight of a loaded CCC occurs with seven DFAs. This weight is 3,900 lb. The storage positions of the CCC are formed by seven steel tubes arranged with six, equally spaced, radial storage positions surrounding the seventh center storage position. There is minimal clearance between storage tubes in the bundle orientation.

The upper portion of the outer tubes is 6.69 in. outside diameter (od) with 109-mil thick walls. There is a saddle section 14 in. above the bottom of the CCC where each outer storage tube transitions to a smaller section measuring 4.0 in. od with 226-mil thick walls. The bottom 12.4 in. of the lower section is fabricated from nickel alloy material for enhanced corrosion resistance. The center tube is 6.54 in. od with 120-mil thick walls. The bottom 10.0 in. of the center tube also is fabricated from nickel alloy material. There is no size reduction at the lower end of the center tube.

The outer storage tubes are suspended from the upper support plate. The center storage tube connects the upper support plate with the lower support plate. The outer tubes are fixed only at their upper ends so they are free to accommodate thermal expansion. This design also permits the outer tubes to stretch slightly to absorb energy during the CCC drop accident onto the ISC internal impact limiter. Drop energy is absorbed by the CCC tubes until the gap between the tube stop and the lower support plate is taken up. At this point, the CCC acts as a rigid body for final interface with the ISC impact limiter during the accident event.

The lower support plate has an 18.0-in. diameter and is 1.5-in. thick. It limits downward travel of the outer storage tubes and also provides radial positioning guidance for inserting the CCC into the ISC using the SWC. The upper support plate has a 20.0-in. diameter with an overall thickness of 3.6 in. It provides support for the outer storage tubes and the seating surface for the cover seal. There are 12 drilled holes in the upper support plate which accommodate the cover bolts.

The cover (see Figures H.3.3-3 and -4) has a 20.0-in. diameter and an overall thickness of 1.63 in. The lower surface of the handling socket extends 8.35 in. below the bottom of the cover. Twelve holes are drilled in the cover and are sized to accommodate the closure

H.3.3.2 PRINCIPAL DESIGN CRITERIA

1. INTRODUCTION

10 CFR 72.236(b) requires that design bases and design criteria be provided for structures, systems, and components important to safety. The criteria for the design, fabrication, construction, testing, and performance of components important to safety are reflected, in part, in the general requirements of 10 CFR 72, Subpart F, "General Design Criteria for Independent Fuel Storage Facilities (ISFSI)," but are more specifically addressed in 10 CFR 72.236(a) through (i).

The following subsections discuss the design criteria applicable to the FFTF spent fuel storage system. The subsection headings generally correspond to requirements in 10 CFR 72, Subpart F, and 10 CFR 72.236.

2. FUEL TO BE STORED

10 CFR 72.236(a) requires that a specification for the spent fuel to be stored in the cask be provided. For the FFTF spent fuel storage system, the design payload for the ISC is a CCC loaded with up to seven spent FFTF reactor fuel assemblies or six Ident-69 pin containers containing spent fuel pins that contain uranium and plutonium. Each core component or pin container has a decay heat generation rate of 250 W maximum (1,500 W maximum for the entire CCC). The radiation source term analyzed represents a combination of seven DFAs, which each have a maximum average burn-up of 150,000 MWD/MTHM, have decayed for at least 4 years, and have an initial plutonium enrichment that constitutes 29.3% of the heavy metal. Any fuel which exceeds this criteria must be analyzed on a case-by-case basis for acceptability. Section H.3.3.10, Limit 11 specifies the fuel that has been analyzed for storage.

3. QUALITY STANDARDS

The quality standards requirements for the FFTF spent fuel storage system are expressed in 10 CFR 72.122(a) and 10 CFR 72.234(b).

10 CFR 72.122(a) requires that structures, systems, and components important to safety must be designed, fabricated, and tested to quality standards commensurate with the importance to the safety function to be performed. 10 CFR 72.234(b) requires that the design, fabrication, testing, and maintenance of the spent fuel storage casks be conducted under a quality assurance program that meets the requirements of 10 CFR 72, Subpart G, or equivalent.

Parameters important to safety for the ISC are stainless steel confinement boundary integrity, concrete density, air entrainment, and compressive strength range; concrete shielding wall thickness, and reinforcement bar material strengths and placement; and structural steel shielding placement, anchor studs, and structural integrity.

The ISC is designated as a storage and onsite only shipping container for FFTF spent nuclear fuel (radioactive material). It also is required to have a design life of 50 years. Consequently, the quality assurance requirements which must be imposed on the design and fabrication of the ISC must also meet the requirements for storage and shipping of radioactive material. The quality assurance program requirements of 10 CFR 71 were imposed on the ISC so that a vendor qualified (and certified by the U.S. Nuclear Regulatory Commission [NRC]) to design and fabricate spent fuel storage or shipping casks could be chosen.

The ISC design, analysis, fabrication, and testing are required to meet the requirements of 10 CFR 72 except for licensing and some additional criteria imposed to allow onsite shipment of the casks. The designer of the ISC was required to have licensed a cask design with the NRC to either the 10 CFR 71 or 10 CFR 72 requirements. The

of yield strength or 1/5 of ultimate strength, whichever is less:

<u>Axis</u>	<u>Peak acceleration (g)</u>
Longitudinal	0.27
Transverse	0.19
Vertical	0.52

- Vertical handling loads including the load due to a set down on a rigid surface at impact velocities of up to 10 ft/min.

