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Page 1 of 2

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13b. Justification Details Leaks through the construction joints following a seismic event could result in a release of radioisotopes to the soil in excess of WHC controls.			
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K Basins Isolation Barriers Summary Report

G. C. Strickland

Westinghouse Hanford Company, Richland, WA 99352
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Abstract: This summary report documents the technical evaluations, design, analysis verification, fabrication, installation, and testing of the basin design modification which closes the USQ.

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K Basins Isolations Barriers Summary Report

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J. P. Schmidt

J. P. Schmidt

**105-K BASINS ISOLATION BARRIERS
SUMMARY REPORT**

WHC-SD-SNF-FDR-001

Rev. 1

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LIST OF TERMS

AEA	<i>Atomic Energy Act of 1954</i>
ASME	American Society of Mechanical Engineers
DBE	design basis earthquake
DOE	U.S. Department of Energy
ERDA	Energy Research and Development Administration
IXC	ion exchange column
IXM	ion exchange module
PUREX	Plutonium Uranium Extraction (Facility)
SAR	safety analysis report
USQ	unreviewed safety question
WHC	Westinghouse Hanford Company
105-K Basins	105-K East and 105-K West fuel storage basins

1.0 INTRODUCTION

The 105-K East and 105-K West fuel storage basins (105-K Basins) were designed and constructed in the early 1950's for interim storage of irradiated fuel following its discharge from the reactors. The 105-K East and 105-K West reactor buildings were constructed first, and the associated storage basins were added about a year later. The construction joint between each reactor building structure and the basin structure included a flexible membrane waterstop to prevent leakage. Water in the storage basins provided both radiation shielding and cooling to remove decay heat from stored fuel until its transfer to the Plutonium Uranium Extraction (PUREX) Facility for chemical processing. The 105-K West Reactor was permanently shut down in February 1970; the 105-K East Reactor was permanently shut down in February 1971. Except for a few loose pieces, fuel stored in the basins at that time was shipped to the PUREX Facility for processing. The basins were then left idle but were kept filled with water.

The PUREX Facility was shut down and placed on wet standby in 1972 while N Reactor continued to operate. When the N Reactor fuel storage basin began to approach storage capacity, the decision was made to modify the fuel storage basins at 105-K East and 105-K West to provide additional storage capacity. Both basins were subsequently modified (105-K East in 1975 and 105-K West in 1981) to provide for the interim handling and storage of irradiated N Reactor fuel.

The PUREX Facility was restarted in November 1983 to provide additional weapons-grade plutonium for the United States defense mission. The facility was shut down and deactivated in December 1992 when the U.S. Department of Energy (DOE) determined that the plant was no longer needed to support weapons-grade plutonium production. When the PUREX Facility was shut down, approximately 2.1×10^6 kg (2,100 metric tons) of irradiated fuel aged 7 to 23 years was left in storage in the 105-K Basins pending a decision on final disposition of the material. The *Hanford Federal Facility Agreement and Consent Order* (Ecology et al. 1994), also known as the Tri-Party Agreement, commits to the removal of all fuel and sludge from the 105-K Basins by the year 2002.

1.1 SUMMARY OF ISSUE

Westinghouse Hanford Company (WHC), prompted by the change in mission for the 105-K Basins to longer-than-anticipated storage of irradiated fuel, reevaluated the structural integrity of the 105-K Basins facilities. The effort involved reconsidering previous analyses, performed to earlier standards, using modern techniques.

The most recent series of dynamic seismic analyses performed on the reactor buildings and basin structures involved more stringent requirements than those in the original design evaluation, and the current facility safety authorization basis. From the results of these analyses, WHC determined that

previous conclusions about the integrity of the 105-K Basins following a design basis earthquake (DBE) did not take into account potential effects from the dynamic analyses.¹

The risk of leakage of basin water was analyzed in the 105-K Basins safety analysis report (SAR) (Meichle 1995) based on a maximum assumed leak rate of 5,680 L/h (1,500 gal/h) caused by a cask drop accident. Based on information available at the time of the analysis, this leak rate bounded leak rates from other events.

