0037064

ENGINEERING CHANGE NOTICE

Page 1 of 3

1. ECN 605351

Proj. ECN

2. ECN Category (mark one)	3. Originator's Name, Organization, MSIN, and Telephone No.		4. Date ~~March 24, 1994~~ May 6, 1994
Supplemental []	R. H. Meichle, Technical Safety, X0-44, 372-1369		
Direct Revision [X]	5. Project Title/No./Work Order No.	6. Bldg./Sys./Fac. No.	7. Impact Level
Change ECN []	STS SUPPORT/K BASINS	K BASINS	SQ
Temporary []			
Standby []	8. Document Numbers Changed by this ECN (includes sheet no. and rev.)	9. Related ECN No(s).	10. Related PO No.
Supersedure []	WHC-SD-WM-SAR-062 Rev 0	NA	NA
Cancel/Void []			

11a. Modification Work	11b. Work Package No.	11c. Modification Work Complete	11d. Restored to Original Condition (Temp. or Standby ECN only)
[] Yes (fill out Blk. 11b)	NA	NA	NA
[X] No (NA Blks. 11b, 11c, 11d)		Cog. Engineer Signature & Date	Cog. Engineer Signature & Date

12. Description of Change

WHC-SD-WM-SAR-062 is to be ~~completed~~ being reissued as Revision 1. Revision 1 is an update of the SAR to revise organizations, procedures, and documents to reflect the current WHC responsibiltiy for operation of the facility. Revise descriptions to reflect current equipment and mission of the facility. Correct known deficiencies and inconsistencies. Provide updated safety analysis or references thereto for current and known future activities. The document as been reformated to facilitate future changes. Refer to continuation sheets for additional details.

USQ Screening and Safety Evaluation K-94-0055

13a. Justification (mark one)						
	Criteria Change	[]	Design Improvement	[]	Environmental	[]
As-Found [X]	Facilitate Const.	[]	Const. Error/Omission	[]	Design Error/Omission	[]

13b. Justification Details

The changes are necessary to correct known deficiencies and inconsistencies and to update the SAR to reflect current organizations, equipment and plans.

14. Distribution (include name, MSIN, and no. of copies)

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1. ECN (use no. from pg. 1) 605351/~~601910~~ R bm 5/6/94

15. Design Verification Required	16. Cost Impact						17. Schedule Impact (days)		
	ENGINEERING			CONSTRUCTION					
[] Yes	Additional	[]	$ NA	Additional	[]	$ NA	Improvement	[]	NA
[X] No	Savings	[]	$	Savings	[]	$	Delay	[]	

18. Change Impact Review: Indicate the related documents (other than the engineering documents identified on Side 1) that will be affected by the change described in Block 12. Enter the affected document number in Block 19.

SDD/DD	[]	Seismic/Stress Analysis	[]	Tank Calibration Manual	[]
Functional Design Criteria	[]	Stress/Design Report	[]	Health Physics Procedure	[]
Operating Specification	[]	Interface Control Drawing	[]	Spares Multiple Unit Listing	[]
Criticality Specification	[]	Calibration Procedure	[]	Test Procedures/Specification	[]
Conceptual Design Report	[]	Installation Procedure	[]	Component Index	[]
Equipment Spec.	[]	Maintenance Procedure	[]	ASME Coded Item	[]
Const. Spec.	[]	Engineering Procedure	[]	Human Factor Consideration	[]
Procurement Spec.	[]	Operating Instruction	[]	Computer Software	[]
Vendor Information	[]	Operating Procedure	[]	Electric Circuit Schedule	[]
OM Manual	[]	Operational Safety Requirement	[]	ICRS Procedure	[]
FSAR/SAR	[]	IEFD Drawing	[]	Process Control Manual/Plan	[]
Safety Equipment List	[]	Cell Arrangement Drawing	[]	Process Flow Chart	[]
Radiation Work Permit	[]	Essential Material Specification	[]	Purchase Requisition	[]
Environmental Impact Statement	[]	Fac. Proc. Samp. Schedule	[]		[]
Environmental Report	[]	Inspection Plan	[]		[]
Environmental Permit	[]	Inventory Adjustment Request	[]		[]

19. Other Affected Documents: (NOTE: Documents listed below will not be revised by this ECN.) Signatures below indicate that the signing organization has been notified of other affected documents listed below.

Document Number/Revision	Document Number/Revision	Document Number Revision
None		

20. Approvals

	Signature	Date		Signature	Date
OPERATIONS AND ENGINEERING			ARCHITECT-ENGINEER		
Cog Engineer R. H. Meichle	R H Meichle	5/6/94	PE		
Cog. Mgr. C. L. Bennett	C L Bennett	5/9/94	QA		
QA G. M. Davis	J J Dietrich for G M Davis	5-12-94	Safety		
Safety R. L. Rathbun	RVS J L Rathbun	5-17-94	Design		
Security			Environ.		
Environ.			Other		
Projects/Programs					
Tank Waste Remediation System					
Facilities Operations J.E. Truax	J E Truax	5-19-94	DEPARTMENT OF ENERGY		
Restoration & Remediation			Signature or Letter No.		
Operations & Support Services					
IRM			ADDITIONAL		
Other J. P Schmidt	J P Schmidt	5/9/94			
V. L. Hoefer	V L Hoefer	05/12/94			

12. Desciption of Change (Continued)

A. Organizational changes & updates were made to the following which are listed by page, section and paragraph, sentence, etc as appropriate.

Page 1 - 1, Section 1.0 Paragraph 1; Page 5 - 19, Section 5.6, 2nd Paragraph; Page 5 - 20, Section 5.6.3, last Paragraph; Page 5 - 22, Section 5.6.5, 1st sentence; Page 6 - 1, Sections 6.1, 6.1.1, 6.1.2.1; Page 6 - 2, Sections 6.1.2.2, 6.1.2.3, 6.1.2.4, 6.1.2.5, 6.1.3, 6.2, 6.2.1; Page 6 - 3, Section 6.2.1.1; Page 6 - 4, Section 6.2.2; Page 6 - 5, Section 6.3; Page 6 - 7, Sections 6.5 & 6.7; Page6 - 8, Sections 6.7 & 6.8; Figures 6.1 & 6.2; Page 8 - 1, Section 8.1.1; Page 8 - 7, Section 8.2.8; Page 9 - 1, Section 9.1.

B. Facility equipment changes and corrections.

Page 1 - 1, Section 1.0, 2nd Paragraph; Page 1 - 2, Section 1.2, 1st Paragraph; Page 1 - 3, Section 1.2, 1st Paragraph; Page 1 - 3, Section 1.2, 4th bullet; Page 1 - 4, Section 1.2; last Paragraph; Page 2 - 2, Section 2.3, 2nd Paragraph; Page 2 - 5, Section 2.11.1 & 2.11.2; Page 4 - 1, Section 4.1, 1st Paragraph; Page 5 - 1, Section 5.1, 4th bullet; Page 5 - 3, Section 5.2, 1st & 2nd Paragraph; Page 5 - 6, Section 5.3.1; Page 5 - 7, Section 5.3.2.3.1 & 3.3.2.3.2; Page 5 - 10, Section 5.3.3.1, 3rd & 4th Paragraph; Page 5 - 11, Section 5.3.3.1, 1st Paragraph; Page 5 - 11, Section 5.3.3.3, 2nd Paragraph; Page 5 - 13, Section 5.4.1, 6th Paragraph; Page 5 - 18, Section 5.5; Page 5 - 21, Section 5.6.3, 4th & 5th bullet; Page 5 - 22, Section 5.6.4, 2nd Paragraph; Page 5 - 23, Section 5.7.1; Page 5 - 26, Section 7.7.4.3, 1st Paragraph; Pages 5 - 27, 5 - 28, & 5 - 30, Table 5.1; Page 5 - 33, Table 5.2; Figures 5.2 & 5.3.

C. Delete reference to response to alarms in N Reactor Control room.

Page 2 - 4, Section 2.9; Page 5 - 5, Section 5.2.3, last Paragraph; Page 5 - 7, 5.3.2.3.1, 2nd Paragraph; Page 5 - 11, Section 5.3.3.4.1; Page 5 - 12, Section 5.3.3.4.3; Page 5 - 12, Section 5.3.2.3.4; Page 6 - 4, Section 6.3.1.1; Page 6 - 5, Section 6.3.1.3; Page 6 - 5, Section 6.3.2; Page 6 - 5, Section 6.3.3; Page 6 - 7, Section 6.5, last Paragraph; Page 8 - 4, Section 8.2.1.

D. Revised minimum depth of water over fuel from 10 to 8 feet. The cited reference concluded that radiation dose rate at 8 feet is less than 1 mr/hr.

Page 5 - 4, Section 5.2.1, 1st Paragraph; Page 5 - 5, Section 5.2.3, 5th Paragraph; Page 5 - 15, Section 5.4.3.1, 4th bullet; Page 5 - 17, Section 5.4.4, 3rd Paragraph.

F. Updated safety analysis.

Page 1 - 1, Section 1.0, 2nd & 3rd Paragraph; Page 2 - 1, Section 2.1.1;

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Name: R. H. Meichle

R H Meichle

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7. Abstract

This safety analysis report (SAR) covers the handling, including load-in and load-out, and storage of predominately N Reactor fuel in the modified 105-KE and 105-KW fuel storage basins to provide the basis for the conclusion that operation does not present an unacceptable risk. This revision updates the SAR to reflect current facility configuration and organization, incorporates basis for hazard classification, and includes by addenda the Safety Evaluation for Encapsulation and Safety Equipment list. This document together with the OSRs serve as the authorization basis for the USQ process.

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ADDENDUMS

Addendum 1 WHC-SD-ND-TA-020 Rev 0, "Safety Evaluation of Fuel Encapsulation in the 105-KE Storage Basin", May 1992

ADDENDUM 2 WHC-SD-NR-SEL-001, "Safety Equipment List", July 14, 1993

1.0 INTRODUCTION

This Safety Analysis Report (SAR) covers the handling, including load in and load out, and storage of N Reactor fuel in the modified 105-KE and KW Fuel Storage Basins. It also covers special safety modifications to railroad facilities and special operating procedures for railroad equipment adjacent to the basins. Operation of the basins is the responsibility of Westinghouse Hanford Company (WHC) Spent Nuclear Fuel Project.

This Safety Analysis Report was originally prepared in accordance with U. S. Department of Energy Order DOE 5480.1A "Environmental Protection, Safety, and Health Protection Program for DOE Operations". Chapter V of DOE 5480.1A required that a SAR be prepared for new or modified nonreactor nuclear facilities. It replaces two Safety Analysis Reports: UNI-341, "Handling and Storage of Irradiated N Reactor Fuel in 105-KE Fuel Storage Facility" and UNI-1072, "Handling and Storage of Irradiated N Reactor Fuel in 105-KW Fuel Storage Facility" and their associated addenda. Information is provided on modifications implemented under Project H-558, "Hangers for N Reactor Irradiated Fuel Storage, 105-KE and KW", Design Change DC-82180, "105-KE Fuel Segregation Facility", and Design Changes 82121 and 82223 which add Decapping/Recanning capability and new ion exchange modules to KW and KE respectively. The equipment authorized by Design Change 82223 for decapping/recanning capability at KE was not installed due to the termination of the fuel segregation activity. New air-cooled chillers, initiated by ECNs 116077 and ECN 116078, 105 KE and 105KW Basin Heat Removal respectively, were installed.

The Safety Evaluation of Fuel Encapsulation in the 105-KE Storage Basin, WHC-SD-ND-TA-020 Rev 0, is attached as an addendum to this SAR. This document provides a description of encapsulation process, the equipment involved and a summary of the safety analysis. The safety analysis demonstrates that the encapsulation imposes no new or increased risk, i.e. it is within the existing safety envelope of the SAR. The identified criticality prevention specifications have been included in the process standards[45]. The issues of chlorine addressed in Section 5.0 of the safety evaluation have been resolved by installation of a new chlorine system that meets industry standards and use of 150 lb. chlorine bottles rather than the 1-ton vessel. Emergency response procedures for chlorine are also in use. The H_2SO_4 and NAOH issue has been resolved by deactivation of the demineralization plant. H_2SO_4 and NAOH are not needed. A D-Sump level indicator has also been installed at KE.

The safety equipment list document, WHC-SD-NR-SEL-001, is also included as an addendum. This document provides the Safety Class category for K Area systems. The bases for the safety classifications are found in WHC-CM-1-3, Management Requirements and Policy, MRP 5.46, Safety Classifications of Systems, Components, and Structures.

For the purpose of abbreviation, the term "KW facility", "KE facility", or just "facility" will be used throughout this document when referring to the

K-Area Fuel Storage Facilities. In addition, the term "lower enriched fuel" will be used when referring to the natural uranium (0.71wt% U-235) fuel elements, Mark IB and IVB, and the 0.95wt% U-235 Mark IC and IV fuel elements, while the term "spike fuel" will be used when referring to the Mark IA fuel elements which have a maximum enrichment of 1.25wt% U-235.

Fuel stored in the basin was destined for chemical processing in the PUREX facility. However, in December 1992, the U. S. Department of Energy directed that the PUREX Plant be shut down and deactivated because it is no longer needed to support the nation's weapons grade plutonium production. Final disposition of stored fuel at this time (1994) is in the Environmental Report preparation phase.

1.1 Site History

The K Reactors are located on the right bank of the Columbia river about three miles upstream from N Reactor. The rail and highway routes between them are about 6 1/2 miles long. The geology, seismology, hydrology and meteorology for the K Reactor sites are discussed in Section 4 and are essentially the same as for N Reactor site, described in Chapter 2 of Reference 1. The K Reactors were built in the early 1950's. They were graphite moderated plutonium producing reactors using once-through cooling. Each reactor was provided with a large water system which provided the once-through coolant. The system included a pump house, filtration plant, clearwell and combined outfall. Each reactor was provided with a large basin for underwater storage of irradiated fuel.

The K Reactors were shut down, KW in February, 1970 and KE in February, 1971. The stored fuel, except for a few loose pieces, was shipped to the 200-E Area for processing. The storage basins were then idle but kept filled with water. The area water system was shut down, except for a small portion periodically activated to provide a reservoir of water for sanitary and fire protection systems, process water for other activities, and make up water for the fuel storage basins.

1.2 Facility Description

The fuel storage basins are rectangular, reinforced concrete basins 125 ft. long, 67 ft wide, and 21 ft deep with a 16 foot water depth (Fig. 1.1). Connected to the south side of the basins and directly behind the reactor is a discharge pickup area, or pit, where fuel elements, as they were discharged from the reactor, were individually placed in baskets using manually operated, flotation-counter balanced tongs. Loaded fuel baskets were moved from the pit to storage locations in the basin by a monorail hoist system. The storage basins have now been modified and reactivated to accommodate storage of N Reactor fuel. The KE Facility was also provided with equipment for segregating N Reactor fuel into weapons stock, low blend stock, high blend stock, and remains stock. Following the segregation program, the equipment

was removed. The KW Facility has been provided with equipment for decapping fuel storage canisters.

Projects H-501 and H-508 modified the fuel storage facilities to provide for storage and handling of the N Reactor fuel design (Mark IA, IB, IC, IV and IVB). Each modification included a recirculating system for the pool water with in-line filters, an ion exchange system, a sand filter system, heat exchangers (which have been disconnected) and instrumentation to monitor radiation levels, temperature and pool water level. Included with the monitoring equipment is a remote alarm system. Storage racks for the Reactor fuel canisters are installed on the floor of the cooling pool.

The 105-KW Basin as modified under Project-H-508, "Supplemental N Reactor Irradiated Fuel Storage - 105-KW," was placed in service February 1981. These modifications were basically the same as those made at 105-KE under Project H-501, "Supplemental N Reactor Irradiated Fuel Storage - 105-KE," which was placed in service June 1975. The only significant differences between them as modified and used are:

- The 105-KW Basin was coated with a pliable epoxy sealant,
- Only 105-KE Basin discharge chute was coated with a pliable epoxy sealant,
- Only encapsulated canisters of fuel elements will be stored in 105-KW,
- The 105-KE Basin storage racks contain open canisters of fuel elements, and
- In KW the decapping station is located in the transfer canal between the south load out pit and the basin western bay.

Project H-558 was developed and implemented under a DOE-RL directive to provide additional spent fuel storage space in the K-Area Basins. Fuel storage capability is potentially increased by 375 MTU (80% Mark IV and 20% Mark IA) in each basin. This is done by providing the capability of hanging fuel over the fuel already stored on the basin floor. This is done in a ratio of one hanging fuel canister over three fuel canisters on the basin floor (See Figures 1.2 and 1.3). Actual hanger storage is confined to the center and east bays of each basin and thus amounts to an additional 250 MTU storage capacity in each basin. Some of the equipment installed for hanging fuel has been removed.

Design Change No. DC-82180 was initiated in 1982 to provide the capability for segregating N Reactor fuel by Pu-240 content into weapons stock, low blend stock, and high blend stock, and remains stock. This activity took place only in the KE discharge-pickup chute area, Figure 1.1. Following the segregation program, this equipment was removed.

Design Change No. DC-82107 was initiated at KE to provide better basin water temperature control during the summer months. A water chiller was installed, replacing Heat Exchanger 1A in KE. The chiller can provide 55°F water to the basin. The resulting lower basin water temperature should decrease the dose rate to personnel in the basin area. A similar project, DC-83057, was completed at KW in FY-84.

Design Change No's 82121 and 82223 were initiated to provide the capability to remove and replace the caps on Mark I and Mark II canisters. Mark I canisters were to be decapped in KE, while Mark II canisters will be decapped in KW. (Note that equipment was not installed in KE.) Additionally, these design changes added new auxiliary ion exchange modules to assist in basin water cleanup, and to control radionuclides release to basin water during canister decapping.

ECNs 116077, 105KE Basin Heat Removal, and 116078, 105KW Basin Heat Removal, provided air-cooled chillers with the capability to cool the basin without the need to use service water. This new unit improves the basin cooling reliability and reduces the quantity of water discharged to the Columbia River.

Figure 1.1

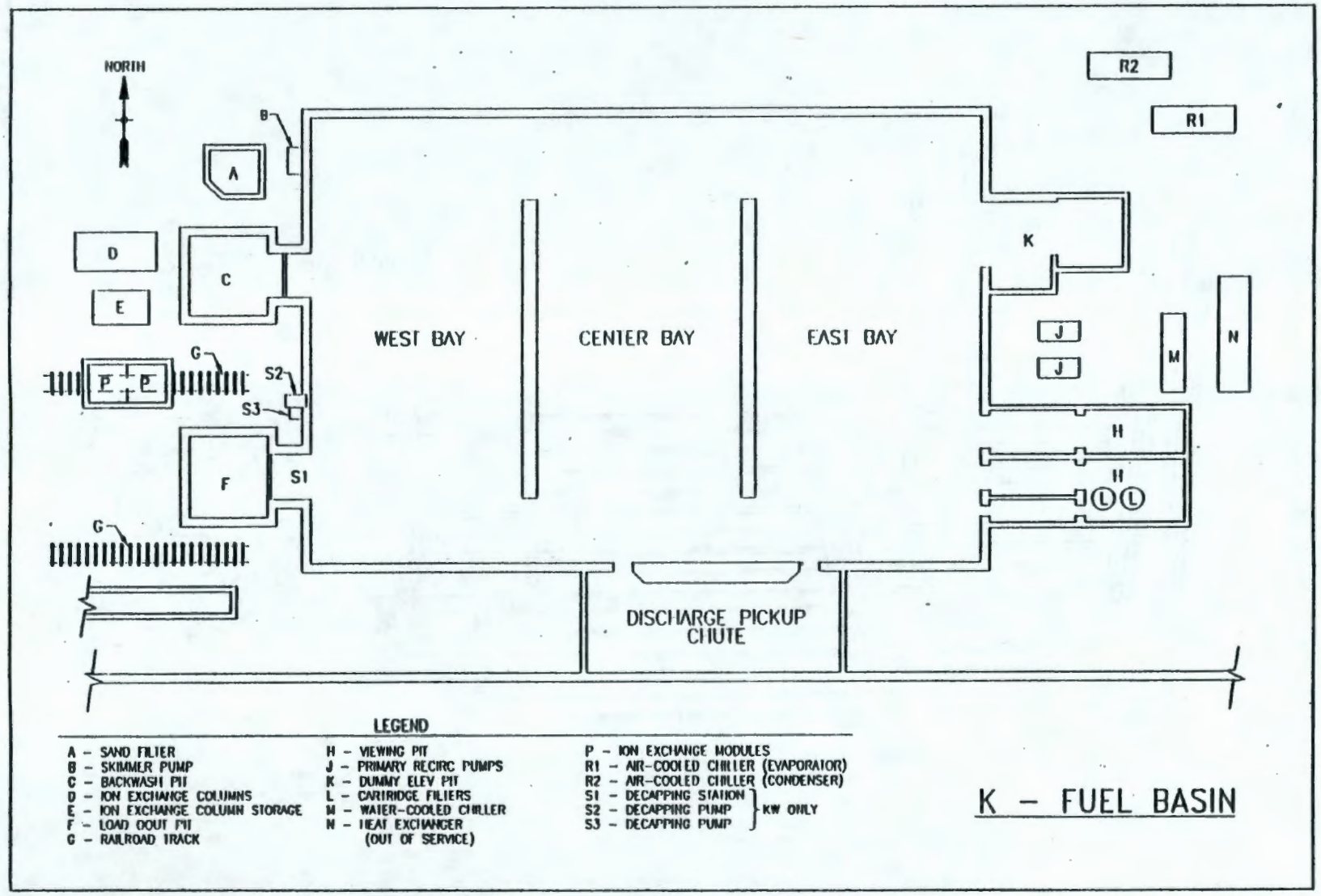

NORTH
A
B
C
D
E
F
G
G
P
P
S1
S2
S3
WEST BAY
CENTER BAY
EAST BAY
DISCHARGE PICKUP CHUTE
R2
R1
K
J
J
M
N
H
H
L
L
LEGEND
A - SAND FILTER
B - SKIMMER PUMP
C - BACKWASH PIT
D - ION EXCHANGE COLUMNS
E - ION EXCHANGE COLUMN STORAGE
F - LOAD OOUT PIT
G - RAILROAD TRACK
H - VIEWING PIT
J - PRIMARY RECIRC PUMPS
K - DUMMY ELEV PIT
L - CARTRIDGE FILTERS
M - WATER-COOLED CHILLER
N - HEAT EXCHANGER (OUT OF SERVICE)
P - ION EXCHANGE MODULES
R1 - AIR-COOLED CHILLER (EVAPORATOR)
R2 - AIR-COOLED CHILLER (CONDENSER)
S1 - DECAPPING STATION
S2 - DECAPPING PUMP
S3 - DECAPPING PUMP
KW ONLY
K - FUEL BASIN

Figure 1.2 Fuel Storage Configuration

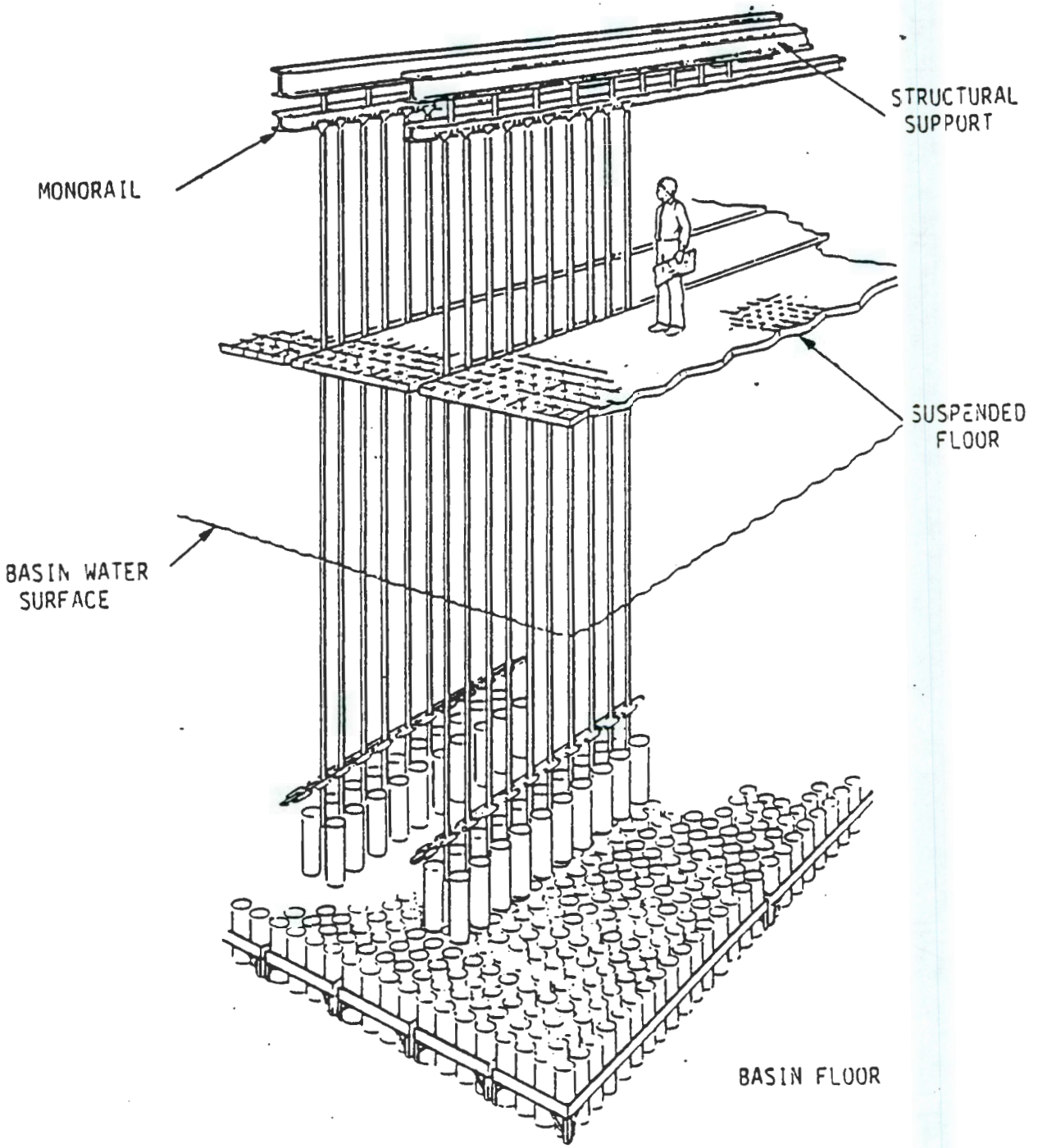
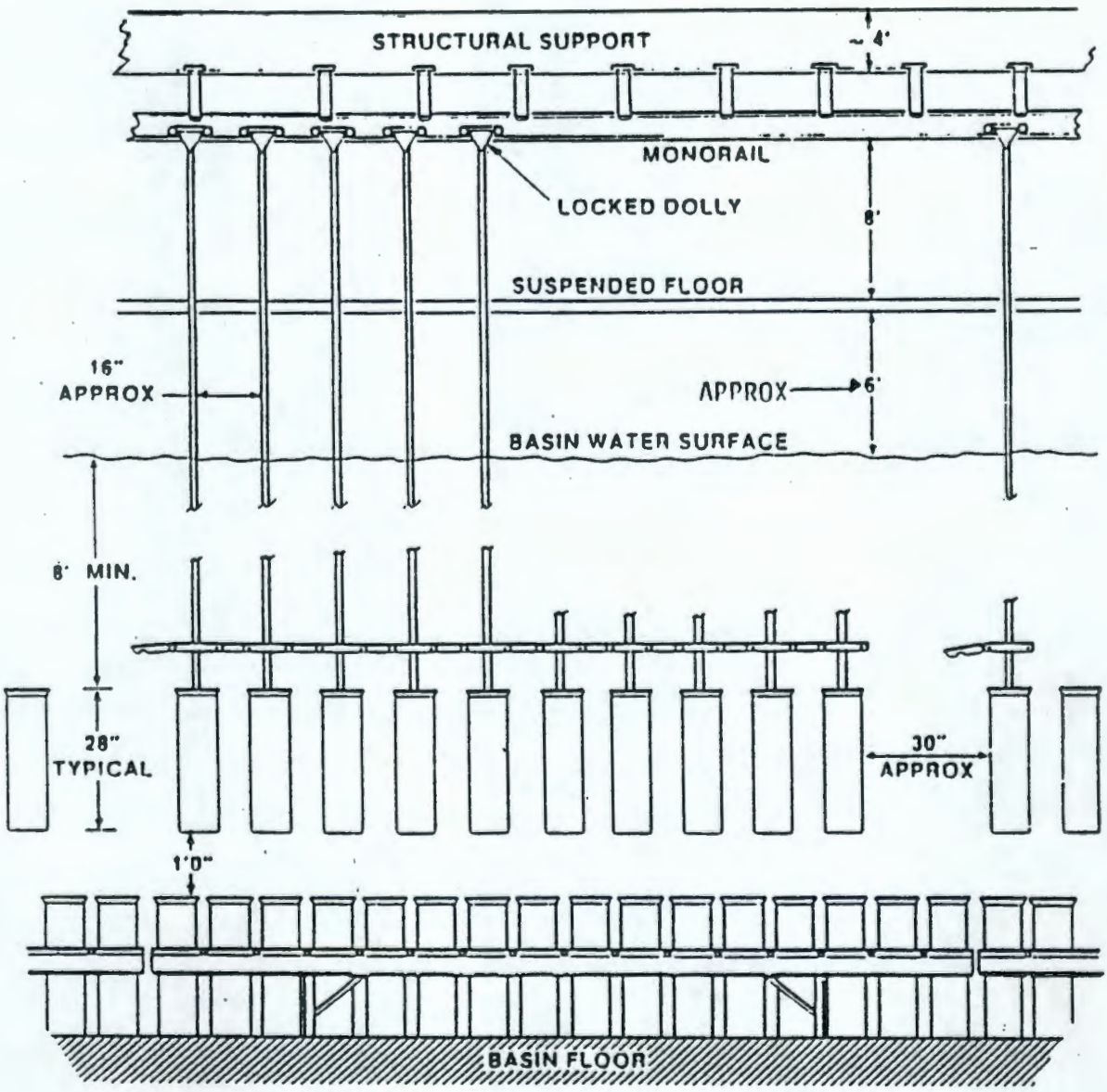


Figure 1.3 Profile of Fuel Storage Configuration



2.0 SUMMARY

2.1 General

The conversion of the K fuel storage facilities to handle, store, decap, and recan the N Reactor irradiated fuel elements packaged in the N Reactor canisters required facility and equipment modifications and changes in some operating parameters. A canister is a pair of connected cylindrical containers as shown in Figure 2.1. These changes and existing facilities have been studied to determine their capability to ensure nuclear and radiological safety. Results show that use of the converted facilities will not present any undue risk to the safety of plant personnel or the general public. Areas of particular significance are summarized below.

2.1.1 Hazard Classification

The hazard classification is based on a worst-case radiological release in a postulated accident scenario without consideration of control or mitigation provided by engineered or administrative barriers. The results of the unmitigated radiological dose consequence calculations indicate a maximum dose of 140 rem Effective Dose Equivalent (EDE) to the Maximum Onsite Individual and 46 EDE rem to the Maximum Offsite Individual.[11] The lower limits for a high hazard facility are 25 EDE rem onsite and 5 EDE rem offsite. The facility is therefore considered a High Hazard Nuclear Facility.[62]

The postulated scenario was that the release of radioactive materials is due to the heat up of the dry sediment in the basin. The sediment contains isotopes and fission products from the corrosion of defective fuel elements. The analyses used 3540 kg of sediment and a release fraction of 0.072

2.2 Nuclear Criticality

The storage, handling, decapping, and recanning of N Reactor fuel in the facilities have been evaluated for potential nuclear criticality accidents and found to be safe and within the nuclear criticality safety criteria and limits established for the fuel storage basins.[3,4,5,6,7] The buildup of fissile concentrations in sludge, ion exchange columns, and filters has been evaluated.[6,7,8] It has been found that when decapping operations are under way, precautions must be taken to ensure components remain critically safe.[7]

The prevention of accidental formation of critical masses in the facility is based primarily on confining the fuel in a critically safe geometry. Further, the control is based on the double contingency criterion which states that at least two unlikely, independent, and concurrent changes in process conditions must occur before a critical configuration is possible. No single contingency shall result in criticality. For conservatism, the safety analyses were based on unirradiated fuel critical mass parameters.

The storage configuration used in the safety analyses is shown in Figure 1.2 and 1.3. Storage racks, shown in Figures 2.2 and 2.3, are installed on the floor of the basin to insure the canistered fuel remains in the proper storage station. Each canister has a storage station that is permanently identified for the purpose of Special Nuclear Material (SNM) accountability. It is physically impossible for any one storage station to hold more than one canister due to dimensional restrictions.

In addition, there is the capability for hanging fuel suspended from the monorail and clamped at designated storage locations. Restraining devices prevent the canisters from hitting each other should a seismic event occur. The hanging fuel configuration is shown on Figures 1.2 and 1.3. No hanging fuel canister storage of 1.25wt% scrap filled canisters is allowed. Hanging canisters of any kind over canisters containing 1.25wt% scrap is not allowed.[7] Note that the equipment for hanging fuel is not installed.

Minor amounts of Single Pass Reactor fuel (SPR) are stored in the basins.

Canisters of 1.25wt % scrap can be stored with only two in adjacent locations, and areas in each basin must be set aside so that no other canisters are stored in adjacent storage rack locations. An effective water thickness of 12 inches is required for neutronic isolation of any two 1.25wt % scrap canisters from any other fissile material[7].

2.3 Radiological Control

There is no planned release of significant quantities of radioactive materials to the environment. Contaminated or potentially contaminated building service floor drains within the facilities boundaries have been intercepted and routed to a liquid effluent sump. Unused or unnecessary drains have been plugged and sealed with concrete. A pliable epoxy sealant has been applied to the floor and walls of the KW Basin to further limit leakage. The existing underbasin leakage collection system composed of an asphalt membrane and a pipeline to a dispersion tile field has been intercepted outside the facility. The contaminated effluents are routed to a sump and pumped back to the facility or to a radioactive waste holding tank. Leakage from the discharge pickup chute area of the basin is not intercepted by the underbasin leakage collection system.

The radiation dose rates above the facility pool from the stored fuel, and during normal fuel movements, is approximately 20 mrem/hr at KE and <1 mrem/hr at KW. A policy of exposure control is applied at the facility to maintain radiation exposure to personnel, from all sources, as low as reasonably achievable (ALARA).[12]

The flow averaged radioactivity concentration in the facility exhaust air is limited to that allowed by Derived Concentration Guides (DCG) of Reference 13 as being ALARA. This results in doses at the facility and site boundary to be at least three orders of magnitude less than the acceptance criteria of 5 rem and .5 rem respectively, consequently filters are not required.

2.4 Maximum Credible Accident

The maximum credible accident has been determined (section 8.2.4) to be the dropping and overturning of a loaded cask in the fuel transfer area with the fuel spilling out of the cask. Analyses show that the fuel would not melt,[14] however, a gaseous release might occur from any broken elements. The potential individual whole body dose at the site boundary from the airborne release would be 7.56×10^{-9} Rem[15] which is only a very small fraction of the guidelines given in 10 CFR 100. It is estimated that the dose of operating personnel would be less than 5 Rem. Section 3.2.4 of Addendum 1 provides the reevaluation of this accident to account for the presence of oxide powders due to fuel degradation. Although the radiological consequences of the maximum credible accident increase in this reevaluation, they remain well within the established risk base dose acceptance criteria of WHC-CM-4-46[73].

2.5 Design Basis Earthquake

The Design Basis Earthquake (DBE) was previously defined as an event producing a maximum horizontal ground acceleration of 0.25 g simultaneously with a maximum vertical ground acceleration of 0.17g at zero period. A 1990 structural analysis was performed whereby the wall between the pickup chute and basin was analyzed as a nonreactor safety class I structure. A 0.2g zero period median response spectra, which is the current DBE, with 7 percent damping was used per SDC 4.1, Revision 11, for evaluating the concrete wall and 0.5 percent damping response curves were used for finding the forces in liquid convective mode. The results indicated a positive margin of safety for the design basis seismic event when the water is maintained at the same level on both sides for water levels up to 19.5 feet.[66] However, the pool level is still limited to a nominal 16.5 feet. The cooling pool, cooling pool water supply and the recirculating systems equipment, equipment foundations and support structures, and the fuel storage racks and hangers will survive the design basis earthquake.

Calculations performed on the K-Area ventilation stacks[17] showed that they could not withstand a DBE (previously 0.25g). Project H-558, therefore provided for the removal of the upper 125 feet of each stack. The resulting height, 175 feet, reduces stress in the outer vertical reinforcing bars of the stacks so that they will be able to withstand a DBE.

Project H-558 also implemented the upgrading of the storage basin monorails for the addition of hanging fuel. Figure 1.2 shows the hanging fuel configuration. The design for this modification which added reinforcements to the monorails included consideration of the DBE. Dynamic testing of hanging fuel canisters provided design data to assure meeting the DBE.[18]

2.6 Tornado

The facilities were not designed to withstand the impact of a tornado. However, the structures were designed to withstand a bomb blast. Tornados would be expected to remove siding, but not collapse the structure.

Tornados in central Washington generally move from southwest to northeast. Items which could become airborne in the immediate vicinity of the facility are all located to the east or north. Along with the low probability of a tornado (6.82×10^{-6}/yr, see section 4.5) no airborne missiles are expected to be generated nearby which could impact on the stored fuel. Administrative controls are instituted to insure that the areas adjacent to the storage basins are kept free from potential tornado missiles.

A tornado is not expected to cause either the hanging fuel to drop or a loss of basin coolant. However, if such damage occurred it would not produce a criticality nor would the loss of coolant be an immediate problem (see section 8.2.7).

2.7 Monitoring Instrumentation

Surveillance and warning instrumentation and alarms are strategically placed to enhance the safety of personnel and plant equipment. These include systems for monitoring pool water level and temperature, pH, normal and high radiation dose rates, radiation levels from the ion exchange columns, and neutron dosimetry. There is also an air monitoring system to detect any airborne radioactivity in the basin area.

2.8 Personnel Safety

Areas in the 105-KW and KE Buildings have been provided to house the operating personnel and supporting crafts. The facilities have been equipped with the necessary sanitary and safety provisions, including fire protection as defined by DOE orders 5480.1A and 5483.1 which were the governing documents at the time of the upgrades. These areas are separated from the pool area, with access through vestibule doors. Exhaust fans in the basin area ensure that any airborne radioactivity will not penetrate the office and personnel support area.

2.9 Industrial Security

A security system is provided to detect unauthorized personnel intrusion during periods that the facility is unmanned by authorized personnel. This system provides audible alarms in the 200W Alarm Monitoring Station (AMS). The system is tested routinely and will alarm upon system failure.

2.10 Quality Assurance Programs[19, 20]

Quality assurance programs were implemented to provide assurance that the design, procurement, construction, testing, inspection, operation, and maintenance activities conducted at the facilities conform to regulatory and contractual requirements.

2.11 Pollution Control

2.11.1 Chemical Pollution

The only chemicals that may be used are those biocides for control of algae and other organisms. Chemicals may also be added for pH control. If needed, these chemicals are introduced into the cooling pool water, which is part of a closed, recirculating filtered system, and no planned discharge of chemicals to the environment will occur. Other chemicals, such as those used for equipment decontamination, may be used occasionally and their use will be specifically authorized and controlled.