ISC Off-Normal Loads: Off-normal loads are evaluated as normal loads and include off-normal severe environmental conditions and off-normal handling. Environmental conditions are discussed in Section H.3.3.2, item 4.b, of this appendix. As described in the preceding section, the design basis loading also shall evaluate the effect of off-normal handling loads. The two cases listed below were evaluated against the Level A ASME service limits (ASME 1989a).

- A 1-ft free drop of an ISC onto a flat, essentially unyielding horizontal surface, striking the surface in a position for which maximum damage is expected
- An impact of the hemispherical end of a vertical steel cylinder of 3.2-cm (1 1/4-in.) diameter and 6-kg (13-lb) mass, dropped from a height of 1.0 m (40 in.) onto the exposed surface of the ISC that is expected to be the most vulnerable to penetration. The long axis of the cylinder shall be perpendicular to the ISC surface.

ISC Accident Handling Loads: The ISC was designed to withstand the accident loads of complete blockage of air inlets, mobile crane boom drop, seismic, wind, tornado, missile, lightning, and fire. In addition to those loads, additional handling accident loads were evaluated to fully envelop all ISC load conditions. The ISC will withstand these design-basis accident loads to the extent that the reduction in shielding is not sufficient to increase the external dose rate to more than 1,000 mrem/h at 1 m from the external surface of the package and the ISC confinement integrity is maintained (i.e., stress levels in the ISC confinement boundary do not exceed the levels specified for Level D Service Limits of the ASME Code [ASME 1989a]).

- ~ A free drop of 20 ft onto the Reactor Service Building-Cask Loading Station (RSB-CLS). The RSB-CLS impact limiter on the support platform is designed to limit this load to a maximum of 70 g to the cask (WHC 1995b).
- ~ A free drop of 40 in. striking the top end of a vertical, cylindrical, mild steel bar mounted on an essentially unyielding horizontal surface, with maximum damage expected. The bar shall be 6 in. in diameter, with a top horizontal and its edge rounded to a radius of not more than 1/4 in. and/or such length as to cause maximum damage to the package, but not less than 8-in. long. The long axis of the bar shall be perpendicular to the unyielding horizontal surface.

This stress is much less than the pre-load stress of 52.72 ksi and is therefore not significant.

Under normal conditions the temperature in the closure bolts increases to 142 °F. The coefficient of thermal expansion of the closure (stainless steel) is more than that of the closure bolts. Therefore, the closure bolts will have additional stress (6.14 ksi) beyond that induced by the preload.

Additionally, the 1-ft drop loading of the ISC on its top induces a bolt stress of 9.08 ksi from the 0.001218 radians of flange rotation. This rotation correlates to 0.0036 in. of temporary seal decompression. The initial compression of the seal is 0.036 in. Manufacturer data state that when separation of the flange occurs, the seal will continue to provide a leaktight joint until the threshold of loss of sealing is reached. The recommended tightening load for these seals is 2,300 lb/in. The tightening load at threshold loss of sealing is 400 lb/in. A 0.003-in. variation in compression will vary the tightening load by only 10%, far above the threshold loss of the sealing load. The rotation stress combined with the thermal and preload stress results in a total closure bolt stress of 67.94 ksi, which is below the allowable bolt stress of 68.6 ksi.

(3) ISC Maximum Thermal Stress Condition

The worst-case thermal stress condition for the ISC occurs for the normal condition thermal case, which assumed the conditions of the hottest summer day with full heat load, full wind, and no solar input as described in

Section H.3.3.4 of this appendix. This thermal case results in the largest thermal gradients between the ISC shield and concrete.

The case of partial fuel loading introduces an unbalanced heat load but was not analyzed because the radial thermal gradient through the ISC shield was lower than the case for the full heat load. The partial fuel-loading case with a circumferential gradient of approximately 12 °F is bounded by the full heat-load case.

Because of the thermal gradient throughout the cask composite shielding wall, thermal stresses are generated in the steel liner and outer shields, in the anchor studs connecting the shields to the concrete, and in the concrete due to the internal constraint of the cylindrical cask and the differential expansion of the steel shields.

The ISC liner is hotter than the remainder of the cask, but because it is not structurally tied to the shielding portion of the cask it is free to expand and no thermal stresses are developed. The inner and outer shields are hotter than the concrete. The concrete constrains the shields, therefore, the shields are in compression and the concrete in tension. In the axial direction, the anchor studs embedded in the concrete constrain the shields. Because the studs bend due to this shear loading and become loaded, an equilibrium condition is reached where the shear load in the studs is equal to the compression in the shields.

The maximum confinement liner-to-shielding differential growth was calculated to be 0.132 in. Additionally, the bottom head and flange radial growth relative to the concrete was evaluated. This differential expansion is

much lower than the 84,000 psi allowable. The seal performance is unaffected by the extra load on the bolts due to the fire, and the seal temperature at 200 °F is well below the 700 °F limit. Because the bolts do not yield, sealing performance is unaffected by the fire. Therefore, the cask will remain within design limits during a fire.

(5) Ashfall Event

Ashfall is not a significant load to the ISC. Additionally, the ashfall event temperatures, with complete blockage of the ISC ventilation ducts, are within normal temperature limits.

f. CCC Structural Analysis

(1) CCC Normal Condition Stress Allowables

The allowable primary membrane and primary membrane-plus-bending stress intensities for normal loads are in accordance with ASME Code, Section VIII, Division 2 (ASME 1989a) and listed below.