The latest dynamic seismic analysis raises question about the ability of the construction joint water seals to restrict leakage following a DBE. The construction joints are located in the floor and walls of the discharge chutes, an area adjacent to and contiguous with the fuel storage basins. The technical challenge of defining the performance of the joint and flexible seal under dynamic seismic conditions was found to be complex, and the results were difficult to substantiate. It could not be shown with the necessary degree of assurance that a dynamic seismically induced leak through the discharge chute construction joint would not exceed the SAR bounding leak rate of 5,680 L/h (1,500 gal/h). Also, WHC concluded that further technical efforts to quantify a potential leak would not be conclusive. Leakage from an assumed failure of the water seal is currently estimated to be between approximately 19,000 and 95,000 L/h (5,000 and 25,000 gal/h). Even with makeup water capability, as assumed in the SAR accident analysis, a leak rate of this magnitude would challenge the basins' capability to maintain fuel coverage.

WHC reviewed the new information and determined that the dynamic seismic analysis results could indicate a condition that was outside the facility safety authorization basis. In accordance with DOE Order 5480.21, *Unreviewed Safety Questions*, WHC evaluated the potential to experience a leak in excess of that analyzed by the existing SAR (Meichle 1995) to determine whether an unreviewed safety question (USQ) existed. The evaluation concluded that a DBE, in addition to potentially causing a leak rate exceeding that analyzed in the SAR, could seriously challenge the ability of the system to maintain fuel coverage. Although the postulated seismic event has a low probability, the potential consequences could be serious. Under these circumstances, and based on the guidance in DOE Order 5480.21, WHC determined that a USQ existed. WHC promptly advised the DOE of the determination by preparing and issuing an unusual occurrence report (Sorenson 1994).

1.2 ISSUE RESOLUTION

WHC actions to resolve the USQ included planning for short-term mitigation activities and long-term resolution of the USQ. A value engineering team was assembled and numerous possible short- and long-term solutions were generated and evaluated (KCM 1994). Short-term mitigation actions taken by WHC included arranging to provide materials for immediately

¹WHC considers a DBE to have a low probability, but the damage from such an event could have potentially serious consequences.

sealing the opening in the construction joint if a large leak occurred and arranging to provide makeup water from the Columbia River using portable diesel-powered generators and pumps.

Of the possible long-term solutions developed by the value engineering team, WHC chose to use isolation barriers to seal the two openings between each discharge chute where the construction joint is located and the fuel storage basin. This activity required the fabrication of two sets of isolation barriers (one set each for 105-K East and 105-K West). Each set is designed to survive a DBE. The openings were originally designed and constructed to allow for the installation of cofferdams, and cofferdams were used successfully several years earlier to allow work to be performed in the discharge chute area of the 105-K West Basin. The new isolation barriers are of a similar design.

A safety assessment of the addition of the isolation barriers was performed (WHC 1995). The assessment concludes that installation of the isolation barriers would not introduce new hazards, that activities associated with installation of the barriers presents no undue risk, and that the activities do not constitute a USQ.

This summary report documents the technical evaluations, design, analysis verification, fabrication, installation, and testing of the basin design modification which close the USQ.

1.3 ORGANIZATION OF SUMMARY REPORT

This summary report provides an overview of the design change implemented to resolve the 105-K Basins construction joint seismically induced leak issue.

Section 1 describes the 105-K Basins, states the issue, and describes the organization of the report.

Section 2 describes the systems associated with the 105-K Basins, including the basin structure, the discharge chute, and supporting systems. The design and function of the isolation barriers are also described in Section 2.

Section 3 discusses the methods used to resolve the basin leakage issue, beginning with a review of the regulatory requirements (upper tier requirements) that govern the basin design, maintenance, and operation and that would apply to the isolation barriers. The section then covers the methods used to develop and evaluate optional approaches to both the short- and long-term resolution of the issues and the process used to select the best option. Section 3 also discusses the design criteria for the isolation barriers and how the design process ensures compliance with these criteria. Because two cofferdams were fabricated as Safety Class 2 equipment to facilitate possible work on the 105-K East discharge chute area before the USQ was identified, the acceptability of their conversion to Safety Class 1 isolation barriers had to be evaluated against the current criteria. This discussion also is found in Section 3. Finally, Section 3 describes the process for fabricating the new isolation barriers.

Section 4 explains the procedures used to install the isolation barriers and the post-installation testing performed to ensure that the barriers will fulfill their intended function. With the barriers installed, the basin systems are operated differently because the barriers isolate the discharge chute area. Section 4 also describes the operation of the systems, as well as the inspection and maintenance procedures that will be used during the life of the barriers to ensure that they continue to meet the leak integrity requirements.