2.11.2 Thermal Pollution

With an average age of 1 year (the fuel has now (1994) decayed for 7 to 23 years) for the hanging fuel (assuming a maximum of 375 MTU per basin at 7,500 BTU per ton per hour) and 2 years for the fuel resting on the basin floor (1650 MTU per basin at 2000 BTU per ton per hour[22]) the estimated heat load is approximately 1790 kilowatts. If each facility were to operate at the thermal capacity of its heat removal system the temperature of the river at its minimum flow rate would be raised about 0.002F using water-cooled heat removal systems. Operations with air-cooled chillers will result in no heating of the river.

2.12 Decommissioning

When the facilities are emptied and are no longer needed for fuel storage they will be returned to an environmental status equivalent to or better than existed prior to modification. Decommissioning will be accomplished at some future time under a decommissioning program encompassing all 105-K structures.

Figure 2.1 Fuel Storage Canisters

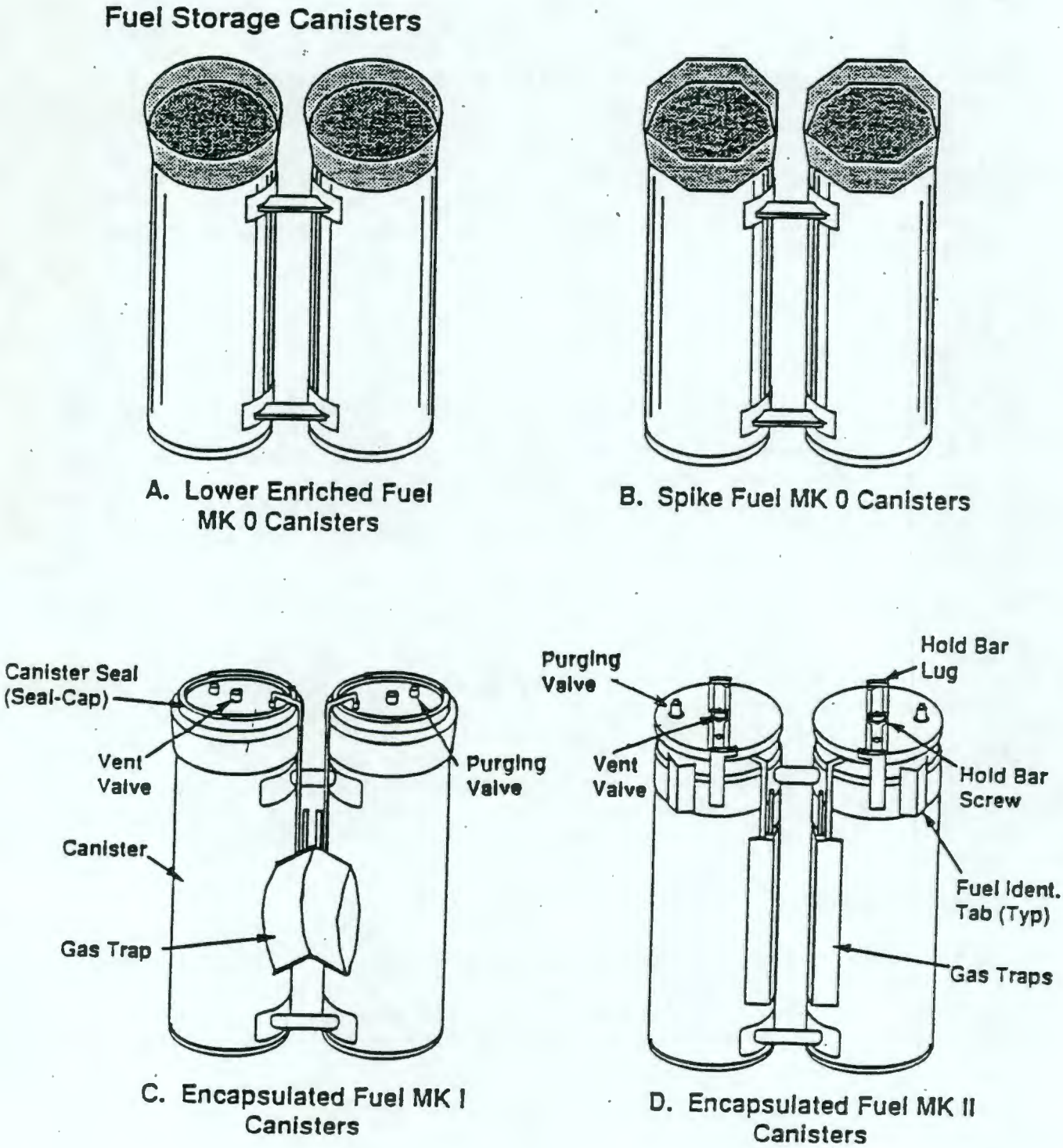


Figure 2.2 Canister Storage Racks - Prior to Installation

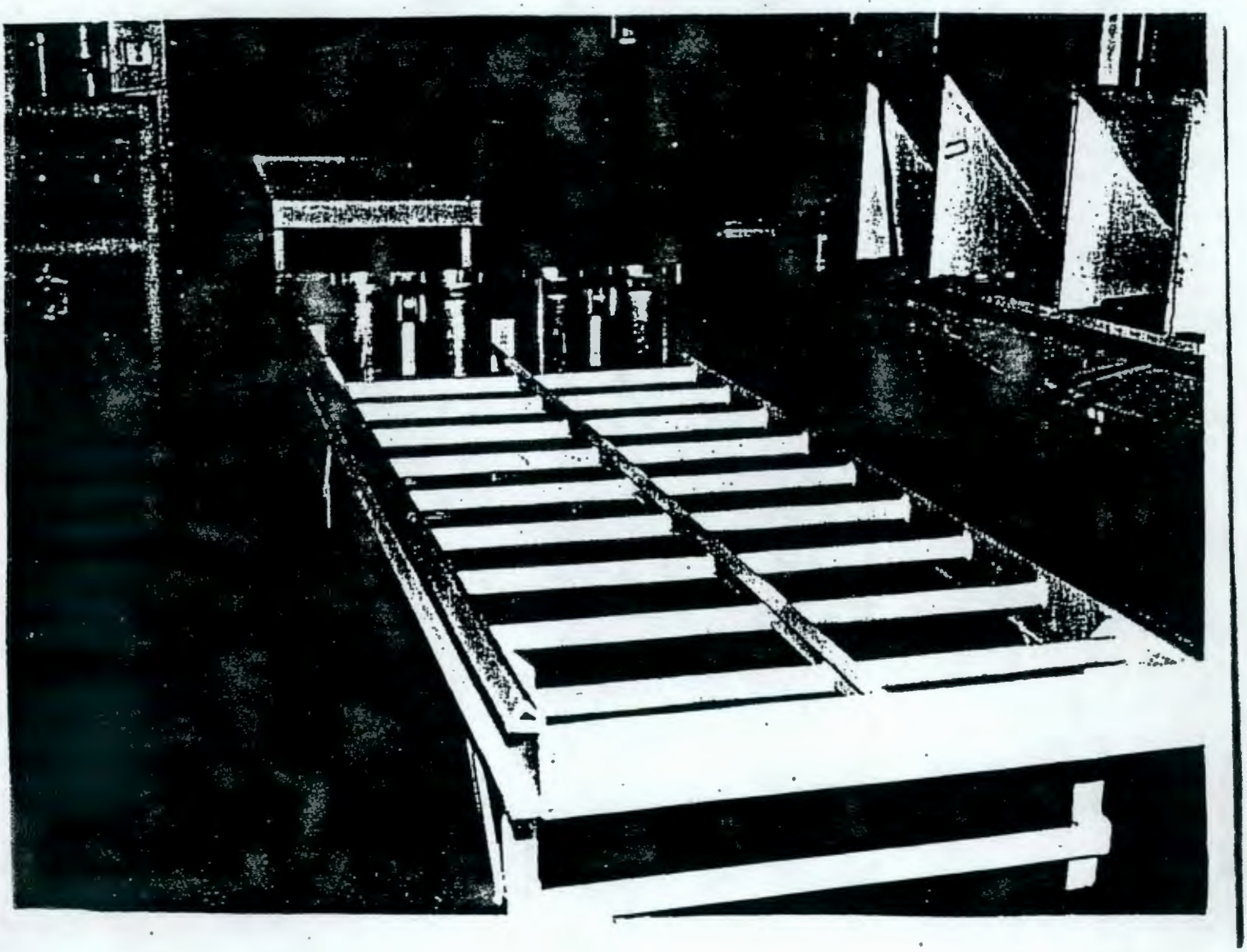
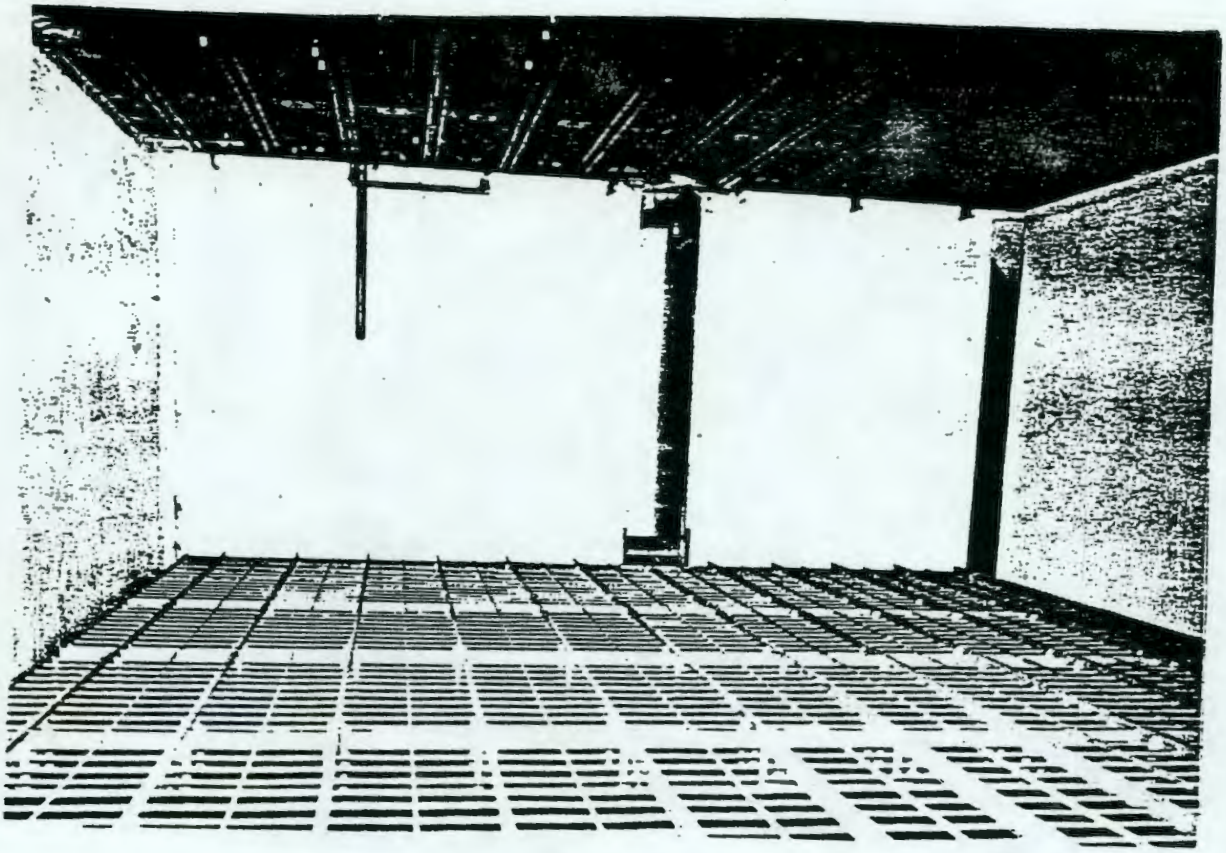


Figure 2.3 Canister Storage Racks - Installed



3.0 CONCLUSION

Irradiated N Reactor fuel, which has been cooled for at least 150 days*, will be safely stored in the K area fuel storage basins in a nominal 16 feet of water, using the storage configuration shown in Figures 1.2 and 1.3, and will present no undue risk to the health and safety of plant personnel or the general public. The water will prevent any significant release of radioactive materials to the environs and limit the radiation exposure in the work area. The potential whole body exposure at the exclusion area boundary from the airborne release due to the maximum credible accident would be 7.56×10^{-9} Rem compared to the 10 CFR 100 guideline of 25 Rem. Section 3.2.4 of Addendum 1 provides the reevaluation of the maximum credible accident to account for fuel degradation. The potential off-site dose was calculated to be 0.29 Rem EDE compared to the WHC guideline of 3.6 Rem (Reference 10 of Addendum 1).

The storage configuration shown in Figure 1.2 consists of full density storage on the basin floor. The full density storage array will allow for storage of up to 1173 MTU of irradiated Mark IA fuel or up to 1648 MTU of Mark IV fuel. The hanging fuel concept provides capacity for storage of up to 291 MTU of spike fuel (Mark IA) or 409 MTU of Mark IV fuel. A normal mix of Mark IA and IV (20% and 80% respectively) hanging fuel is 375 MTU. These quantities will be reduced by one third as long as hanging storage is not allowed in the west bay of each basin. Any combination of enrichments of fuel can be stored in the full density (floor) array and one-over-three hanging array, except that hanging fuel canister storage of 1.25wt% scrap filled canisters are not allowed, nor are any hanging canisters of any kind allowed over canisters containing 1.25wt% scrap.

Addendum 1, provides the conclusion, and its bases, that the encapsulation of stored fuel in KE Basin will present no undue risk to the health and safety of plant personnel or the general public.

*The basis for the original safety analysis; the fuel has now (1994) decayed for 7 to 23 years.

4.0 SITE CHARACTERISTICS

4.1 Location

The facility is located in the 100-K Area in the northwest quadrant of the 570 square mile DOE Hanford Reservation in the south central part of the State of Washington along the right bank of the Columbia River (see Figure 4.1). In addition to the two shutdown production reactors and the KE and KW Fuel Storage Facilities within the 100-K Area, the Hanford Reservation contains the following major facilities or activities: five additional reactor areas designated 100-B, 100-N, 100-D, 100-H, and 100-F, which contain six shutdown production reactors and one shutdown dual purpose reactor (N Reactor); the 300 Area which contains a shutdown fuel fabrication facility and laboratory facilities supporting all of DOE's Hanford Programs; two areas for waste processing and waste storage designated 200-E and 200-W Areas; a commercial nuclear waste burial operation on land leased to the State of Washington; the Fast Flux Test Facility (FFTF) currently in the shutdown process; and two partially constructed and one operating Washington Public Power Supply System nuclear power plants.

Major metropolitan areas within the broad vicinity of the plant (see Figure 4.2) include Spokane, Washington, about 120 miles to the northeast; Seattle, Washington, about 130 miles to the northwest; and Portland, Oregon, about 150 miles to the southwest. Two other areas of significant population density include Moses Lake, Washington, about 30 miles north of the K-area and the Yakima Valley, in Washington, extending from Yakima, about 45 miles west of the plant, to the Tri-Cities, in Washington, about 35 miles southeast of the plant.

A more detailed description of the general site location and additional information on the surrounding population density may be found in the N Reactor Updated Safety Analysis Report (NUSAR) Reference 1, Section 2.1.

4.2 Associated Facilities

Both the KW and KE Facilities are located within the 100-K Area perimeter as shown in Figure 4.3. Table 4.1 lists the active and inactive buildings/facilities located at K area. The following are the major buildings which contain supporting equipment in terms of water and electrical supply to both of the fuel storage basins:

The 183-KE building is the water treatment plant where the water is filtered and chemically treated for removal of solids. Sedimentation basins with a total capacity of 6.3×10^6 gallons of water have been available for servicing the KE and KW Facilities.

The 181-KE river pump station contains the two raw water pumps used to pump water from the Columbia River to the filter plant (183-KE building). These pumps each have a 10,000 gpm capacity.

The 1706-KE Building contains the demineralization plant (capable of producing 100 gpm) and a 16,000 gal demineralized water storage tank. The demireralization plant is no longer used. The 1706-KE building houses the Engineering/Environmental Demonstration Laboratory which is independently operated.

The 190-KE Building contains the three service water pumps that supply the treated water to the KE and KW facilities. Each of these pumps has a 5,000 gpm capacity.

The 165-KE Control Building contains the station service transformers and the temperature indicators and annunciator equipment for monitoring the 181-KE River Pump Station and 190-KE pumps and motor operation, and additional office space.

The 1717 Building houses maintenance shops and craft personnel.

The backup cooling water for the fuel storage basins is supplied from large underground storage structures commonly called the clearwells. Each of the two clearwells can contain approximately 9.0×10^6 gallons of treated water. If electrical power is lost to the area (only one source of power is available), analysis of fuel heatup shows that at least 60 hours are available to restore the heat removal equipment to service before a basin water temperature of 130°F would be reached (see Section 5.3.3.1). If necessary, once-through cooling water, supplied from the clearwells by portable pumps or fire trucks, would provide temporary emergency cooling. The overflow water would be collected in a crib outside the facility (see section 5.3.3.1).

4.3 Floods

The highest recorded flood stage of the Columbia River at the facility site occurred in 1894 and was 415 feet above mean sea level (MSL). The Probable Maximum Flood (PMF) as defined by the Corps of Engineers and described in section 2.4.3. of Reference 1 (NUSAR) will reach 423 feet MSL. The 181-KE River Pump Station is located at an elevation of 421 feet MSL, the fuel storage basin is located at an elevation of 465 feet MSL and the vital equipment located in the 165-KE, 183-KE and 190-KE are at an elevation of 459 feet MSL or higher.

Thus, if the PMF did occur, the two raw water pumps located at the 181-KE River Pump Station would be submerged in about 2 feet of water (these pumps are powered by electric motors which would probably be destroyed). However, the fuel storage facility and the vital supporting equipment would be at least 42 feet and 36 feet above the maximum flood level. In this instance, the clearwells are capable of supplying water to the cooling pools as needed.

Realistic modes of upstream dam failures or damage would produce a flood not exceeding the PMF.

For further information on the PMF, the general hydrology of the area, or the upstream dam failure analysis see Section 2.4 of Reference 1.

4.4 Seismology

The Hanford Reservation is in a region of low to moderate seismicity. The historical record of seismic activity within a 200 mile radius of the 100-K Area site dates from 1827. During this time there have been only 14 earthquakes which may have been felt or were recorded at or near the 100-K Area site. Five of these may have affected the site with an intensity of IV or greater on the Modified Mercalli Scale (MM). The most severe of these occurred on December 14, 1872 with a probable epicenter located about 85-100 miles north of the facility. No historical record of the effect of the 1872 quake on the 100-K site is available, but it is postulated to be in the intensity range of V to VI MM. For more information concerning the seismology and geology of this area, see Section 2.5 of Reference 1 (NUSAR).

4.5 Wind and Tornado

The site is subject to frequent strong westerly winds. The all-time peak gust of 80 mph was recorded January 11, 1972. The 80 mph gust is expected to occur once every 30 years. A peak of 96 mph would be expected to occur once every 500 years[1].

The site is well outside of established tornado alleys. The probability of a tornado in any year at any point within the 100 mile radius of the Hanford Meteorology Station is 6.8×10^{-6}/yr.[24]

The Hanford design base tornado is defined as having a 150 mph rotational velocity and a 25 mph translational speed. When the facilities were designed and constructed this criterion had not been established. Consequently the facilities are not designed to withstand the impact of a tornado. The structural parts of the facilities were designed to withstand a bomb blast. Tornados would be expected to cause some structural damage but not collapse the structure. However, even if the structure were to collapse, the hanging fuel would be dropped, but would not cause a criticality.[3]

4.6 Ashfall

The Hanford reservation is in a region subject to ashfall from volcanic eruptions. The three major volcanic peaks closest to the project are: Mt. Adams about 100 miles away, Mt. Rainier at about 110 miles away, and Mt. St. Helens approximately 130 miles away.

Important historical ashfalls affecting this location were from eruptions of Glacier Peak about 10,000 BC, Mt. Mazama about 4000 BC, and Mt. St. Helens about 6000 BC. The most recent ashfall resulted from the May 18, 1980 eruption of Mt. St. Helens.

As a result of the 1980 Mt. St. Helens eruption, the site design criteria were modified to include ashfall. The 105-KW roof was replaced in 1988 due to concern for degradation of original roof panels at that facility. The 105-KW roof has been analyzed and will survive the limiting live load combination consisting of the design basis snow and ashfall loads of the current site design criteria.[64]

Deterioration of the 105-KE roof structure membrane has occurred and is to be replaced as part of the N Reactor Facilities Roof Repair Program Management Plan.[68] The replacement will not change the loading criteria, so the original Uniform Building Code Criteria involving live snow and seismic loads remain applicable. The potential roof collapse due to deterioration has been reviewed and would not result in consequences to the safety class structures or equipment in the KE Basin.

TABLE 4.1 100 K Area Buildings/Facilities

ACTIVE BUILDINGS/FACILITIES		INACTIVE BUILDINGS/FACILITIES	
105KE	FUEL STORAGE BASIN AREA	105KE	PROCESS AREA
105KW	FUEL STORAGE BASIN AREA	105KW	PROCESS AREA
115K	SWITCHING STATION	107KE	WATER RETENTION BASIN
151KE	SUBSTATION 230-KV	107KW	WATER RETENTION BASIN
151KW	SUBSTATION 230-KV	115KE	GAS RECIRCULATION BUILDING
165KE	POWER CONTROL BUILDING	115KW	GAS RECIRCULATION BUILDING
165KW	POWER CONTROL BUILDING	116K-1	EMERGENCY CRIB
167K	CROSS TIE TUNNEL BUILDING	116K-2	.8 MILE WASTE TRENCH
181KE	RIVER PUMP HOUSE	116KE	REACTOR STACK - STRUCTURE
183KE	HEADHOUSE/CHLORINE VAULT	116KW	REACTOR STACK - STRUCTURE
190KE	MAIN PUMP HOUSE	117KE	EXHAUST AIR FILTER BUILDING
1706KE	WATER STUDIES SEMIWORKS FACILITY	117KW	EXHAUST AIR FILTER BUILDING
1706KEL	DEVELOPMENT LABORATORY	118K	SOLID WASTE BURIAL
1706KER	WATER STUDIES RECIRCULATION BUILDING	119KE	EXHAUST AIR SAMPLE BUILDING
1717KE	MAINTENANCE SHOP FACILITY	119KW	EXHAUST AIR SAMPLE BUILDING
------	YARD EQUIPMENT INCLUDING RAILROAD TRACK SYSTEM AND ASSOCIATED SWITCHES AND DERAILERS USED DURING THE TRANSPORT OF FUEL INTO AND OUT OF THE 105KE AND 105KW FUEL STORAGE BASINS	150KE	HEAT RECOVERY FACILITY
		150KW	HEAT RECOVERY FACILITY
		166KE	OIL BUNKER
		166KW	OIL BUNKER
		181KW	RIVER PUMP HOUSE
1713KE	SHOP BUILDING	182K	WATER PUMP HOUSE
1713KER	WAREHOUSE	183KW	HEADHOUSE/CHLORINE VAULT
1718K	MOBILE OFFICE (MO401)	190KW	MAIN PUMP HOUSE
		1614K	MONITORING STATION

Figure 4.1 Site

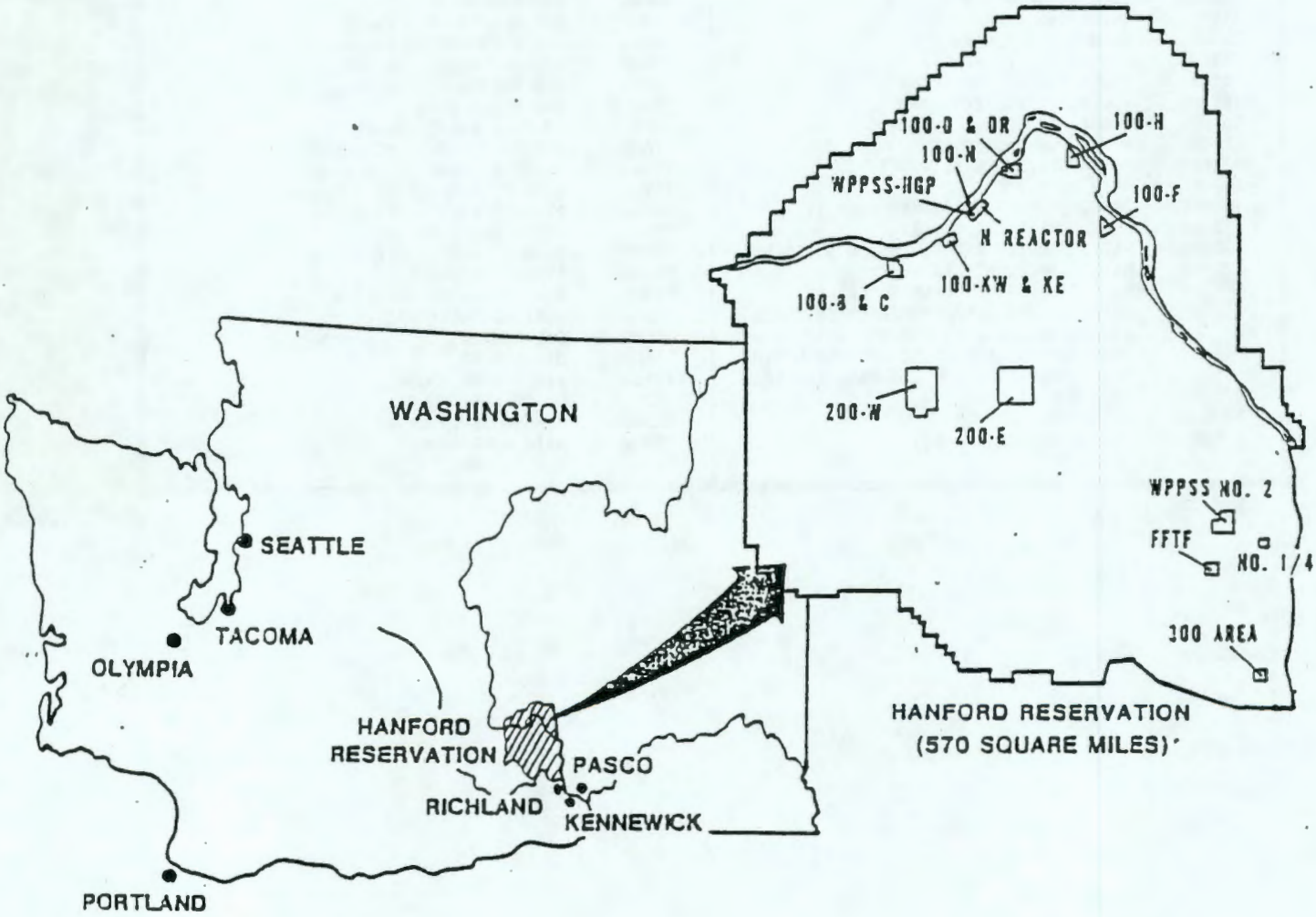


Figure 4.2 Location of Plant Relative to Major Metropolitan Areas

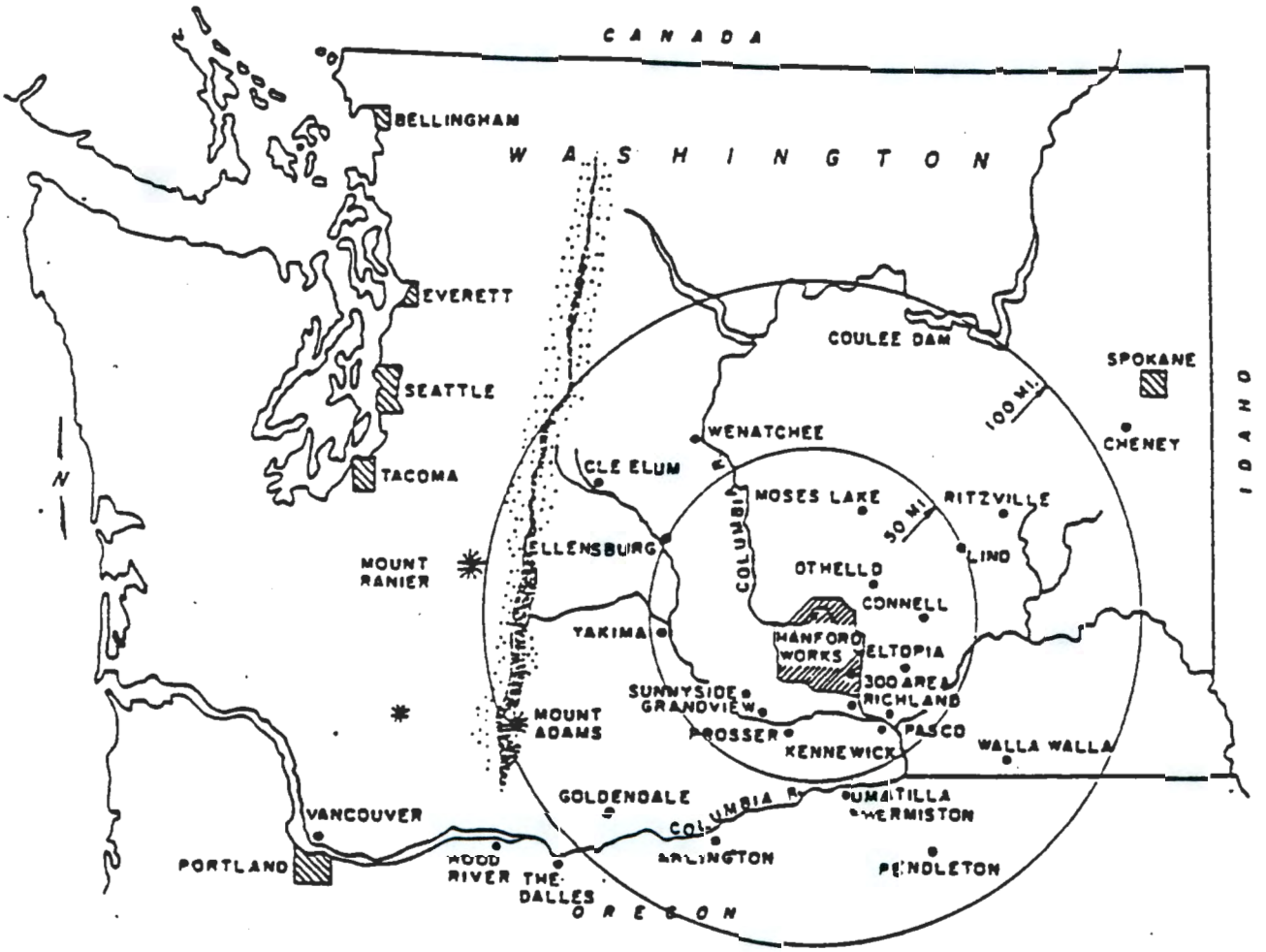
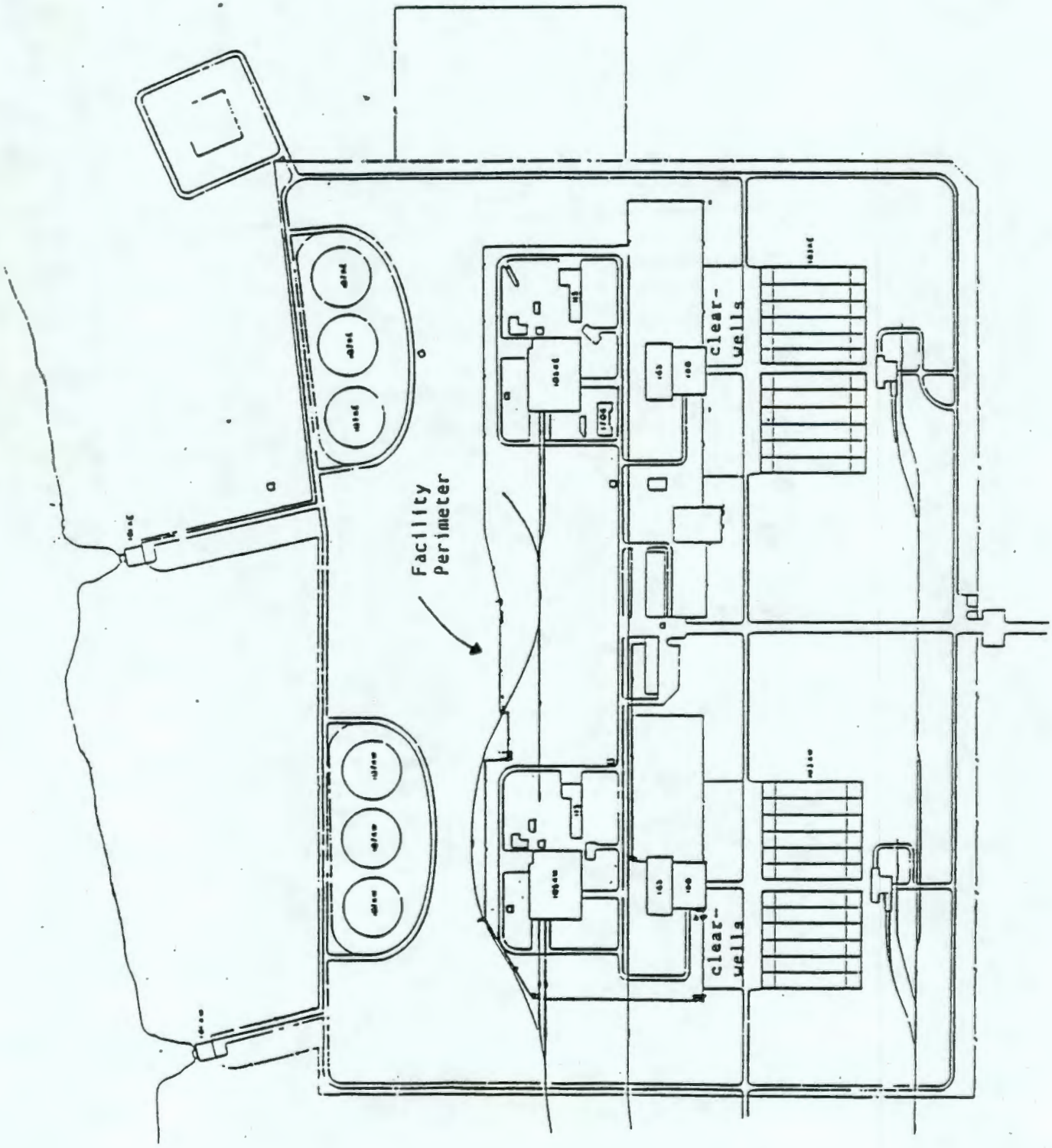


Figure 4.3 100-K Area



5.0 STRUCTURE, COMPONENTS, AND SYSTEMS

5.1 General

The KW and KE reactor facilities were originally of essentially the same design and the modifications made to the KW facility under Project H-508 and to the KE facility under Project H-501 were basically the same.

Significant differences between the KE and KW facilities are:

- A pliable epoxy seal has been applied to the KW concrete basin,
- A pliable epoxy seal has been applied to the KE discharge chute,
- The KE storage racks have been filled with open (unencapsulated) canisters,
- Only encapsulated canisters of fuel elements are stored in KW,
- The decapping station in KW.
- A new roof was installed over the basin in KW.
- The KW basin has tapered walls.

The epoxy sealing of KW Basin and the use of encapsulated canisters are improvements made due to experience at KE. The epoxy seal reduces the probability of basin leakage and will facilitate decontamination on deactivation. The encapsulated canisters will significantly reduce the release of radioactive contaminants into the cooling water and consequent load reduction on ion exchange and filtering systems. It also provides a significant reduction in personnel radiation exposure.

5.1.1 Comparison to 10 CFR 50 Appendix A

Although compliance with 10 CFR 50 is not required, a comparison with 10 CFR 50 Appendix A was made to aid in judging the adequacy of the facility. It was concluded that only Paragraph VI, "Fuel and Radioactivity Control", Criteria 60, 61, 62, 63 and 64 are applicable for comparison to the fuel storage basins and that the facility design is in conformance with these criteria except No. 60 which is discussed below.

Criterion 60 - "Control of Release of Radioactive Material to the Environment"
The facility design does not include means to control the release of radioactive gaseous effluents nor provide any holdup capacity for retention of gaseous effluents containing radioactive materials.

The fuel storage basin design criteria specifies transportation and storage of N Reactor irradiated fuel, cooled for a minimum of 150 days after reactor discharge. Based on previous irradiated fuel storage experience at N Reactor and at the other Hanford Reactors, a small amount of radioactive noble gases may be released due to corrosion of broken fuel, but the amounts are so small that no provisions for containment of radioactive gases are included in the functional design criteria. However, monitoring equipment is provided to monitor the facilities atmosphere for airborne radioactivity that may be released during normal operation, including anticipated operational occurrences and from postulated accidents. The flow-averaged airborne radioactive concentrations averaged for a year for all discharges points will not exceed the Derived Concentration Guides (DCG), of DOE Order 5400.5 Radiation Protection of the Public and the Environment[13]. This release is considered ALARA and ensures that the public and occupational airborne dose limits of DOE Order 5400.5 and 5480.11, respectively, are not exceeded. Consequently filters are not required.

5.1.2 Comparison to NRC Regulatory Guide 1.13 - "Spent Fuel Storage Facility Design Basis"

NRC Regulatory Guide 1.13 provides design details based on the requirements of 10 CFR, Part 50, Appendix A, 1973, Paragraph VI, Criterion 61. Although compliance is not required, a comparison with the requirements of Regulatory Guide 1.13 was made to aid in judging the adequacy of the facility, and it's design was found to be in substantial conformance.

A k-effective of 0.98 is used in lieu of 0.95 because the Regulatory Guide is directed at commercial power reactor fuels that are more highly enriched than N Reactor fuel. The mass difference between a k-effective of 0.98 and a k-effective of 1.0 for N Reactor fuel enrichments is greater than the mass difference between a K-effective of 0.95 and a k-effective of 1.0 for power reactor fuel enrichments. In addition, subcriticality measurements, detailed in Reference 4, and other data developed over the years at Hanford give a high degree of confidence in the results of the criticality analysis.

5.1.3 Basic Design Requirements

The basic design requirements for Project-501 modifications to the facilities and associated equipment under the three primary modification projects were:

- Store irradiated N Reactor fuel for a period of 15 years.
- Limit radioactive release to the environment to within established limits.
- Protect personnel from undue physical and radiological hazards during normal or abnormal operation of the facility and related equipment.

The more specific design requirements may be found in the Functional Design Criteria, References 25, 26, and 27.

There is a program plan[63] for facility life extension. The Program Plan anticipates providing a facility suitable for interim storage of the fuel for 20 years from 1992. A program is being developed to remove fuel and water from the basin by the year 2002. This program is in the environmental report preparation phase.

5.2 Facility Cooling Pool

As discussed previously, the cooling pools are existing structures that were modified for the purpose of storing irradiated N Reactor fuel. They are rectangular reinforced concrete structures 125 feet long, 67 feet wide and 21 feet deep. The pools are divided into three sections by concrete walls which are open at each end (see Figure 1.1). The bottom of the pools are approximately 20 feet below grade. The cooling pools are filled to a nominal depth of 16 feet. The water circulates through the pool by drawing water from one end of each basin section, circulating the water through filters, the water and/or air-cooled chiller, and the ion exchange columns, then discharging it back at the opposite end of the sections. The maximum operational water temperature will be 100°F.[26,27] The water provides radiation shielding and cooling for the stored fuel. The storage racks and hangers provide individual storage stations for the canisters and insure the storage configuration remains intact during a design basis earthquake.