Primary membrane stress - $"P_m" \leq K S_m$

Primary membrane-plus-bending stress - $"P_l + P_b" \leq 1.5 K S_m$

The 304 stainless steel tubes govern for normal loads

where:

$K = 1.0$ for pressure, dead loads, and external attachment loads

S_m = Design stress intensity allowable value.

304 stainless steel:

$$P_m \leq S_m$$

$$P_m \leq 20,000 \text{ psi at } 200^\circ\text{F}$$

$$P_m \leq 16,400 \text{ psi at } 600^\circ\text{F}$$

$$P_l + P_b \leq 1.5 S_m$$

$$P_l + P_b \leq 30,000 \text{ psi at } 200^\circ\text{F}$$

$$P_l + P_b \leq 24,600 \text{ psi at } 600^\circ\text{F}.$$

(2) CCC Normal Conditions Analysis

The CCC normal condition analyses show that the CCC can withstand all normal loading conditions without compromising its structural confinement "canning" integrity.

(3) CCC Internal Pressure Analysis

A finite element stress analysis was performed for the pressure load of 70 psi (WHC 1995a). The results of this analysis show that the maximum membrane stress of 3,325 psi in the central tube and the maximum membrane-plus-bending stress of 4,968 psi are both well below the allowable stresses of 16,400 psi and 24,600 psi, respectively.

(4) CCC Vertical Lift and Set Down by Crane

The CCC was evaluated (WHC 1995a, 1995c) for vertical lift and set down by a crane. This analysis used a dynamic load factor of 2.0 to encompass the expected crane loadings. The maximum stress was 11,600 psi and occurred at the interface weld of the center support cylinder to

Nickel Alloy UNS 000625 material (200 °F)

Elastic analysis:

$$\begin{aligned} \text{At } 200 \text{ }^{\circ}\text{F, } P_m &= \text{lesser of } 2.4 \times S_m &= 87,840 \text{ psi} \\ &0.7 \times S_u &= 84,000 \text{ psi (govern)} \end{aligned}$$

$$P_I + P_b = 1.5 \times 84,000 = 126,000 \text{ psi}$$

Plastic analysis:

$$\begin{aligned} \text{At } 200 \text{ }^{\circ}\text{F, } P_m &= \text{larger of } 0.7 \times S_u &= 84,000 \text{ psi (govern)} \\ &S_y + 1/3 (S_u - S_y) &= 78,600 \text{ psi} \end{aligned}$$

$$\begin{aligned} \text{Maximum primary stress} &= 0.9 S_u &= 108,000 \text{ psi (200 }^{\circ}\text{F)} \\ &&= 52,326 \text{ psi (600 }^{\circ}\text{F)} \end{aligned}$$

ASTM A574 Bolt Material (200 °F)

$$\begin{aligned} \text{At } 200 \text{ }^{\circ}\text{F, } P_m &= \text{lesser of } 0.7 S_u &= 119,000 \text{ psi (govern)} \\ &S_y &= 150,000 \text{ psi} \end{aligned}$$

h. CCC Accident Analyses

The CCC accident analyses show that the CCC can withstand the accident conditions without compromising its structural integrity. Additionally, the CCC is shown to retain its confinement "canning" integrity.

(1) CCC 18-Ft Vertical Drop into ISC

The ISC internal impact limiter is used to absorb the impact of a fully loaded CCC from a height of 18 ft. The impact limiter consists of a 1.5-in. layer of aluminum

honeycomb with a compressive strength of 4,400 psi and a CCC impact area of 153 in².

The dynamic finite-element analysis (WHC 1995a) shows that the CCC will remain intact during and after the 18-ft vertical drop into the ISC, and the CCC boundary will provide adequate sealing during and after the drop. The results of the stress evaluation indicate that the stresses in some of the components exceed the yield strength but they are within the stress limits of the ASME Code, Section III, Appendix F (ASME 1989a). The magnitude of the CCC tube plastic deformation does not affect the integrity of the CCC or the seals. The critical component of the CCC for membrane stress is the central tube of the CCC. The maximum membrane compressive elastic stress is 37,313 psi, which is less than the allowable 39,360 psi.

If an assembly is stored in the center tube of the CCC, it will rebound and hit the CCC cover and the bolts, and the seal region will remain elastic. The impact limiter over the central tube limits the load to the socket region of the CCC cover to 8,000 lbf. The metal o-ring seal of the CCC cover will have a maximum opening of 0.005 in., which is within the manufacturer's deformation limit to retain sealing effectiveness. The seal will neither lose contact with the cover nor with the upper support flange. The CCC cover bolts are subject to a tensile stress of 61,666 psi, which is less than the allowable 119,000 psi.

(2) CCC Horizontal Side Drop Within the ISC

The CCC is subject to a 96.1-g maximum side loading while it is stored within the ISC (GA 1995). The ISC remains leaktight during this event. The CCC was evaluated for this 4-ft drop condition (WHC 1995a). If it is assumed that the CCC rests on the ISC wall, the results show that the CCC material will not exceed the yield strength of the materials thus no permanent deformation will occur.