The original modifications were made to the cooling pools under Projects H-501, H-508, and H-558. Installation of the segregation equipment was done under design change DC-82180. Following the segregation program, this equipment was removed. Heat exchangers 1A were replaced with water chillers in each basin under DC-82107 for KE and DC-83057 for KW. Decapping facilities were added to KW under DC-82121. ECN 116077 for KE and ECN 116078 for KW installed air-cooled chillers with sufficient capacity to replace water chillers. The ability to operate water chillers is retained.

5.2.1 Design Description

Fuel storage racks, shown in Figure 2.2, were installed in the cooling pools as shown in Figure 2.3. These storage racks were developed under Projects H-501 and H-508 and are designed with individual storage stations large enough for only one fuel canister per station. The individual racks are interlocked to form one solid unit across the pool floor and are butted tight to the pool walls.

Project H-558[25] provides an additional capacity of 1240 fuel canister storage positions per basin. The fuel will be hung from the monorails with 40 canisters to a monorail. There are 32 monorails of which 31 could be used for storage. The remaining monorail is to be maintained as a spare and for operating flexibility. However, only 21 monorails are now usable since

hanging storage is not currently allowed in the west bay of either basin. Additional bracing was added to the monorail supporting structure under Project H-558 to meet seismic criteria.

The hanging fuel canisters on each monorail will be interlocked into four groups of 10 to prevent them from knocking together should a seismic event occur. Thirty inches of space between groups are provided to prevent them from bumping each other. There will be about 44 inches between rows of hanging canisters, a sufficient distance to prevent them from colliding during a seismic event. The hangers will be attached to trolleys and each trolley will be clamped to the monorail at a specific marked location.

All the pool floor drains are plugged and sealed with concrete. The floor and walls of the KW pool are coated with a pliable epoxy sealant. Water provides the radiation shielding over the irradiated fuel in the pool. The level in the pool is maintained at a nominal 16 feet which provides for a nominal 8 feet of water over the fuel during all normal operation conditions in the basins. Figure 5.2 shows the flow diagram of the water system which provides for basin makup, heat removal from water-cooled chiller, sanitary water, and fire protection water.

5.2.2 Design Evaluation

The installed fuel storage racks and hangers in the cooling pool provide 6133 canister storage locations in each facility plus an additional 174 spaces in the KW discharge chute area. Calculations, supplemented with subcritical mass experiments provided in PT-506,[4] show that by using the fuel storage configuration shown in Figures 1.2 and 1.3, a criticality could not occur with the failure of a single contingency. The approved storage configuration consists of full density storage for fuel located on the basin floor. Hanging fuel, in a 1 over 3 array can consist of all spike fuel. The full density Storage array plus the hanging fuel locations will allow for storage of a maximum of 1925 MTU (2125 tons) of irradiated fuel in each facility plus an additional 55 MTU (60 tons) in the KW discharge chute.

Upon deactivation of the single-pass reactors, some irradiated single-pass reactor fuel elements remained stored at the KE fuel storage pool. Single-pass reactor fuel can safely be stored in the same facility with the Mark IA and Mark IV N Reactor fuel in quantities up to 5% of the minimum critical mass (MCM) in each single-pass reactor fuel container.[28] This limit is compatible with analyses performed for any accident that can occur with the N Reactor fuel stored in the facility. Each N Reactor fuel canister contains up to 5% MCM. Therefore, if a container of single-pass reactor fuel were dropped or spilled into the storage array of N Reactor fuel canisters, the change in the k-effective value of the storage array would be no more than for the dropping or spilling of a canister of N Reactor fuel into the storage array. (See Section 8.1.3.)

The container weight limit of 545 pounds(5% MCM) of single-pass reactor fuel, identification, labeling and isolation requirements are specified in the Process Standards for the facility.

5.2.3 Discussion

The following discussion concerns operations in the main basin area only. Decapping operations are discussed in section 5.7.

The fuel cask is physically prevented from traversing over the irradiated fuel storage pool by the building structure and the absence of crane rails into this area.

Assurance against the storage configuration being altered by a cask dropping onto the racks is provided by the absence of rails that would allow crane travel in this area. The crane cannot position a cask over the racks and stored canisters.

The canister hoists and hangers are the only installed mechanical lifting devices capable of moving objects over the irradiated fuel racks. The rated capacity of the hoist is 0.9 metric tons (one ton) working load. A steel grating platform covers the entire cooling pool area which serves as a platform to work from and as a guide for installing canisters of fuel into individual storage rows (see Figure 5.1). The grating also prevents a hoist from dropping onto the irradiated fuel storage racks and canisters. Lighting fixtures and other components of the building above the cooling pool are not sufficiently massive to destroy the grating and cause damage to the fuel racks or canisters below this level.

The consequence of a transfer cask dropping into the cooling pool loadout pit has been analyzed[16] and is discussed in Section 8.2.3.

By maintaining the water level in the cooling pool at approximately 16 feet at least 8 feet of water will be over the irradiated fuel during all normal operating modes. Assurance against loss of water, which would result in an increase in radiation levels, is provided by sealing all floor drains in the pool with concrete. Additional protection is provided at KW by coating the walls and floor of the pool with a pliable epoxy sealant. The only escape route for the water from the pool is through overflow weirs located approximately 2.5 feet above the pool water level. With 8 feet of water over the irradiated fuel, the radiation dose rate from the fuel was calculated to be less than 1.0 mR/hr.[9] However, actual experience at KE has shown dose rates to be higher than predicted due to the presence of dissolved radionuclides in the basin water. The dose rates are still within acceptable limits for controlled radiation zone work.

A radiation monitoring system has been provided at the facilities for personnel protection and general surveillance. Continuous monitoring and recording readouts and high radiation level alarms have been provided in the

facility control rooms, and the 200 W Alarm Station. Audible and visual indicators are also provided in facility pool areas.

5.3 Facility Pool Cooling and Cleanup System

Each facility pool cooling and cleanup system is designed to perform three primary functions:

1) Remove the decay heat generated by the irradiated fuel stored in the basin.
2) Maintain water quality and clarity.
3) Control the concentration of soluble and particulate radioactive nuclides in the water to minimize personnel radiation exposure.

Table 5.1 describes the design parameters for the system, the design data are included in Table 5.2. The more specific design requirements for this system may be found in the Functional Design Criteria, References 26 and 27.

5.3.1 System Description

Each pool cooling and cleanup system consists of a recirculation system, an ion exchange system, a sand filter system, and several other small subsystems (see Figure 5.2 for flow diagram). The recirculation system includes the recirculation pumps, water-cooled or air-cooled chiller, ion exchange columns, cartridge filters, and the necessary valves and piping. Table 5.1 provides the design and operating parameters for this system.

The recirculation system draws water from the three cooling pool sections through a common header to the suction side of the recirculation pumps. The water is then pumped through a 5-micron disposable cartridge type filter to the water-cooled or air-cooled chiller heat exchanger and then returned to the cooling pool. Part of the discharge, up to 150 gpm , passes through the ion exchange system before returning to the pool.

Water is supplied to the ion exchange columns through a common header connected to the recirculation system. Flow from the ion exchange system is discharged into the cooling pool through return piping.

Water is drawn from the cooling pool through three skimmers to the suction side of the skimmer pump, and then pumped through the sand filter, through the ion exchange modules (IXM), and back into the basin (see Figure 5.3). Included with the sand filter system is a settling pit for backwash water.

A demineralized water supply line with a manual control valve supplies the cooling pools with normal makeup. The makeup line discharge is above the normal water level and visible so that makeup can be confirmed. An emergency makeup valve that can supply 1500 gpm of filtered water to the cooling pool is available to recover from an abnormal reduction in water level.

5.3.2 Component Description

Following is a brief description of the major components that make up the fuel pool cooling and cleanup system.

5.3.2.1 Cooling Pool Recirculation Pumps

The cooling pool recirculation system in each basin includes two pumps which are the primary units for moving the pool coolant through the recirculating system.

5.3.2.2 Cooling Pool Recirculation Filters

Two cooling pool filters were installed to improve the pool water clarity by removing particulate matter. The filters can use various pore-sized cartridges. Currently five-micron size cartridges are used. These filter assemblies can be used in parallel or as single units. Each has a flow capacity of 500 gpm at an operating pressure of 50 psig.

5.3.2.3 Cooling Pool Chillers

5.3.2.3.1 Water Chiller

A water chiller has been installed in the recirculation system to remove heat from the irradiated fuel cooling pool water. The cooling pool chiller is a water-cooled type capable of handling the total heat load. The chiller is capable of emitting water at 42°F.

Filtered water is circulated to the chiller condenser and then discharged to the river, see Figure 5.2. The fuel storage pool water is pumped from the pool through the primary side of the chiller evaporator and then discharged back to the storage basin. The chiller filtered water cooling line is equipped with instrumentation to indicate that cooling water is flowing.

5.3.2.3.2 Air-Cooled Chiller

An air-cooled chiller has been installed to decrease the service water demand which reduces the quantity of water discharged to the Columbia River. The unit consists of an evaporator and condenser/compressor and associated piping electrical, instrumentation and control. The functional design criteria for the chiller is as follows.

> The air-cooled chiller shall have the cooling capacity to replace the water-cooled chiller and discharge water from the evaporator at 42° F for the following conditions:
>
> > Evaporator inlet water of 50° F.
> > Total basin heat load of 119kW for KE and 138kW for KW as of 1/1/91.

Maximum available water flow rate through evaporator of 550 gpm at 25-40 psig.
Maximum pressure drop through evaporator of 25 psid.
Outside ambient air temperature of 115° F.

The air-cooled chiller shall have 10 ton excess capacity at an outside ambient temperature of 95° F, at the conditions stated above.

Installation shall not eliminate the ability to operate the current water-cooled chiller.

5.3.2.4 Ion Exchange System

The ion exchange system located in the transfer area, consists of three small carbon steel ion exchange columns operating in parallel. To provide adequate radiation shielding, the ion exchange columns are installed in a large concrete block with three wells. The wells are 3 feet square with 18 inches of shielding around each well. The 60 gallon capacity columns may be charged with up to 5 ft^3 mixed cation and anion organic resin ion exchange material. The ion exchange tanks are disposable and are removed and disposed of as a complete unit when resin is depleted.

As part of the decapping design changes (DC No.'s 82121 and 82223), a new ion exchange system was installed to supplement the existing system. The new system uses ion exchange columns similar to those already in use; however, in the new system the ion exchange columns are permanently sealed in a large concrete block that is approximately 7'x7'x6'. The assembly is referred to as an ion exchange module (IXM). Each basin is provided with two IXMs. The block serves as both shipping container and radiation shield. The new system is installed on the north rail spur coming into the basin, protected by a bumper capable of stopping a one car train with a speed of 4 mph.

5.3.2.5 Sand Filter System

Each sand filter system, located adjacent to the ion exchange system in the transfer area, consists of a skimmer system, pump, piping, and a pressurized sand filter with shielding. A settling pit is also provided for the backwash water. During filter operation, the skimmer pump delivers 400 gpm of storage basin water through the sand filter and back into the basin. During backwash periods, the same pump provides a back flow into the settling pit. The sediments accumulate in bottom of the settling pit from successive backwash cycles. The sand filter system is designed to operate independently of, or in conjunction with, the present cartridge filters described in Section 5.3.2.2.

5.3.2.6 Liquid Waste Transfer Pump

The liquid waste transfer system utilizes a 50 gpm electric pump. This system was used to empty liquid waste from the underground holding tank to

transportation vehicles for shipment to and disposal at the 200 Areas. This capability has been removed.

5.3.2.7 Rear Face Sump (B Sump)

A 60 gpm pump is located adjacent to the discharge chute and under each reactor's rear face elevator. This pump transfers collected coolant leakage from the discharge chute area back to the basin.

5.3.2.8 Cooling Pool Leakage Catch Tank Pumps (D Sump)

Two 80-gpm vertical, extended-shaft-centrifugal sump pumps have been installed to return the leakage water collected by the under basin asphalt membrane back to the cooling pool. The pumps are equipped with electrical controls to alternate the pumping cycle from one pump to the other and operate both pumps when the demand exceeds the capacity of one pump.

5.3.2.9 Irradiated Fuel Cask Transfer Area Sump Pumps (C Sump)

Two 40 gpm vertical centrifugal sump pumps were installed outside the cask transfer area to either return drainage water to the cooling pool, or pump it (when contaminated with detergents or other chemical agents) to a holding tank located outside the facility building.

The pumps are equipped with electrical controls to alternate the pumping cycle from one pump to the other and operate both pumps when the demand exceeds the capacity of one pump.

5.3.2.10 Valves and Piping

Manually operated gate, globe and ball valves are used to isolate equipment and control the water flow. All piping in contact with the cooling water is either Schedule 40 or Schedule 80.

5.3.3 Design Evaluation

5.3.3.1 Availability and Reliability

Each irradiated fuel pool cooling and cleanup system is a manually controlled system that can be shut down for periods of time for maintenance or replacement of malfunctioning components. Shutdown time limitations are based on a maximum differential temperature across the concrete basin walls and bottom of 54°F, and a process control document specified 42°F to 90°F temperature range. The system has some redundant equipment (e.g., two cartridge filters, two primary pumps, two separate ion exchange systems, etc.). Loss of electrical power could be a serious but not critical event.

During any power outage all normal fuel handling activity will cease, since normal instrumentation will be lost. Radiation levels, coolant temperature,

and coolant level can be determined with portable instruments or other means. The most important need during a power outage is to maintain adequate cooling and water coverage of the stored fuel.

A sixty hour quiescent cooling period (no forced circulation or addition of coolant) for a basin full of fuel has been established.[23] Sixty hours is sufficient time to bring in emergency engine-driven pumps. This cooling period, when established in 1977, was based on the 150-day-old fuel being processed at the time, and maximum water temperature of 130°F considered acceptable at that time. The fuel has now (1994) decayed for 7 to 23 years. Basin heat loads are such that the quiescent cooling period with the current basin water temperature limit of 100°F, is considerably greater than 60 hours. The cooling limit associated with basin operation is the maximum basin water temperature of 100°F, see Section 5.3.3.4.2.

A more urgent cooling problem would arise if the power outage occurred during the short time interval when the crane lifts the cask from the well car and lowers it into the basin. However, the probability of a power outage occurring during the period that the cask is in aerial transit is estimated at about 1.3×10^{-7}/yr, based on historical data. Alternate cooling is not necessary for the first 4 hours for a cask in such a position (based on 150 day fuel cooling) [29]. The few power outages occurring in recent years have been short, not approaching 4 hour duration, indeed most lasted only a few seconds. For these reasons an emergency power source is not considered cost effective.

5.3.3.2 Cooling Pool Dewatering

Hypothetically, the most serious failure of each system would be complete loss of water in the cooling pool. However, it has been concluded that loss of water from the pool sufficient to uncover the fuel is not credible. The basin structure was analyzed for its ability to withstand the DBE. Also, the adjoining cask transfer pits were analyzed for a postulated cask drop accident.[16] In both cases, the loss of coolant from the pool would be insignificant (less than 25 gpm).[30] A railroad cask car falling into the basin and causing an uncontrolled leak is discussed in Section 8.2.8, which concludes this also is not a credible accident. The cooling pool bottom drains have been sealed closed with concrete. Additionally, a pliable epoxy sealant has been applied to the walls and floor of the KW cooling pool. An emergency water makeup valve is immediately available to the operating crews and can supply 1500 gpm to the cooling pool, to recover from a non-design-basis abnormal reduction in water level.

In the unlikely event of a leak beyond the capacity of the emergency make-up system the basin would be expected to drain and personnel would be evacuated due to high radiation levels.

A peak temperature of 1697°F would be reached for encapsulated fuel which had been stored for the minimum 150 days at N Reactor prior to shipment to the

facilities.[31] At these temperatures, aluminum canisters used in hanging storage could be expected to fail and drop the suspended fuel. This would not cause a criticality.[3] The fuel currently stored in the facilities would not exceed a temperature of 781°F and would not melt aluminum canisters.

5.3.3.3 Water Quality

Prior to storing N Reactor fuel in the cooling pools, the pool floors were cleaned. It is expected that some debris will be introduced to the pool during the transfer and storage of the irradiated fuel elements. KW should be the cleaner of the two facilities since only encapsulated canisters will be stored in it. However by controlling the pH and temperature and passing the water through filters in the recirculating system, the ion exchange columns and the sand filter, the water in each facility will be maintained at an acceptable quality as defined in the facility process standards.

The process and construction materials of the fuel storage basins have been reviewed.[10,21,41,55] A coolant pH range from 5.0 to 10.0, is acceptable for a remaining design life of 20 years.[63]

5.3.3.4 Instrumentation

The instrumentation provided for each pool cooling and cleanup system is discussed below. Alarms and indicators are provided as noted. A system alarm for one facility annunciates and identifies the system and problem in that facility. It annunciates in the other facility only as a "system failure" without identifying the particular system or problem. However, a radiation alarm originating in one facility annunciates in the other facility as a "high radiation" alarm even though the indication in the originating facility alarms as low, intermediate or high radiation.

5.3.3.4.1 Temperature

Local instrumentation is provided to measure the temperature of the water in the pool and give indication and annunciation in the KE and KW Facility control rooms. Local instrumentation is also provided to give indication of the temperature of pool water as it leaves the chiller. Pool temperature shall not exceed 100F.[26,27]

5.3.3.4.2 Pressure

Local instrumentation is provided to measure and give indication of the pressures in the pool pumps suction and discharge lines. Instrumentation is also provided to measure pressure differential on the pool filters, the cask transfer area sump pump filters and the underbasin leakage collection sump pump filters.

5.3.3.4.3 Flow Rate

The cooling water discharges into each of the three separate bays or sections of the pool. The discharge lines are equipped with flow indication instruments and flow throttling valves to permit balancing of the cooling water flow rate through the three separate cooling pool areas. Instrumentation has been provided to indicate that the secondary cooling water is flowing.

5.3.3.4.4 Level

Instrumentation is provided to measure the pool coolant level and to alarm on a high or low water level condition. A high or low alarm is annunciated in both facility control rooms.

5.3.3.4.5 Ion Exchange Column Radiation Monitors

Portable instruments are used to monitor radiation levels in the ion exchange columns.

5.3.4 Test and Inspections

Each project (Projects H-501, H-508, and H-558) design included an acceptance test procedure that demonstrated the ability of the cooling pool systems and auxiliary equipment to operate as designed. These acceptance test procedures are: ATP No. 3158 for Project H-501,[32] ATP No. 3404 for Project H-508,[33] and ATP No's 4386 & 4387 for Project H-558 [34,35].

Active components of the pool cooling and cleanup systems are in continuous or intermittent use during normal system operation. Periodic visual inspection and preventive maintenance are conducted as outlined in the 100-K Irradiated Fuel Storage Operating Procedure System.

5.4 Fuel Handling System

The fuel handling system consists of equipment and structures utilized at both the KW and KE facilities for handling irradiated fuel elements in a safe manner during storage operations and transfer into and out of the basins. The design requirements for the KW and KE facilities[22] are very similar.

5.4.1 System Description

The fuel handling system consists of equipment required for the transfer and storage of N Reactor irradiated fuel elements in the facility cooling pool. The structures associated with the fuel handling equipment are the cask loadout areas and the canister trolley & track systems at both the facilities.

Fuel handling equipment was designed to handle the spent fuel under water from the time it is loaded into shielded casks until it is stored in canister racks or on hangers in the facility cooling pools. Underwater transfer and movement of the fuel provides an effective, economical and transparent radiation shield as well as a reliable cooling medium for removal of decay heat.

The associated fuel handling structures may be generally divided into two areas: (1) the facility cooling pools, which are full of water and are always accessible to operating personnel, and (2) each facility cask loadout area which houses a 30-ton overhead crane for transferring the shielding casks from the railroad cars to the cooling pools and vice versa.

In the facility pools, canisters of irradiated fuel elements are moved about by 1-ton capacity electric hoists hanging from trolleys on monorails. When lifting canistered fuel, the monorail hoists are limited by an electrical uptravel switch and backed up by a mechanical system which will prevent lifting the fuel elements any closer than 8 feet from the cooling pool surface. This ensures that sufficient radiation shielding is always maintained. Long handled tongs with adjustable jaws are available to handle individual fuel elements when the occasional need arises. Operating and radiation practice procedures govern the use of tongs for handling irradiated material.

Fuel is transported by loading into a shielded cask and transported by railroad cars. Each railroad car has three wells (compartments). Each well can contain only one cask.

All N Reactor and Single Pass Reactor fuel elements have now been transferred to the 100K Fuel Storage Basins with the following exceptions. N reactor and Single Pass Reactor fuel currently stored at PUREX may be shipped to the K Basins sometime in the future. In addition any irradiated fuel located during the restoration activities at N Reactor will be transferred to the K storage basins.

When a railroad car arrives at one of the facilities, the casks are unloaded one at a time into the fuel transfer pit by using the 30-ton overhead crane. The canisters are then removed from the shielded casks, one at a time, using a 1-ton electric hoists and placed in storage racks or on hangers in the facility cooling pools. To transfer the canisters out of the facilities, the canisters are brought in one at a time and loaded into the shielded casks which are loaded, when ready, into the railroad cars for shipping.

5.4.2 Component Description

Following is a brief description of the major components that make up each fuel handling system.

5.4.2.1 Storage Canister (Figure 2.1)

The irradiated fuel storage canisters are 28 inches high and consist of two upright cylinders approximately 8.2 inches in diameter. Each cylinder will hold seven N Reactor fuel elements. The canisters were originally constructed of aluminum; because of long term corrosion problems new canisters are constructed of stainless steel.

The original N Reactor fuel storage canisters, designated Mark 0 (shown in parts A and B of Figure 2.1), were modified by welding a seal plate on the bottom and a sealing cap on the top, as shown in part C of Figure 2.1. A vent was provided on the cap to permit escape of gases generated by corrosion, without water interchange between the canister and the basin. Fittings were also provided for the introduction of solutions or gases into the canister or for sampling the solution in the canister.[36,37] These modified Mark 0 canisters were designated Mark I canisters. The Mark II encapsulated canister (part D of Figure 2.1) is currently being used. It is similar to the Mark I encapsulated canister except that it uses new stainless steel body cylinders and an improved cap, closure and gasket assembly.

The tops of the Mark 0 and Mark I lower enriched fuel storage canister are flared into a circle, as shown in part A of Figure 2.1, while the tops of the spike fuel canisters are flared into an octagon, as shown in part B of Figure 2.1. Mark II canisters have an identification tab welded on to indicate the presence of spike fuel. These visual differences provide an additional check on the storage locations of spike and lower enriched fuel.

5.4.2.2 Fuel Shipping System (Figure 5.3)

Several railroad well cars have been provided to ship the casks used for transferring N spent fuel to and from the facility cooling pools. Each car is compartmented to handle three casks. Each compartment is an independent well with its own water pipes so that fuel coolant can be removed without disturbing the adjoining wells. The existing railroad track system is used to move the railroad cars to and from the facilities, the 105-N fuel handling area, or PUREX. The safety analyses for transport is covered in References 38 and 39.

5.4.2.3 Cask Crane - KE and KW Facilities

The design capacity of each facility cask load-out crane is 30 tons. Weight of a loaded cask is approximately 24 tons. The crane is a traveling bridge type with motorized trolley bridge and hoist.

5.4.2.4 Storage Canister Hoists

One-ton electrically operated hoists are used at each facility to raise and lower loaded fuel canisters and baskets from and into shipping casks and cooling pool storage stations.

5.4.2.5 Storage Canister Hooks

Canister hooks are provided for moving canisters in and out of storage in each facility cooling pool. Two types of hooks are used. One, a straight bar with a hook on the end permits storage in the rows directly under the slots in the floor grating. The other type has an offset mechanism that permits storage in the alternate rows which are underneath the grating (Figure 5.15).

The trolley and storage hook for suspended fuel storage have a spacing bar which prevents the canisters from hitting each other during seismic events. This spacing bar is locked in place and unlocked using a disengaging tool.

5.4.2.6 Canister Elevator

A small hydraulic elevator (21 inch maximum lift) operating on demineralized water is used when unloading canisters designated for suspension storage. The elevator receives the canisters (one at a time) from the storage canister hoist and then raises and lowers the canister for engaging the canister trunion on the storage hook.

5.4.3 Fuel Handling Sequence

The general fuel handling sequence starts with spotting the railroad car at the unloading station and ends with each canister of fuel placed in a specific location in a facility cooling pool. The sequence of operation is:

5.4.3.1 *Unloading the Railroad Shipping Car*

- At the facility cask load-out areas, the well lids are manually cranked opened and the 30-ton overhead crane hooks engage the trunions of the first cask.
- The hooks safety latches are properly engaged for safe lifting.
- The first cask is then lifted from the railroad car and lowered into the south cask transfer pit.
- Lowering of the cask in the pit is stopped where the cask lid mechanism can be seen.
- At this point the lid is unlocked using an air-operated impact wrench or manual wrench.

- The cask is then slowly lowered into the water filled transfer pit where the lid overhang comes in contact with two wall mounted steel structures that hold the lid in suspension while the rest of the cask continues down toward the pit floor.

- The cask is stopped prior to contacting the pit floor but low enough to pass under the supporting structural steel in the pit.

- The cask is moved in a easterly direction until the cask cavity is in full view for removing the canisters of fuel.

- Using a one-ton electrically operated hoist, the canisters are removed from the cask one at a time and moved via the monorail system into the irradiated fuel cooling pool area and stored individually in predesignated racks. When unloading casks containing canisters designated for suspended storage, the hoist is used to set canisters (one at a time) on the canister elevator. The canister elevator is activated to raise the canister to the up position and a support hook is moved over to engage the canister trunion. The elevator is then lowered back to its original position and the trolley with the suspended canister is moved to its storage position. The entire operation is accomplished under at least 8 feet of water.

- After removing all the fuel filled canisters from the cask, it is moved in a westerly direction with the overhead crane until it is directly under its suspended lid.

- The cask is slowly raised and as it ascends from the pit it will pick up the suspended lid.

- The cask is stopped at or near the water surface, where it is confirmed that the lid is in the proper position to be locked and radiation levels are measured if the cask is loaded for shipping.

- The cask is raised from the pit and moved west over a raised platform with a drip pan which will return any excess dripping water back into the loadout pit. A spray of water may be used to wash radioactive particulate matter back into the transfer pit.

- The cask is then transferred via the overhead crane system back into the railroad car.

- The above steps are repeated until all three casks are unloaded and returned to the railroad car.

- The railroad car is then surveyed by Health Physics Technicians to insure the radiation level and smearable contamination are within limits, prior to releasing the car.

5.4.3.2 Loading the Railroad Shipping Car

The loading out procedure is essentially the reverse of the unloading described above in Section 5.4.3.1.

5.4.4 Safety Evaluation

Each facility transfer area 30-ton crane design includes several features to assure safe unloading of casks containing N Reactor irradiated fuel. Two independent brakes are supplied on the hoist. The mechanical load brake is a double-disc screw type and is applied to and automatically sustains the load and assists the motor to control the operation of lowering. The electric brake provides positive stoppage and control. It is held open by hoisting current and is instantly applied when current is interrupted. A safety switch is provided to automatically prevent overtravel. After it is thrown, stopping the hoist, the load may be lowered without resetting the switches. Track type limit switches are provided to prevent bridge and trolley overtravel. A screw type limit switch is provided to limit motion at top and bottom of lift. In addition, there is an extra limit switch in the system to prevent overtravel at the top of the lift.

The facility cooling pool hoists and related equipment design includes the following features to assure safe handling of canistered fuel elements in the irradiated fuel cooling pool and transfer areas.

Each electrical hoist and fuel hanger is equipped with a long steel canister hook that is designed to handle only one canister at a time. The upper geared limit switch on the hoist is backed up with mechanical protection that prevents the inadvertent lifting of fuel above the 8 feet of water shielding. Each hoist and fuel hanger is securely pinned to a four wheel trolley that rides a monorail system. Mechanical stops are provided at ends of the monorail and at transfer points to prevent losing the hoist or fuel hangers from the rails.

5.4.5 Tests and Inspections

Each project (Projects H-501, H-508, and H-558) design included an acceptance test procedure that demonstrated the ability of the project fuel handling system to function as designed. As part of the normal facility operation, the irradiated fuel handling equipment is inspected and maintained according to practices outlined in the 100-K Irradiated Fuel Storage Operating Procedures.

Cask cranes are given third party inspection annually. Load tests are performed at facility startup as specified in the crane and hoist inspection program. Periodic inspections and lubrication on all cranes and hoists are performed under the preventive maintenance program.

5.5 Fuel Segregation

The fuel segregation activity consisted of separating the Mark IV fuel stored in the KE Facility into weapons stock, low blend stock, high blend stock, and remains stock. Subsequently, the equipment was removed from the basin. The system description and its operation has been removed from the SAR. Segregated fuel and scrap fuel associated with the segregation process are stored in the basins.

In the segregation program, fuel was removed from the canisters and the inner fuel element removed. Following disassembly a detector and computer system were used to determine the Pu-240 content of each element. The elements were separated according to their Pu-240 content (weapons stock, low blend stock, high blend stock, and remaining stock elements, 12wt % Pu-240). The elements were reassembled and placed in canisters and returned to storage.

Fuel element debris accumulated at the canister dumper and disassemble detector. The debris consists of segments of broken fuel elements, support feet, and "W" springs. Such debris was placed in canisters (broken fuel separate from other debris), which when full were stored in the main basin floor storage racks.

5.5.1 Safety Analysis

The safe hemispherical masses established in Reference 6 for unirradiated Mark IV fuel (unirradiated fuel is more reactive than irradiated fuel) are:

1. 7722 pounds of uranium for assembled fuel assemblies, which is equal to 149 of the heaviest assemblies, or

2. 6963 pounds of uranium for outer elements, which is the equivalent of 198 of the heaviest outer elements, or

3. 3927 pounds of uranium for inner elements, which is the equivalent of 238 of the heaviest inner elements, or

4. 3451 pounds of uranium for broken fuel segments, which is the equivalent of 66 of the heaviest fuel assemblies.

The equipment was arranged such that the major components were separated by more than the 10 inch neutronic separation under water required for safe masses as established in Reference 6.

The safe mass assures that double batching of two safe masses under accident conditions will not exceed k-effective of 0.98.

The segregation activity did not significantly affect the amount of radioactive material in solution, consequently there was no increased load on the ion exchanger (demineralized resins). The canisters have holes in the

bottom, and since the particulate matter settles to the bottom it partially flushed out as the canisters are lifted and moved. The rest of the particulate matter was dumped along with the fuel on the canister dumper. Some additional particulate matter was released when the fuel was disassembled. Most of it quickly settled to the floor. A small amount of suspended particulate ended up in the filters, which, at the most, increase the frequency of filter cartridge change-out or backwash of the sand filter. Filter cartridge change-out and sand filter backwash was performed when the differential pressure across the filters indicated that the maximum filter loading had been reached, or when the operational frequency limit is reached, as described in facility operating procedures. The impact of sludge and dissolved corrosion products on criticality were considered in the criticality evaluation described in Reference 8. It concluded criticality is not possible in such low enriched (0.95wt % U-235) uranium solutions and sludges. (See Section 5.7.4.2 for a discussion of the effects of decapping 1.25wt % U-235 fuel.)

A canister full of broken Mark IV fuel scrap is more reactive than a canister full of unbroken Mark IV fuel. However, it is less reactive than a canister full of unbroken Mark IA fuel. Therefore, dropping a single canister of broken fuel segments will be less of a reactivity perturbation than dropping multiple canisters of Mark IA in a one over three configuration, which was analyzed in Reference 3. Since such a canister drop would occur under water, there would be no releases to the environment.

Storage of canisters with broken fuel segments in a one over three configuration has not been analyzed for a multiple canister drop. Because of the limited amount of broken fuel segments expected, such an analysis is not planned. Therefore, canisters of broken fuel segments are limited to floor storage only.

Low exposure fuel is more reactive than higher exposure fuel or the average unsegregated fuel. However, low exposure Mark IV fuel is less reactive than Mark IA fuel[3]. In addition, all criticality calculations are based on unirradiated fuel. Therefore storage of low exposure fuel is safe even in a one over three configuration.

5.6 Railroad Track and Equipment

The railroad track and equipment are under the control of ICF Kaiser Hanford train crewmen. The derailers, heavier railstops, and bumpers were installed under Project H-501, H-508, and H-558. The rail bumper on the northern spur was removed and replaced under DC-82121 in KW.

Procedures are in place requiring a Facility Manager or other qualified person (as defined in Facility Operational Safety Requirements) to control entry into either facility.

5.6.1 System Description

A railroad switch, which is kept in the position for entry into KE, controls entry to the facility. The switch to the auxiliary railspur, located in front of the derailer on the KE line will normally be kept open, diverting any runaway trains away from the KE Facility. Derailers are provided outside of the facilities to derail trains unless a Facility Representative unlocks the derailer and the derailer is deactivated. Within the derailers is a second switch which allows the rail car to enter the facility on one of two tracks. This switch is kept in position so that the south track will normally be used. Within the facility at the edge of the basin, rail stops and/or bumpers are provided on both tracks to prevent a rail car from entering the basin should the engineer fail to stop the train.

5.6.2 Components

5.6.2.1 Derailers

The derailers are installed 520 feet from the rail terminations at the side of the basin. The derailers are designed to derail a one-car train traveling at a speed of up to 5 mph. A derailed train would enter a loose gravel area which would then completely stop it. The derailers are locked in the active position and are unlocked by a Facility Representative and opened only after the train has stopped adjacent to the derailers.

5.6.2.2 Rail stops

Rail stops are provided on the rail spurs entering each facility. Stops are designed to lift the train thereby absorbing kinetic energy. The maximum amount of energy that can be absorbed is estimated to be 31,500 ft-lbs.

5.6.2.3 Bumpers

In addition to the rail stops a spring loaded bumper is mounted on a beam connected to two building columns. The combination of bumper and structural member deformation will absorb about 59,000 ft-lbs.

5.6.2.4 Bumping Posts

Bumping posts permanently embedded into the rails are installed on the northern rail spurs entering each facility, to protect the auxiliary ion exchange modules. The posts are designed to absorb a maximum energy of 252,000 ft-lbs.

5.6.2.5 Rail Spur

A switch to the auxiliary rail spur may be locked open, to divert an uncontrolled train away from the basin except for entering or exiting trains or track maintenance activities.

5.6.3 Sequence of Operation

- The railroad track switch west of KW area which routes the train to either the KW or KE Storage Basin shall always be in the KE position except when the train is actually entering, within, or exiting the KW Storage Basin area.

- An additional rail spur is located at the entrance to KE. The switch for this rail spur shall be locked open, to divert any runaway train from the KE Basin.

- The train derailer devices shall not be opened until the train has come to a full stop and the locomotive Pace Setter controls have been verified to be in the required positions, see Section 5.6.4 Safety Evaluation.

- When entering the storage basin transfer area, the fuel transfer train shall be controlled at a speed not greater than 2 mph. A Facility Manager, or other qualified person, shall ride the locomotive cab whenever a cask car is being moved into the storage basin transfer area to ensure by a trip of the emergency stop that the train speed does not exceed 2 mph.

  The Facility Manager, or other qualified person, is essentially a safety backup for the engineer. Best judgement will be exercised in determining when speed is excessive and when the engineer is not responsive for timely application of the emergency stop switch.

5.6.4 Safety Evaluation

A railroad cask car entering the basin could damage the building structure, the fuel handling equipment, and the basin. Possible consequences are loss of coolant or a criticality caused by geometrical rearrangement of stored fuel. Procedures and mechanical controls are in place to the extent that the probability of such an event is negligible.

The GM locomotive used to deliver the cask cars to the facility has two modes of operation, manual and automatic. A master switch must be in the "Manual" or "Pace Setter" position. The Pace Setter control unit is an electronic device capable of very exact speed controls. The Pace Setter is equipped with a three position speed range selector (0.1 to 1 mph, 1 to 10 mph, and 10 to 100 mph). A speed dial can be set to any speed in any of the three ranges. The Pace Setter also has a "Manual-Auto" switch which must be in the Manual position in order for the train to start moving, it must then be put in the "Auto" position for automatic speed control. The Facility Manager or other qualified person boards the locomotive at the derailer. This individual

ensures that the Engineer has all switches and dials in the proper position and that the Engineer flips to "Auto" appropriately. An emergency stop switch is available if necessary. Excessive speed is determined by a digital speedometer reading in one mph increments.

The railstops and bumpers combined on the southern rail spur are designed to stop a one car train at a speed of 4 mph. The rail stops will absorb half the kinetic energy (31,500 ft-lbs) of a one car train moving at 2 mph[40]. The shock absorbers are designed to absorb about 59,000 ft-lbs. Total kinetic energy of a one car train moving at 2 mph is 63,100-ft-lbs[40].

The bumping posts on the northern rail spur are designed to stop a one-car train moving at 4 mph. Total kinetic energy of a one-car train at 4 mph is 252,000 ft-lbs.

The derailers are designed for 5 mph train speed. Procedures require the train to stop in front of the derailer. The Facility Manager or other qualified person will then unlock the derailer, the derailer is deactivated, and the individual will board the locomotive for the slow speed entry into the facility.

To prevent an uncontrolled train from entering the KW Facility the switch diverting to KW is locked in the KE position. At KE the switch will divert the train onto the auxiliary railspur away from the basin. Procedures direct the relocking of the switch when the locomotive leaves the KE Facility.

A two-car train is prohibited from entering either facility beyond the derailers.

5.6.5 Testing & Inspection

Testing and inspection of rail equipment and track are the responsibilities of Transportation/Waste Handling of ICF Kaiser Hanford Company.

5.7 Decapping/Recanning

The KW decapping station is in the transfer canal between the south loadout pit and the western bay of the basin.

The equipment is designed to:

- provide a contamination control system for decapping operations in the transfer channel;
- decap Mark II canisters in the KW transfer canal;
- provide a contamination control system for decapping operations in the transfer canal.

5.7.1 System Description

A decapping station[42] is provided in the KW transfer canal, to be used for the decapping of MK II canisters. A purging tool is used over the decapped canisters to flush out contamination. An under-water manifold and associated pumps and piping is used as a suction device to remove the contaminated water from the canal and process it through the auxiliary ion exchange modules. These modules are installed on the center rail tracks in the transfer bay. A barrier door is placed at the transfer canal entrance to provide additional contamination control to the main basin.

5.7.2 Components

Following is a brief description of the major components of the decapping system.

5.7.2.1 KW Decapping Station

The decapping station in KW consists of a receptacle capable of holding only one canister with a single storage rack capable of holding one additional canister attached (additional storage racks capable of holding two canisters are located at the entrance to the transfer canal, see section 5.7.2.4). The suction manifold for the ion exchange system is attached to the top of the receptacle where it can remove the contaminants flushed out by the purging tool.

5.7.2.2 KW Auxiliary Ion Exchange System

The auxiliary ion exchange system (IXM) consists of the suction manifold in the receptacle, a 150 gallon per minute shielded pump (for decapping only), two ion exchange modules which contain six tanks each and a return line to the west bay of the basin. A composite sampling system is used to determine radionuclide concentrations in the modules.

5.7.2.3 KE Auxiliary Ion Exchange system

The ion exchange system in KE is similar to KW, with the ion exchange return line outlet located close to the north loadout pit.

5.7.2.4 KW Purging Tool

Removal of soluble contaminants from the decapped canisters is accomplished by lowering a showerhead type purging tool over the open canister and spraying basin water into it. This will cause the soluble contaminants to be flushed from the canister and into the manifold where they will be transported to the IXMs. A separate pump and associated piping is used to provide the necessary water pressure for this system.

5.7.2.5 Peripheral Equipment

At KW there are two single canister storage racks at the entrance to the transfer canal from the south loadout pit.

A bumper capable of stopping the impact of a one-car train, with a speed of up to four mph, protects the ion exchange modules at each basin.

Other peripheral equipment includes miscellaneous handling tools as described elsewhere (e.g. Sec. 5.5.2.4), for each facility.

5.7.3 Sequence of Operation

The following sequences will be followed to process canisters.

5.7.3.1 KW Basin Decapping Sequence of Operation

At the beginning of each operation, flow will be established through the contamination control manifold, the ion exchange modules, and the purging tool. The water sampler will be set up. Necessary flow and pressure checks, tool inspections, and radiation monitoring will be completed.

The desired canister will be located in the storage basin, transferred through the barrier wall at the entrance to the transfer channel, and placed in the decapping station in the transfer channel. The canister will be decapped and the lids placed in a temporary storage container for later disposal. The purging tool will be positioned onto the open canister to flush contaminants from the canister via the contamination control manifold, into the IXM. When the flush cycle has completed, the purging tool will be removed from the canister. The canister will be moved from the decapping station and placed into the temporary storage rack or into the cask, if available.

At the end of each operation, the purging tool will be shut off. The contamination control system may be left on to remove soluble contamination from the transfer channel area, or valved to remove soluble contamination from the storage basin.

5.7.4 Safety Analysis

Introduction of Mark IA fuel into KE has been evaluated for nuclear criticality safety limits in those areas in which this higher enriched fuel will be processed, stored, or transported. Additionally, an analysis of the impact of sludge and dissolved corrosion products on criticality was performed to take into account the higher enriched sludge which could result from decapping canisters containing Mark IA fuel.[7] Finally, a safety analysis of the new auxiliary ion exchange modules, located on the tracks of the northern rail spur, was completed.

5.7.4.1 Effect of Decapping Station Activity on Basin Operations

5.7.4.1.1 KW Basin

The decapping station must meet the hemispherical safe mass storage requirements of Table 5.3. All canisters containing fuel must be encapsulated before being returned to the main basin.

Canisters containing Mark IA scrap may be significantly more reactive than canisters filled with unbroken fuel. They require separate storage areas and may not utilize hanging storage. Operational Safety Requirements and Process Standards specify how these requirements are met.

5.7.4.1.2 KE Basin

If decapping of KW canisters is initiated in KE the criticality limits will require changes because of the introduction of the higher enriched fuel assemblies (Mark 1A) into the basin. The limits currently established in the segregation area are based on Mark IV fuel only. Therefore, introduction of Mark IA fuel into KE Basin was evaluated to take the higher enriched Mark IA fuel into account. Table 5.3 shows how the hemispherical safe mass limits have changed. As can be seen, there is a significant decrease in the safe mass of assemblies, outers, and scrap. Specific requirements must be given governing the storage of canisters containing Mark IA scrap. These canisters will be the most reactive in the basin. They will require separate storage areas and may not utilize hanging storage. Operational Safety Requirements and Process Standards will specify how these requirements will be met.

5.7.4.2 Analysis of Effect of Higher Enriched Dissolved Fuel

If the uranium to plutonium ratio, as found in the spent fuel, is maintained, any basin operation, including corrosion of fuel elements and sediment buildup, can be conservatively covered by using unirradiated uranium critical mass limits.

However, it may be possible for certain chemical reactions to occur whereby the uranium to plutonium ratio in solutions and sludges is perturbed. Therefore, conservative assumptions must be made to protect against possible buildup of higher concentration of plutonium in vessels, pumps, and piping. Since no pre-knowledge of how much the uranium to plutonium ratio may be perturbed is available, the most conservative plutonium limits must be followed.

Table 5.4 shows the nuclear criticality safety limits[65] established using the above criteria. Operational requirements for periodic monitoring of gamma radiation from all pumps, and piping is performed to provide trend information to indicate any radioactive material buildup.

Sludge layer limits apply to KW Basin only, even though use of encapsulated canisters in KW has significantly reduced the sludge buildup problem. These limits apply to the storage of 1.25 percent enriched fuel. At the present time, storage of 1.25 percent enriched fuel is limited to the 105-KW Basin.

These limits are very conservative and will ensure safe operation of the equipment and areas analyzed.

5.7.4.3 Auxiliary Ion Exchange Modules and Columns

Table 5.5 is an estimate[43] of the maximum potential radionuclide buildup in the ion exchange modules and columns used in the K-Basins. These estimates are based on weekly and monthly analyses of the radionuclide content of the inlet and outlet flow through the modules/columns during their operation and calculation of the retained radionuclides.

These modules are located on the northern rail spurs coming into both of the basins. Thus, there is a potential for a one car train to impact on these modules, with a resulting release of radioactive contamination. Even if only 1% of the burden were released, this would be a significant release, with all the associated cleanup problems.

To eliminate the possibility of this type of accident, a rail bumper, capable of withstanding an impact of a one car train (235 tons) traveling 4 mph, has been installed. Since the train size is limited to the engine and one car and the train speed is limited to 2 mph (Sec. 5.6), the credible possibility of this accident is essentially eliminated.

TABLE 5.1 DESIGN AND OPERATING PARAMETERS FOR EACH COOLING POOL SYSTEM

Recirculation Pumps		
Number of Units	2	
Design Pressure, psig	75	
Design Temperature, °F	100	
Design Flow, gpm	500	
Material	Cast Grey Iron	
Skimmer Pump		
Number of Units	1	
Design Pressure, psig	60	
Design Temperature, °F	100	
Design Flow, gpm	400	
Material	Cast Grey Iron	
Heat Exchangers (Not in service)		
Number of Units	1	
Design Heat Transfer, BTU/hr (Kw)	7.85×10^6 (2300)	
	Shell	Tube
Design Pressure, psig	100	100
Design Temperature, °F	200	250
Design Flow, gpm	1000	1500
Inlet Temperature, °F	100	70
Outlet Temperature, °F	86	79

TABLE 5.1 cont.

	105 KW	105 KE
Fluid Circulated	Water	Water
Material	Steel ASTM A-285 Gr "C"	ASTM Bill cupro-nickel (90-10 annealed)
Water Chiller		
Number of Units		1
Design Heat Transfer, BTU/hr (Kw)		2.4×10^6 (700)
Design Flow, gpm		500
Outlet Temperature °F		55
Air Cooled Chiller		
Number of Units		1
Design Heat Transfer, BTU/hr (KW)		7.2×10^5 (2.1×10^2)
Design Flow, gpm		250
Outlet Temperature oF		42
Recirculation Cartridge Filters		
Number of Units		2
Design Pressure, psig		150
Design Flow, liters/sec (gpm)		1000
Filtration Requirement		5 micron
Material (Structural)		Carbon Steel
Ion Exchange Columns		
Number of Units		3
Design Pressure, psig		75
Design Flow, gpm		50
Resin Capacity, ft^3		5
Material (Structural)		Carbon Steel

TABLE 5.1 cont.

Auxiliary Ion Exchange Modules	
Number of Units	2
Design Pressure, psig	60
Design Flow, gpm	160
Resin Capacity, ft^3	21
Sand Filter	
Number of Units	1
Design Pressure, psig	90
Design Flow, gpm	400
Material (Structural)	Carbon Steel
Skimmers	
Number of Units	3
flow Rate, gpm	120

TABLE 5.2 COOLING POOL SYSTEM DESIGN DATA

Maximum Racked Irradiated Fuel Storage Capacity,	metric tons	-	1648
Maximum Hanging Irradiated Fuel Storage Capacity,	metric tons	-	375
Volume (Water),	ft^3	-	1.37×10^5
	gal		1.02×10^6
Normal Water Level,	feet		16.0

TABLE 5.3 IMPACT OF THE INTRODUCTION OF MKIA ON HEMISPHERE MASS LIMITS

	Hemisphere Safe Mass (lb U)*	
	MKIV	MKIA
Assemblies	7722	2052
Outers	6963	1524
Inners	3927	3927
Scrap (chunks)	3451	744

*The safe masses listed assures that if two such safe masses are brought together under accident conditions, the combined masses will not exceed a K-eff of 0.98. The safe masses listed in the table range from 33% to 45% of a hemispherical critical mass. The range depends on the fuel geometry and fuel enrichment.

TABLE 5.4 NUCLEAR CRITICALITY SAFETY LIMITS, FOR EQUIPMENT AND AREA, BASED ON DISSOLVED FUEL CONSIDERATIONS

Nuclear Criticality Safety Limits

Equipment	Mass	Dimension	Sludge Layer (KW only)
Pumps	---	ID 15.9 inches	---
Piping	---	ID 15.9 inches	---
Canister Receptacle (KW)	---	---	Thickness 5.8 in
Filtration Component Including IXMs	225 g Pu/ Component	---	---
Area			
Main Basin (KW)	---	---	Thickness 5.8 in
Decapping Station (KW)	---	---	Thickness 5.8 in

TABLE 5.5 RADIONUCLIDES IN SPENT ION EXCHANGE MODULES AND COLUMNS

Isotope	Mixed bed module	Mixed bed column
Cs-137	125 Ci	45 Ci
Sr-90/Y-90	65 Ci	25 Ci
Pu-238	0.15 Ci	0.03 Ci
Pu-239	0.5 Ci	0.1 Ci
Am-241	0.5 Ci	0.1 Ci

Figure 5.1 Fuel Pool Hangers

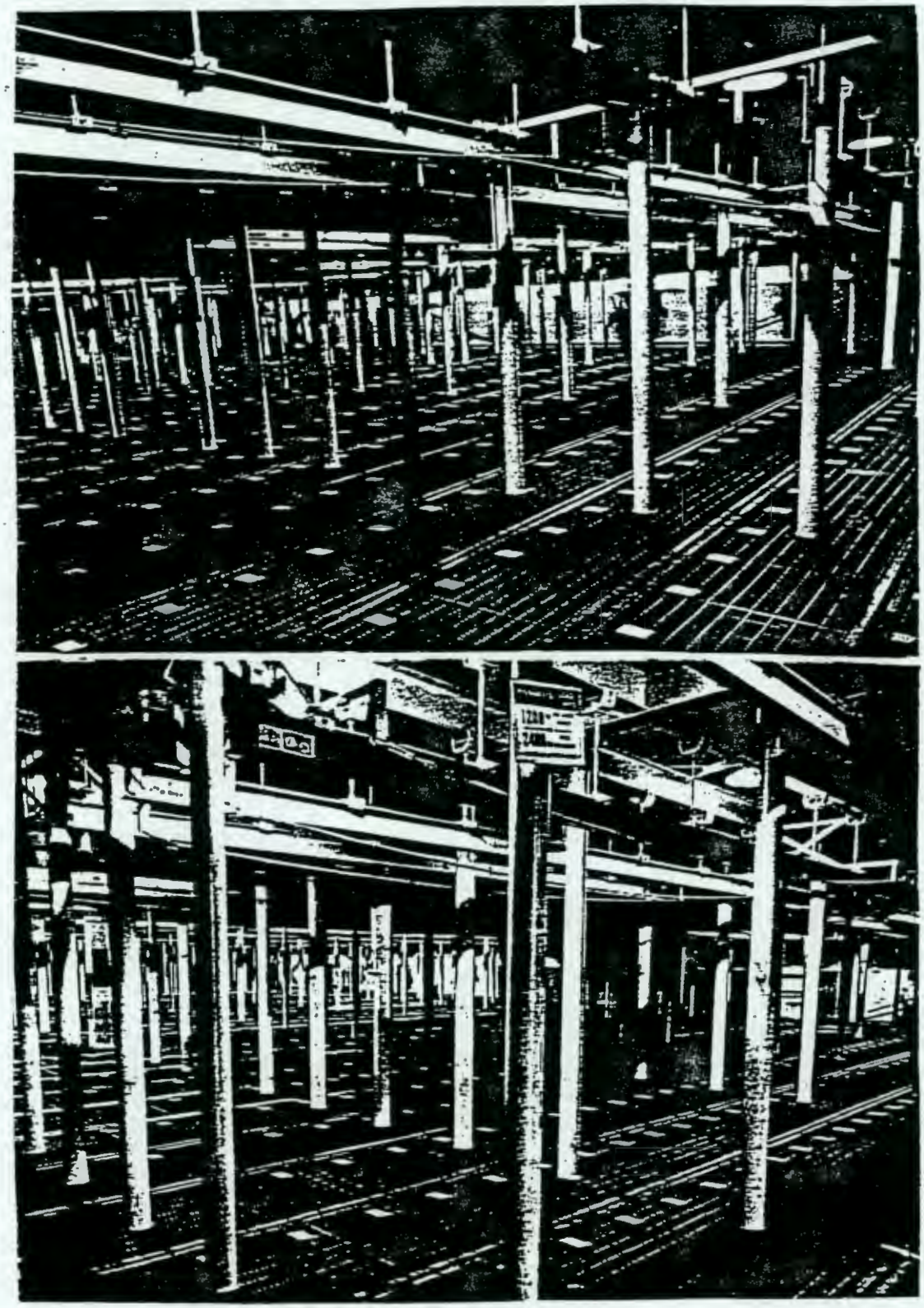


Figure 5.2 Water Systems Flow Diagram - General

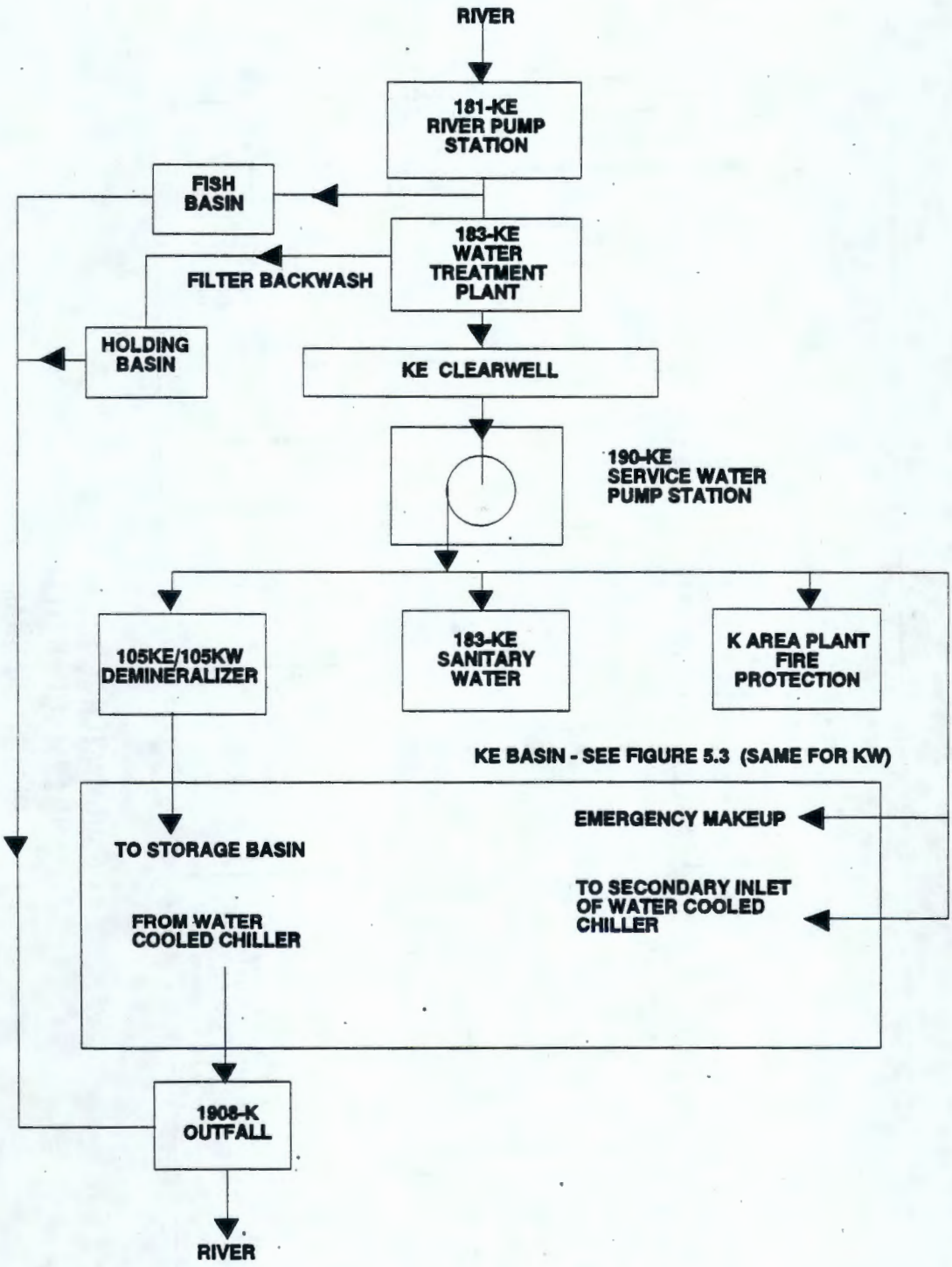
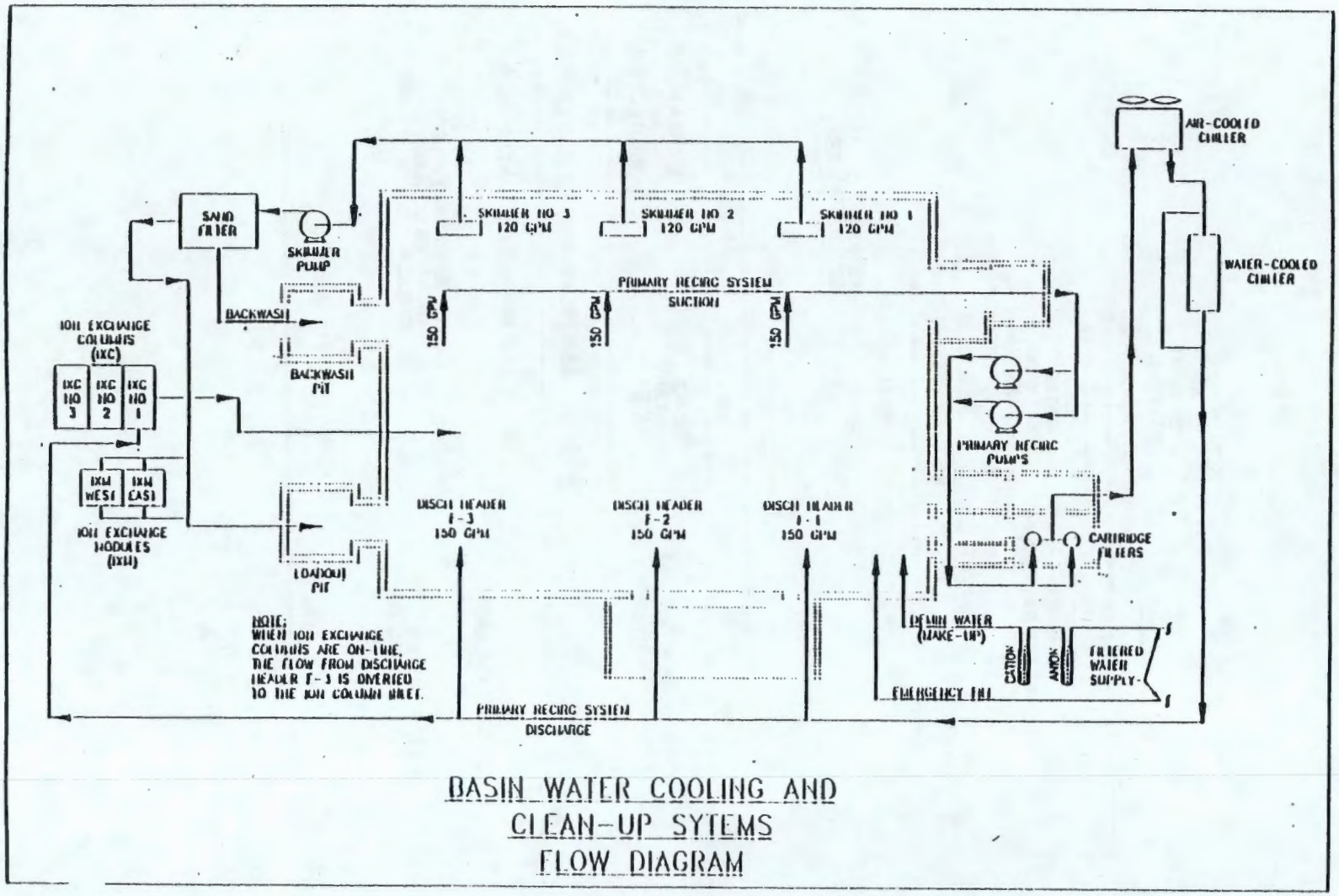
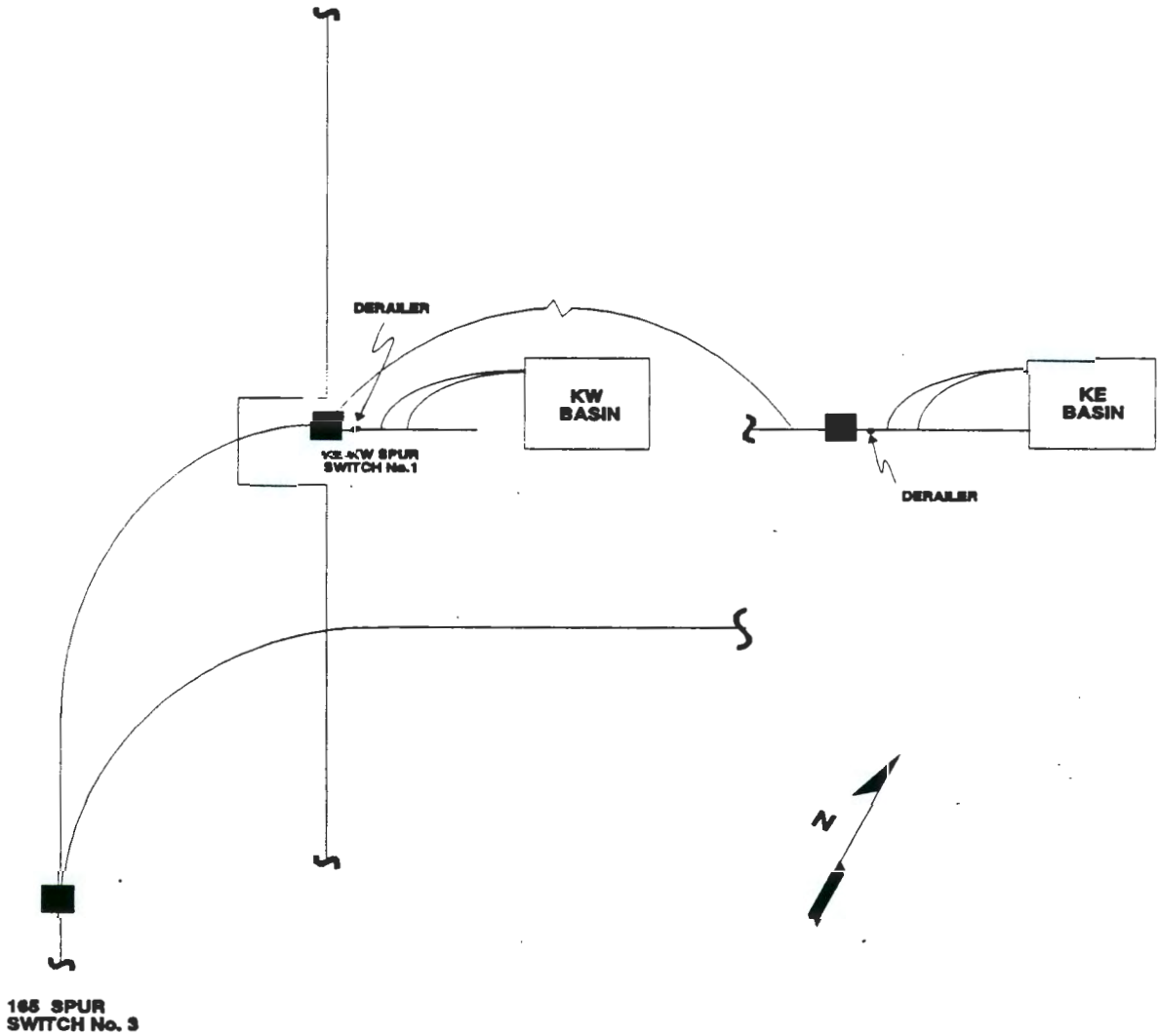


Figure 5.3



BASIN WATER COOLING AND CLEAN-UP SYTEMS FLOW DIAGRAM

Figure 5.4 Railroad Track Layout to K Area Storage Basins



6.0 CONDUCT OF OPERATIONS

6.1 Organizational Structure and Responsibilities

The K Basins organization of the Spent Fuel Disposition Project is responsible for the operation of the 100K Fuel Storage Facilities. See the organizational charts on Figures 6.1 and 6.2.

6.1.1 K Facility Operations

The Director, K-Basins, is the onsite director overseeing each subordinate manager at K Basins. The managers are responsible for operations, engineering and maintainance of the facilities in accordance with the appropriate approved procedures. References 44, 45, 58 and 72.

The Health Physics Technicians are supplied by Health Physics organization of Emergency, Safety, Quality Services (ESQ), and the craft services (instrument, electrical, millwright, etc.) required to maintain the facilities are supplied by K Basins Maintenance organization.

The Manager, K-Basins Operations, is directly responsible for the training, safety, and conduct of all personnel involved in handling and storage of the irradiated N Reactor fuel elements at the facilities. The manager is also responsible for the accountability of the fuel after it arrives at the facilities as well as operating and emergency procedures for the facilities. The manager has the responsibility to ensure that all personnel assigned duties involving the handling of irradiated fuel are qualified to perform that work.

The Manager, Engineering Support Services, is responsible for providing written maintenance procedures for the facilities.

The Shift Managers have the responsibility for taking immediate action on all facility alarms received. They will dispatch qualified personnel to take corrective action as outlined in the Operating and Emergency Procedures. Also, they will notify facility personnel of any condition of a serious nature.

6.1.2 Technical Support to the 105K Facilities

6.1.2.1 Nuclear Criticality Operating Limits

The facility is operated under the limits provided by Process Standards[45] that are issued by the Technical Safety Section in conformance with the Facility Operational Safety Requirements[54] and WHC-CM-4-29.[46] The Process Standards addressing criticality are the Criticality Prevention Specifications for the Facility.

6.1.2.2 Environmental Control Limits

The requirements for environmental monitoring to ensure permit compliance and assessment of operational releases are established by WHC-EP-0497, *Facility Effluent Monitoring Plan for K Area Fuel Storage Basins*[56] in conformance with DOE Order 5400.1.[57] All effluent releases, actual or potential, will be within the bounds of WHC-CM-7-5, *Environmental Compliance Manual*.

6.1.2.3 Radiological Control Limits

The radiological control program for Westinghouse Hanford Facilities is established by WHC-CM-1-6, *WHC Radiation Control Manual*.[12] Health Physics manages the program.

6.1.2.4 Changes in Plant System Design

Configuration control management for the K Fuel Storage Facilities is controlled by WHC-CM-6-1, *Standard Engineering Practices*.[58] Capital modifications are controlled by WHC-CM-6-2, *Project Management*.[59]

Physical facility modifications and procedure changes are subject to the Unreviewed Safety Question process[72].

6.1.2.5 Work Control

Work control requirements are defined by the WHC-CM-8-8, *Job Control System*.[60]

6.1.3 Quality Assurance Programs

Quality assurance program elements are defined in WHC-CM-4-2, *Quality Assurance Manual*.[20] Quality Assurance manages the program.

6.2 Training Program

Training requirements for personnel involved in support of the K Fuel Storage Facilities are derived from numerous sources.[67, 46, 47, 12, 61, 69] These include criticality safety, radiological safety, emergency preparedness, hazardous waste, industrial safety, security, and work control. The training and qualification requirements for respective individuals are maintained on Soft Reporting, a company-wide computer program. The training features both general and facility-specific elements. These requirements are routinely updated to reflect changes in frequency and/or content. Individual training must be current to qualify the individual to perform the governed tasks. Training requirements and qualifications are reviewed on a continuing basis.

6.2.1 Nuclear Criticality Safety Training

The Nuclear Criticality Safety Training Program[46] ensures that operators, supervisors, and managers involved in the control, handling, storage, processing, or transferring of fissile materials in quantities that could support a chain reaction possess the knowledge and experience commensurate with the responsibility for their assigned tasks and that they are aware of the risks involved with a nonreactor nuclear facility criticality. The Nuclear Criticality Safety Training Program fulfills the requirements set forth in DOE Order 5480.20.[69]

6.2.1.1 Basic Program

WHC Technical Training provides formal criticality Safety Training classes for fissionable material handlers an managers/engineers. This formal classroom training that includes examinations and a passing score is required every two years. General topics covered in the formal class room training includes:

- Basic criticality principles (consequences, factors, contingency, prevention, and responsibilities)
- Labeling and posting
- Emergency procedures and their responses

On the job facility specific training is provided on a continuing basis and in periodic safety meetings as considered necessary by the supervisors/managers and as required by the Nuclear Criticality Safety Manual[46]. As a part of this training, the facility Criticality Safety Representative (CSR) provides quarterly criticality safety training information to the managers/operators for their review. Annually, the CSR conducts a safety meeting for managers and operators to review this criticality safety training information and present new material as appropriate.

Time is allowed in the facility training for the discussion of unusual occurrences and work-oriented problems. Changes to procedures and Process Standards (which are the Criticality Prevention Specifications), Operational Safety Requirements, WHC Management Requirements and Procedures, Emergency Procedures, and Radiation Work Permits are also addressed.

In addition to the periodic training sessions, each manager responsible for handling nonreactor fissionable materials is responsible for on-the-job training of assigned personnel. The managers not only train workers in the use of procedures and Process Standards, but also in the actual use of equipment to perform the various jobs. Training in evacuation procedures through the use of periodic practice plant evacuations is also provided.

6.2.1.2 Record Keeping and Testing

Nuclear Criticality Safety Training Program classes are scheduled regularly. Personnel handling fissile material are required to attend one training program every two years. Each person signs the attendance sheet when attending training sessions.

A written test is given at the end of each formal training session. Retraining and retesting are given to those who fail the test. A summary check sheet contains the names of the personnel successfully completing the training program, the date attended, and their pass-fail test grades. Formal training records that verify satisfactory completion of the formal training are maintained by Technical Training. The records for Job-Specific-Orientation and periodic training are established and maintained by facility management .

6.2.2 Radiation Worker Training

Before an employee enters a radiation area, Radiation Worker training is required. The Radiation Worker Training Program is summarized below.

Employees who work in radiological areas receive several hours of training on radiation protection that includes: WHC Radiation Control Manual[12] requirements, Radiation Work Procedures, respiratory protection, personnel dosimeters, emergency signals, radiation controls, contamination control, radiation effects, and as low as reasonably achievable (ALARA) principles.

Each manager provides on-the-job training for assigned personnel who work in radiological areas. Each manager also holds staff and safety meetings in which radiation control and radiation safety topics are covered.

Information letters concerning radiation protection topics are furnished periodically to Company workers.

Facilities Health Physics Technicians furnishes information on radiological conditions in radiological areas to employees before they enter a radiation area.

Radiological area workers are required to attend a radiation worker refresher training every 2 years.

6.3 Monitoring Equipment

6.3.1 Radiation Instrumentation

6.3.1.1 High Level Gamma Instrumentation

High level gamma instrumentation has been provided to actuate visual and audible alarms located in each facility work area, both facility control rooms, and the facility entrances upon actuation of a high level gamma alarm. The high radiation alarm set point is controlled and is kept locked.

6.3.1.2 Nuclear Criticality Dosimeters

Criticality dosimeters are located within the fuel basin areas to provide data in the unlikely event of a nuclear criticality accident.

6.3.1.3 Gamma Radiation Monitors

Instrumentation is provided to monitor the radiation levels in the recirculation equipment areas, in the railroad car cask loadout transfer areas, and at several locations within the cooling pool work areas. The instruments are equipped with adjustable high radiation setpoints. Radiation data will be transmitted and recorded in the K Facility control rooms. A radiation alarm will actuate a visual alarm located in each cooling pool work area and actuate annunciators in both facility control rooms.

6.3.1.4 Ion Exchange Column Radiation Monitoring

The ion exchange column radiation levels are monitored with portable instruments.

6.3.2 Water Level Instrumentation

Redundant electrode operated alarms have been provided to monitor each cooling pool water level. Any high or low level trip provide annunciation in both facility control rooms. The alarm system is functionally tested routinely. In addition, a 24-hour drawdown test of each cooling pool is performed monthly to determine leak rates. A water level recorder is also provided.

6.3.3 Water Temperature Instrumentation

The cooling pool water temperatures are monitored and recorded in the facility operations offices. A high temperature alarm is set to annunciate in both facility control rooms. A high temperature alarm will annunciate. This system is functionally tested routinely.

6.3.4 Water pH Instrumentation

The cooling pool water pH levels are monitored and are recorded in the facility operations offices.

6.3.5 Corrosion Monitoring

There are two areas in which corrosion may become a factor. The cooling system is a concern because the piping used is made of carbon steel. Periodic inspections of this piping ensure its integrity. The other area of concern is the aluminum canisters and encapsulation components. It is not known what effect the pool environment will have on the aluminum or how long the canisters will last.

The aluminum components are subject to pitting corrosion. Extensive pitting would initially cause leakage of radioactive material into the pool and eventually reduce the ability of the canister to hold the fuel.

A program has been developed to track the integrity of piping to ensure that enough lead time is available to take corrective action should corrosion become a problem.

6.3.6 Testing and Calibration

All instrumentation discussed in this section is functionally tested routinely and is periodically calibrated in accordance with approved procedures.[44]

6.4 Operating Procedures

Listed below are the primary procedures that will be in effect at the facilities for handling and storage of irradiated fuel.

- Cask unloading at K Facility irradiated fuel storage basin.
- Cask loading and well car loading for fuel shipment.
- Changing and disposal of primary coolant filters.
- Operation of secondary coolant system.
- Operation of the liquid waste pump and disposal of waste from the liquid waste holding tank.
- Operation of the auxiliary exchange modules.
- Canister decapping.
- Addition of water to primary coolant system.
- Operation of 181-KE raw water submersible pumps, 183 filters, and 190 service water pumps.
- Functional test of all cooling pool monitoring systems.

- Functional test of Transfer Area cranes and electric hoists.
- Operation and functional check of air monitoring system.
- Primary and secondary cooling water sampling program.

6.5 Emergency Procedures

The *Emergency Response Procedures*, WHC-IP-0702[61], include the following emergency guides for the KE and KW Basins:

- Area Evacuation
- Bomb Threats and Terrorist Acts
- Chlorine Release
- Response to High Radiation Event, Oil Spill, or Hazardous Chemical or Radioactive Release
- Emergency Rescue
- Fire and Explosion Control
- Loss of Electrical Power
- Natural Events Mitigation

Copies of these Emergency Response Procedures will be available in the KE/KW area for personnel use in the event they are needed when the facilities are not manned.

6.6 Fuel Inventory Control and Accountability

Special Nuclear Material (SNM) and inventory control records will be handled in accordance with instructions issued by the Manager, K Basins before each transfer campaign. It is essential that the fuel handlers maintain absolute control over material received from N Reactor for storage at the facility, transferred from one basin to the other, or to some other destination.

Any discrepancies noted when loading or unloading canisters from the casks will be resolved before another canister is unloaded or any cask movement is made.

The facilities' irradiated fuel storage accountability and summary records will be routinely checked against stored canisters in the cooling pool.

6.7 Fire Protection

A wet sprinkler system is installed in the facility control rooms, offices, lunchrooms, and clothing change rooms; any operation of this system annunciates at the Hanford Fire Department Station No. 2, 609A Building. Fire alarm boxes are located inside and outside the building, as well as telephones located at several locations inside the building which are available for contacting the Fire Department.

The normal water supply to the facility fire protection systems, both the sprinklers and outside hydrants, is the same filtered water source (1,500 gal/min) that provides emergency water to the cooling pools.

The fire protection alarm system is part of the Hanford Fire Department Station No. 2 radio fire alarm reporting system. All the detectors at the facility are in series and tied into the system. The power supply is dc (batteries) located in the 1720-K Building equipped with ac rectifier charging units. The batteries are rated for 72 hours and if they drop below 85 percent of full charge, an appropriate alarm will sound in the Hanford Fire Department Station No. 2 alerting people in this continuously manned area of this condition.

The Hanford Fire Department Station No. 1 is located about 3 miles away. Operations Support Services is responsible for operating this station and its equipment. Fire Fighting personnel periodically inspect and test the fire protection systems and equipment provided in each facility.

General housekeeping for fire prevention and training for those individuals involved in fire risk work is provided in accordance with WHC-CM-4-3, *Industrial Safety Manual*.[47]

6.8 Review and Appraisal

DOE Order 5480.5, Section 9, Contractor Independent Review and Appraisal System[2], requires the establishment and maintenance of an internal review, auditing and appraisal system. This requirement is met within Westinghouse Hanford by WHC-IP-0860, *Nuclear Safety Manual*.[48]

6.9 Effects of Normal Operation

6.9.1 Effluents

6.9.1.1 Liquids

Beneath each basin is an asphalt membrane which was installed during the original construction. This membrane was intended to collect any basin leakage and divert it to a tile drainage field. The diversion line has been intercepted and connected to a sump. Since repair of a leak in the discharge pit area at KE, no leakage has been detected in this system at either

facility.[49] The basin leak collection membrane will not collect leaks from the discharge pit. An additional check for leakage is provided by seven monitoring wells around each basin. Quarterly sampling from these wells also indicate no leakage. Any leakage collected would be pumped from the sump back into the basin.

Filtered water is pumped through the water-cooled chiller to provide backup cooling for the basin water, then discharged directly back to the Columbia river. This discharge is covered by the DOE NPDES permit for outfall No. 004. At the rated capacity of one chiller at each basin and the river at its minimum flow rate the thermal discharge could increase the river temperature less than 0.002F. Currently (1994) the basin water is cooled by air-cooled chillers to minimize the effluent for outfall No. 004.

6.9.1.2 Gaseous Waste

Ventilation is provided by four powered roof vents, two over the basin rated at 10,000 cfm each and two over the transfer area rated at 7500 cfm each. The basin vents operate as needed. The discharge is provided with a continuous air monitor. The radiological emission totals for a year would not result in producing a dose of 100 mrem at the point of discharge. Emissions are primarily of ^{60}Co, ^{137}Cs, ^{90}Sr, ^{106}Ru, ^{241}Am and Pu isotopes.[49] Therefore the onsite and offsite doses are well within all limits of DOE Order 5400.5A and 5480.11.

6.9.1.3 Solids

The water filter cartridges are changed out about every 6 months and shipped to the 200 areas for burial. The ion exchange columns are changed out approximately every 6 weeks. The spent resins are shipped to the 200 areas for burial.

6.9.2 Personnel Radiation Exposure

All operations within the facilities will be conducted in accordance with procedures and practices developed specifically to assure that occupational radiation exposures are kept as low as reasonably achievable (ALARA).