If it is assumed that the CCC has 0.5 in. that can be deflected within the ISC liner (unrealistic case because in actuality the CCC will rest on the ISC wall), the dynamic finite-element analysis case shows that the elastic stresses in some of the components exceed the yield strength but they are within the stress limits of the ASME Code, Appendix F, Section III, Division 1 (ASME 1989a). The conditions of this analysis assume that the fuel element within the CCC hits the storage cylinder wall, and then the wall and element move together and hit the side of the ISC. The stress results show the bending stress in the storage tube is 51,608 psi, which is less than the membrane-plus-bending stress allowable of 59,040 psi. The magnitude of the plastic deformation is not expected to affect the integrity of the CCC because stresses are within allowable limits.

3. CONCLUSION

The ISC was analyzed with respect to stress, criticality, shielding, radiological consequences, and thermal considerations. The analyses show that the ISC will perform its function as the secondary confinement barrier for public protection and retain adequate

shielding under all design basis normal, off-normal, and accident conditions. The ISC is designed to interface with FFTF handling equipment and facilities.

The ISC confinement boundary was demonstrated to remain leaktight for all design basis normal, off-normal, and accident load conditions. Even under the design-basis accident, the stresses in the ISC confinement liner remain below normal condition allowables. Therefore, the ISC confinement demonstrates a large margin of safety for storage of the spent fuel.

The ISC reinforced-concrete shielding was designed and evaluated to conform to the requirements of ACI 349-1990 (ACI 1990). For the normal load conditions, the concrete was evaluated and only small areas of local crush damage were allowed as these small localized areas could easily be repaired and would not impact the shielding integrity of the cask. For the accident conditions, the concrete damage was evaluated, and the acceptance criteria limited shielding damage such that the dose rate at 1 m from the surface of the cask did not exceed 1,000 mrem/h. The maximum damage for the accident loads occurred during a 4-ft drop onto an unyielding surface in which 6.19 in. of concrete was crushed on the corner. The hypothetical loss of shielding accident analysis described in Section H.3.3.6 of this appendix shows that removal of 19 in. of concrete from the entire surface of the ISC shield structure results in one-half the allowable accident dose rate of 1,000 mrem/h at 1 m. Therefore, there is a large margin of safety in the ISC shielding design.

The CCC was analyzed with respect to its structural and confinement effectiveness as a canning structure for the FFTF spent fuel. The CCC results demonstrate that it retains its confinement and structural integrity under all normal and accident load conditions. Even during the 18-ft drop accident into the ISC, the CCC was shown to remain intact and retain its sealing effectiveness.

Therefore, the CCC demonstrates a large margin of safety for confinement and handling of the spent fuel.

H.3.3.4 THERMAL EVALUATION

1. DESIGN DESCRIPTION

The FFTF storage system is designed to provide adequate spent fuel heat removal capacity without active cooling systems. The main component in the fuel storage system which ensures passive removal of decay heat is the ISC. The ISC is designed to provide natural convection (ventilation) and conduction through the reinforced concrete and steel shielding to the external environment. The inlet flow configuration for the ISC convection cooling is shown in Figure H.3.3-6. Two 4-in. od (3.75-in. inner diameter) inlet ducts located 180° apart admit air into the cask bottom. These feed into a channel approximately 1.4 in. by 3.6 in., which encircles the liner and serves as a bottom plenum. Six 2-in. by 2-in. windows in the inner shield admit air from this plenum into the 0.75-in. wide annulus between the confinement liner and the inner shield. The air flows up this annulus, picks up heat, and exits through a window at the top into another plenum channel. The top windows and plenum channel are identical to those at the bottom. Two 4-in. outlet ducts, also 180° apart, connect to the top plenum and exhaust air to the environment.

The 1.5-in. thick stainless steel ISC confinement liner and 3-in. thick inner carbon steel shield plate extend the entire length of the fuel cavity; the 4-in. thick outer carbon steel shield plate is 92-in. long and located at the cask mid-plane. These high-conductivity components act as fins in the axial direction and

increase the efficiency of the conduction heat transfer by minimizing axial gradients.

The principal operating features of the FFTF fuel storage system for normal storage are as follows.

- The decay heat within the fuel assemblies is transferred by thermal radiation and gas conduction through the pin bundle, out to the stainless steel fuel duct.
- The decay heat is transferred by thermal radiation and gas conduction to the inner surface of the CCC support tubes; the support tubes then conduct the heat to the outer surface of the tubes.
- The decay heat of the payload is transferred from the CCC support tubes to the inner surface of the inner liner (confinement vessel) of the ISC by radiant heat transfer as well as by gas conduction via the internal atmospheric gas, argon or helium. The payload will have a maximum decay heat of 1,500 W.
- The decay heat is transferred to the exterior of the cylindrical steel shell of the ISC by conductive heat transfer.
- This decay heat is transferred to the exterior ISC surface by radiant heat transfer and conduction through various air gaps, steel, and concrete radiation shields, and the outer surface coating. The ISC has two radial air gaps, one is 0.75 in. (between the confinement liner and the inner shield) and one is 0.25 in. (between inner and outer shields).
- The outer ISC surface dissipates the decay heat to the ambient environment by combined thermal radiation and convection heat transfer. The ambient temperature is assumed to vary in the

cooling ducts do not significantly increase the surface dose rates because they contain two sharp bends which are above or below the ends of the fuel assemblies to minimize streaming. Table H.3.3-17 provides a compilation of all the relevant materials used for the cask shielding analysis. Shielding calculations were performed to obtain the total dose rate from all contributing source components, including the following:

- Primary neutron source from the spent fuel
- Secondary neutron source from additional fission in fuel
- Primary gamma source from fuel and associated hardware
- Secondary gamma source from neutron interaction with fuel assemblies and cask materials
- Scattering source in air.