Personnel working in the facilities will be exposed to radiation from the spent fuel during unloading and transfer. The maximum dose rate expected at the operating floor level from these operations is 50 mrem per hour at the transfer and segregation stations in KE. Personnel in the KE Facility will experience dose rates on the order of 20 mrem per hour. The dose rates at the KW Facility are less than 1 mrem per hour. The radiation exposure control program records and tracks all personnel dosimetry results to ensure that all individual exposures are maintained as far below the quarterly and annual limits as possible. The collective radiation exposure to the work force is also minimized.

Area radiation monitors and air particulate activity monitors will continuously operate in the facilities to provide warning of high radiation conditions. Regular surveys by radiation monitoring personnel and posting of high radiation or contamination areas will help reduce or eliminate unplanned exposures. Internal exposures to operating personnel will be controlled by use of respiratory protection if required.

Figure 6.1 Spent Fuel Disposition Project Organization

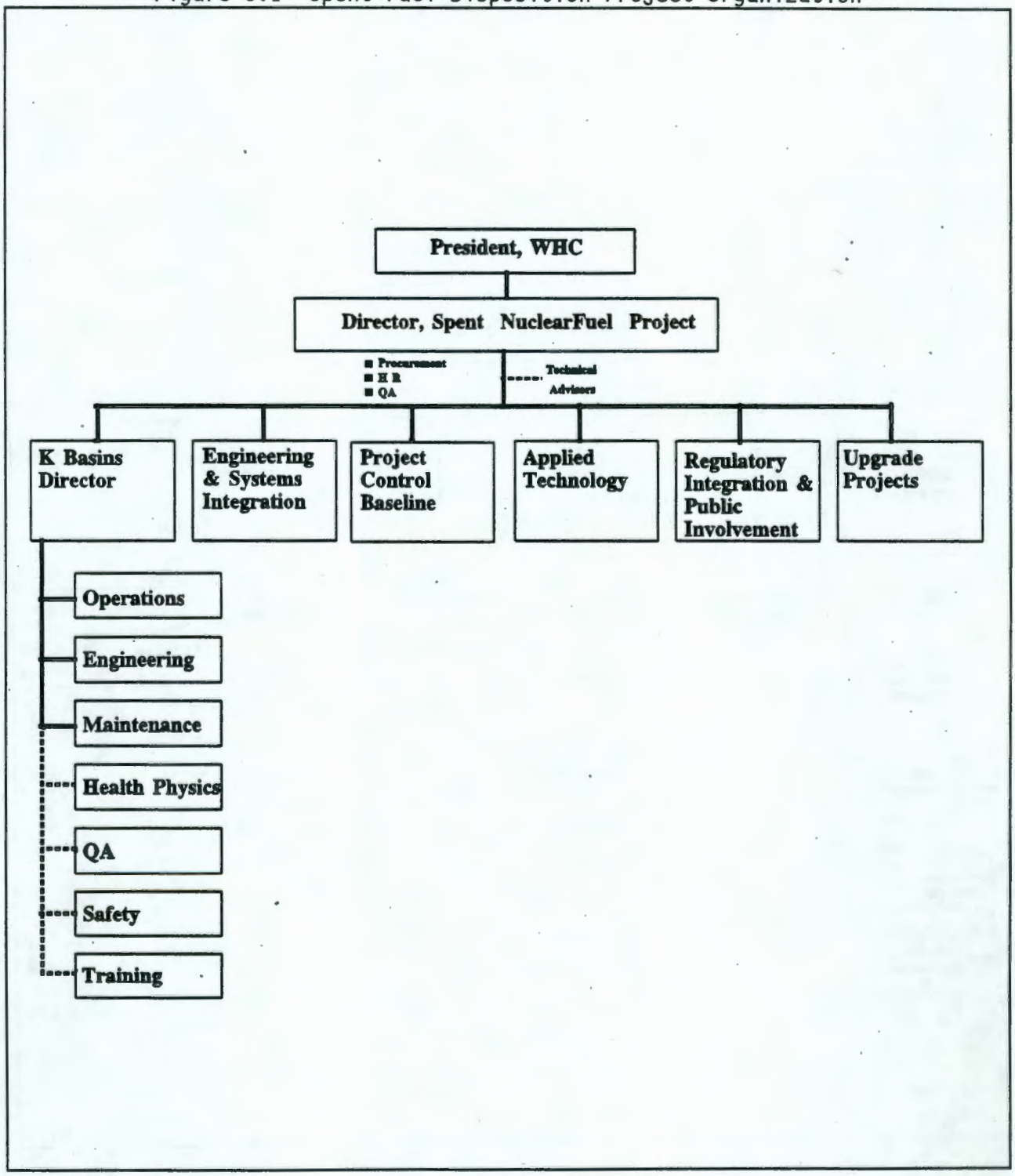
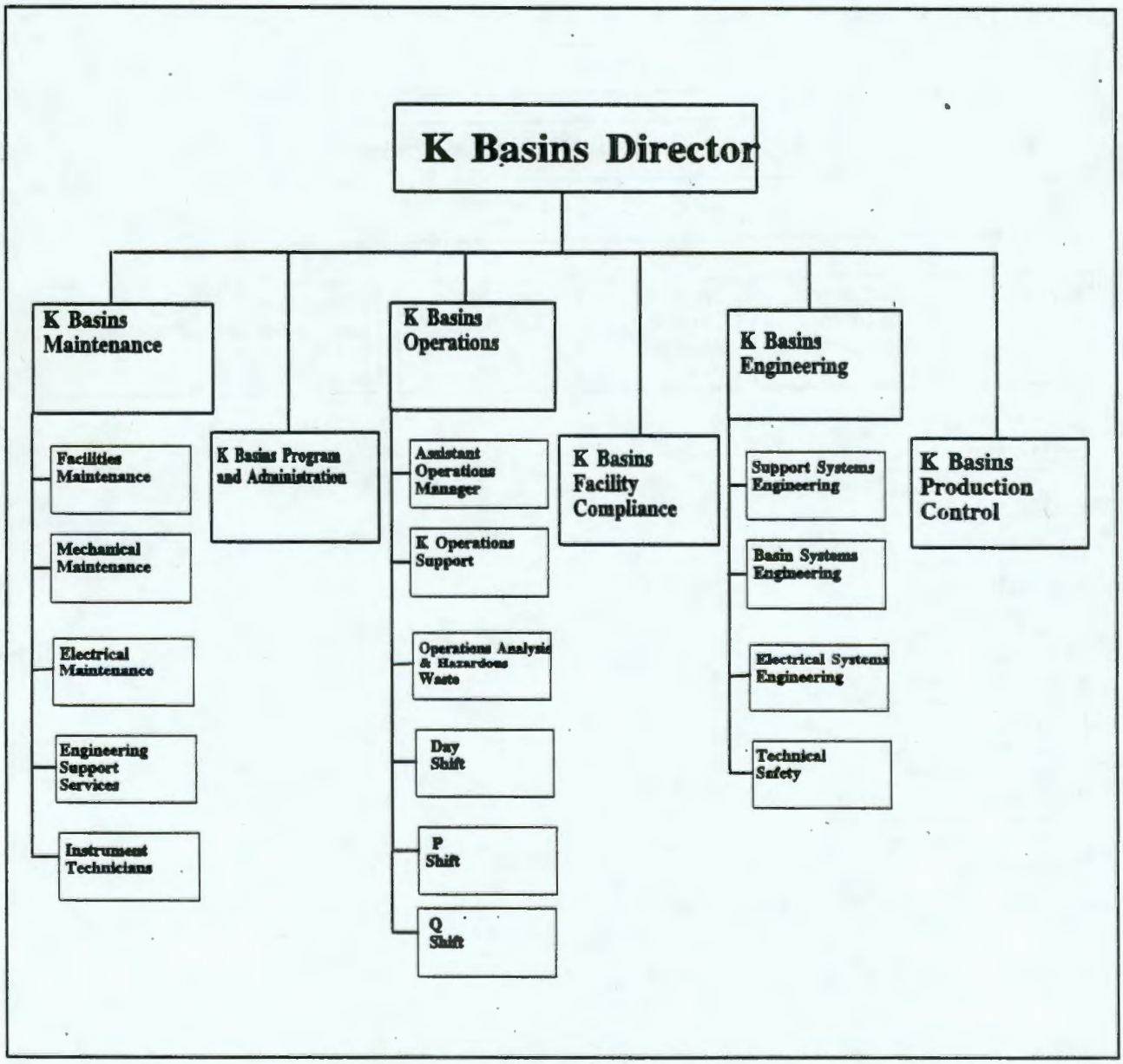


Figure 6.2 K Basin Organization



7.0 CRITICALITY SAFETY ANALYSIS

The storage of irradiated N Reactor fuel in the KE and KW Fuel Storage Facilities has been evaluated for potential nuclear criticality accidents and found to be safe and within the nuclear criticality safety criteria and limits established for the fuel storage basins operational activity.[3,4,6,7] The basis of the nuclear safety limit is that the neutron multiplication factor, or k-effective, will remain below 0.98 for all postulated accidents. The fuel storage configuration used in the analysis is shown in Figures 1.2 and 1.3 and consists of full density storage of fuel stored on the basin floor. This fuel can be any combination of the various enrichments used at N Reactor. The hanging fuel, hung at a ratio of one canister over three floor storage canisters, may also be of a single enrichment or any combination of irradiated N Reactor fuel. However, no hanging fuel canister storage of 1.25wt % scrap is allowed. Hanging canisters of any kind over canisters containing 1.25wt % of scrap is not allowed.[7]

The criticality analysis supporting the fuel storage configurations is based on calculations assuming use of unirradiated fuel of the highest authorized enrichment in the storage arrangements and postulated accident conditions. The assumption of unexposed Mark IA fuel in these calculations is the most conservative and most limiting basis for the calculations from a standpoint of nuclear criticality safety. Further conservatism in the calculations was achieved by assuming use of only aluminum canisters. The use of stainless steel canisters would provide additional safety margins as discussed in Reference 55.

The calculational models of fuel stored in the unique fuel storage geometry were compared to the measured data in PT-N-506[4]. The results of this test confirmed that the calculational models would adequately ensure that the postulated accident conditions would not exceed nuclear safety limits.

Additional criticality evaluations were made regarding the accumulation of dissolved fissile isotopes[7,8] in various pieces of equipment (ion exchange columns, sand filter, cartridge filter, etc.). The criticality analysis for the decapping design change[7] identified several criticality concerns for the Decapping/Recanning Project. Some of these concerns require that administrative controls be applied to this operation, while others require modification to equipment to implement physical constraints to supplement administrative controls.

Criticality control limitations and procedural requirements, including routine samples for fissile isotope concentrations, will be specified in the facility Process Standards.

The criticality evaluation for the basin ion exchange modules and ion exchange columns[70] provide the interim safety basis to support operations without a Criticality Alarm System (CAS). The DOE approval letter[71] identifies issues that will be addressed in future SAR amendments to provide the final safety basis for operations without a CAS.

8.0 ACCIDENT ANALYSIS

The postulated accidents associated with this facility are discussed in this section. Also included are the potentials for and the consequences of radioactive releases associated with the accidents. Administrative procedures and specific equipment design limit the potential frequency of accidents considered from "infrequent" to "not expected to occur during the lifetime of the facility". Reference 1 provides the bases for these frequency classification. The potential consequences of the accidents considered were minimal.

8.1 Criticality Related Accidents[3,4,5]

A series of incidents with a potential for accidental nuclear criticality were evaluated. In all instances, more than one unusual independent and simultaneous event would have to take place to exceed the storage basin safety limit. The accidents analyzed are summarized below. All criticality safety analyses of storage configurations considered the highest enrichment fuel currently used (Mark IA).

8.1.1 Railroad Car Fuel Spill

Transport of irradiated fuel by rail is the responsibility of the Transportation/Waste Handling of ICF Kaiser Hanford Company. Their safety analysis concludes that in a railroad well car accident the fuel casks could be thrown from the wells. The casks could lose their coolant but contain the fuel.[38,39] Even if it is assumed that the fuel spills from the casks and into a hemispherical configuration, criticality could not occur, since a well car carries only 126 fuel elements, 42 in each of the wells. A minimum of 169 Mark IA fuel elements are required for a criticality to occur in an optimally moderated and reflected hemispherical configuration.[51]

A railroad car fuel spill accident is judged to fall within the category of "not expected to occur during the lifetime of the facility."

8.1.2 Cask Fuel Spill

The minimum number of Mark IA fuel elements needed to form a hemispherical critical mass is 169.[51] Consequently, spilling a cask (which contains only 42 fuel elements) will present no criticality problem.

A cask spill is judged to fall in the category of infrequent.

8.1.3 Canister Drop

Dropping multiple canisters containing spike fuel from the one over three storage configuration onto a full density storage array of spike fuel, is a minimal perturbation. Such an incident would not raise the k-effective above

0.833. This value is significantly below the k-effective safety limit value of 0.98.

Special criticality concerns associated with capped canister (Mark I and II) storage were evaluated in Reference 3. Based on the analysis in that document, limits on the amount of fuel suspended above a full density array was established at a ratio of 1 over 3.

Equipment design limits the suspended fuel storage to the 1 over 3 configuration, therefore a 2 over 3 or 3 over 3 configuration is not possible. Administrative controls prohibit hanging fuel in the west bay of each basin, so that a railcar accidently entering the basin could not drop its contents and the hanging fuel onto the fuel stored on the floor.

A canister drop accident is judged to have an occurrence frequency of "infrequent."

8.1.4 Broken Fuel Elements

Some of the fuel elements shipped to the facility for storage will be broken and cracked. An estimated quantity of such elements is approximately five percent of the total fuel in storage. Calculations have assumed that one percent of the fuel has gone into solution and remains in the water immediately surrounding the fuel elements. This is a conservative assumption since the water circulates and passes through filters and ion exchangers for removal of radionuclides.

N Reactor spent fuel elements in the KW Basin are encapsulated in either:

(1) The standard N Reactor fuel storage canisters (designated Mark 0) modified by welding a plate on the bottom and installing a sealing cap on the top of the canister to become a Mark I canister, or

(2) The Mark II, a canister of improved design and made entirely of stainless steel. See Figure 2.1.

The KE Facility storage racks currently store fuel in the open Mark 0 canisters. However, fuel in Mark I or Mark II canisters, with or without encapsulation, may also be stored in KE.[50]

Encapsulation of the spent fuel elements in the Mark I and II canisters is intended to prevent the escape of solid and dissolved radionuclides into the K Facility basins. Gaseous fission products are vented from the fuel loaded canisters and dissipated through the building ventilation system. Release of non-gaseous radionuclides into the KW storage basin water can take place only through a leak in a canister or the cap assembly.

At the KE Facility more than five percent of the fuel is damaged, however the release of the uranium and plutonium is very gradual. Any suspended

particulate uranium and plutonium are continually being removed by the filters and ion exchange columns. The presence of dissolved fissile isotopes in the cooling pool should not present any potential criticality problems as these are being constantly removed by the ion exchangers. See section 5.7 for a discussion of the effect of decapping operations. Decapping 1.25 wt% fuel will require administrative limits on sludge buildup (see table 5.4).

8.1.5 Water Loss and Partial Moderation

The nuclear interaction between adjacent fuel canisters is reduced by the water separating them. Reducing the density of water could increase the reactivity of the storage array. During normal operating conditions, water density will vary with temperature. The changes are small and will have an insignificant effect on the reactivity of the storage array. However, upon loss of the basin water and during the process of reestablishing it, the fuel could go through stages of partial moderation. Calculations using the KENO computer code were carried out to evaluate partial moderation and reflection of fuel.

The results are contained in References 3, 4, and 5. Calculations in Reference 4 indicate that the k-effective value for a partially moderated storage array will not exceed that of a fully moderated storage array. For spike fuel, the k-effective value for the full density array with one-over-three hanging fuel array is within the k-effective safety limit of 0.98 for all normal and credible single contingency accident situations. It is apparent that even during a loss of coolant, multiple events would have to take place for a criticality to occur.

The loss of coolant in the facilities is judged to fall within the category of "not expected to occur during the lifetime of the facilities."

8.1.6 Loss of Storage Array Geometry

Collapse of the support structure with fuel stored in a one-over-three array would cause a fuel concentration of 443.4 lbs. uranium/sq. ft. on the basin floor. For unirradiated Mark IA fuel with optimum moderation and reflection, the minimum critical mass of fuel would be 401 pounds of uranium per square foot. Thus, for unirradiated Mark IA fuel in the one-over-three configuration, a loss of storage array geometry accident would pose a potential criticality problem. However, unirradiated fuel will not be stored that way. The minimum critical mass for Mark IA exposed to 6% Pu-240 is approximately 25% greater than for unexposed Mark IA[5]; consequently, the minimum critical floor loading for 6% Mark IA is about 500 lbs./sq. ft. basin floor. Thus, for one-over-three storage, the loss of storage array geometry accident does not pose a criticality problem for exposed fuel.

A 2-over-3 or 3-over-3 configuration of spike fuel is not safe, however, hanging storage facility design is intended to prevent this configuration.

For a criticality to occur with lower enriched Mark IV fuel, the fuel would have to be removed from the canisters and then be configured in an optimum hemispherical array.

Dropping a loaded cask car into the basin has been considered. It has been determined that only one car could enter the basin.[52] If the car could drop the suspended fuel and its own load (of all spike fuel) in the same 1-over-3 configuration of spike fuel, criticality would be a possibility. The possibility of a criticality is prevented by authorizing only the lower enriched MKIV fuel to be stored in the first (west) bay of each facility's basin. Also, storage by hanging fuel in the west bay of each basin is prohibited.

A cask car dropping into the west bay of either basin would not cause a criticality in the east and middle bays because of the 3 foot thick concrete wall divider between bays.

Collapse of the hanging fuel support structure is judged to fall within the category of "not expected to occur during the lifetime of the facilities."

8.1.7 Well Car Drop Into the Fuel Storage Basin

The possibility of a well car accidentally dropping into a K Facility basin has been effectively eliminated. The measures taken to ensure that it does not occur are discussed in section 8.2.8. In addition, criticality from such an event is prevented by fuel storage restrictions as discussed in Section 8.1.6.

8.2 Noncriticality Related Accidents

8.2.1 Loss of Monitoring Instrumentation

The loss of a critical monitoring instrumentation would cause an annunciation in the K Facility control rooms. Procedures are provided for appropriate response and corrective action.

8.2.2 Liquid Waste Car Accident

A review of possible impact accidents at railroad crossings indicates that in all cases there is a possibility of a spill.[53] The severity of any spill will depend on the mode or combination of modes of accidents postulated. The analysis referenced indicates the probability and consequences of an accidental spill from this cause are at an acceptable level. See Reference 58 for details.

8.2.3 Crane Failure and Cask Dropped Into the Fuel Loadout Pit[16]

A fully loaded transfer cask, dropped accidentally onto the floor of the transfer pit through a vertical distance of 25 feet, would fracture the pit

floor slab. However, the extent of slab failure and accompanying cracking would not permit water to leak from the pit at a very fast rate (25 gpm).[30] There would be a minimum environmental release from such an accident because of the underbasin collection membrane. A cask drop into the fuel loadout pit is judged to have a frequency classification of "infrequent".

8.2.4 Crane Failure and Cask Dropped onto the Floor of Transfer Area (Maximum Credible Accident)

A fully loaded transfer cask, dropped accidently on to the floor of the transfer area from a height of 15 feet or more and overturning would result in the cask lid coming off and the irradiated fuel spilling out of the cask. The maximum temperature of the fuel elements would be less than 910°F as shown by water loss fuel heatup calculations for fuel in canisters in the cooling pool.[14] An analysis of the radiological consequences from the gaseous release of volatile solids involving a fully loaded cask (42 elements) of irradiated fuel in such an accident was evaluated.[15] The potential individual dose at the site boundary is calculated to be 7×10^{-6} mrem whole body and 1.5×10^{-3} mrem thyroid.

A cask containing three canisters of broken fuel segments dropped onto the transfer area floor would be expected to release about 30 times more radioactivity than a cask of all unbroken fuel elements. The original analysis covering unbroken fuel assumed that three fuel elements were broken during the cask drop and spill.[15] The breaks were assumed to be at a 45 degree angle, with radioactive release predicted on resultant exposed surface and 100% of the broken fuel noble gas inventory. A canister of broken fuel segments is estimated to have a factor of 30 more exposed uranium surface. Consequently the offsite population exposure would be about 2×10^{-5} mrem - whole body and 4.5×10^{-2} mrem thyroid. These are well within the DOE limits of 500 mrem whole body and 1500 mrem thyroid.

Such a cask drop as described above could result in unshielding the fuel and cause a high radiation alarm requiring immediate evacuation of personnel. In an incident at N Reactor, reported in Abnormal Occurrence Report AO No. 2-77, personnel were exposed to fuel as it was being discharged from the reactor. The quick response (evacuation) of the affected personnel resulted in individual doses of 4 to 15 rem. The radiation levels for fuels stored 150 days, as in the cask drop situation, is much lower than freshly discharged fuel. Based on this experience, personnel exposed to unshielded irradiated fuel from a cask drop would receive a radiation dose not exceeding the 5 rem annual limit. The personnel dose following evacuation would be higher but comparable to the offsite dose and well within the limit.

A cask drop is judged to have a classification of "infrequent". Such a drop containing all broken fuel segments is judged to have a classification of "not expected to occur during the lifetime of the facility" because of the small quantity of broken fuel segments (5% of the total).

8.2.5 Loaded Canister Dropped from a Lifting Hoist to the Pool Floor

A fully loaded canister, accidentally dropped onto the floor of the fuel storage basin from a height of 3.5 feet, would not fracture the bottom floor slab, and would not result in an environmental release.[16]

8.2.6 Fire and Explosion Hazards

The three modification projects (Projects H-501, H-508, and H-558) required only minor alterations and additions to the facility storage pool areas or other parts of the 105-K buildings. None of these should affect the fire or explosion risk to the facilities. The facilities are of noncombustible construction and no additional combustible or explosive materials are required for the operation of the facilities. An adequate supply of water will be available for fire protection needs. The existing fire sprinkler systems in the lunch rooms and change rooms were extended to cover all the shop and office locations which are in service in the facilities. The fire and sanitary water systems have been reactivated including the fire hydrants around the 105-K buildings.

The use of the ion exchange systems will not present any hazard of explosion since there will be no attempts to regenerate the ion exchange material. A mixed cation and anion organic resin may also be used in the ion exchange columns, if necessary, to reduce the radionuclide concentration in the basin water. If a mixed organic resin is used, the residence time will be administratively limited to prevent a possible degradation of the resin.

8.2.7 Loss of Pool Coolant

Early in the development of the KE-SAR the complete loss of storage pool coolant was considered as the maximum credible accident. However, with the issuance of UNI-287, "Extreme Load Analysis of the 100-K Fuel Storage Basins and Clearwells", (Ref. 16), the complete loss of water from the storage pool due to an earthquake or an accidental drop of the fuel shipping cask into the transfer pit is no longer considered credible. A railroad well car falling into the basin is also considered not credible as explained in Section 8.2.8. The maximum credible accident is now defined as a cask drop in the transfer area that results in spilled fuel outside of a transfer cask. Further discussion of the maximum credible accident is provided in Section 8.2.4.

However, two analyses have been made of a hypothetical loss of coolant. The first study, Reference 14, is based on a full floor loading of 150 day stored (decayed or aged) fuel. This study determined a maximum fuel temperature of 910°F. At this temperature the fuel cladding will remain intact. There would be release of volatile solids from damaged fuel as discussed in Section 8.2.4. With less than 5% of the fuel damaged the release would be less than 200 times the release from a cask drop. The second analysis[31] was based on heat generation of the fuel as actually stored in the nearly filled KE Facility. This fuel has been stored or aged much longer than 150 days. The maximum fuel

temperature would be 781°F for a full floor load with 16 1/2 rows of hanging fuel.

8.2.8 Uncontrolled Railroad Train

A hypothetical accident involving a runaway train has been identified. It is postulated that the train engineer becomes physically incapacitated as the train approaches or enters the facility. The rails at the K facilities line up directly with the fuel storage pool and terminate approximately 6 feet from the edge of the fuel storage pool. The weight of one loaded cask car is approximately 120 tons and it is not inconceivable that a car could enter the pool in this kind of accident. Without specifically calculating the effects, it was determined that the weight of one cask car would be well in excess of the amount necessary to damage the floor.[30] Based on these determinations, it was concluded that to prevent such a hypothetical accident from becoming a reality, protective devices and procedures should be in place to prevent a runaway or uncontrolled train from entering the facility.

Derailers, railstops, and bumpers are utilized as protective devices. The combined weight of the heaviest loaded cask car and locomotive is 470,000 pounds. The kinetic energy of this combination at 2 mph is 63,100 ft-lbs, at 4 mph 252,000 ft-lbs. The rail stops on the southern-most rail spur coming into each basin can absorb 31,500 ft-lbs, the bumpers 59,000 ft/lbs for a total capacity of absorbing 90,500 ft-lbs. This is sufficient to stop a train traveling 2 mph, but clearly is inadequate for a train with a velocity of 4 mph. The bumping posts used to protect the northern-most rail spur coming into each facility are designed to absorb 252,000 ft-lbs, which is adequate to stop a one-car train going 4 mph. Derailers capable of derailing trains traveling at speeds up to 5 mph are located 520 ft from the end of the track. The derailers are double locked in the active position with Facility Manager ~~Supervisor~~ and train crewman locks. These are unlocked after the train has uncoupled all but one cask car and stopped adjacent to the derailer. The locks are then removed from the derailer by the train crewman and the Facility representative.* The Facility representative then rides in the locomotive cab while the locomotive is within the derailer location.

The GM locomotives are provided with a Pace Setter electronic controller (see Section 5.6.4) which can be accurately set for any low speed. In this locomotive the Facility representative ensures that the Engineer properly sets and activates the Pace Setter. An emergency stop switch is also available on these locomotives. Excess speed is determined by an electronic speedometer reading in one mph increment.

There are a number of protective features in place to prevent an out of control train from entering the facilities. The track switch which allows trains to enter KW is kept in the closed position, routing any out of control train to KE. The entrance to KE is protected either by a crash car with locked brakes or the switch which will route trains to the auxiliary siding.

Both KE and KW have locked derailers which must be removed before trains can enter. The Pace Setter control unit, the emergency stop switch, and the presence of the Facility representative in the cab of the locomotive ensures that the train will remain under control as it approaches the basin.

*As defined in the Facility Operational Safety Requirements.

9.0 QUALITY ASSURANCE

9.1 Construction Phase

Projects H-501, H-508, H-558, DC-82180, DC-82121 and DC 82223 were completed in accordance with a documented quality assurance program plan which encompasses project activities from the issuance of design criteria through acceptance of the completed facilities. This plan defined the requirements applicable to UNC (now Westinghouse Hanford), Vitro (now Kaiser), the architect-engineer and J.A. Jones Construction Company, the construction contractor. In defining these requirements, the plan incorporated appropriate elements of the quality assurance programs which were already in place for DOE sponsored work at Hanford by these three organizations.

Compliance with the quality assurance program plan for Projects H-501, H-508, H-558, DC-82180, DC-82121 and DC-82223 was verified by the UNC Quality Assurance Department through auditing and review and approval of (1) documents which defined the specific quality assurance measures to be applied, (2) project design criteria, drawings, specification documents and acceptance test procedures, (3) procurement documents for engineered items, and (4) inspection plans. Audits consisted of reviewing the adequacy and findings of the audits conducted by the architect-engineer and the construction contractor of their activities, as well as auditing for compliance with the requirements applicable to UNC.

Quality records in the form of properly approved plans, procedures and instructions; design and procurement documents; manufacturer's certifications; reports of inspections, tests and audits; and other quality related documents are available for review in the project files.

9.2 Operating Phase

The operation and maintenance of the facilities, as modified by Projects H-501, H-508, H-558, and equipment installed under DC-82180, DC 82223 and DC-82121 will be governed by the quality assurance program document.[20] The program encompasses all applicable elements that relate to operation, maintenance, and any future modification of the K Basin facilities.

The quality assurance program document has been reviewed with the appropriate management of the facility to assure the implementation of the requirements. Quality Assurance will conduct periodic audits to assure compliance with all requirements of the program and to determine the effectiveness of the program.

10.0 OPERATIONAL SAFETY REQUIREMENTS

Limits, operating conditions and other requirements have been established to assure that the K Area Fuel Storage Facilities are operated in a manner that provides for the protection of the health and safety of the public. The requirements are included in the latest issue of the Operational Safety Requirements[54] and are implemented through the Process Standards[45].

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LIST OF TERMS

ALARA As Low As Reasonably Achievable
CSER Criticality Safety Evaluation Report
DBA Design Basis Accident
DBE Design Basis Earthquake
DBT Design Basis Tornado
DOE U. S. Department of Energy
DOE-RL U. S. Department of Energy-Richland Field Office
DRL Division of Reactor Licensing
EDE Effective Dose Equivalent
EIS Environmental Impact Statement
EPA U. S. Environmental Protection Agency
ERPG Emergency Response Planning Guidelines
gpm gallons per minute
HAZOP Hazard and Operability Study
IDLH Immediately Dangerous to Life and Health
Keff Neutron Multiplication Factor for a finite, defined system
Kgs Kilograms
Kw Kilowatts
LCO Limiting Condition for Operation
mcm Minimum Critical Mass
MK Mark
mRem millirem
MSL mean sea level
MTU Metric Ton Uranium
mw Megawatts
OSR Operational Safety Requirement
PHA Preliminary Hazards Assessment
PMF Probable Maximum Flood
PNL Pacific Northwest Laboratory
POC Patrol Operation Center
psig Pounds per Square Inch
Pu Plutonium
SAR Safety Analysis Report
SNM Special Nuclear Materials
SPR Single Pass Reactor
SSE System Survivability Earthquake
TRU Transuranic
UBC Uniform Building Code
USQ Unreviewed Safety Question
WDOH Washington State Department of Health
WHC Westinghouse Hanford Company

1.0 INTRODUCTION

As part of the preparations for encapsulation of irradiated fuel in the KE fuel storage basin, a comprehensive evaluation of the activities and conditions associated with the planned fuel encapsulation was performed. The primary purpose was to determine the adequacy of the current K basin safety envelope for these activities, as provided in the Safety Analysis Report (SAR) and the Operational Safety Requirements (OSR), and, if inadequate, to identify any appropriate revisions to this safety envelope. To further support the evaluation, a preliminary hazards analysis (PHA) was conducted of K Basin fuel storage operations. This evaluation included an examination of PHA events not specifically described in the SAR and an assessment of existing K Basin conditions to identify any additional appropriate actions. This evaluation indicated that the SAR adequately describes and analyzes the encapsulation activities, that the SAR and OSR provide an adequate safety envelope for these activities, and that the fuel encapsulation activities do not represent an Unreviewed Safety Question. The evaluation is documented in this report.

The 105-KE Basin has been in use for interim storage of irradiated N Reactor fuel assemblies since June 1975. There are three types of metallic uranium spent fuel presently stored in the 105-KE Basin. The first type is the Mark IV fuel which is enriched to 0.95 weight percent. The second type consists of natural uranium (0.71 weight percent). The third consist of two baskets (the equivalent of five canisters) of aluminum-clad Single Pass Reactor (SPR) fuel assemblies, that are 0.95 weight percent material, also stored in the 105-KE Basin. The fuel elements are presently stored in open aluminum Mark (MK) 0 canisters, stainless steel MK I canisters and open stainless steel MK II canisters. A small percentage of this fuel has cracked cladding and is releasing radionuclides into the basin water as the uranium metal continues to corrode. Therefore, approximately 1,150 MTU of fuel presently stored in the 105-KE Basin will be encapsulated. The scope of the encapsulation program is to replace the open canisters with new MK II fully sealed stainless steel canisters that are filled with water, sealed, and provided with an

inert gas expansion chamber and a corrosion inhibitor. Fully encapsulating all of the irradiated fuel in sealed stainless steel containers will effectively eliminate the future release of soluble or particulate radionuclides into the basin cooling water and will therefore reduce occupational radiation exposure. The encapsulation activities are currently scheduled to be performed over a period of 16 to 28 months in the Discharge-Pickup Chute area of the 105-KE Basin.

1.1 SAFETY EVALUATION PROCESS DESCRIPTION

The Safety Evaluation that was performed for the encapsulation process is shown in the Program Plan Flowsheet (Appendix I of Reference 1) and entailed the following steps:

- Identify Task Team
- Assess Existing and Potential Accident Scenarios
- Perform Preliminary Hazards Assessment (PHA)
- Review of Operational Safety Requirements (OSRs)
- Prepare Safety Analysis Report (SAR) Supplementary Documentation
- Prepare Safety Evaluation

Identify Task Team

A team of knowledgeable and dedicated personnel was established for the development of the Safety Evaluation to support encapsulation of the fuel stored in the 105-KE Basin. The team consisted of representatives from N Reactor, Facility Operations Programs, Safety Analysis and Regulation, Health and Safety Assurance, and Westinghouse Nuclear and Advanced Technology Division in Pittsburgh.

Assess Existing and Potential Accident Scenarios

The N Reactor personnel identified-a list of existing and potential accident scenarios that needed to be evaluated relative to the 105-KE Basin safety

envelope. These potential accident scenarios were reviewed by members of the Task Team to determine if the existing SAR (Reference 2) is bounding. The purpose of this review was to determine if any other accident scenarios may exist. If such a scenario was identified, action was taken to perform the necessary safety analysis to determine any potential consequences.

Perform PHA

A team of engineers and PHA experts performed a PHA of the 105-KE Basin normal operations and the planned 105-KE Basin encapsulation process. The purpose of the PHA was to identify and qualitatively assess the consequences of any abnormal conditions that potentially result in hazards to the facility workers, on-site personnel, and the off-site population. This was accomplished by performing a Hazard and Operability Study (HAZOP) of the normal operation of the 105-KE Basin including the encapsulation process, and preparing a report documenting the results of the studies (Reference 3).

Review of Operational Safety Requirements

A review of the existing facility OSRs was performed (Reference 5) to confirm that administrative requirements are provided such that the facility will be operated within the existing safety analysis envelope. Where necessary, modifications to the OSRs or facility operating documentation resulting from this safety evaluation, are identified. Any concerns identified as a result of this review and the PHA review were brought to the attention of the Task Team. The Task Team took the appropriate actions necessary to address these concerns in the OSRs or related facility operating documentation.

Review Existing SAR Documentation

A review of the existing 105-KE Basin existing SAR was performed to identify any potential changes required for encapsulation. No SAR changes were identified that

would be required as a prerequisite for encapsulation. Changes to incorporate during routine updating include identification of the equipment necessary for encapsulation, the encapsulation process, criticality Safety Evaluation, and waste generation. In addition, the Environmental Assessment (Reference 4) submitted for the encapsulation process was reviewed to ensure that the environmental impacts are included in the existing SAR. The associated tasks of preparing the PHA and OSR review identified additional accident analyses, facility operating documentation changes, and the bases to ensure that encapsulation does not exceed the allowable safety envelope established by the existing facility SAR and OSR. These results are incorporated into this Safety Evaluation report.

Prepare a Safety Evaluation

This Safety Evaluation assesses the encapsulation process to determine if the existing safety analysis is bounding. The Safety Evaluation encompasses the results of the tasks identified above.

It should be noted, that in the process of validating the existing safety envelope, the 105-KE Basin was used as the subject of the analysis and evaluations. The 105-KE Basin is the most restrictive by virtue of the unencapsulated fuel and the associated environmental and radiological consequences.

1.1.1 Safety Evaluation Scope

The scope of this Safety Evaluation is limited to an assessment of the existing safety envelope (Reference 2); evaluation of the encapsulation process relative to the safe operation of the facility; and, to provide the technical basis for addressing the criteria defined in MRP 5.12, (Reference 6) for determining an Unreviewed Safety Question (USQ). These limitations in scope were discussed and agreed to with DOE-RL as documented in the Program Plan (Reference 1) and accepted by DOE-RL in Reference 7.

1.1.2 Safety Evaluation Methodology

This Safety Evaluation was developed and reviewed in accordance with the following acceptance criteria (Section IV of Program Plan, Reference 1):

- Is the safety envelope as defined in the existing SAR valid?
- Are the existing OSRs sufficient to maintain facility operation within the existing safety envelope?; and
- For the encapsulation process:

  1) Could encapsulation increase the probability of occurrence of an accident previously evaluated in the existing safety analysis?

  2) Could encapsulation increase the consequences of an accident previously evaluated in the existing safety analysis?

  3) Could encapsulation increase the probability of occurrence of a malfunction of equipment important to safety, previously evaluated in the existing safety analysis?

  4) Could encapsulation increase the consequences of a malfunction of equipment important to safety, previously evaluated in the existing safety analysis?

  5) Could encapsulation increase the possibility of an accident of a different type than any previously defined in the existing safety analysis?

  6) Could encapsulation increase the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the existing safety analysis?

7) Does encapsulation reduce the margin of safety as defined in the basis for any existing Operational Safety Requirement?

The above criteria will serve as a basis for determining the acceptability of the encapsulation process relative to the existence of an adequate safety analysis envelope as discussed in. DOE Order 5481.1B (Reference 8).

1.2 105-KE BASIN FACILITY

As described in Section 1.2 of the existing SAR, the Fuel Storage Basin was designed to be an underwater storage facility for irradiated fuel. The basin is a rectangular reinforced concrete structure 125 feet long, 67 feet wide, and 21 feet deep, that is divided into three sections. Modifications that were made to the basin in 1975 included a recirculating system for the basin water with pumps, in-line filters, an ion exchange system, a sand filter system, heat exchangers, instrumentation to monitor radiation levels and instrumentation to monitor the water level and temperature in the basins. In addition, storage racks for the N Reactor fuel canisters were installed on the floor of the basin.

The basin is filled with water to a nominal depth of sixteen feet to cool the fuel and to provide a radiological shield for the facility workers. The water circulates through a closed water-cooling system by drawing water from one end of each basin section, circulating the water through filters, the heat exchanger or water cooler, and the ion exchange systems, then discharges it back at the opposite end of the basin.

The basin's cooling system was designed for a maximum heat load of 2500 kw and is based on storing N-Reactor fuel following a minimum 150 day period of heat decay. As of January 1991, with the fuel that is presently stored in the 105-KE Basin ranging in age from 4-20 years, the total basin heat load generation was 119 kw (Reference 9). A description of the Basin Cooling and Cleanup System and its respective components is provided in Section 5.3 of the existing SAR. The

specific design and operating parameters for each component is provided in Table 5.1 of the existing SAR.

1.2.1 Facility Layout

The encapsulation activities will be performed in the Discharge-Pickup Chute area of the 105-KE Basin. The Discharge-Pickup Chute area is connected to the south side of the basin and is directly behind the reactor building as shown in Figure 1.1 of the existing SAR. The Discharge-Pickup Chute area has the lowest exposure rate, approximately 5 mrem/hr, in the 105-KE Basin.

1.2.2 Facility Modifications to Support Encapsulation

The active components of the Basin Cooling and Cleanup System, including all monitoring instrumentation, will be in operation as required during encapsulation. In addition to the current cooling and cleanup capabilities, supplemental ion exchange flow that is currently available only in the transfer pit will be extended to carry any radionuclides released during the encapsulation process directly to the ion exchange columns. Spray nozzle heads and lances will also be added to flush the suspended sludge particles away from the work stations in order to improve water clarity. These features will utilize existing piping employed for previous purposes with slight modifications thereto. The features also will use existing pumping capacity of the basin recirculation system.

1.2.3 Encapsulation Equipment

Since the fuel encapsulation process is similar to the fuel segregation activities that were performed in 1983-84, the equipment for the encapsulation process will be similar to the equipment that was designed for fuel segregation activities. The segregation effort emptied fuel from canisters and separated fuel elements that would produce a weapons grade plutonium assay from a stored fuel inventory that was irradiated to an average exposure greater than that corresponding to weapons

grade plutonium. The segregation activity was also performed in the Discharge-Pickup Chute area of the 105-KE Basin.

Specialized equipment that will be utilized for the encapsulation process include the following:

- a work table for dumping fuel from the open canisters
- a canister splitter device for emptying canisters that will not dump
- a packager assembly to load fuel into the new canisters
- spray nozzles and lances for submerged flushing of sludge/particles from the work stations
- underwater lights to improve visibility
- a mechanical scoop for scraping up and depositing fuel debris into canisters
- scales for weighing canisters that contain fuel fragments, disassembled fuel, or sludge
- a nitrogen supply system for establishing an inert gas water seal in the canisters
- a corrosion inhibitor injection system to prevent the further deterioration of fuel assemblies in the new MKII canisters
- an encapsulation station that includes a lid transport system that lowers a pair of lids from above the water down to the level of the canister in the encapsulation station

- underwater cleaning and crushing station for emptied excess canisters

This equipment will be installed in the Discharge-Pickup Chute area of the 105-KE Basin. The encapsulation process will be performed in accordance with detailed operational and radiation protection procedures. An overview of the operations that will be performed is provided below.

1.3 ENCAPSULATION SEQUENCE OF OPERATION

Fuel canisters will be moved from their storage location in the 105-KE Basin to a dump table in the Discharge-Pickup Chute area using an existing trolley hoist and the facility's fuel handling procedures. Water clarification jets (spray nozzle heads and lance) will be used to move any suspended particles away from the working area. The fuel elements will be emptied onto the dump table. Once emptied, the old canister will be moved to a designated temporary discard storage area in the main storage basin. All intact fuel elements will be moved into the curved troughs of the Packager Assembly by the use of long handled tongs. With the new MK II canister laying flat in the Packager Assembly, the hydraulic plungers will insert the fuel into the canister.

In the event that fuel in either or both sides of a spent fuel canister will not dump, a splitter device will be used to open the canister. If the entire inventory of swollen fuel from a split canister can not be loaded into a single new MK II canister the remaining fuel elements will be placed in another new MK II canister. The basin inventory fuel accountability records will track and record these types of fuel transfers.

Fragments and oxides that remain on the work table after the new MK II canisters have been loaded will be gathered and encapsulated in a new MK II canister and handled and stored under a special criticality control limit (i.e., the weight of 14 of the longest MK IV fuel elements). Broken pieces of fuel elements that are a few inches long and can be handled readily with tongs, will be packaged into the new

canister. Pieces of fuel elements that are deemed large enough will be roughly reassembled into a fuel element configuration and encapsulated with intact fuel. Pieces and fines too small to be retrieved and placed in a canister will be allowed to fall and mix in with a limited sludge accumulation on the floor. Total fuel inventory of the discharge chute area including sludge will be controlled to a mass inventory limit based on incoming and outgoing canister contents.

The newly loaded canisters will be capped by installing a cover assembly that contains a GRAFOIL seal. Nitrogen will be inserted into the canister to provide an inert gas blanket in the canister above the fuel that isolates the water within the canister from the basin water. A predetermined quantity of corrosion inhibitor will also be added to the canister to provide a water/corrosion inhibitor cover over the encapsulated fuel elements. The MK II canisters of encapsulated irradiated fuel will then be moved back to a storage location in the 105-KE Basin. The old, emptied aluminum and stainless steel canisters will be cleaned, compacted, packaged and shipped to a disposal facility.

The SPR fuel (fuel that was not irradiated at N-Reactor) will be encapsulated in new MKII canisters under special weight limits and storage requirements. The SPR fuel canisters will be uniquely identified, kept separate, and neutronically isolated from all N-Reactor fuel. Empty SPR containers will be cleaned and prepared for disposal with the other emptied fuel canisters.

2.0 REGULATORY BASIS

The existing SAR describes a variety of fuel handling activities including those which comprise encapsulation as described in this evaluation. However, it does not describe encapsulation as a specific process utilizing the equipment and process described herein. MRP 5.1 2 Identification and Resolution of Unreviewed Safety Questions (Reference 6), allows evaluations of such activities. It is the purpose of MRP 5.1 2 to assess the potential for an Unreviewed Safety Question (USQ). This Safety Evaluation implements the MRP 5.1 2 requirements and

provides the supporting documentation. This Safety Evaluation will assess the encapsulation process relative to the safety analysis envelope as defined by the current authorization basis. For the encapsulation process, the Safety Evaluation will assess each existing SAR event, the events identified in the PHA and other events identified within this report to validate the existing safety envelope. The results will be evaluated relative to the existing safety envelope and the WHC acceptance criteria as documented in References 1 and 10. This regulatory approach has been discussed with and accepted by DOE-RL (Reference 7).

In conjunction with the above, this Safety Evaluation assesses on-site and off-site radiological and off-site toxicological events as identified in Appendix I of the Program Plan (Reference 1).

3.0 EVALUATIONS

Accidents that could result in criticality and/or radiation exposure at the site boundary during the handling and storage of irradiated N Reactor fuel in the 105-KE Basin have been postulated and analyzed or evaluated in Sections 2, 4, 5 and 8 of the existing SAR. The existing SAR describes the analyses or evaluations for the following postulated accidents:

Criticality Related Accidents

- Railroad Car Fuel Spill (SAR Section 8.1.1)
- Cask Fuel Spill (SAR Section 8.1.2)
- Canister Drop (SAR Section 8.1.31
- Broken Fuel Elements (SAR Section 8.1.4)
- Water Loss and Partial Moderation (SAR Section 8.1.5)
- Loss of Storage Array Geometry (SAR Section 8.1.6)
- Well Car Drop Into the Fuel Storage Basin (SAR Section 8.1.7)
- Sludge Criticality (SAR Section 5.5.4)

Non-Criticality Related Accidents

- Loss of Monitoring Instrumentation (SAR Section 8.2.1)
- Liquid Waste Car Accident (SAR Section 8.2.2)
- Crane Failure and Cask Drop Into the Fuel Load out Pit (SAR Section 8.2.3)
- Crane Failure and Cask Dropped Onto the Floor of Transfer Area (Maximum Credible Accident, SAR Section 8.2.4)
- Loaded Canister Dropped From a Lifting Hoist to the Pool Floor (SAR Section 8.2.5)
- Fire and Explosion Hazards (SAR Section 8.2.6)
- Loss of Pool Coolant (SAR Section 8.2.7)
- Uncontrolled Railroad Train (SAR Section 8.2.8)

External Events

- Loss of Off-Site Power (SAR Section 5.3.3)
- Loss of Pool Cooing (SAR Section 5.3.3)
- Design Basis Earthquake (SAR Section 2.5)
- Tornado (SAR Section 2.6)
- Floods (Probable Maximum Flood - PMF, SAR Section 4.3)
- Ashfall (SAR Section 4.6)

In addition to the accidents that were addressed in Sections 2, 4, 5 and 8 of the existing SAR, a PHA was performed to identify process related events which could potentially challenge the existing facility safety envelope.

Section 3.1, 3.2, and 3.3 of this report evaluate the existing SAR accident scenarios and Section 3.4 of this report evaluates specific PHA events identified by the S2 and S3 PHA screening criteria (Reference 3).

3.0.1 Existing SAR-Criticality Events

A nuclear criticality safety analysis was performed and is documented in the existing SAR (Reference 2) to demonstrate the acceptability of the present fuel storage configuration for both normal operation and credible accident conditions. The existing SAR criticality analysis analyzed various accidents involving fuel storage and handling. The existing SAR analysis determined that for all normal operations and credible single contingency accident situations the Keff Safety Limit of 0.98 is not violated.

3.0.2 Existing SAR Non-criticality and External Events

The non-criticality and external events reviewed in Sections 3.2 and 3.3 of this safety evaluation and addressed in the existing SAR met the applicable acceptance criteria for credible events. Sections 3.2 and 3.3 evaluations conclude, that relative to encapsulation, the non-criticality and external events discussed in the existing SAR either remain bounding with respect to the encapsulation process or the encapsulation process does not impact the existing accident scenario.

3.0.3 Criticality Safety Evaluation for Encapsulation

To demonstrate that the reactivity values are maintained within the current safety envelope during the encapsulation program, a Criticality Safety Evaluation (Reference 11) was performed for the 105-KE fuel encapsulation process. This evaluation provides the basis for encapsulation process controls. These controls will assure compliance with the facility criticality prevention criteria, that Keff not exceed 0.98 for any allowed condition and the occurrence of a single contingency, during encapsulation activities.

Since additional degradation of the 105-KE fuel may have occurred since the last time any major packaging or repackaging activity occurred in 1988, a conservative approach was taken to the analysis. The analysis assumed all fuel entering the

discharge chute area could be comprised of the most reactive sized and water moderated and water reflected .95% enriched uranium metal rods in a hemispherical shape on the basin concrete floor. In reality, the fuel will be comprised of integral fuel assemblies or elements, a percentage of uranium oxide fines and an unknown but probably small percentage of metal fragments approaching the size of the most reactive rod configuration upon which the analysis is based. In addition, as per usual practice, the unirradiated fuel enrichment was assumed and no credit was taken for fission products or other actual system poisons. Cylindrical rods were selected to represent the most reactive geometry as they are a conservative representation of the physical system being analyzed and extensive experimental data exists to bench mark the analysis.

Based on the above modeling approach, a discharge chute inventory mass limit equivalent to six fully loaded fuel canisters (723 lb/canister) or about 0.6 the mass of a hemisphere required to reach a keff 0.98 was selected (42% of the minimum critical mass). Various postulated contingencies were examined and the corresponding masses were added to the hemisphere, as optimally spaced and moderated uranium rods, to determine the increase in keff. None of the contingencies resulted in keff values approaching 0.98.

Thus the basis for criticality prevention of the actual encapsulation process reflects a conservative, mass based approach and wide margins of safety. Mass control in the discharge-pickup chute area during encapsulation is readily achieved. Incoming canister contents can be established by accountability records or weighing in water and applying a conservative buoyancy correction. Outgoing canister contents will be established by weighing in water. Upon reaching the mass inventory limit for the discharge-pickup chute area, either fuel will have to be removed in the form of encapsulated fuel or in the form of packaged sludge that is expected to accumulate on the discharge-pickup chute area floor during encapsulation. When sludge cleanup is performed, established cleanup requirements are satisfied and the packaged sludge removed to storage, zeroing the discharge-pickup chute area

inventory, if all other fuel is also removed, or subtracting the weight of sludge removed from the inventory will be procedurally authorized.

The Criticality Safety Evaluation, (Reference 1 1) demonstrates that encapsulation can be performed within the criticality prevention criteria established for the facility by the SAR (Reference 2) and OSR (Reference 29). Further, this analysis demonstrates that limiting the mass inventory permitted in the discharge-pickup chute area to the equivalent of six fully loaded canisters (723 lb/canister) allows for conservatively selected contingencies to occur while still satisfying the criticality prevention criteria with wide margins of safety. This approach is consistent with the generally conservative safety margins reflected in the facility SAR. Therefore, it is concluded that facility operating controls can be readily established that will ensure the fuel will remain subcritical during encapsulation operations and during postulated contingency events.

3.1 EVALUATION OF EXISTING SAR CRITICALITY SAFETY ENVELOPE

With respect to the criticality related accidents presented in the existing SAR, the effect that the encapsulation activities will have on each specific analysis is provided in the following evaluations. See Table 3.1 - Summary of Safety Evaluation Event Results.

3.1.1 Railroad Car Fuel Spill (SAR Section 8.1.1)

This postulated accident scenario assumes that the entire contents of three transport casks are spilled into an optimum configuration while transporting the irradiated fuel from the N-Reactor to the 105-KE Basin. Since one shipping cask can hold three canisters of fuel, it is conservatively assumed that the content of all nine canisters are spilled. It was determined that the amount of fuel carried in one well car is insufficient for a critical mass, even if the fuel were rearranged and optimally moderated and reflected (Reference 12).

The encapsulation process does not impact the Railroad Car Fuel Spill accident scenario as analyzed in Section 8.1.1 of the existing SAR.

3.1.2 Cask Fuel Spill (SAR Section 8.1.2)

The existing SAR concludes that spilling a cask will present no criticality problem, and on the basis that the Reference 12 letter determined that the minimum number of Mark IA spike fuel assemblies (1.25 weight percent outer element, 0.95 weight percent inner element) needed to form a hemispherical critical mass is 169.

During the encapsulation activities canister movement will be under OSR control with the Discharge-Chute Area considered as a bay. In addition, the encapsulation program will utilize standard size MK II canisters that can contain no more than 14 full length fuel elements in each two-cylinder canister assembly and the uranium mass in canisters containing scrap material will continue to be limited to 328 Kgs (723 lbs.). Finally, the Discharge-Pickup Chute Area fuel inventory will be limited to less than a critical mass of fuel material. Therefore, the amount of fuel that could be involved in accidents during the encapsulation process will be insufficient for a critical mass. As such, the encapsulation program does not adversely affect any of the assumptions used in the Cask Fuel Spill and the Cask Fuel Spill accident as described in Section 8.1.2 remains bounding.

3.1.3 Canister Drop) (SAR Section 8.1.3)

This accident analysis as presented in the existing SAR, assumed that the fuel is stored in the one-over-three suspended fuel storage configuration (this capacity enhancement storage configuration is no longer permitted by facility specific procedures). The existing SAR concludes that dropping multiple canisters containing spike fuel (worse case consideration, because the 1.25 weight percent Uranium spike fuel is only in the 105-KW Basin) from the one-over-three storage configuration onto a full density storage array of spike fuel would not raise the Keff

above 0.833. The fuel stored in the 105-KE Basin has a lower enrichment than the spike fuel in the 105-KW Basin.

The encapsulation program involves repackaging the existing fuel stored in open canisters into new stainless steel MK II canisters. Upon completion of the encapsulation activities the repackaged fuel canisters will be returned to a storage location in the 105-KE Basin. However, the storage configuration will remain the same after encapsulation. Furthermore, the existing SAR accident as analyzed in the Reference 14 nuclear criticality safety analysis assumed that the spent fuel was stored in open aluminum canisters. This criticality analysis determined that the use of steel canisters rather than aluminum canisters reduced the basin Keff. It therefore can be concluded that the assumptions in the Reference 14 analysis remains valid and that the Canister Drop analysis as described in Section 8.1.3 of the existing SAR remains bounding.

3.1.4 Broken Fuel Elements (SAR Section 8.1.4)

Section 8.1.4 of the existing SAR assumes that approximately five percent of the total fuel in storage is broken. The Reference 13 criticality analyses conservatively assumed that one percent of the fuel in storage corroded away and the Pu and U isotopes went into solution and remained in the water immediately surrounding the fuel elements. The assumption that one percent of the fuel in storage is dissolved and remains in the water immediately surrounding the fuel elements is conservative since the suspended and dissolved radionuclides have been continuously removed by filters and ion exchangers. It is currently estimated that approximately seven percent of the total fuel in storage is broken (Reference 4). Moreover, the Reference 16 criticality analyses, that was performed for the fuel segregation activities determined that even when it is assumed that ten percent of the fuel elements were completely dissolved criticality was not possible for such low enriched uranium solutions.

- With respect to particles that are removed by the filtration system, the filters will continue to be changed out in accordance with requirements specified in facility operating procedures (i.e., when the differential pressure across the filters indicates that the maximum filter loading has been reached or when the operational frequency limit is reached). In addition, the basin water and geometrically favorable components in the pool cooling and recirculation system will continue to be monitored periodically, as specified in the OSRS, for Pu to assure that a critical mass cannot be accumulated. Furthermore, the requirements specified in Safety Limit 2.2.1 and LCOs 3.7.1 and 3.7.2 (Reference 29) for the control of fissile material in various components of the cooling and recirculation system will be followed for the encapsulation program. Based on the above, it is concluded that Keff, will remain below 0.98 during encapsulation and that the Broken Fuel Elements analysis presented in Section 8.1.4 of the existing SAR and the References 16 and 17 criticality analysis is bounded.

3.1.5 Water Loss and Partial Moderation (SAR Section 8.1.51

As stated in Section 8.1.5 of the existing SAR, calculations in Reference 15 determined that the Keff value for a partially moderated storage array will not exceed that of a fully moderated storage array. Furthermore, the existing SAR concluded that for spike fuel, the Keff value for the full density array with one over three storage is within the Keff Safety Limit of 0.98 for all normal operation and credible single contingency accident situations.

As previously stated, the Reference 14 nuclear criticality safety analysis conservatively assumed that the spent fuel was stored in open aluminum canisters. This criticality analysis determined that the use of steel canisters rather than aluminum canisters reduced the basin Keff. The encapsulation process will replace the open aluminum canisters with new stainless steel canisters and the encapsulation process involves only lower enriched fuel that is presently stored in the basin. It therefore can be concluded that the assumptions in the Reference 14

analysis remain valid and that the accident as described in Section 8.1.5 of the existing SAR remains bounding for the planned encapsulation process.

3.1.6 Loss of Storage Array Geometry (SAR Section 8.1.kl

The Reference 14 criticality analysis determined that a one-over-three suspended fuel storage configuration is within the Keff Safety Limit of 0.98 for all normal operation and credible single contingency accident situations. Furthermore, the Reference 13 criticality analysis determined that for unirradiated fuel critical mass parameters the normal storage configuration and storage arrays perturbed due to a number of accidents would have Keff values significantly below a 0.98.

The encapsulation process involves repackaging lower enriched (0.95 weight percent and less) fuel that ranges in age from 4 to 20 years. The encapsulated fuel will then be returned to storage locations that are currently established in the 105-KE Basin. As such, the encapsulation process does not adversely affect any of the assumptions in the Reference 14 analysis and the Loss of Storage Array Geometry accident as described in Section 8.1.6 of the existing SAR remains bounding for the encapsulation process.

3.1.7 Well Car Drop) Into the Fuel Storage Basin (SAR Section 8.1.7)

Section 8.1.7 of the existing SAR states that the possibility of a well car accidentally dropping into a 105-KE Basin has been effectively eliminated. The existing SAR also concluded that criticality from such an event is prevented by fuel storage restrictions as discussed in Section 8.1.6 of the existing SAR.

Following the encapsulation process the encapsulated fuel will be returned to the 105-KE Basin and stored in accordance with present practices to ensure subcriticality. Therefore, it can be concluded that the encapsulation process will not have any impact on the Weil Car Drop Into the Fuel Storage Basin accident analysis described in Section 8.1.7 of the existing SAR.

3.1.8 Sludge Criticality (SAR Section .5.4

The 105-KE Basin is authorized for the storage of lower enriched (0.95 weight percent or less) fuel only. Since the highest enrichment of fuel in the 105-KE Basin is 0.95 weight percent, it is used as the maximum enrichment for any analysis pertaining to the sludge in the 105-KE Basin.

Accumulations of sludge consisting of finely divided 0.95 weight percent uranium material, i.e., homogeneous sludge, is not a criticality safety concern since reactivity calculations have shown that homogeneous sludge at this enrichment cannot become critical (References 16 and 17). Analysis of sludge containing pieces of fuel has also been made for the basin canister storage bays and shows criticality safety concerns do not exist until incredible sludge accumulations occur (Reference 37). The finely divided sludge which has originated from past fuel handling in the discharge chute pickup area and is present in the storage bay area from handling canisters with closed or coarse screened bottoms may not typify the sludge generated as a result of fuel handling in the discharge-pickup chute area during encapsulation. Therefore, discharge-pickup chute area criticality safety controls during encapsulation activities will be based on an even more conservative representation of the sludge and fuel (Reference 11). The entire discharge-pickup chute area fuel inventory will be assumed to consist of optimally sized and spaced rods, water moderation and reflection. The allowable discharge-pickup chute area fuel inventory during encapsulation will be limited to the equivalent of six fully loaded fuel canisters (723 lb/canister) or about 0.6 of the mass required to reach a keff of 0.98. When postulated, conservative, single contingencies were analyzed, their assumed occurrences were shown to still result in wide reactivity margins compared to the facility criticality prevention criteria for normal operations and the occurrence of single contingencies. Therefore the analysis described in the existing SAR Section 5.5.4 remains bounding.

3.2 EXISTING SAR NON-CRITICALITY RELATED ACCIDENTS

(See Table 3.1 - Summary of Safety Evaluation Event Results)

3.2.1 Loss of Monitoring Instrumentation (SAR Section 8.2.1)

As discussed in Section 5.3.3.4 of the existing SAR, instrumentation is provided for monitoring temperature, pressure, flow rate and water level for the components in the Basin Cooling Pool Recirculation System. In addition, radiation monitoring systems are provided that annunciates in both the facility and at another remote location. Operating limits for these systems and the surveillance requirements are specified in the OSRS. Monitoring instrumentation will be in operation as required during the encapsulation process. The encapsulation process does not impact any of the existing monitoring systems or the radiation monitoring instrumentation. Consequently, the encapsulation process does not impact the Loss of Monitoring Instrumentation accident as analyzed in Section 8.2,1 of the existing SAR.

3.2.2 Liquid Waste Car Accident (SAR Section 8.2,21

As discussed in Section 3.2.2 of the existing SAR, the Reference 18 engineering study evaluated credible accidental hazards that could cause leakage of the tank cars during transportation of low level liquid waste from the 100-K area to the 200 area for disposal. This evaluation concluded that although postulated accidents could cause a spill the occurrence of a spill is minimized by safe handling and operating procedures. Low level contaminated waste water will continue to be packaged and transported in accordance with the current established handling and operating procedures. The encapsulation process will not generate any additional low level liquid waste. Furthermore, the existing SAR accident analyses assumed the worst case liquid waste concentrations based on. the waste generated at the K Basins. The encapsulation process does not impact the Liquid Waste Car Accident as analyzed in Section 8.2.2 of the existing SAR.

3.2.3 Crane Failure and Cask Dropped Into the Fuel Loadout Pit (SAR Section 8.2.3)

This accident, as described in Section 8.2.3 of the existing SAR, postulates the dropping of a fully loaded transfer cask onto the floor of the transfer pit from a vertical distance of 25 feet. The existing SAR concluded that there would be a minimum environmental release from such an accident because of the underbasin collection membrane and tile field.

The encapsulation process does not involve the moving of fully loaded transfer casks. Therefore, the quantity of fuel involved in postulated accidents would be bounded by that assumed in the existing SAR accident scenario. All movement of fuel will be done under water such that any postulated drops during the encapsulation process would be from a height which is significantly less than that postulated in the existing SAR Section 8.2.3 analysis. Therefore, the encapsulation process does not impact the Crane Failure and Cask Drop into The Fuel Loadout Pit as analyzed in Section 8.2.3 of the existing SAR.

3.2.4 Crane Failure and Cask Dropped onto the Floor of Transfer Area (Maximum Credible Accident. SAR Section 8.2.4)

As stated in Sections 2.4 and 8.2.4 of the existing SAR, the maximum credible accident is the dropping and overturning of a loaded cask in the fuel transfer area resulting in a site boundary whole body dose of 7 E-06 mrem and a 1.5 E-03 Mrem dose to the organ (thyroid). The dropping of a cask containing three canisters (42 elements) of all broken fuel elements is expected to result in a site boundary whole body dose of 2 E-05 mrem and a 4.5 E-02 mrem dose to the organ (thyroid). The results of the accident analyses indicate that the radiation exposures to the general public will be very low. The radiological consequences are within 25 Rem whole body dose and 300 Rem to any specific organ guidelines established by the WHC-CM-4-46 criteria. However, since this analysis was originally performed, the condition of the fuel in the basin has changed.

Over the years the broken fuel segments have been subject to oxidation in the basin water environment. These oxidation products have considerable activity which could be released during a cask drop accident. Therefore, this postulated accident has been reanalyzed to account for oxidation product release, as well as the gaseous release assumed in the.original analysis (Reference 19).

The cask is assumed to have 3 canisters with a total of 42 broken fuel elements with the exposed surface corrosion involved in the release. The source terms were based on ten year old 12% Pu fuel which gave the worst results. The analysis assumed a time dependent release with an exposure time of 2 hours for on-site workers and 8 hours for the off-site public prior to evacuation.

The on-site worker' r dose was calculated to be 0.22 Rem EDE and 4.0 Rem limiting organ. The off-site dose was calculated to be 0.29 Rem EDE and 5.4 Rem limiting organ. Furthermore, this accident is judged to be extremely unlikely "not expected during plant lifetime" with a maximum frequency of 10-4 (Reference 2). From the Radiological Risk Acceptance Guidelines for EDE (Reference 10), the acceptable EDE on-site dose is 11.3 Rem and acceptable EDE off-site dose is 3.6 Rem. The acceptable limiting organ dose is ten times the corresponding EDE; therefore, the acceptable limiting organ on-site dose is 113 Rem and the acceptable limiting organ off-site dose is 36 Rem. Therefore, even though the radiological consequences from the postulated maximum credible accident scenario are higher than those previously calculated, they remain well within the established risk base dose acceptance criteria.

During the process of encapsulation, some accumulated sludge will be gathered and will be packaged in the new MK II canisters. Therefore, an additional analysis was performed for a cask drop with 3 canisters of sludge. Since the sludge is mobile, it is assumed that it will escape the ruptured canisters and begin to dry and release radioactive materials. The source terms are based on 1985 sludge data decayed for 6 years. As discussed above, the applicable acceptable EDE on-site dose is 11.3 Rem and acceptable off-site dose is 3.6 Rem (Reference 10). The

acceptable limiting organ dose is ten times the corresponding EDE; therefore, the acceptable limiting organ on-site dose is 113 Rem and the acceptable limiting organ off-site dose is 36 Rem. The onsite dose was calculated to be 0.25 Rem EDE and 4.7 Rem limiting organ, and the off-site dose was calculated to be 0.083 Rem EDE and 1.6 Rem limiting organ. Even though, the radiological consequences from the postulated accident scenario increased substantially as a result of the updated methodology, they remain well within the established frequency related dose acceptance criteria. In addition, they do not require other than a numerical change to the existing SAR and no change to the existing OSR or other operating documentation. Therefore, these updated dose consequences do not represent an unreviewed safety question.

As previously stated, the encapsulation process does not involve the moving of fully loaded transfer casks. In addition, all movement of fuel will be done under water and in accordance with established handling procedures. Therefore, the encapsulation process does not impact the Crane Failure and Cask Dropped onto the Floor of Transfer Area accident scenario as analyzed in Section 8.2.4 of the existing SAR.

3.2.5 Loaded Canister Dropped From a Lifting Hoist to the Pool Floor (SAR Section 8.2.5)

This analysis, as described in Section 8.2.5 of the existing SAR, assumes that the fully loaded canister is dropped onto the floor of the fuel storage basin from a height of 3.5 feet. The analysis concluded that the integrity of the bottom floor slab would be maintained and that no environmental release would result.

During encapsulation, the fuel canisters will be moved in accordance with the existing facility fuel handling procedures. These procedures maintain at least eight feet of water over the spent fuel during all normal operating modes. Furthermore, the monorail hoists are limited by an electrical uptravel switch and backed up by a mechanical system to prevent lifting the fuel elements out of the basin water. As

such, any postulated load drop during encapsulation would be bounded by the existing SAR Section 8.2.5 Loaded Canister Dropped From a Lifting Hoist to the Pool Floor accident.

3.2.6 Fire and Explosion Hazards (SAR Section 8.2,61

Section 8.2.6 of the existing SAR states that the facilities are of noncombustible construction and no additional combustible or explosive material are required for the operation of the facility. It is recognized that the encapsulation process will require an increased number of workers in the facility and will undoubtedly result in an increase in the amount of bagged trash from that area; however, current housekeeping procedures require the area to be kept free of combustible material. The encapsulation process does not introduce any new or different combustible or explosive materials into the basin. Furthermore, the encapsulation process does not impact any fire detection and suppression capabilities and the encapsulation process will be performed under water. Consequently, the encapsulation process does not impact the Fire and Explosion Hazards accident scenarios as analyzed in the existing SAR.

3.4-.7 Loss of Pool Coolant (SAR Section 8.2.7)

The existing SAR determined this accident to be non-credible. The credible Loss of Pool Coolant Accident is described in Section 3.4.3 of this Safety Evaluation. The encapsulation process does not impact the Loss of Pool Coolant Accident scenario as analyzed in Section 8.2.7 of the existing SAR.

3.2.8 Uncontrolled Railroad Train (SAR Section 8.2.8)

The existing SAR analysis determined that the weight of one cask car would be well in excess of the amount necessary to damage the basin floor. Therefore, redundant and diverse features were installed to prevent a runaway train from entering the facility.

It is not anticipated that a railroad train will be in the immediate vicinity of the 105-KE Basin during the encapsulation process since no fuel is being shipped. Moreover, the protective features will be maintained in place to prevent an out of control train from entering the facility. Encapsulation does not impact any of the redundant and diverse features that are in place to prevent a runaway train from entering the facility. Consequently, the encapsulation process does not impact the Uncontrolled Railroad Train accident scenario as analyzed in the existing SAR.

3.3 EXISTING SAR EXTERNAL EVENTS

(See Table 3.1 - Summary of Safety Evaluation Event Results)

3.3.1 Loss of Off-Site Power (SAR Section 5.3.3)

The accident scenario for the existing SAR analysis for Loss of Off-Site Power is a Loss of Pool Cooling. A Loss of Pool Cooling for the existing SAR analysis and the encapsulation process is described below.

Relative to encapsulation the impact of a Loss of Off-Site Power was evaluated for the encapsulation operations (Reference 22). Based on this evaluation, it was concluded that the loss of power will not result in any accident conditions. Specifically, the encapsulation operations supported by off-site power will cease in the event of a Loss of Off-Site Power and no adverse safety conditions will result.

Therefore, the existing SAR analysis continues to be valid and bounds the encapsulation process. Furthermore, the Reference 23 procedure addresses monitoring during a loss of power and will be used for an extended power outage to monitor vital parameters during the event.

3.3.2 Loss of Pool Cooling (SAR Section 5.3.31

Current cooling is provided to the 105-KE Basin by the chiller units and/or by natural phenomena. As noted in the existing SAR analysis, cooling is provided to maintain the basin water temperature between 42 and 90°F during normal operation to preclude adverse effects on the concrete structure and to help control radionuclide release to the atmosphere. The existing SAR assumes a loss of cooling with a maximum heat load in the pool and takes no credit for natural phenomena. Postulating this worse case scenario, Reference 21 has shown that with an initial temperature of 100OF and using a maximum heat load of 1.45 MW the resulting heat up rate would be 0.5@F/Hr. This would result in a maximum temperature of 130OF after 60 hours. Given this amount of time, fuel pool cooling could be restored. However, with existing heat generation loads an evaluation has been performed (Reference 24) to demonstrate that the maximum fuel pool temperature of 130OF is not exceeded when only taking credit for natural phenomena. Therefore, a Loss of Pool Cooling accident has no significant consequences and the assumptions used in the existing SAR analysis remain valid.

Furthermore, the encapsulation process has no impact on the loss of Fuel Pooling Cooling accident scenario as analyzed in the existing SAR.

3.3.3 Design Basis Earthquake (SAR Section 2.5)

The design basis earthquake (DBE) is defined in the existing SAR as an event producing a maximum horizontal acceleration of 0.25g simultaneously with a vertical acceleration of 0.17g.

The Basin cooling pool, the cooling pool water supply and the cooling pool recirculation equipment were all designed to withstand a DBE. The accident analysis (Reference 25) has shown that if the water level of 16'+8" is maintained in the fuel pool there will be no loss of structural integrity to the fuel pool, storage racks, cooling pool water supply or recirculation equipment. A subsequent analysis

re-evaluated the discharge chute wall and determined its acceptability for normal and overfilled water conditions (Reference 43).

As documented in Reference 42, an updated and more detailed seismic analysis was performed. As a result of this analysis, it was identified that cracking at the corners of the basin at approximately the 12 foot level will occur resulting in minor leakage. The loss of pool inventory is discussed in Section 3.4.3.

The new seismic analysis (Reference 42) identified a seismic concern relative to the basin joint between the bottom of the basin and the old KE reactor building structure. This item is further discussed in Section 3.6.9 of this report.

The encapsulation equipment is not safety related and is not required to mitigate any postulated accident, therefore seismic considerations were not required in the initial design (Reference 20).

Should a DBE occur during the encapsulation process it would not create any additional hazards. Therefore, the existing SAR analysis remains bounded such that no increased radiological consequences will result from the encapsulation process during a DBE.

3.3.4 Tornado (SAR Section 2.6)

As noted in the existing SAR, tornado loadings were not included in the original design basis. However, the steel building which forms the walls and roof over the fuel storage basin was designed to resist lateral forces caused by a bomb blast. These forces are more severe than those caused by a tornado (Reference 25). Therefore, although a tornado may result in some peripheral damage to the facility its structural integrity will be maintained.

The existing LCO 3.6.1 requires a monthly inspection to determine that there are no objects in the area (as shown on the diagram in the LCO) which could

potentially become missiles. Therefore, tornado generated missiles are not required to be considered as credible events.

The encapsulation process does not impact the Tornado event as analyzed in Section 2.6 of the existing SAR.

3.3.5 Floods (Probable Maximum Flood - PMF. SAR Section 4.3)

The PMF as defined in the existing SAR is 423 feet above mean sea level (MSL) as determined in Section 2.4.3 of the existing SAR. All of the equipment associated with the facility are located above this level with the exception of the two raw water pumps which are located at 421 MSL. All realistic modes of upstream dam failure cause a level below the PMF.

The only impact the PMF would have on the facility would be a loss of raw water which supplies cooling to the basin. As noted in Section 3.3.2 of this report, cooling can be maintained by natural phenomena. It therefore can be concluded the encapsulation process has no impact on the Floods accident scenario as analyzed in Section 4.3 of the existing SAR.

3.3.6 Ashfall (SAR Section 4.6)

The K Basin is located in a region where volcanic activity is of concern when discussing the safe operation of the facility. Volcanoes of concern include Mt. Adams-located approximately 100 miles from the site, Mt. Rainier-located approximately 110 miles from the site, and Mt. St. Helens-located approximately 130 miles from the site. The most violent eruptions of volcanoes in the area can be expected to exhibit the same type of actions as the Mt. St. Helens eruption of May 18, 1980. Although mudflows, landslides, earthquakes, ground deformation and flooding can all be attributed to volcanic eruptions, ashfall is a major concern due to the proximity of the site to the volcanoes.- The recent eruption at Mt. St. Helens on May 18, 1980 resulted in approximately a 1/2 inch of ash accumulation

on the site. However, history has shown that the eruption of Mt. Mazama, approximately 6000 years ago, resulted in an ash accumulation of up to six inches.

As a result of the 1980 Mt. St. Helens eruption, the site design criteria were modified to include ashfall. The 105-KW roof was replaced in 1988 because of concern for degradation of original roof panels at that facility. The 105-KW roof has been analyzed and will survive the limiting live load combination consisting of the design basis snow and ashfall loads of the current design criteria.

Deterioration of the 105-KE roof structure has not occurred. The roof structure remains as originally constructed. Since the roof structure has not been modified or replaced, the original Uniform Building Code Criteria involving live snow and seismic loads are still applicable.

The encapsulation process does not impact the Ashfall event as analyzed in the existing SAR.

3.4 Radiological Events Identified by the PHA Potentially Not Bounded by the Existing SAR

A PHA was conducted on the 105-KE Basin (Reference 3). The hazard screening technique used was a Hazard and Operability Study (HAZOP). This study identified numerous potential hazards. A complete listing of the hazards along with the HAZOP is presented in Reference 3. Not all of these hazards are considered significant. The criteria for selecting HAZOP items for this Safety Evaluation were based on consequences only. With the exception of a potential criticality case (sludge) only those items ranked S2.or S3 were considered significant enough to warrant detailed evaluations. S2 was defined in the PHA as a potential for radiological or chemical dose consequences to the on-site worker in excess of WHC-CM-4-46 (Reference 10) risk acceptance limits for expected events (events with an annual frequency of 1.0). S3 is the potential radiological or chemical dose consequences to the off-site population in excess of WHC-CM-4-46 risk

acceptance limits for expected events (events with an annual frequency of 1.0). These S2 and S3 hazards have been identified in the HAZOP and are presented in Table 3.4-1 of this report.

Nineteen specific hazards have been screened for further analysis and disposition. Eighteen of these hazards have been placed into two categories: those having a potential radiological impact and those hazards having a potential chemical impact. Items 4, 5, 6, 7, 8, 9, 10, 11, 13, and 14 of Table 3.4-1 have a potential radiological impact and are addressed in Section 3.4 of this evaluation. Items 1, 2, 3, 15, 16, 17, 18, and 19 of Table 3.4-1 have a potential chemical impact and are addressed in Section 3.5 of this evaluation. The nineteenth hazard, Item 12 of Table 3.4-1 which pertains to OSR fissile mass limits, is addressed in Section 3.6.1 of this evaluation. See Table 3.1 - Summary of Safety Evaluation Event Results.

3.4.1 Low/No Flow of Service Water (Item 4 of Table 3.4-1)

A pipe break in the 105-KE building causes an overflow of the fuel storage basin water, with possible release of radioactivity to the environment.

Filtered service water is supplied to the K fuel storage facilities for secondary side cooling for the heat exchangers (these have been removed from service) and chillers, and for emergency makeup to the basin through an eight inch supply line. The distribution lines inside the facility are also eight inch. It is postulated that a pipe breaks in the 105-KE Building and results in a flow rate of 2000 gpm into the basin. A response time of eight hours to terminate the flow from the break was also conservatively assumed. It was estimated that overflow from the basin would result in a discharge of 90,900 gallons to the river via overflow to the 1908-K Outfall and the formation of a 5.4 acre shallow lake outside the KE storage facility (Reference 27).

The off-site consequences from the release to the river and airborne effluent release from the lake have been calculated to be 4.1 E-1 mrem/yr total off-site dose. The potential on-site dose due to airborne release from the lake was determined to be 3.4 mrem/yr. The results of the analyses indicate that the radiation exposures to the general public and to the on-site workers will be very low. The radiological consequences from the postulated scenario are well within the 100 mrem/yr whole body off-site dose and 500 mrem/yr for on-site workers limit established by the DOE (Reference 10).

As discussed in Section 5.3.3 of the existing SAR, there are several diverse means available to detect increased basin water inventory. Basin level instrumentation provides remote high level alarms which currently annunciate at both 105-K Basin facility control rooms and a remote location outside the facility. There are remote alarms for basin secondary cooling flow loss which annunciate at both 105-K Basin facility control rooms and a remote location outside the facility. There are several remote alarming gamma radiation monitors which could detect an increase in airborne activity (Reference 28). In addition, emergency response procedures are in place which provide the necessary instructions to isolate a pipe rupture. Therefore, this postulated event can be readily detected and approved corrective procedures are in place to mitigate this low probability event.

The encapsulation process does not affect the operability of any of the above instrumentation nor does encapsulation increase the demand on the service system such that any design limits a@re exceeded. Therefore, the encapsulation process will have no impact on the Low/No Flow of Service Water event.