The shielding analysis used validated computer codes, including the 3-D PATH point-kernel code and 2-D TWODANT transport code. These codes have been benchmarked and validated against the equivalent industry standard codes. A comparison of the calculated-to-measured dose rate for a similar storage cask was performed for the PATH code to determine an adjustment factor.

2. DESIGN EVALUATION

The principal shielding design criteria for the FFTF spent fuel storage system is based on maintaining the exposure of the radiation worker to <500 mrem/yr, which is the limit imposed by HSRCM-1, *Hanford Site Radiological Control Manual* (WHC 1994a). The ISC design requirement (WHC 1992a) also requires that the ISC be designed to limit the radiation dose to 1,000 mrem/h at 1 m from the external surface of the cask following accident scenarios. This requirement ensures that the ISC design is compatible with the requirements imposed by 10 CFR 71.51(a)(2). In

addition, 10 CFR 72.106 requires that the shielding maintain occupational exposures to radiation to ALARA.

As previously discussed, the shielding is designed to limit exposure to 2.0 mrem/h at the cask surface in accordance with WHC (1992a) and WHC (1994b), which allow localized higher dosage areas of up to 5.0 mrem/h to account for shielding imperfections and localized hot spots. The 2.0-mrem/h design limit was based on the assumption that an individual's average exposure time is <1 h/day. The bottom surface is limited to 200 mrem/h contact because it is normally inaccessible. The 2.0-mrem/h design level at the cask surface also was used in the analysis that determined the minimum fence distance (22 ± 1 m) at which the dose rate would be at or below the ISA perimeter fence design limit of 0.05 mrem/h (WHC 1995j).

A major assumption made in the shielding design is the source strength of the nuclear fuel to be stored in the FFTF storage system. Table H.3.3-18 provides the bounds for the source terms used in the shielding analysis for the ISC.

Table H.3.3-19 provides a summary of the ISC dose rate-shielding analysis. The ISC shield design was analyzed in GA (1995) by calculating the dose rates at the surface of the cask based on an assumed neutron and gamma radiation source from DFAs. The source was assumed to have been irradiated for 150,000 MWD/MTHM (plutonium taken as 29.3% of the heavy metal) and allowed to decay for 4 years resulting in a decay heat of 250 W/assembly. The PATH computer code was used in GA (1995) to calculate gamma flux at the cask surface, and the TWODANT code was used to calculate neutron flux.

Dose rates at the surface of the cask were determined to be <2.0 mrem/h at all accessible surfaces. The highest dose rate calculated was 1.99 mrem/h at the top closure of the cask. The highest dose rate at a non-accessible surface was 99.3 mrem/h, calculated at the bottom closure, which is less than the 200 mrem/h limit.

Objective:

- To preclude a loaded ISC drop from >4 ft in the RSB or >8 ft in the ISA.
- To prevent an accidental drop of equipment onto a loaded ISC.

Action:

In the event that the ISC handling limits are inadvertently exceeded or the equipment is inadvertently moved over a loaded ISC, the ISC shall be lowered immediately to a permissible height or the equipment immediately moved from over the loaded ISC.

In the event of a drop of a loaded ISC from any height, handling of the ISC shall be stopped, appropriate radiological surveys shall be made to assess the cask integrity and major damage, and an engineering evaluation shall be made concerning the ability to use the ISC for further storage.

Surveillance:

Administrative procedures shall provide precautions and limitations for cask handling to ensure that the ISC is not handled at a height greater than the limits defined above, and that the movement of equipment other than ISC rigging will not require transfer over a loaded ISC.

Basis:

The ISC structural analysis shows that the ISC can sustain the above drops without breaching the confinement boundary, preventing removal of spent fuel, or causing a criticality accident. This specification ensures that handling height limits will not be exceeded in transit to or at the storage pad. The handling limit in the RSB is lower than at the ISA as the analysis assumes the drop in the RSB is onto an unyielding surface while the drop in the ISA is assumed to be onto a yielding surface <1-ft thick. Limiting movement of unspecified or unanalyzed equipment over the ISC prevents damage from accidents due to drops of these loads.

10. Title: IEM Cell Handling Limits

Specification: The following limits shall be observed when handling fuel in the IEM Cell.

IEM Cell atmosphere	IEM Cell oxygen and water vapor concentrations shall be maintained at <200 ppm whenever a CCC is open in the IEM Cell.
Storage condition	Sodium-wetted assemblies shall not be placed into a CCC until they have been cleaned and dried in the IEM Cell sodium removal system in accordance with normal IEM Cell operating procedures.
Number of assemblies	<p>The loaded CCC shall contain one of the following:</p> <ul style="list-style-type: none">◦ either six or seven fuel assemblies,◦ six pin containers,◦ or a mixture of fuel assemblies and pin containers such that the total is either six or seven and the number of pin containers is five or less <p>prior to removal from the IEM Cell.</p>
CCC closure	<p>The CCC containing assemblies shall not be transferred from the IEM Cell unless the following actions have been taken.</p> <ul style="list-style-type: none">◦ The CCC cover bolts have been installed and torqued in accordance with approved IEM Cell operating procedure(s)◦ The CCC cover has successfully passed a leak check in accordance with approved IEM Cell operating procedure(s) signifying that the cover is properly installed.