3.4.2 Fuel Storage Basin Water Level Low (Item 5 of Table 3.4-1)

There are many low basin water level initiators that can be postulated, such as a malfunction in the makeup system or a basin leak plus evaporation loss that exceeds the makeup system capability,

Basin water level above 15' 4" ensures that sufficient water shielding exists to limit the above basin radiation dose rate to less than one mrem/hr and prevents basin wall contamination from becoming airborne. The increase in radiation doses from contamination and/or reduced water shielding would be of no significant safety consequence but would present ALARA (As Low As Reasonably Achievable) concerns.

The key to prevention of this event is detection of low water level and the capability to reestablish normal basin water level. There are several diverse means available to detect reduced basin water inventory. Basin level instrumentation provides remote low level alarms which annunciate at both 105-K Basin facility control rooms and a remote location outside the facility. There are remote alarms for basin secondary cooling flow loss and local instrumentation to monitor secondary cooling pump suction and discharge pressure, which could indicate loss of pump suction from reduced level in the basin. There are several remote alarming gamma radiation monitors which could detect reduced water shielding or an increase in airborne activity (Reference 28).

The encapsulation process does not effect the operability of any of the above instrumentation. Therefore, the encapsulation process does not impact the Fuel Storage Basin Water Level Low event.

3.4.3 Fuel Storage Basin Water Level Very Low or None (item 6 of Table 3.4-1)
Fuel Storage Basin Leakage - Catastrophic (item 9 of Table 3.4-1

Postulated events that could be severe enough to significantly degrade the basin's capability to contain water were examined. Consequences of such events could potentially be extreme due to heat-up of fuel, loss of shielding, and boiling of remaining water with resulting airborne releases of radionuclides.

A postulated event might be dropping of a major piece of equipment, such as a shipping cask or train, into the basin. However, Section 8.2 of the existing SAR

discusses the shipping cask drop and concludes that only minor water leakage would occur. The uncontrolled railroad train is an incredible event because of the design and procedural controls associated with basin operations.

As noted in Section 3.3.3, Design Basis Earthquake, of this report, the seismic event could result in minor cracking in the corners of the KE basin structure. This cracking is analyzed to occur only at approximately 12 feet above the basin floor level. Assuming that the basin drains down to this elevation there is still approximately eight feet of water above the fuel storage canisters which is consistent with the guidance provided in ANSI/ANS 57.7-1988 and the existing SAR analysis (Section 3.2.5 of this report).

As a result of a seismic event, failure of the water stop is also postulated. This failure could result in increased basin leakage, a reduction in the basin water level, and a potential uncovering of the fuel and sludge. This scenario is being evaluated as part of the long term existing SAR upgrade program. It should be noted that this scenario is not impacted by encapsulation.

Encapsulation does not involve equipment, that if dropped, would cause a basin leak in excess of previously. bounding analyses. Therefore, encapsulation process does not impact the Fuel Storage Basin Water Level Very Low or None event or the Fuel Storage Basin Leakage - Catastrophic event.

3.4.4 Fuel Storage Basin Temperature Very.High (Item 7 of Table 3.4-1)

As noted in Section 3.4.3 of this report, water level in the basin remains approximately 12 feet above the basin floor level for all credible accidents. Consequently, the temperature in the basin will not exceed the existing SAR analysis of 130"F as long as the ability to add water exist (Reference 24).

The effect of fuel heatup has been evaluated for current basin conditions and found not to be a significant concern for the current safety analysis envelope or

encapsulation. Therefore, the encapsulation process does not impact the Fuel Storage Basin Temperature Very High event.

3.4.5 Fuel Storage Basin Leakage (Item 8 of Table 3.4-1)

Environmental contamination could occur due to small leakage from the basin into the surrounding soil that goes undetected or unmitigated. There are two independent means to detect small basin leaks in effect at the 105-KE/KW facilities. A periodic (monthly) drawdown test is performed on the basin which measures the rate of evaporation plus leakage. By trending this data, a leak can be detected as small as 4 gpm and be repaired based on past experience. There are four groundwater wells located near the 105-KE building which are monitored quarterly for increase in activity. Water samples from these wells indicate stable to decreasing radionuclide activities since 1981, when the last known leak from 105-KE Basin was detected and repaired. Furthermore, considerable soil retention data indicate that radionuclide migration is limited and poses a minimal radiological threat to the Columbia River, approximately 2000 feet away.

Based on independent and diverse leak detection capability, and soil retention data, any small basin leak will be detected and repaired with insignificant environmental impact. Therefore, this event is considered bounded by the current safety analysis envelope. The encapsulation process does-not comprise any activities which would impair the ability to detect and repair leakage from the basin; therefore, the encapsulation process does not impact the Fuel Storage Basin Leakage event.

3.4.6 Contaminated Building Atmosphere (Item 10 of Table 3.4-1)

This potential accident could,be caused by loss of fuel cooling or activities that create large quantities of airborne material. As discussed in Section 3.3.2 of this report, the loss of basin cooling would not cause significant building airborne contamination because cooling can be maintained by natural phenomena.

Moreover, a complete loss of fuel coolant is not considered credible per Section 3.4.3 of this report and Section 8.2 of the existing SAR.

There are postulated accidents such as dropping a canister loaded with sludge, which could potentially create significant amounts of contamination. However, activities involving radioactive materials are performed at the bottom of the basin (except cask handling) with at least 8 feet of water shielding. Therefore, the water would prevent a rapid dispersion of contaminated material into the air. Furthermore, there are sufficient numbers of gamma monitors and air particulate activity monitors to detect and alarm for airborne contamination.

As part of the PHA a scenario associated with encapsulation process was identified which could result in an increase in airborne activity. This scenario involves the removal of a canister, assumed to be empty, from the basin cooling water for disposal. The canister is assumed to contain an unnoticed fuel assembly, an increasing dose rate is not detected by the assigned Health Physics Technician, and the control for the lifting hoist fails in the "on" position. The event terminates with the activation of the radiation alarm system and prompt personnel evacuation per procedural requirements and training. The accident represents an extreme scenario that involves malfunctioning of equipment safety features and more than one concurrent personnel error. Thus, the probability of this accident is very low (not expected to occur during the life of the facility with a maximum frequency of < 10-4 (Reference 3).

Because the fuel assembly involved in this postulated accident could be damaged, the off-site consequences could be as great as 1/42nd of the facility maximum credible accident discussed in Section 3.2.4 of this report, which involved 42 damaged fuel assemblies. The on-site dose would be 0.005 Rem EDE and 0.09 Rem limiting organ. The off-site dose would be .007 Rem EDE and 0.13 Rem limiting organ. The results are much less than non frequency related acceptance criteria of 0.5 Rem EDE and 5.0 Rem limiting organ for on-site, and 0.1 Rem EDE and 1.0 Rem limiting organ for off-site. Therefore, it is concluded that

encapsulation will not pose an undue risk to either on-site personnel or the public, and is bounded by the WHC-CM-4-46 (Reference 10) risk acceptance criteria.

3.4.7 Confinement of Gaseous Radiolysis Products (Item 11 of Table 3.4-1)

Radiolysis of water and contaminants in the sludge creates a gas mixture which could be explosive. The energetic reaction could eject radioactive material from the basin causing high airborne activity and personnel contamination.

Sludge is defined as oxidized uranium and metallic fuel fragments emanating from discharge damaged fuel which has accumulated on the storage basin floor. In areas where appreciable fuel handling has occurred, significant accumulations exist. Such accumulations are known to trap some free oxygen and hydrogen, i.e., the potential for hydration of uranium fragments is increased relative to storage configurations where oxidation products can escape. Such sludge accumulations have been moved by water lance and/or mechanical means at various times to support routine basin operations. At no time has any observable indication been noted that suggests a vigorous oxidation reaction of uranium hydride has occurred. Based on the experience to date, the PHA postulated event is extremely unlikely and thus not considered a credible event (Reference 30).

The PHA also identified a potential concern over accumulation of radiolytically produced gases within canisters. Pressure buildup within canisters was analyzed assuming plugging of the canister vent. A burn of hydrogen assumed to have displaced the nitrogen cover gas in the top portion of sealed canisters was also analyzed. Under static pressure buildup, both sealed MKI and MKII canisters will relieve pressure buildup via seal leakage. MKI canister lids are routinely removed for shipping by injection of 115 psig service air into the canister to displace the lid. Analyses and cold water pressure tests of the MKII design indicated seal leakage occurred at around 100 psig internal pressure. In the event of a hydrogen burn, internal pressures of 297 psig and 1600 psig were calculated for open and plugged vent tube cases, respectively. It is judged that these pressures will not result in

rupture or structural failure of the canister because seal leakage will result in pressure bleedoff once pressures of about 100 psig are attained. A rapid burn would be expected to excite the natural frequencies of the canister and result in pressure relief beginning at a lower pressure (Reference 32).

Although the encapsulation process will potentially create more sludge from handling fuel, the process will not produce any sludge with significantly different characteristics. Therefore, the encapsulation process is bounded by the existing safety analysis envelope.

3.4.8 Nitrogen Cylinder Rupture (Item 13 of Table 3.4-1)

As part of the PHA a concern was identified regarding the generation of a missile from stored Nitrogen bottles. An evaluation was performed on the location of the Nitrogen source and it was concluded that relocation of all compressed gas bottles would eliminate this hazard.

All compressed gas bottles will be relocated outside the facility to remove the possibility that a gas bottle missile could enter the basin (Reference 31). Therefore, on the basis that the gas bottles will be removed from the area and prohibited inside the facility, this event is considered to be non-credible. Consequently, the encapsulation process does not impact the Nitrogen Cylinder Rupture event.

3.4.9 Oxidation of Uranium Hydrides (Item 14 - Failure to Add Corrosion Inhibitor of Table 3.4-1)

A potential energetic reaction may occur when canisters of fuel are opened or the fuel processed, in the case where hydration of the uranium in the canisters has occurred. Failed fuel exposed to basin water oxidizes readily and uranium hydride may also be formed during the process of oxidation. The energetic reaction could

eject radioactive material from the basin causing high airborne activity and personnel contamination.

Although no analytical work has been performed that would preclude an energetic reaction of uranium hydride, the following observations can be made from the extensive experience at handling failed fuel over several years at 105-KE and. KW facilities (Reference 30):

1. Metallographic examination of failed fuel confirm limited hydride formation visible in the uranium matrix. Visual examination of failed fuel also identified the likely presence of visible hydride in the corrosion products of fuel stored and shipped in rupture cans.

2. Extensive underwater handling of fuel material suspected to contain some uranium hydrides has never resulted in observable effects from a uranium hydride oxidation reaction.

3. Hotcell examination of fuel cooled greater than 30 days did not result in any observable effects from uranium hydride oxidation reactions.

4. The percentage of uranium hydrides formed from various storage conditions would not be expected to change markedly with time and experience to date would appear to bound the range of hydride content that could be expected to occur. Thus, the likelihood of uranium hydride reactions during underwater fuel storage and handling activities would not be expected to change in the future based on experience to date.

5. A sealed MK II canister pressure test was performed based on a scoping calculation which predicted an internal pressure loading of up to 1600 psig as a result of a postulated hydrogen burn in a canister with a plugged vent. It is judged that this pressure will not result in rupture or structural failure of the canister because deformation of the cover lid and cover hold bar will permit

gradual leakage by the canister seal. The test results indicated that the canister seal began to leak at less than 105 psig (Reference 32).

Based on the experience to date, the PHA postulated event is extremely unlikely and thus not considered a credible event. Since encapsulation process will not involve fuel handling activities which are significantly different than those performed to date at 105-KE and 105-KW facilities, the encapsulation process does not impact the Oxidation of Uranium Hydrides event.

3.5 NON-RADIOLOGICAL (CHEMICAL) EVENTS IDENTIFIED BY THE PHA NOT
BOUNDED BY THE EXISTING SAR

The HAZOP identified Items 1, 2, 3, 15, 16, 17, 18, and 19 of Table 3.4-1 as having a potential chemical impact. The following evaluations address each of these potential hazards. See Table 3.1 - Summary of Safety Evaluation Event Results.

3.5.1 Release of Chlorine (Items 1, 2, and 3 of Table 3.4-1)

The PHA identified three potential events, Items 1, 2 and 3 in HAZOP summary Table 3.4-1, involving the release of chlorine from the KE water treatment facility. The first event involved an off-site release (at the Columbia River intake) of a chlorine water mixture backflowing from the water treatment system. Items 2 and 3 have been combined as one accident scenario - the rupturing of one chlorine cylinder. The overriding concern here is death or injury as a result of inhaling chlorine fumes. A complete analysis of chlorine releases is presented in ET-SOAR-HM-91-[illegible] (Reference 33). Only the scenarios and their consequences are presented in this evaluation.

Item 1 involves backflow of chlorine to the Columbia river. Chlorine is normally injected into the 105-KE Basin cooling water from the river at a rate of 12.5 lbs./hr

(300 lbs./day). The References 33 and 34 analyses postulates two possible chlorine release initiators:

1. A procedural failure to cease chlorine addition upon shutting down the river pumps such that chlorine would backflow and be released at the Columbia River KE Area pump station at the injection rate of 12.5 lbs./hr.

2. The instantaneous failure and release of the contents of a one ton chlorine cylinder as used for chlorination of raw river water at the KE water treatment facility. The consequences of this event bound those possible from either the damage due to dropping or a failure of a one ton chlorine cylinder as identified by the PHA.

The Immediately Dangerous to Life and Health (IDLH) value for chlorine is 30 ppm. Both of the chlorine release scenarios yield chlorine concentrations at the site boundary that are greater than the IDLH value. However, the dispersion analyses indicate that the instantaneous release of chlorine, Items 2 and 3 in Table 3.4-1, have a much greater impact than the chlorine backflow scenario. Site workers including encapsulation personnel are at greater risk due to their proximity to the point of release and due to the fact that the cloud can reach them within a few minutes.

The complete cylinder failure results in significant concentrations of chl4orine at the river in about ten minutes. A level of 30 ppm (IDLH value), for which exposures of less than thirty minutes are not felt to be harmful, extend almost three miles from the release point approximately 1.1 hours into the event. Levels of 20 ppm (ERPG-3), for exposures below which individuals exposed for up to one hour are not expected to receive health threatening effects, extend about 3.5 miles and peaks at this value approximately 1.4 hours after the event. Levels of 3 ppm (ERPG-2), for exposures below which nearly all individuals would come to no permanent harm after an one hour exposure, are predicted to extend slightly more than nine miles and peak at this level some 3.1 hours after the event.

The use of one ton chlorine cylinders at the Hanford Site water treatment facilities has been recently reviewed (Reference 35). This review concluded that the risks associated with one ton chlorine cylinders can be maintained at an acceptably low level by the conscientious application of Chlorine industry standards for facility design and construction, product handling, operations, and emergency preparedness measures (Reference 34).

The encapsulation process at 105-KE Basin has no impact on the frequency or consequences of the release. That is, it does not increase the risk of a chlorine accident. However the postulated chlorine tank rupture accident does put site workers at risk. It follows that 105-KE Basin encapsulation will have more people working in the area and therefore, more workers will be at risk from the postulated accident. The above results are sensitive to modeling assumptions. Relatively stable conditions with slight wind directions towards the river would be expected to produce greater spread along the river trough.

Higher wind speeds results in quicker spreading of the chlorine cloud but decreases the range at which the IDLH, ERPG-3 and ERPG-2 level occur. These results, nonetheless, emphasize the need for prompt emergency action in the event of a major release (Reference 34). Furthermore, protection of on and off-site personnel can be improved by implementing the following recommendations identified in Reference 34:

1 Train the 183-KE Building Operators on the importance of immediate notification to the 100 Area Shift Manager of any known or suspected large chlorine leak from a one ton cylinder or from failure of fitting or connections thereto.

2. Familiarize the 100 Area Shift Managers and 100 Area Facility Representatives (including on-call representatives) with the following considerations involving an unimpeded leak from a one ton chlorine cylinder (i.e., other than clearly a fitting leak).

   a. The results of the consequence analysis of a major chlorine leak and the general dependency of the consequences on meteorological conditions, particularly wind direction.

   b. The need to immediately evacuate 100 Area site personnel based on wind direction.

   c. The need to request the Event Notification Center or Patrol Operation Center (POC) implementation of public evacuations from the Columbia River and adjacent public lands and Grant County residents within 10 to 15 miles in the affected direction. The benefits of using helicopters and Grant County phone lists in these evacuations shall be emphasized in such communications.

3. Revise WHC-IP-0702, 100 Area Emergency Response Procedures, Procedure 4.0, Chlorine Release to reflect these upgrades.

4. Discuss these upgrades with the Northern Area Emergency Center staff, the POC staff, and the Event Notification staff. Emphasize to the Northern Area Emergency Staff the need to consider City of Richland notification of a chlorine release to the river. Notification directed by the Northern Area Emergency Center as judged sufficient due to the approximately eight hour transport time involved.

5. Determine the need for a Safety Evaluation of the 183-KE chlorination installation for conformance to chlorine industry standards and any follow on

upgrades recommended in lieu of the planned replacement of the 183-KE water treatment facility.

3.5.2 Release of Sulfuric Acid/Sodium Hydroxide (Items 15, 16 17, and 18 of Table 3.4-1)

Items 15, 16, 17, and 18 of HAZOP summary Table 3.4-1 pertain to sulfuric acid and/or Sodium Hydroxide release from the appropriate storage tanks. Potential accident scenarios include:

a. Release of sulfuric acid due to failure of the sulfuric acid storage tank drain,

b. Release of sulfuric acid and sodium hydroxide from a rupture of the storage tanks from corrosion or seismic event,

The rupture of the sulfuric acid and sodium hydroxide tanks results in the largest spill. Additionally, the rapid mixing of the majority of the contents of both tanks produces the largest amount of heat and generates the most steam/vapor. Therefore, the above sulfuric acid and sodium hydroxide accident scenarios are bounded by the seismic event scenario. Since the seismic event is the bounding scenario, an assessment of the tanks' seismic capacity and assessment of the mixing of sulfuric acid and sodium hydroxide was performed (Reference 36).

A two dimensional seismic evaluation was performed consistent with older vintage commercial nuclear plants. Only one horizontal component is considered active at one time. Further, no consideration of the vertical component was made.

The seismic evaluation determined that the tanks have very low seismic capability. The level of excitation that the tanks can withstand is far lower than the original site seismic requirement of 0.25g, and the current site seismic requirement of 0.2g. It is also noted that these seismic capacities represent maximum capacities since the vertical seismic component has not been considered, and the tanks may

experience "walking" as they rock and slide. It is further noted that the tanks will most likely experience large displacements prior to collapse. This will cause the loss of piping resulting in spillage at nozzle locations at seismic levels lower than those given above. It can be concluded that the tanks have essentially no seismic resistance.

An analysis of the mixing of sulfuric acid (H2SO4) and sodium hydroxide (NaOH) was also performed (Reference 36). The failure of the storage tanks would result in approximately 3000 gallons of sodium hydroxide solution and 3000 gallons of sulfuric acid solution spilling out and mixing together. The mixture of sulfuric acid and sodium hydroxide is an exothermic reaction and would therefore result in some boiling of the water. The majority of the vapor produced is expected to be water and some sulfuric acid. Very little of the sodium hydroxide is expected to vaporize due to the high boiling point of sodium hydroxide, the relative high reactivity between sulfuric acid and sodium hydroxide, and the excess amount of sulfuric acid present. However, some sodium hydroxide solution as well as sulfuric acid may be entrained in the steam. These entrained liquids are conservatively estimated to travel as far as 100 feet. The consequences of a release of sulfuric acid and sodium hydroxide are fairly localized and include:

- Localized health hazard due to potential severe eye and skin burns from vaporized sulfuric acid and entrained sulfuric acid and sodium hydroxide droplets.

- Localized environmental insult from absorption of sodium hydroxide and sulfuric acid.

- Potential for increased corrosion of 1706-KE Building from sulfuric acid.

The encapsulation process at 105-KE Basin will have no impact on the frequency or consequences of the release of sulfuric acid and sodium hydroxide. Encapsulation does not increase the risk of a sulfuric acid and sodium hydroxide

accident. However, the postulated rupture accident does put site workers at risk and since 105-KE Basin encapsulation activities will require more people working in the area, more workers will be at risk. Therefore, in order to improve the protection of both on and off-site personnel it is recommended (long term) that either the sulfuric acid tank or the sodium hydroxide tank be moved to a different location. This recommendation is based on the guidance provided in the Fire Protection Guide to Hazardous Materials, 10th Edition by the National Fire Protection Association, that sodium hydroxide be stored separately from acids. Implementing this recommendation will reduce the consequences of a potential accident because the majority of fumes produced are due to the heat generated when sodium hydroxide and sulfuric acid combine. Finally it is recommended that a dike be installed around both the sodium hydroxide tank and the sulfuric acid tank so as to entirely contain the release to a postulated spill.

3.5.3 Sulfuric Acid or Sodium Hydroxide Solution Sent Directly to Basin (Item 19 of Table 3.4-1)

Item 19 of HAZOP summary Table 3.4-1 pertains to a sulfuric acid and/or sodium hydroxide release from the appropriate storage tanks to the basin or pipe trench. Although, a postulated sulfuric acid and/or sodium hydroxide release to the pipe trench is bounded by sulfuric acid and sodium hydroxide mixing scenarios evaluated in Section 3.5.2, the impact on the basin water quality due to a pH change from the sulfuric acid or sodium hydroxide must be evaluated.

If sulfuric acid or sodium hydroxide is added through the pipe trench, all the liquid will not enter the basin. Therefore, adding sulfuric acid and/or sodium hydroxide directly is the bounding case since no loss of liquid occurs. The evaluation (Reference 36) has determined that adding 3000 gallons of 96% sulfuric acid to the 1.2 million gallons of water in the basin would change the pH of the water from 7.0 to approximately 1. Adding 3000 gallons of 50% sodium hydroxide solution to the 1.2 million gallons of water in the basin would change the water pH from 7.0 to 12.7. The addition of the acid or the base after the other would have

approximately the same effects as mixing the two directly (See Section 3.5.2). The final pH of the basin water would be 1.4. Therefore it is concluded that the postulated release of sulfuric acid or sodium hydroxide into the basin can have a significant effect on the pH of the water in the basin and could result in the increased corrosion of components within the 105-KE basin. Because the potential does exist for sulfuric acid or sodium hydroxide to be released directly into the basin, it is recommended that an Emergency Response Guide be written to address this issue. However, this is not a prerequisite for encapsulation. As stated in Section 3.5.2, the encapsulation process at 105-KE Basin will have no impact on the frequency or consequences of the release of sulfuric acid and sodium hydroxide.

3.6 ADDITIONAL EVALUATIONS

The events and/or facility conditions included in Section 3.6 were identified from various sources and were not included in Sections 3.1 through 3.5. Included are one HAZOP finding related to criticality Tiger Team findings, issues raised during the independent OSR review (Reference 5) and some additional conditions known to exist at the 105-KE Basin that require evaluation. See Table 3.1 - Summary of Safety Evaluation Event Results.

3.6.1 OSR Fissile Mass Limits (Item 12 - Addition of Sludge to Open Canisters. of Table 3.4-1)

During the encapsulation process, canisters loaded with sludge may be transported to storage above existing canisters in the storage array. Since some of the canisters stored at 105-KE Basin are open canisters, it is postulated that a canister of sludge could be dropped and result in sludge being added to an open canister, thereby exceeding the fuel canister mass limit.

An analysis was performed using a more realistic yet conservative characterization of sludge (Reference 37). The analysis showed that any sludge spilled into open

canisters would make canisters filled with fuel less reactive. Therefore the postulated accident would not cause criticality concerns and therefore is bounded by the existing SAR analysis.

3.6.2 0.98 vs 0.95 (Tiger Team)

The use of limiting Keff value of 0.98 for less than 1.25 weight percent enriched N-Reactor fuel criticality safety criteria was noted by the Tiger Team. Based on the Tiger Team concern, the use of this value was reexamined and found to be valid for fuel fabrication and basin storage criticality safety analyses, (Reference 41).

WHC-CM-4-29, Nuclear Criticality Safety Manual, permits the use of a limiting subcritical Keff value of 0.98 for spheres and cylinders if reliable experimental data are available for closely similar systems and adequate calculational techniques exist for relatively small extrapolations of the data. The reference review noted that the geometry requirements are met. Experimental data is available, extrapolations are slight and validated computational techniques are available for such extrapolations. The review further noted that the double contingency criteria is applied to handling and storage limits and double batching is unlikely because of the well defined storage and handling units. Finally, the review noted that for the N-Reactor fuel, the margins of safety in terms of mass are greater than those available in higher enriched LWR fuel systems at which the industry standard of 0.95 Keff is directed. The WHC response to the Tiger Team finding therefore was that the Keff value of 0.98 was justifiable and appropriate.

3.6.3 DBE Basin Water Level (OSR Review, Reference 5)

The 5.5 ft thick pickup wall between the reactor discharge-pickup chute and the main basin at 105-KE Basin is a free standing wall in the basin that has been analyzed for stability during a DBE.

A 1975 analysis concluded that the wall was under reinforced so that, for a water depth of 16 ft., the safety margin against overturning during the DBE was about 1.07. For a water depth of 19.5 ft, the overturning moment was somewhat greater than the resisting moment with a safety factor of about 0.85. The analysis was performed assuming that the pickup area water level was 0 ft. It was concluded that the maximum basin water depth should be limited to approximately 16 ft.

The existing SAR Section 2.5 concluded that the facility would survive a DBE with the basin water depth limited to 16.5 ft; however, LCO 3.2.1 (Reference 29) specifies a basin water level range of 16'+ 8" above the basin floor. At a maximum water depth of 16' 8" there is little or no margin for the pickup wall during a DBE.

In 1990 additional analysis (Reference 42) was performed to determine the structural adequacy of the pickup wall using the latest DOE design criteria SDC 4.1. This analysis concluded that with water on both sides of the wall and no seismic restraints installed that a safety margin of 1.17 exists at a basin level of 19.5 ft. This margin will increase with lower water levels in the basin.

Therefore, the pickup wall is safe during a DBE provided that the water level on both sides of the wall are the same. Also the 1990 analysis concluded that the wall would be safe for any water level combination provided that the specified seismic restraints were installed. Therefore, administrative controls must be maintained to prevent the pickup chute area from being drained down until the seismic restraints can be installed or the seismic adequacy of this wail with water on one side only is otherwise resolved (Reference 42).

The process of encapsulation will not require this area to be drained down. As such, the above conclusions for DBE remains bounding for the encapsulation process.

3.6.4 Spent Ion Exchange Shipment for Burial (Tiger Team)

The encapsulation process will potentially increase the rate at which ion exchanger modules will be required to be replaced. The Tiger Team finding OP.5-1 concerns the inability to ship resin modules because the shipping cask no longer meets current DOE standards. This does not pose a direct problem with the encapsulation program since the Ion Exchange modules are not located in the fuel pool nor are they stored in the fuel pool while awaiting transport for permanent disposition. Since the encapsulation process involves increased Ion Exchanger activity, a release of ion exchange medium was analyzed (Reference 19) to address the consequences of dropping a resin module.

Reference 44 addressing the solutions to this problem determined that the best solution was to work with packaging development to revise the Safety Analysis Report for Packaging (SARP) which would support the continued use of the existing cask, "Big Bertha", for resin shipment. Since this issue deals with the need for proper documentation relative to the SARP, the encapsulation process does not impact the Spent Ion Exchange Shipment for Burial event.

3.6.5 Basin Leakage Monitoring (T;@er Team)

At the 105-KE and KW fuel storage facilities, no provision exists for continuous redundant monitoring of basin water for leakage. The current capability for detecting small basin leaks is discussed in Section of 3.4.5 of this report. The two methods are periodic drawdown tests and four groundwater test wells which are monitored quarterly to detect increased activity from a basin leak. Based on studies from known leaks in the 105-KE Basin, the resulting activity becomes fixed in the soil very near the basin. This soil contamination could increase the scope of future decommissioning projects at the 105-KE Basin; however, there has been no personnel exposure problem or off-site environmental problems to date.

The intent of basin leakage monitoring is to ensure that a leak can be detected and corrective action initiated in time to prevent unacceptable contamination of the environment or unacceptable individual exposure. The current basin leak detection systems are independent and sufficient to detect basin leaks before significant impact on the environment or personnel occurs. However, because of the delay between basin leak initiation and detection an additional basin leak detection system is being developed for installation at the 105-KE Basin (Reference 45). This system would consist of two separate D-Sump level monitoring instruments. D-Sump is a collection point that will detect main basin leakage almost immediately.

The encapsulation process does not involve any activities which would result in the possibility of increased leakage or impair the ability to detect and repair leakage from the basin for current or proposed leak detection methods. Therefore, the installation of the additional basin leak detection system is not considered a prerequisite for the encapsulation process, and the encapsulation process does not impact the Basin Leakage Monitoring issue.

3.6.6 Release of Ion Exchange Medium (Existing condition)

It is postulated that a spent ion exchange module or two spent ion exchange columns is dropped from a crane while being transported for shipment to burial. It is assumed that one percent of the radioactive inventory is released from the resin as a result of the spill. The source terms used for this analysis were equal to the maximum amount of Cs-137, Sr-90, and Y-90 recorded for each type of ion exchange unit. Analyses were performed using the GENII computer code for the maximum on-site (3.2 km NE) and maximum off-site (9.4 km NNW) receptors using an acute release in the 100 area (Reference 43). A finite plume integration was used for submersion doses. Ingestion pathway doses were also calculated for the maximum off-site receptor (Reference 43). The on-site and off-site doses were calculated as effective dose equivalents (EDE) and limiting organ doses. Inhalation and submersion doses are 50 year committed doses based on acute uptake.

The maximum on-site dose was 2.9 E-03 Rem EDE and 2.8 E-02 Rem limiting organ. The maximum off-site dose was 9.2 E-04 Rem EDE and 8.6 E-03 Rem limiting organ. The calculated doses for this unlikely event are less than one percent of the allowable of 0.5 Rem EDE and 5.0 Rem limiting organ for on-site personnel and 0.1 Rem EDE and 1.0 Rem limiting organ for the off-site public applicable to an annual probability of occurrence of one (Reference 10). Therefore, it is concluded that encapsulation will not pose an undue risk to either on-site personnel or the off-site public.

Although the encapsulation process will potentially increase the rate that ion exchange modules will have to be replaced, the increased risk to on-site workers and the public is small based on the low consequences for this accident and is bounded by WHC acceptance criteria (Reference 10).

3.6.7 Clearwell Integrity During DBE (Existing Condition)

Raw water is pumped from the Columbia River, treated, and stored in underground concrete clearwells. These clearwells are the water supply for the basin makeup and cooling, fire protection, and various other facility activities. In 1975 an analysis was performed to verity the structural integrity of the basins during a DBE. However, over the years the concrete has degraded in the clearwells to the extent that basin structural integrity may not be maintained during a DBE (Reference 46).

With a full load of relatively fresh spent fuel from N-Reactor, the basin requirements for makeup and cooling were significant. Therefore, the loss of the clearwell could result in a loss of basin inventory and increased basin temperatures. However, the current heat load is such that cooling can be accomplished without makeup from the clearwells (See Section 3.3.2 of this report). Furthermore, the integrated upgrade plan for 105-KE Basin includes a new demineralized water plant that will eliminate the current need for the clearwells.

Since the encapsulation process will not increase the current requirements for basin makeup and cooling water, it can be concluded that the encapsulation process will not impact the Clearwell Integrity During a DBE event.

3.6.8 Ventilation stack. Failure (Existing Condition)

Section 2.5 of the existing SAR states that the K-Area ventilation stacks would sustain a DBE provided the upper 125 feet was removed, this was accomplished by Reference 47; however, an additional external event was evaluated for stack failure. If an airplane should hit the stack and it falls in the right direction, the upper part of the stack could impact the basin causing considerable damage and potential on-site and off-site radiological consequences.

An evaluation was performed (Reference 48) and it was concluded that the frequency of this type of event was less than 10-7 . The Non-Reactor Facility Safety Analysis Manual (Reference 10, 4.0 Table 4-3) defines frequency less than 10-" as an incredible event, for which no consequence analysis needs to be performed.

The encapsulation does not increase the probability of a ventilation stack failure, the encapsulation process will not impact the Ventilation Stack Failure event.

3.6.9 Unsupported Basin Construction Joint (Existing Condition)

In the 105-KE Basin, the construction joint in the bottom of the basin between the old KE reactor building structure and the basin has been determined by surveillance
to lack rebar or doweling across the joint. The integrity of this joint during a DBE has not been determined. An analysis was performed to determine the relative motion across the joint but the results were inconclusive due to the difficulty of modelling the elastomer water stop.

The postulated response to a DBE may be some leakage from the basin through the damaged water stop; however, a leakrate greater than the ability to makeup to the basin is not expected. The leakrate from the failure of a 17 sq. ft. area of the basin floor was estimated at 2 gpm (Reference 49). The leakrate was limited by the soil percolation rate assumed. A two inch gap in the joint sixty feet in length would provide approximately 10 sq. ft of area for leakage.

The construction joint is scheduled for repair in Fiscal Year 1996. Since the estimated frequency of a DBE is 5000 years and expected leakrate from the basin is limited, there is reasonable assurance that the unsupported construction joint will not pose a significant hazard to on-site personnel or the public prior to its repair.

The encapsulation process will have no impact on the integrity of this construction joint.

3.7 EQUIPMENT/SYSTEM EVALUATION

3.7.1 Basin Cooling and Cleanup System

As described in Section 5.3 of the existing SAR, the primary functions of the Basin Cooling Cleanup System are to:

a. Remove the decay heat generated by the irradiated fuel stored in the basin,

b. Maintain water quality and clarity, and

c. Control the concentration of soluble and particulate radioactive nuclides in the water to minimize personnel radiation exposure.

Since the encapsulation process will not introduce any new sources of irradiated fuel, the heat load requirements on the Basin Cooling Cleanup System will not be

affected by the encapsulation process. However, the encapsulation process could potentially increase the demand on the cooling cleanup system with respect to water quality and radioactive nuclides particulate concentrations. Therefore, an evaluation of the potential impact of the encapsulation process on the Basin Cooling Cleanup System was performed.

The major components of this system are described in Section 5.3.2 of the existing SAR and as listed below, will remain in normal operation as required during the encapsulation program.

- Cooling Pool Recirculation Pumps
- Cooling Pool Recirculation Filters
- Chillers
- Ion Exchange System
- Sand Filter System
- Liquid Waste Transfer Pump
- Cooling Pool Leakage Catch Tank Pumps
- Irradiated Fuel Cask Transfer Area Sump Pumps

In addition to the current capabilities of the Basin's Cooling and Cleanup System, the following enhancements will be implemented for the encapsulation process:

- The supplemental ion exchange flow that is currently available only in the transfer pit will be extended to carry any radionuclides released directly to the exchange columns.

- Spray nozzles heads and lance for submerged flushing of sludge from the work stations.

Radiation levels in the storage basin water are controlled by removing the radionuclides that are released to the water from corrosion of the bare irradiated

uranium in broken fuel. The majority of the radionuclide released is collected by the spent filtration and ion-exchange system components.

Although it is expected that the encapsulation process will result in increased levels of suspended and dissolved radionuclides in the storage basin water, the increased levels are expected to be temporary. The proposed activities for the encapsulation of all open canisters presently stored in the 105-KE Basin are similar to those that were required by the previously established s6gregation activities which were analyzed in Section 5.5.4 of the existing SAR. As was the case for the segregation activities, a small amount of suspended particulate will end up in the filters as a result of the encapsulation process. This will result in a temporary increase in the filter cartridge change -out or backwash of the sand filter frequency. However, the filters will continue to be changed out in accordance with requirements specified in facility operating procedures (i.e., when the differential pressure across the filters indicates that the maximum filter loading has been reached or when the operational frequency limit is reached). Furthermore, the requirements specified in the OSRs for the control of fissile material in various components of the cooling and recirculation system will be followed for the encapsulation process. The functional design criteria specified in References 52 and 53 will be maintained. The long term effects of the encapsulation process will be a significant reduction in the release of radioactive contaminants into the cooling water and consequently a reduction in the load on the ion exchange and filtering systems.

With respect to the clarity of the basin, a water clarification system will be installed in the Discharge-Pickup Chute Area to blow the lighter sludge that remains suspended in the water away from the work area in order to improve visibility. Although the criticality analysis that was performed for the fuel segregation activities (Reference 17) determined that fuel suspended in the repackaging area (assuming one out of ten fuel elements is completely dissolved) is not a criticality concern, suspended sludge in the basin water hindered segregation activities by reducing the visibility of the sorting table. Therefore, the experience

gained from the segregation activities has dictated the use of spray nozzles heads and lance for submerged flushing of sludge from the work stations to improve visibility.

3.7.2 Specialized Equipment

The dump table, packager, and all related equipment will be designed for normal load capacity. However, the equipment will not be seismically designed or qualified since their individual failure or their concurrent failures would not create an accident array with a Keff greater than 0.98. The functional design requirements for the equipment are defined in Reference 26. The specific requirements established for encapsulation are as follows:

1. The design of equipment for encapsulation of N Reactor spent fuel will be based on a maximum of 0.95 weight percent U-235 enrichment. It must resist radiation effects resulting from exposure to gamma radiation.

2. All criticality safety physical arrangements and procedural controls will adhere to the double contingency criterion that at least two unlikely, independent, and concurrent changes or contingencies must occur before criticality is possible.

3. Standard size MK II canisters that can contain no more than 14 full length (26.1 inch) fuel elements in each two-cylinder canister assembly will be used.

The encapsulation equipment will be located underwater and will be supported from the fuel storage basin floor structure. The initial installation of the equipment will be accomplished without draining the discharge pit.

Similar to the segregation equipment, the encapsulation equipment will use demineralized water as the hydraulic fluid and up to five percent ethylene glycol additive may be added to the water for lubrication. As stated in Section 5.5.4 of

the existing SAR for the segregation program, leakage from the hydraulic system into the basin coolant is expected. However, Section 5.5.4 of the existing SAR concluded that the effect of the ethylene glycol on the filters or demineralizer resins is considered negligible.

3.7.3 Spray Nozzle Heads and Lance

This system will be installed to blow the sludge and/or particles that are suspended in the water away from*the work area to improve the visibility. This system will be designed to operate in a recirculation mode and will be supplied with water from a clean location in the basin water.

3.7.4 Radiological Consequences

The Environmental Assessment of 105-KE and 105-KW Basin Fuel Encapsulation and Repackaging, 100-K Area, Hanford Site, Richland Washington (Reference 4) has calculated the total occupational radiation exposure for the encapsulation process to be 114.8 person-Rem. The Environmental Assessment (EA) then used a conversion factor of 1 person Rem = 4.0 x E-04 latent cancer fatalities (Section 4.1.3 of the EA) and determines that no latent cancer fatalities are expected over the workers lifetimes as a result of the encapsulation process.

With respect to non-occupation radiation exposure, the Environmental Assessment concludes that no significant increase in airborne releases is anticipated. Due to the time that has elapsed since the irradiated fuel was discharged from N Reactor, the only noble gas remaining in the inventory is Kr 85 . The small amount of noble gas vented continuously from the fuel that has lost its cladding integrity in the open canisters is below detection limits in basin exhaust air that is sampled using standard air sampling techniques.