Applicability: This specification is applicable to the processing of all FFTF spent fuel in the IEM Cell into the CCC.

Objective: These limits ensure that the assemblies loaded into the CCC are cleaned of residual sodium and that the CCC atmosphere contains a limited amount of oxygen and water vapor. These limits ensure that the loading of the CCC is within the bounds of the analyses. Finally, these limits ensure that the CCC cover is properly installed.

Action: If the IEM Cell atmosphere is outside the above limits or if monitoring capability is lost, the IEM Cell atmosphere must be restored prior to CCC closure. Engineering must then evaluate the length of time the CCC must remain open to ensure that the CCC atmosphere is at equilibrium with the cell atmosphere based on the length of time the cell was over the limit and the magnitude of the excursion. (This may be performed on a case-by-case evaluation or proceduralized to cover a range of conditions.)

The CCC shall not be removed from the cell until the inventory has been verified and the cover has been installed in accordance with the IEM Cell Work Plan.

Surveillance: The IEM Cell atmosphere shall be monitored any time a CCC is in the cell with the cover removed. CCC loading and installation of the CCC cover shall be controlled in accordance with the IEM Cell Work Plan.

Basis: WHC (1995i) specifies that components loaded into the CCC have only a minimal amount of residual sodium to minimize the potential for fuel cladding caustic corrosion and hydrogen production. Operational experience has demonstrated that fuel washing in the IEM Cell sodium removal system will leave only trace amounts of residual sodium at best. The limitation on CCC loading ensures that the number and types of assemblies loaded into the CCC are within the bounds of the analyses. Finally, the

inert atmosphere requirement and the sealed CCC provide additional protection/margin against cladding degradation.

11. Title: Fuel Specification

Specification: The following limits shall be observed for fueled components to be stored in the FFTF spent fuel storage system:

Type	<ul style="list-style-type: none">DFAs as described in the FFTF FSAR, Chapter 17, Section 17.5.3.1. In addition, driver evaluation (DE), core characterizer assemblies (CCA), and run-to-cladding-breach (RTCB) assemblies are includedExperimental fuel assemblies listed in Table H.3.3-21.Ident-69 pin containers listed in Table H.3.3-21.
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Decay power (per assembly)	250 W maximum
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Decay power (per CCC)	1,500 W maximum
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Applicability: This specification is applicable to fueled components that are covered by analysis to be stored in

the ISC. Fuel types outside the specification listed above may not be loaded into a CCC/ISC until appropriate analyses have been performed and reported in a change to this

Appendix H section.

Objective: This specification was derived to bound the majority of the FFTF fuel that is to be transferred from sodium pool storage to dry storage. This design basis source term has been evaluated to ensure that the peak fuel pin temperatures, maximum surface dose rate, and nuclear criticality loadings are below the design values for the storage system. Furthermore, the fuel type ensures that the structural conditions in the safety analysis bound those of the fuel to be stored.

Action: If this specification is not met, additional case-by-case evaluation is necessary to transfer the fuel assembly to an ISC for dry inert gas storage.

Surveillance: The fueled components selected to be stored in the ISC shall meet the above specification, and shall be verified and documented prior to fuel loading. The identity of each fuel assembly and pin container shall be independently verified and documented prior to installation of the CCC cover in the IEM Cell. Loading of fuel assemblies and loaded Ident-69 pin containers will be performed in accordance with the Detailed Refueling Plan (DRP) and/or the IEM Cell Work Plan, which administratively control the inventory control system (ICS) described in the FFTF FSAR, Chapter 11.

Basis: The ISC/CCC storage system is designed to provide adequate heat removal, criticality, and structural margin for safe operation and response to accident conditions. When FFTF fuel storage in the ISA was initially authorized, there was not time to analyze all FFTF fuel types. Therefore,

the initial approval of Appendix H limited the FFTF fuel to standard driver fuel assemblies and other assemblies that are essentially the same as driver fuel assemblies. Although it was expected that this authorization would cover a large percentage of FFTF fuel, the uncertainty of restart prevented much of the driver fuel from being washed and loaded into ISCs.

In order to provide additional fuel for offload, further analyses (WHC, 1996a) provided a list of fuel experiments (Table H.3.3-21) that can now be loaded into CCC/ISCs. Areas of review in that analyses included radiological accident releases, radiological shielding, criticality, thermal and stress. The analyses concluded that storage of these new components is bounded by the existing safety envelope.

In most sections of H.3.3, the CCC/ISC design still refers to the basis of 7 DFAs with maximum pre-irradiation enrichment of 29.3%, burnup of <150,000 MWD/MTM and more than 4 years since last irradiation. Those are still valid design basis criteria so, in most cases, design sections were not modified for this change to the fuel specification.

In the change that added the new experiments in Table H.3.3-21, the 4-year post-irradiation time limit was deleted. That is because it has now been over 4 years since the reactor has operated. The 150,000 MWD/MTM and 29.3% enrichment limits have also been deleted. No DFAs exceeded those limits. Each test assembly listed in Table H.3.3-21 has been evaluated for the cask parameters, and the fact that some had greater burnup or enrichments is no longer a limitation.