The estimated off-site dose due to encapsulation of 3.4 E-4 mrem EDE is comparable to the off-site dose that occurred during the segregation activities. It

is a negligible contributor to the annual Hanford Site off-site dose limit of 1.0 E2 mRem EDE and is essentially the same as the measured airborne release from 105-E Basin of 3.2 E-4 mrem EDE in 1984 when segregation occurred. Although, as stated in Section 5.1.1 of the existing SAR, the facility design does not include means to control the release of radioactive gaseous effluents nor provide any holdup capacity for retention of gaseous effluents containing radioactive materials, the concentrations of radioactive gases in the effluents will be a small fraction of that allowed by established DOE limits. Furthermore, all radioactive gases released to the atmosphere via the building ventilation system will continue to be well below federal and state limits as defined in WHC-7-5, Environmental Manual (Reference 54), and the Department of Health notification requirement for releases will continue to be met.

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TABLE 3.1 - SUMMARY OF SAFETY EVALUATION EVENT RESULTS

SAFETY EVALUATION SECTION	EVENT (Reference)	IMPACT OF ENCAPSULATION PROCESS
3.1	Evaluation of Existing SAR Criticality Safety Envelope	
3.1.1	Railroad Car Fuel Spill (SAR Section 8.1.1)	No Impact
3.1.2	Cask Fuel Spill (SAR Section 8.1.2)	Bounded
3.1.3	Canister Drop (SAR Section 8.1.3)	Bounded
3.1.4	Broken Fuel Elements (SAR Section 8.1.4)	Bounded
3.1.5	Water Loss and Partial Moderation (SAR Section 8.1.5)	Bounded
3.1.6	Loss of Storage Array Geometry (SAR Section 8.1.6)	Bounded
3.1.7	Well Car Drop Into The Fuel Storage Basin (SAR Section 8.1.7)	No Impact
3.1.8	Sludge Criticality (SAR Section 5.5.4)	Bounded
3.2	Existing SAR Non-Criticality Related Accidents	
3.2.1	Loss of Monitoring Instrumentation (SAR Section 8.2.1)	No Impact
3.2.2	Liquid Waste Car Accidents (SAR Section 8.2.2)	No Impact

TABLE 3.1 - SUMMARY OF SAFETY EVALUATION EVENT RESULTS

SAFETY EVALUATION SECTION	EVENT (Reference)	IMPACT OF ENCAPSULATION PROCESS
3.2.3	Crane Failure and Cask Dropped Into the Fuel Loadout Pit (SAR Section 8.2.3)	No Impact
3.2.4	Crane Failure and Cask Dropped Onto the Floor of Transfer Area (Maximum Credible Accident SAR Section 8.2.4)	No Impact
3.2.5	Loaded Canister Dropped From a Lifting Hoist to the Pool Floor (SAR Section 8.2.5)	Bounded
3.2.6	Fire and Explosion Hazards (SAR Section 8.2.6)	No Impact
3.2.7	Loss of Pool Coolant (SAR Section 8.2.7)	No Impact
3.2.8	Uncontrolled Railroad Train (SAR Section 8.2.8)	No Impact
3.3	Existing SAR-External Events 3.3.1 Loss of Off-Site Power (SAR Section 5.3.3)	Bounded
3.3.2	Loss of Pool Cooling (SAR Section 5.3.3)	No Impact
3.3.3	Design Basis Earthquake (SAR Section 2.5)	Bounded
3.3.4	Tornado (SAR Section 2.6)	No Impact

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TABLE 3.1 - SUMMARY OF SAFETY EVALUATION EVENT RESULTS

SAFETY EVALUATION SECTION	EVENT (Reference)	IMPACT OF ENCAPSULATION PROCESS
3.3.5	Floods (Probable Maximum Flood - PMF, SAR Section 4-3)	No Impact
3.3.6	Ashfall (SAR Section 4.6)	No Impact
3.4	Radiological Events Identified by the PHA Potentially Not Bounded by the Existing SAR	No Impact
3.4.1	Low/No Flow of Service Water (Item 4 of Table 3.4-1)	No Impact
3.4.2	Fuel Storage Basin Water Level Low (Item 5 of Table 3.4-1)	No Impact
3.4.3	Fuel Storage Basin Water Level Very Low or None (Item 6 of Table 3.4-1)	No Impact
	Fuel Storage Basin Leakage - Catastrophic (Item 9 of Table 3.4-1)	No Impact
3.4.4	Fuel Storage Basin Temperature Very High (Item 7 of Table 3.4-1)	No Impact
3.4.5	Fuel Storage Basin Leakage (Item 8 of Table 3.4-1)	No Impact
3.4.6	Contaminated Building Atmosphere (Item 10 of Table 3.4-1)	Bounded

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TABLE 3.1 - SUMMARY OF SAFETY EVALUATION EVENT RESULTS

SAFETY EVALUATION SECTION	EVENT (Reference)	IMPACT OF ENCAPSULATION PROCESS
3.4.7	Confinement of Gaseous Radiolysis Products (item 11 of Table 3.4-1)	Bounded
3:4.8	Nitrogen Cylinder Rupture (Item 13 of Table 3.4 1)	No Impact
3.4.9	Oxidation of Uranium Hydrides (Item 14 - Failure to Add Corrosion Inhibitor of Table 3.4-1)	No Impact
3.5	Non-Radiological (Chemical) Events Identified by the PHA Not Bounded by the Existing SAR	
3.5.1	Release of Chlorine (Items 1, 2, and 3 of Table 3.4-1)	No Impact
3.5.2	Release of Sulfuric Acid/Sodium Hydroxide (items 15, 16, 17, and 18 of Table 3.4-1)	No Impact
3.5.3	Sulfuric Acid or Sodium Hydroxide Solution Sent Directly to Basin (Item 19 of Table 3.4-1)	No Impact
3.6	Additional Evaluations	
3.6.1	OSR Fissile Mass Limits (Item 12 - Addition of Sludge to Open Canisters of Table 3.4-1)	Bounded
3.6.2	Keff 0.98 vs 0.95 (Tiger Team)	Bounded

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TABLE 3.1 - SUMMARY OF SAFETY EVALUATION EVENT RESULTS

SAFETY EVALUATION SECTION	EVENT (Reference)	IMPACT OF ENCAPSULATION PROCESS
3.6.3	DBE Basin Water Level (OSR Review, Reference 5)	Bounded
3.6.4	Spent Ion Exchange Shipment for Burial (Tiger Team)	No Impact
3.6.5	Basin Leakage Monitoring (Tiger Team)	No Impact
3.6.6	Release of Ion Exchange Medium (Existing Condition)	Bounded
3.6.7	Clearwell Integrity During DBE (Existing Condition)	No Impact
3.6.8	Ventilation Stack Failure (Existing Condition)	No Impact
3.6.9	Unsupported Basin Construction joint (Existing Condition)	No Impact

NOTES

Bounded - The encapsulation process is bounded by the existing SAR or WHC acceptance criteria.

No Impact - The encapsulation process cannot in any way impact the postulated event.

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Table 3.4-1 Summary of S2, and S3 HAZOP Items

ITEM NUMBER	DEVIATION/PROCESS PARAMETER	DESCRIPTION
1	Release of Chlorine	Chlorine gas is released to atmosphere as a result of failure to shut off Chlorine flow before shutting down 181 KE River pump and service water flow. The quantity of Chlorine released has not been determined.
2	Release of Chlorine	Chlorine is released to atmosphere as a result of dropping Chlorine cylinder during transport or placement. This event is considered unlikely due to the DOT requirements for Chlorine cylinders being capable of withstanding handling accidents.
3	Release of Chlorine	Chlorine is released to atmosphere as a result of the rupture of a 2000 lb. Chlorine cylinder.
4	Low/No Flow of Service Water	Pipe break in the 105-building causing an overflow of the fuel storage basin water, with possible release of radioactivity to the environment.
5	Fuel Storage Basin Water Level Low	Low level in the basin caused by conditions such as evaporation with inability to add make-up, small to moderate leak without make-up, joint leak without make-up, or any other such events. Implies no normal corrective actions are taken. This condition has the potential for allowing basin wall contamination to become airborne with resultant airborne dose to personnel inside or outside the facility.
6	Fuel Storage Basin Water Level Very Low or None	Very Low/No level in basin caused by event severe enough to significantly degrade basin's capability to contain water. Consequences would be extreme due to heat-up of fuel, loss of shielding, and boiling of remaining water with resulting airborne releases of radionuclides.
7	Fuel Storage Basin Water Temperature Very High	Very high basin water temperature, at or near boiling. This results from very low water level. Consequences would be loss of shielding and airborne release of radionuclides. Some potential for fuel combustion if the basin water all evaporated.
8	Fuel Storage Basin Leakage	Environmental contamination occurs due to leakage from the basin that goes undetected or unmitigated.
9	Fuel Storage Basin Leakage	Catastrophic basin leakage resulting in low or no water level. Consequences would be loss of shielding and airborne release of radionuclides.

ITEM NUMBER	DEVIATION/PROCESS PARAMETER	DESCRIPTION
10	Contaminated Building Atmosphere	Significant contamination in the fuel storage basin atmosphere caused by loss of fuel cooling or activities that create large quantities of airborne material.
11	Confinement of Gaseous Radiolysis Products	Radiolysis of water and contaminants in the sludge creates a gas mixture that may be explosive. Consequence of an explosion have not been determined but may be significant. Characterization of the gas and its capability to support combustion has not been evaluated.
12	Addition of Sludge to Open Canisters	Dropping a loaded canister of sludge over an open canister of Fuel could result (if the sludge canister opened) in an addition of sludge to the fuel canister, thereby exceeding the fuel canister mass limit. This criticality case has not been specifically analyzed.
13	Nitrogen Cylinder Rupture	Nitrogen cylinder rupture could result in a projectile condition. The possibility of a high velocity cylinder entering the basin and disrupting fuel storage is small, but this event has not been evaluated to determine if this is a significant risk.
14	Failure to Add Corrosion Inhibitor	Formation of uranium hydrides which may result in energetic reactions when canisters are opened or fuel is reprocessed.
15	Release of Sulfuric Acid	Significant quantity of sulfuric acid is released to the river or environment due to failure of the storage tank drain that drains to a small open sump. Consequences are not fully understood and need further analysis. Possibility exists for significant personnel hazard as well as environmental insult.
16	Release of Sulfuric Acid	Significant quantity of sulfuric acid is released due to failure of the storage tank either from corrosion or seismic event. Consequences are not fully understood and need further analysis. Possibility exists for significant personnel hazard as well as environmental insult.
17	Release of Sulfuric Acid	Significant quantity of sulfuric acid is released due to leak through valve and operator error. Consequence is significant if hydroxide solution has previously spilled to the clearwell or pipe trench and concentrated acid is added. Large quantities of heat and steam could be produced.

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ITEM NUMBER	DEVIATION/PROCESS PARAMETER	DESCRIPTION
18	Release of Sodium Hydroxide	Significant quantity of sodium hydroxide is released due to valve leak through, human error or pipe leak. Consequence is significant if sulfuric acid previously spilled to clearwell or pipe trench. Potential exists for a violent reaction to occur upon mixing of these two materials with possible hazard to personnel and insult to the environment.
19	Sodium Hydroxide Solution Sent Directly to Clearwell	Significant quantity of sodium hydroxide sent to clearwell due to valving error or pipe leak. Consequence is significant impact on basin water quality due to low pH.

4.0 ASSESSMENT OF UNREVIEWED SAFETY QUESTION

The purpose of this section is to evaluate the encapsulation of the fuel stored in the 105-KE Basin to determine the potential for an Unreviewed Safety Question (USQ). The following USQ determination was based on the information provided in Section 3.0, Evaluation, and the definition and requirements delineated in Section 2.0, Regulatory Basis, of this report, and is consistent with the requirements of MRP 5.12.

1. Could encapsulation increase the probability of an accident previously evaluated in the existing safety analysis?

   Encapsulation of the fuel stored in the 105-KE Basin does not increase the probability of an accident previously evaluated in the existing SAR, which includes normal 105-KE fuel handling activities. Encapsulating all of the open canisters of fuel provides an additional confinement barrier and will significantly reduce any future releases of solid and dissolved radionuclides into the basin cooling water. Encapsulation will significantly reduce the buildup of sludge on the basin floor. Furthermore, the addition of a corrosion inhibitor and an inert gas seal in each new MK II canister is intended to minimize any future corrosion of the fuel. The dropping of a loaded canister from a lifting hoist is not anticipated during the encapsulation activities, and is bounded by the load drops assessed in existing SAR Sections 8.1.3, 8.2.3, 8.2.4 and 8.2.5.

   Although some particulate matter may be dumped out of the original fuel storage canisters during the encapsulation process, the criticality evaluation performed for the encapsulation program has determined that the accumulation of suspended Sludge in the discharge pickup chute is not a criticality safety concern. In addition, conservative criticality Safety Limits have been established to ensure that Keff remains below 0.98 for all encapsulation activities. The existing fuel handling equipment and procedures

will be used to move the fuel canisters to and from the encapsulation work stations. No interaction between the encapsulation work stations and canister storage racks is expected to occur. The encapsulation operations do not require modification to basin water level, temperature or pH. Therefore, the probability of a criticality event is not increased as a result of encapsulation activities. Based on the above, the probability of occurrence of accidents and malfunctions which could result in release of radioactivity is bounded by similar accidents and malfunctions evaluated in the existing SAR.

2. Could encapsulation increase the consequences of an accident previously evaluated in the safety analysis?

   All accidents previously described in the existing SAR have been evaluated. As noted in Table 3. 1, the consequences of the accidents are not increased due to encapsulation of the fuel stored in the 105-KE Basin. The new MK II canister will provide an additional confinement barrier between the encapsulated fuel and the basin water and thus will confine fission products that escape from damaged fuel cladding. The addition of an corrosion inhibitor will minimize the future corrosion of fuel and will therefore significantly reduce any future releases of solid and dissolved radionuclides into the basin cooling water. The encapsulation program does not adversely affect the ability of any safety system/component to mitigate the consequences of any accidents analyzed in the existing SAR. The most limiting accident which can occur would be.that of a dropped canister containing broken fuel elements. However, the consequences of the canister drop previously analyzed in section 8.1.3 of the existing SAR remain bounding. In addition, all criticality safety physical arrangements and procedural controls will adhere to the double contingency criterion that at least two unlikely, independent, and concurrent changes or contingencies must occur before criticality is possible. Criticality Safety Limits have been defined and will be implemented to ensure that the criticality analysis remains bounding.

The preceding evaluation has determined that postulated accidents involving encapsulation activities would be bounded by the results of existing SAR accident analyses. Therefore, accidents that could occur during the encapsulation program are bounded by the current design basis accident analyses, such that no increase in radiological consequences will result.

3. Could encapsulation increase the probability or occurrence of a malfunction of equipment important to safety, previously evaluated in the existing safety analysis?

   Encapsulating the fuel stored in the 105-KE Basin will not increase the probability of a malfunction of equipment important to safety, previously analyzed in the existing SAR. The proposed encapsulation equipment is not safety related, and all equipment necessary to mitigate any possible accident conditions during execution of the encapsulation operations would be functional. All original design criteria for equipment and systems important to safety continue to be met. The encapsulation activities do not place plant systems in configurations conducive to the occurrence of accidents or malfunctions not previously evaluated. The fuel will remain subcritical during credible accident situations. It is therefore concluded that the ability of any safety related component or system to perform their intended safety function is not adversely affected by the encapsulation activities.

4. Could encapsulation increase the consequences of the malfunction of equipment important to safety, previously evaluated in the existing safety analysis?

   Encapsulating the fuel stored in the 105-KE Basin will not increase the consequences of a malfunction of equipment important to safety previously evaluated in the existing SAR. The preceding Safety Evaluation has determined that encapsulation activities do not affect in-place safety systems.

As such, the consequences of the existing SAR analyses will not increase as a result of the encapsulation activities.

5. Could encapsulation create the possibility of an accident of a different type than any previously defined in the existing safety analysis?

   Encapsulating the fuel stored in the 105-KE Basin does not create the possibility of an accident which is different than any already evaluated in the existing SAR (see Table 3. 1). Many of the categories of accidents, or accident types, evaluated in the existing SAR are not possible during encapsulation. Since all irradiated fuel is already in the basin, accidents related to transporting fuel from N-Reactor to the basin, and spilling the entire inventory of a casks are not a concern during the encapsulation process. A dropped canister event is similar to the drop canister event described in the existing SAR. Furthermore, no new failure modes have been defined for any system or component important to safety nor has any new limiting single failure been identified.

6. Could encapsulation create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the existing safety analysis?

   Encapsulating the fuel stored in the 105-KE Basin will not create the possibility of a malfunction of equipment important to safety different than any already evaluated in the existing SAR. No safety systems are being modified. The original design intent and performance criteria continues to be met such that the ability of any safety related system to perform its intended safety function is not affected. There is no unanticipated malfunction of equipment expected that is not bounded by the existing SAR accident analyses.

7. Does encapsulation reduce the margin of safety as defined in the basis for any existing Operational Safety Requirement?

   The assembly and operation of the encapsulation equipment will not decrease the margin of safety as defined in the bases to any Operational Safety Requirement. All facility systems which will be relied upon during encapsulation are described in the existing SAR and are governed by the Operating Safety Requirements. The evaluation of the effect of the encapsulation has verified that the Keff Safety Limit of 0.98 is not exceeded and the conclusions presented in the existing SAR remain valid. As such, there is no reduction in the margin of safety as defined in the basis for any Operating Safety Requirements. In order to properly and safely administer the encapsulation process, criticality prevention specifications governing encapsulation will be provided in facility operating documentation to ensure that encapsulation is performed within the safety envelope defined by the safety analysis report and imposed by the OSRS.

   An underwater criticality is the maximum hypothetical accident associated with encapsulation. Analysis has shown a facility worker standing directly above such an event would receive the maximum consequence consisting of less than 1 mr exposure due to delayed gamma radiation from the event (Reference 55). Therefore, the criticality prevention specification may be applied at the facility operating documentation level as prescribed by the WHC administrative system (Reference 40) since OSR level control is not required by DOE Orders unless the public or workers can be exposed to a radioactive material release or radiation exposure from an inadvertent criticality.

   Based on the above discussion, it is determined that the encapsulation of the fuel stored in the 105-KE Basin does not represent an Unreviewed Safety Question.

5.0 CONCLUSION

The preceding Safety Evaluation has determined that it is safe to proceed with the encapsulation of the fuel stored in the 105-KE Basin, that the current safety envelope as defined in the existing SAR remains valid, and the existing OSRs will impose an adequate safety envelope on facility configuration and operation. Additional criticality prevention specifications will be established for encapsulation in facility operating documentation to ensure encapsulation activities are conducted in compliance with the OSR'S. The purpose of this section is to 1) recap the fundamental reasons why the encapsulation of the fuel stored in the 105-KE Basin is considered to be safe, and, 2) to identify the process controls which will be implemented during the encapsulation process to ensure that the existing safety envelope continues to be bounding.

The fundamental reasons why the encapsulation of the fuel stored in the 105-KE Basin is considered to be a safe operation are:

- The SAR adequately describes and evaluates encapsulation steps.
- The Keff safety limit of 0.98 will not be exceeded during the en-capsulation of the fuel stored in the 105-KE Basin (encapsulation).
- Neither system actuations nor accident mitigating capabilities are adversely affected by encapsulation.
- Encapsulation does not adversely affect the associated 105-KE Basin systems.
- Encapsulation does not require a change to the existing Operational Safety Requirements. It does, however, require the implementation of additional criticality prevention specifications in facility operating documentation.

- All safety analysis acceptance criteria continues to be met and all safety analysis assumptions remain valid.
- Encapsulation does not represent an Unreviewed Safety Question.
- Encapsulation will reduce the risks from potential accidents.
- Encapsulation will reduce the risk associated with passive fuel storage.
- Encapsulation will not exceed the bounds of the existing safety envelope as documented in the existing SAR.
- Encapsulation will not exceed bounds of the WHC acceptance criteria as defined in WHC-CM-4-46.

The additional facility documentation procedural controls which will be implemented during the encapsulation process to ensure that the existing safety envelope continues to be bounding include:

- Limit the total fuel mass in the discharge-pickup chute area during encapsulation to the equivalent of six fully loaded fuel canisters (723 lb/canister) or about 0.6 of the mass of optimally sized and water moderated rods in a hemispherical geometry required to achieve a keff of 0.98.
- Limit the number of fuel elements in a fuel canister to 14 (or the equivalent mass of 14 fuel elements). The SPR fuel limit of 248 Kgs (545 lbs.) per container remains unchanged.
- Limit the uranium mass in a single canister to 328 Kgs (723 lbs.).

With respect to Chlorine release:

1. Train the 183-KE Building Operators on the importance of immediate notification to the 100 Area Shift Manager of any known or suspected large chlorine leak from a one ton cylinder or from failure of fitting or connections thereto.

2. Familiarize the 100 Area Shift Managers and the 100 Area Facility Representatives (including on-call representatives) with the following consideration involving an unimpeded leak from a one ton chlorine cylinder (i.e., other than clearly a fitting leak):

   a. The results of the consequence analysis of a major chlorine leak and the general dependency of the consequences of meteorological conditions, particularly wind direction.

   b. The need to immediately evacuate the 100 Area site personnel based on wind direction.

   c. The need to request the Event Notification Center or Patrol Operation Center implementation of public evacuations from the Columbia River and adjacent public lands and Grant County residents within 10 to 15 miles in the affected direction. The benefits of using helicopters and the Grant County phone lists in these evacuations shall be emphasized in such communications.

3. Revise WHC-IP-0702, 100 Area Emergency Response Procedures, Procedure 4.0, Chlorine Release, to reflect these upgrades.

4. Discuss these upgrades with the Northern Area Emergency Center staff, the POC staff, and the Event Notification staff. Emphasize to the Northern Area Emergency Staff the need to consider City of Richland notification of a chlorine release to the river. Notification directed by the Northern Area

Emergency Center is judged sufficient due to the approximately eight hour transport time involved.

5. Determine the need for a Safety Evaluation of the 183-KE chlorination installation for conformance to chlorine industry standards and any follow on upgrades recommended in lieu of the planned replacement of the 183-KE water treatment facility.

In the process of performing this Safety Evaluation, several items were identified that did not impact the safety significance of the encapsulation process; however, it is recommended that these items be considered for incorporation into the long term K Basin existing upgrade effort. These include:

- Seismic qualification of the H2SO4 and NAOH tanks.
- Install dikes to contain potential spills around hazardous chemical tanks.
- Relocate NAOH away from facility.
- Write Emergency Response Guidelines to address NAOH or H2SO4 release into Basin.
- Installation of D-Sump level monitoring instrumentation.

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Safety Equipment List for K Area

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Safety Equipment List for K Area

Introduction:

This document was generated in an attempt to fill the previously identified need for a K Area safety equipment list.[1] This list is intended to provide a brief overview of only those systems at K Area that fall under the direct responsibility of K Area Management. A description of the major equipment components contained within each of the various K Area systems is provided. The K Area system number designators and their associated safety classes had been previously established.[2] This list does not identify all equipment in use at K area;only equipment that has some safety related issues or functions associated with it.

System #02 Annunciators Safety Class #3

The annunciators comprise an electrical system which provides an indication of the alarm status of specific basin equipment components. The annunciators also serve to alert basin operators to equipment malfunctions. This system includes the remote alarm summary for both basins.

System #05 Building Radiation Safety Class #3

The radiation monitoring system consists of all instrumentation used for monitoring the building radiation levels in the immediate area of the 105KE and 105KW fuel storage basins. This instrumentation consists of PIOPS dataloggers, building radiation detectors, continuous air monitors, egress portal monitors and fixed head samplers.

System #06 Building Structure

Building structures consist of all buildings and portions of buildings not specifically covered under the basin structure described in system #07.

Structural Items: Safety Class #3
Included within this classification are building roofs and all load bearing or structural components of the building.

Architectural Items: Safety Class #4
Architectural items consist of building components such as siding, doors windows, cabinets, partitions, floor coverings, trim etc. that do not specifically fall within the definition of a structural component or otherwise perform a structural function.

System #07 105K Fuel Storage Basin

This system covers all components directly involved with the storage and handling of spent reactor fuel within the fuel storage basin.

Basin Structure: Safety Class #1
In the context of system #07, the basin structure consists of the entire concrete pool area which is responsible for retaining the basin water.

Underwater Storage Racks: Safety Class #1
The underwater storage racks are steel frames situated upon the concrete floor at bottom of the fuel storage basin. Although the racks are not anchored to the basin floor, they are interlocked with each other at their top edges to form one solid unit across the floor. The rack units are butted tightly to the pool walls. The openings within the rack structure retain the fuel storage canisters which are placed into them. They are sized to retain only one fuel storage canister per opening.

Railcar Control System: Safety Class #1
The railcar control system consists of bumpers affixed to building structural columns inside of the basin along with derailing devices and rail switches located outside of the building. The derail devices and switches are located on the railroad track which allows railroad cars carrying fuel to enter and exit the basin building. The rail switches provide rail car position control;they are locked in place to direct the routing of the train. The railcar control system serves to prevent an accident that might otherwise allow a railcar to leave the tracks and fall into the basin.

Discharge Chute Isolation Doors: Safety Class #2
The discharge chute isolation doors are an original fixture of the basin. The doors are hinged to the basin wall and were designed to be closed in order to isolate the passageway that connects the basin proper to the discharge chute area. The doors were equipped with pneumatic seals that inflated to seal the edges of the doors against the surface of the basins concrete wall, thus helping to minimize water movement from the main basin into the discharge chute. With the underwater fuel storage racks in place it is not possible to open or close the doors without first removing the fuel storage rack sections that are situated in the swing path of the doors. The doors have not been operated for a number of years and it is unlikely that they could be moved without experiencing some damage. The rubber seals have deteriorated to the point where they could not be depended upon to seal the doors effectively. Any future isolation of the discharge chute area will be need to be performed using temporary doors rather than trying to employ the use of the existing doors.

Fuel Storage Canisters: Safety Class #3
The fuel storage canisters that hold the spent reactor fuel are made of either aluminum or stainless steel. The newer type of canisters seal the spent fuel within them from the basin water while the older type of canister is open to the basin water. The sealed canister variety acts as a barrier to separate the fuel which is inside of the canister from the basin water which surrounds the outside of the canister.

Fuel Handling Equipment: Safety Class #3
Fuel handling equipment includes all equipment and hardware, including the stiffbacks, used to transport and handle fuel within the basin. It does not include the hoists and trollies which are covered under System #14.

The fuel handling equipment also includes all of the equipment developed to support fuel encapsuation. These items are the dump table, packager, seal conveyer, canister crusher, canister cleaner, fuel handling tongs and the nitrogen dosing station.

Demineralized Water Supply System: Safety Class #3
A self contained system composed of commercial deionization unit which converts service water supply into demineralized water for fuel storage basin makeup water. The spent demineralizer cartridges utilized on the unit are periodically replaced by the vendor of the demineralizer.

Basin Recirculation System: Safety Class #3
The Basin Recirculation System is used to filter, cool and supply basin water to the basin cleanup system. The system consists of the following equipment:

Recirculation pumps: These pumps draw water from the basin and route it through the cartridge filters, chillers and ion exchange columns before returning it to the basin. Only one pump is operated at any one time.

Cartridge Filters: There are two cartridge filter units situated on the floor of the basin. Disposable, five micron capacity filter inserts are placed into the housing units and periodically removed when differential pressure readings indicate the filter is no longer functioning effectively.

Chillers: There are two different types of mechanical chillers used for basin water cooling. The older type rejects waste heat to a secondary water supply and the newer unit rejects waste heat to the outside air. Only one of the chillers is used at any particular time

Basin Cleanup System: Safety Class #3
The basin clean up system is used to both remove particulate matter and radionuclides from the basin water. It consists of the following equipment.

Ion Exchange Columns: These disposable columns consist of carbon steel tanks which contain within them the ion exchange resins used for treating the basin water. When the tanks are in service they are situated in concrete enclosures for radiation shielding. The water supply to the tanks is provided by the recirculation pump. Up to three of the columns can be in service at one time. When the resin within the tanks is depleted the tanks are removed and disposed of as a complete unit.

Skimmer Pump: The skimmer pump is a 400gpm pump single pumping unit which draws basin w[illegible] from the north side of the basin from locations near to the waters surface. The water from this pump is supplied to the sand filter and ion exchange modules.

Sand Filter: The sand filter is a steel tank which is filled with two different grades of sand and situated within a concrete enclosure for radiation shielding purposes. The sand filter acts as a prefilter for the Ion Exchange Modules by removing any debris from the water discharged from the skimmer pump. The sand filter is manufactured by the Baker Filtration Company and is a model 2 HRB-78.

Ion Exchange Modules (IXM): These modular units consist of six carbon steel tanks containing ion exchange resin for water treatment connected in parallel which are in turn encased within a block of concrete to form a single unit. The units are disposable like the ion exchange columns, but, unlike the ion columns, they have their own integral radiation shielding provided by the concrete that surrounds them. The water which is supplied to the IXM first passes through the sand filter.

Basin Instrumentation: Safety Class #3
The basin instrumentation system is used to monitor the status of critical operating parameters of the basin. The instrumentation currently in use consists of the following:

Basin Water Level Alarms: A device used for monitoring high and low basin water level trip points.

Basin Water Level Indicator: A device used for monitoring the level of water in the basin. The basin water level must be maintained within established limits in order to comply with Operational Safety Requirement

Basin Water pH Monitor: An electronic device for measuring the amount of H or OH ions in the basin water. Basin water pH must be maintained within established limits in order to comply with Operational Safety Requirements.

Basin Water Temperature Monitor: An electronic device for measuring the temperature of the basin water. Water temperature is an important parameter and must be maintained within established limits in order to comply with Operational Safety Requirements.

System #09 Chemical Handling System Safety Class #3

The chemical handling system contains all of the chemicals and equipment used for water treatment purposes. This equipment includes:

183KE Alum System:

Alum Storage Tank: The alum storage tank is a covered tank constructed of stainless steel which is located on the south side of the 183KE building. The lower portion of the tank is trace heated and insulated to protect the alum solution within it from freezing. The alum is used as a water treatment to remove suspended solids from raw river water.

Alum Transfer Pump: Located in the lower level of the 183KE building the pump transfers alum solution from the alum storage tank to a location where it is mixed with water and then injected into the raw water stream.

183KE Separan System:

The separan system is used to inject Separan, a coagulant aid, into the water entering the sand filters of the water treatment plant. The system consists of a storage tank, transfer pumps and flow meters.

183KE Chlorine System:

The chlorine system injects the chlorine that is stored in the standard ton size chlorine containers into both the raw water and also into the potable water. The basic equipment consists of the ton containers and the chlorine injection system that transfers the chlorine from the containers into the water streams. This includes all tubing and piping, valves, fittings and gauges containing liquid or gaseous chlorine.

165KW Hypochlorinator: The hypochlorinator unit is located in the lower level of 165KW building. It is used for injecting liquid bleach into the 100KW potable water distribution system

System #11 Communications Safety Class #3

This system consists entirely of a two way radio base station.

System #12 Compressed Air Safety Class #3

The main components of the compressed air system are air compressors and air storage tanks or receivers. The compressed air equipment is located in the following buildings.

Building 105KE:
The main components are a 7-1/2 Hp electric air compressor, a diesel compressor located outside, an air cooled aftercooler and one air receiver. These components supply a compressed air loop that passes through the basin area. The air is used for general shop uses and for operating some of the encapsulation equipment.

Building 105KW:
The main components are a 7-1/2 Hp electric air compressor and an air receiver. This supplies a compressed air loop that passes through the basin area. The air is used for general shop uses.

Building 183KE & 190KE:
The main components are a 5 Hp electric air compressor in 183KE and an air receiver in the tunnel. The air is supplied to the 183 KE and 190 KE buildings for water plant level instruments. The air is also used for general shop uses.

Various Buildings:
Compressed gasses are used for a variety of purposes including portal monitor gas supply. Gas bottles are typically located in racks on the outside of the buildings where they are used. These gasses are supplied in commercial gas cylinders.

System #14 Cranes & Hoists Safety Class #3

Cranes and hoists of various sizes are used throughout the K area facility for various purposes. An itemized listing of all cranes and hoists in use at K Area is included in Appendix A1 and A2

System #18 Drains Safety Class #3

This system contains all of the active drains at K Area; they include:

Outfall Flume: The outfall flume is the only permitted conduit for returning K area effluent water to the river. The outfall primarily serves to collect both the sand filter backwash water from the 183KE water treatment plant and the service water used for collecting waste heat from the water cooled chillers located in the fuel storage basins. The drains from the 1706KE building that used to discharge into the outfall flume have been plugged.

Sanitary Drains: The sanitary drains collect drainage from the various plumbing fixtures in the K area buildings. There is no central drainage system for all the sanitary drains but there are several different septic tanks that are used to collect the drainage from the various buildings that have sanitary facilities in use.

Fuel Storage Basin Sub Basin Collection System: There is a drain system underneath the bottom of the fuel storage basin. This system is used to collect any leakage originating from within the basin which should happen to get past the bottom of the basin. The collector for this drainage system discharges into a location called the D Sump. The sump is equipped with two sump pumps and filters. These are used to return any water which leaks from the basin back to the basin after passing it through filters.

Basin Transfer Area Floor Drains: The transfer area floor drains at the working level of the basin have been rerouted so that they drain into a location called the C Sump which is located just outside of the 105K building. Two sump pumps remove the water which collects in the sump, pass the water through filters, and then return it back into the basin.

Basin Chiller Pad Floor Drains: The floor drains in the chiller pad area of the basin have been rerouted to allow them to drain into the basin proper.

System #19 DC Electrical Safety Class #3

Included within this system are batteries used for switchgear, rectifiers and distribution panels. DC circuits on the load side of the DC panels are part of the system they serve. The AC line between the AC distribution center and the rectifier is in the DC Supply System.

System #20 AC Electrical Safety Class #3

This system covers the distribution of AC power within the K Area. The system starts with the 230 KV switchgear at the 151K Building and the 13.8 KV switchgear in the 165K Building. The system ends at the load side of the motor control center or at the secondary side of the transformers for voltages below 480 V.

Also included as part of this system is the lighting system which includes all wiring, switches, fixtures and panels for normal and emergency lights. The system starts at the secondary side of the 480 V transformer.

System #23 K-Area Water Supply Safety Class #3

This system is involved with the supply of all the water used at K Area.

Structure: The primary structural components of the water supply system include the 181KE river pump station, the 183KE building with its water treatment basin, and the clearwell reservoirs which adjoin the 190KE building. The water treatment basin of the 183KE building includes within its confines both the coagulation and sedimentation portions of the basin as well as the sand filters.

Travelling River Screen: The river screen, which is powered by an electric motor, is located at the inlet to the river pump intake structure. The screen prevents debris and fish from entering the intake to the river pump. During operation the screen is sprayed with water to clean it of accumulated debris. The original design of the screen wash called for the screen wash water to be returned to the river via the 1904 outfall.

River Pumps: There are two types of river pumps: Large 1500 Hp turbine pumps of 32,000 gpm capacity and smaller 600 Hp submersible pumps of 10,000 gpm capacity.

Backwash Pumps: These are 600 Hp turbine pumps which are used to backwash the sand filters.

Service Water Pumps: 300 Hp pumps used to pressurize all water streams in the K area including the potable, service water, and fire system water.

Potable water booster pump: A centrifugal pump used to provide differential pressure for potable water chlorine injection.

Air Valves and Vacuum Breakers: Automatic air valves vent air from the large water pipes which connect the river pumps to the 183KE water plant. The air valves and vacuum breakers allow air to enter the piping system when the water drains from them.

Water Supply Instrumentation: There are two significant items of instrumentation associated with the water supply system:

Level Transmitters: These units monitor the levels in the coagulator basins, sand filters and clearwells.

Chlorine Detectors: The detectors are used to warn personnel of a chlorine leak. They are located in the 183KE chlorine room and in the lower level of the 165KE building.

System #24 Fire Protection Safety Class #3

This system contains all equipment involved with fire protection at K Area.

Automatic Sprinkler Systems: 105KE and 105KW; A wet pipe sprinkler system is installed in the active portion of these buildings, including the facility control rooms, offices, lunchrooms, and clothing change rooms; any operation of this system annunciates, through a Radio Fire Alarm Reporting (RFAR) box, at the 100N Area Fire Station. Fire Alarm pull boxes are located both inside and outside of the buildings; they annunciate in the same way as the sprinkler system. The 1717K building is protected throughout by a wet pipe automatic sprinkler system that annunciates, through a RFAR box, at the 100N Area Fire Station.

Automatic heat detectors: The 165KE building is equipped with automatic heat detectors which are connected to a RFAR box that annunciates at the 100N Area Fire Station.

Fire Hydrants: A series of hydrants are placed throughout the K Area to act as a source of water for fire fighting activities.

Water Supply: The normal water supply to both the buildings and the fire hydrants is the filtered water which both originates from, and is pressurized by, the service water pump.

System #30 Heating & Ventilation Safety Class #3

This system contains all the equipment used for ventilation and temperature control within the buildings at K area.

Electric Hot Water Boilers: There is a hot water boiler located in both the KE and KW basin area. A glycol solution is circulated through the boilers and supplies heat to the unit heaters in the basin which are connected to it

Evaporative Air Coolers: These coolers are used for cooling occupied buildings throughout the area. They are predominately roof mounted units connected to a water supply within the building.

Refrigeration Air Conditioners: Various sizes of refrigerated air conditioners are used for localized cooling purposes. The units can be either situated in a fixed installation or can be portable.

Exhaust Fans: The fans located on the roof of the 105K basin area are responsible for providing air circulation and air removal within the basin area. The fans do not have filters installed on them. There are also exhaust fans situated in other buildings which provide a source of ventilation for exhausting gasses from battery storage rooms or provide for general air circulation within a building or an enclosed area.

Unit Heaters: Electric unit heaters are used to provide a heating source for both entire buildings or portions of buildings which do not utilize either boilers or heat pumps.

Heat Pumps: Heat pumps supply both the heating and cooling functions for localized areas within various buildings where the centralized heating or cooling unit is either entirely nonexistant,,nonfunctional or less than adequate to meet all heating and cooling needs.

System #43 Water Sampling & Analysis

The water sampling system consists of all the equipment used for collecting water samples from the following sources.

183KE Water Sampling System: Safety Class #4
The sampling system for 183KE consists of piping, sample pumps and instrumentation primarily used for obtaining and measuring the turbidity of the water.

Outfall Water Sampling: Safety Class #3
The sampling system including sample pumps and instrumentation used to monitor various parameters of the water which is being returned to the river via the 1908 outfall structure. These parameters include temperature, flow and radiation levels.

Fuel Storage Basin Water Sampling System: Safety Class #3
The basin water sampling system includes the piping and instrumentation used to collect fuel storage basin water for subsequent labratory analysis of pH, radionuclide level and suspended solids.

System #50 Environmental Monitoring Safety Class #3

The environmental monitoring system is the sampling system with its associated instrumentation that monitors exhaust air departing the 105K Basin buildings via the exhaust air fans. This exhaust air is monitored for radionuclide content.

System #54/55 Security Safety Class #4

The security system consists of all equipment used to monitor the movement of nuclear materials and personnel within the 105K building basin area.

References

1. WHC-SD-NR-PD-005, Rev. 0, Plan for Development of Safety Equipment Lists for 100N and 100KE/KW, by S. A. Slinn, dated June 25, 1992.

2. WHC-SD-NR-SD-003, Rev. 0, K-Area Systems - Identification of System Number and Safety Classification, by M. J. Langevin, dated Feb. 13, 1991.

Appendix A1

System #14: Listing of Cranes and Hoists at K East Area

Appendix A2

System #14: Listing of Cranes and Hoists at K West Area