When the remainder of the FFTF fuel is analyzed, it will be added to this specification.

H.3.3.11 ACCIDENT ANALYSIS

This section summarizes analyses of off-normal and accident conditions which clearly demonstrate the safety of the CCC/ISC system for FFTF fuel. The accident scenarios evaluated are consistent with the 10 CFR 72 criteria and other commercially licensed systems (NRC 1991a).

The ISC design is totally passive. Because of the low decay heat, no active cooling systems or monitoring are required. Although a convective cooling system was designed into the cask, total blockage results in only a slight increase in temperature that does not approach safety limits. Helium is used as the gas inside the ISC, but heat dissipation does not depend on the heat transfer of that gas. Thus, there are no credible cask failures that can lead to releases, and the safety issues shift to "What external events can affect the cask?"

As stated in Section H.3.3.3 of this appendix, "the *FFTF Spent Fuel Interim Storage Cask Design Analysis Report* (GA 1995) demonstrates that the ISC can withstand the design basis normal, off-normal, and accident conditions and still maintain the ability to perform its safety function." During all design-basis accident scenarios, there are no releases.

In addition to design-basis accidents, analyses of hypothetical cask accidents--with no known cause--were made. Even these hypothetical releases do not exceed established Hanford Site boundary limits. Therefore, it can be concluded that FFTF fuel storage is not a public health and safety hazard.

WHC, 1995h, *Thermal Analysis of the Core Component Container Within the Interim Storage Cask*, WHC-SD-FF-ER-100, Rev. 1, Westinghouse Hanford Company, Richland, Washington.

WHC, 1995i, *Residual Gases in Interim Storage Cask/Core Component Container*, WHC-SD-FF-DA-079, Westinghouse Hanford Company, Richland, Washington.

WHC, 1995j, *FFTF ISC Platform Minimum Fence Distance*, WHC-SD-FF-TA-037, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

WHC, 1995k, *Radiation Shielding Impact of a Seventh DFA in the Core Component Container for the FFTF Fuel Off-Load Program*, WHC-SD-FF-TA-038, Westinghouse Hanford Company, Richland, Washington.

WHC, 1995l, *Handling Socket of a FFTF ISC*, WHC-SD-FF-TA-036, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

WHC, 1995m, *Structural Analysis of Interim Storage Area (ISA) Concrete Pad and Crane Drop on ISC*, WHC-SD-FF-DA-078, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

WHC, 1996a, *Review of FFTF Fuel Experiments for Storage at ISA*, WHC-SD-FF-RPT-005, Rev. 1, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-4-46, *Nonreactor Facility Safety Analysis Manual*, Westinghouse Hanford Company, Richland, Washington.

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DOE-RL, 1993, "Design Loads for Facilities," HPS-SDC-4.1, Rev. 12, *Hanford Plant Standards*, U.S. Department of Energy-Richland Operations Office, Richland, Washington.

²Guttenberg, S., 1993, *Agreement on the Requirements Necessary to Assure that the Phenomenon of Hot Cell Rot During Interim Storage of FFTF Spent Fuel is Properly Addressed* (internal memo 8300-SG-012 to L. R. Besel, March 24), Westinghouse Hanford Company, Richland, Washington.

WHC, 1993, *Thermal Hydraulic Evaluations of the FFTF Core Component Container*, WHC-SD-FF-ER-079, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

²For retrievability, these references have been combined into:

WHC, 1995, *Fuel Offload Documentation*, WHC-SD-FF-RPT-003, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

Table H.3.3-3. Diurnal Ambient Temperature at the Hanford Site for the Peak Summer Month.

Time	Temperature (°F)	Time	Temperature (°F)
12:00 a.m.	82	12:00 p.m.	103
2:00 a.m.	78	2:00 p.m.	111
4:00 a.m.	75	4:00 p.m.	115
6:00 a.m.	74	6:00 p.m.	113
8:00 a.m.	85	8:00 p.m.	100
10:00 a.m.	97	10:00 p.m.	89

Table H.3.3-4. Summary of Interim Storage Cask Normal Condition Load Combinations.

Loading condition	Case	Applicable initial condition							
		Ambient temperature		Insolation		Decay heat		Maximum internal pressure	Maximum weight of contents
		Maximum	-27 °F	Maximum	0	Maximum	0		
Hot environment maximum ambient temperature	1			X		X		X	
Cold environment -27 °F ambient temperature	2				X	X			
	3				X		X		
Minimum external pressure	4	X		X		X		X	
Maximum external pressure	5		X		X		X		
Vibration and tie-down loads	6	X		X		X		X	X
	7		X		X		X		X
Free drop: 1-ft drop	8	X		X		X		X	X
	9		X		X		X	X	X
Lifting and handling loads	10	X		X		X		X	X
	11		X		X	X		X	X
	12		X		X		X	X	X

**Table H.3.3-5. Interim Storage Cask Confinement Boundary
Normal Condition Allowable Stresses (ksi)**

STRESS CATEGORY	70°F	100°F	200°F	300°F
COMPONENT OTHER THAN BOLTS				
MATERIAL	304 STAINLESS ASTM A-240 AND A-182			
Primary membrane stress intensity	20.0	20.0	20.0	20.0
Primary membrane + bending stress intensity	30.0	30.0	30.0	30.0
Range of primary + secondary stress intensity	60.0	60.0	60.0	60.0
Primary shear stress	12.0	12.0	12.0	12.0
BOLTS				
MATERIAL	ASTM A-540			
Membrane stress	70.0	70.0	66.8	64.8
Membrane + bending stress	105.0	105.0	100.2	97.2

ASTM = American Society for Testing and Materials

**Table H.3.3-6. Summary of Interim Storage Cask Load
Combinations Hypothetical Accident Loads.**

Accident condition	Case	Applicable initial condition							
		Ambient Temperature		Insolation		Decay Heat		Max. Internal Pressure	Max. Weight of Contents
		Max.	-27°F	Max.	0	Max.	0		
Free Drops; 8 ft 4 ft	1	X		X		X		X	X
RSB-CLS Tipover	2		X		X		X		X
Thermal: Fire Ashfall	3	X		X		X		X	

RSB-CLS = Reactor Service Building – Cask Loading Station

**Table H.3.3-7. Interim Storage Cask Confinement Boundary
Accident Condition Allowable Stresses (ksi)**

Stress Category	70 °F	100 °F	200 °F	300 °F	70 °F	100 °F	200 °F	300 °F
Component other than Bolts								
Material	A240 Type 304 Stainless Steel				A 182 Grade F304 Stainless Steel			
Primary membrane stress intensity	48.0	48.0	48.0	46.2	48.0	48.0	46.3	43.1
Primary membrane + bending stress intensity	72.0	72.0	71.0	66.0	70.0	66.2	61.5	61.5
Primary shear stress	31.5	31.5	29.8	27.7	29.4	29.4	27.8	25.8
Bolts								
Material	A 540							
Membrane stress	84.0	84.0	84.0	--	--	--	--	--
Membrane + bending stress	120	120	120	--	--	--	--	--

Table H.3.3-8. Material Properties for Interim Storage Cask Thermal Analysis.¹

Material	Thermal conductivity (Btu/h-ft-°F)	Specific heat (Btu/Lbm-°F)	Density (lb/ft ³)	Emissivity
Carbon steel	25	0.11	490	0.9 ^a
Stainless steel	b	0.11	488	0.5
Concrete	1.0 ^c	0.156	146	0.9 ^a
Al honeycomb	0.64 ^d 2.8 ^e	0.22	11.5	0.5
Air	f	0.25	N/A	N/A
Argon	g	N/A	N/A	N/A
Volcanic ash	0.12	N/A	N/A	N/A
CCC/DFA	0.5/3.35 ^h	0.090	169.7	N/A

^aEmissivity = 0.9 for thermal radiation from a painted surface. For solar wavelengths the absorptivity is based on a white, painted surface and ranges from 0.18 - 0.25. For conservatism, the ISC analysis used 0.3.

^b $8.08 - A*(T-95)$ where $T = ^\circ\text{F}$ and $A = 0.0052$

^cLocally increases to 1.25 adjacent outer shield and 1.97 at inner shield due to studs (radial direction only)

^dRadial direction

^eAxial direction

^f $0.0174 + 1.81 \text{ E-}5*(T-200)$

^gSee Table H.3.3.D-3

^hRadial/axial thermal conductivity²

¹GA, 1995, *FFTF Spent Fuel Interim Storage Cask Design Analysis Report*, Document No. 910683, Rev. A, General Atomics, San Diego, California.

²WHC, 1995h, *Thermal Analysis of the Core Component Container Within the Interim Storage Cask*, WHC-SD-FF-ER-100, -Rev. 1, Westinghouse Hanford Company, Richland, Washington.

CCC = Core component container

DFA = Driver fuel assembly

ISC = Interim storage cask

N/A = Not applicable

Table H.3.3-21. Experimental Fuel Assemblies and Ident-69 Pin Containers Analyzed for Loading into a CCC/ISC

FUEL ASSEMBLIES	IDENT-69 PIN CONTAINERS		
AAD-1, 2, 3, 4, 6, 7	ID-69 S/N 1	ID-69 S/N 23	ID-69 S/N 43
ACO-2, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16	ID-69 S/N 2	ID-69 S/N 24	ID-69 S/N 44
AW-1	ID-69 S/N 3	ID-69 S/N 25	ID-69 S/N 45
CRBR-1, 3, 5	ID-69 S/N 5	ID-69 S/N 26	
CV-1	ID-69 S/N 7	ID-69 S/N 27	
D9-3	ID-69 S/N 8	ID-69 S/N 28	
DE-HTD	ID-69 S/N 9	ID-69 S/N 30	
DEA-2	ID-69 S/N 10	ID-69 S/N 31	
DIPRESS	ID-69 S/N 11	ID-69 S/N 32	
FOTA-1, 2	ID-69 S/N 12	ID-69 S/N 33	
GF001, GF002	ID-69 S/N 13	ID-69 S/N 35	
MW-1, 2, 3, 4, 5, 6	ID-69 S/N 14	ID-69 S/N 36	
PO-2, 5	ID-69 S/N 15	ID-69 S/N 37	
RNTT-1	ID-69 S/N 17	ID-69 S/N 38	
RTCB-4	ID-69 S/N 18	ID-69 S/N 39	
SRF-1, 2	ID-69 S/N 21	ID-69 S/N 40	
WF004, WF005	ID-69 S/N 22	ID-69 S/N 42	