Section 5 provides the conclusions arrived at regarding the long-term resolution of the USQ.

Section 6 summarizes the technical tasks.

Section 7 contains the references, which provide additional supporting information.

2.0 SYSTEM DESCRIPTIONS

This section describes the 105-K Basins structures, supporting systems, and isolation barrier system.

2.1 105-K BASIN STRUCTURES AND SYSTEMS

The 105-K East and 105-K West Basin facilities are designed to provide underwater storage for irradiated reactor fuel. Approximately 2.1×10^6 kg (2,100 metric tons) of irradiated N Reactor fuel is currently stored in the two basins. Fuel stored in the 105-K West Basin is contained in sealed canisters while fuel stored in the 105-K East Basin is stored primarily in open canisters.

2.1.1 Main Basin Structures

The main basin structures consist of the storage basin, the discharge chute, the leak/drainage collection system, and the superstructure.

2.1.1.1 Basin Structure. The basin structures (Figure 2-1) consist of the entire concrete pool that retains the water covering the fuel. Both the 105-K East and 105-K West fuel storage basins are rectangular, reinforced-concrete basins located below ground level. The basins are approximately 38.1 m (125 ft) long, 20.4 m (67 ft) wide, and 6.1 m (20 ft) deep with a nominal 4.9-m (16-ft) water depth. The rectangular main basin is subdivided into three equal sections by two concrete walls that run north and south for most of the width of the basin. There are openings between the dividing walls and the basin walls at each end of the dividing walls. The south wall of each basin contains two openings into an area known as the discharge chute; the openings are separated by a center island.

Each basin is constructed with a subbasin asphalt leak collection system, which collects any water that may leak through the main basin floor.

There are several significant differences between the 105-K East and 105-K West Basins.

- The 105-K West Basin is coated with an epoxy sealant that was pliable when installed.
- The fuel elements stored in the 105-K West Basin are encapsulated in closed canisters; the 105-K East Basin storage racks contain open canisters of fuel elements.
- The 105-K West Basin has a canister decapping station located in the transfer canal between the load-out pit and the basin's west bay.

Figure 2-1. Configuration of the 105-K Basins.

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- The walls of the 105-K West Basin taper, thinning from bottom to top; the 105-K East Basin has uniformly thick sidewalls.
- The 105-K West discharge chute center island is anchored to the 105-K West reactor building wall by steel beams.

2.1.1.2 Discharge Chute. An area known as the discharge chute is connected to the south side of each rectangular main basin. The discharge chute is the area where fuel elements discharged from the adjacent reactor were collected. The discharge chute encompasses a region of basin water approximately 16.2 m (53 ft) long by 2.7 m (9 ft) wide with a sloped rear face. The discharge chute has been used for conducting fuel segregation work.

The discharge chute is separated from the main basin by a freestanding wall or island. The wall is approximately 12.8 m (42 ft) long and 1.8 m (6 ft) thick. The top of the wall is at the same elevation as the grating that forms the floor surface at the working level of the main basin. The north surface of the island is common with the surface of the main basin south wall. At the east and west ends of the island are channels or openings to provide access from the main basin to the area between the south surface of the island wall and the rear wall of the discharge chute. These openings provide for transport of discharged fuel from the reactor discharge chute to the basin for storage.

The discharge chute contains construction joints between the basin structure and the reactor structure. These unreinforced joints, between construction pours for the reactor building and the storage basins, run the length of the discharge chute. The construction joints incorporate flexible membrane water stops, which have leaked in the past. Repairs to the leaking 105-K East construction joint were completed in 1979.

2.1.1.3 Leak/Drainage Collection System. Both basins have under-basin leak collection systems. Each collection system is composed of an asphalt membrane and a collection field. However, the asphalt membrane and collection field do not extend under the discharge chute area. Collected leakage is routed from the collection field to a sump located at the bottom of a 20-cm- (8-in-) diameter, steel-lined caisson located on the outside of the building on the north side. The sump uses two 227-L/min (60-gal/min) sump pumps located about 7.6 m (25 ft) below ground level. The sump pumps return collected drainage water to the basin or to a radioactive waste holding tank. The pumps are equipped with electrical controls that alternate the pumping cycle from one pump to the other. The control system can operate both pumps simultaneously in the unlikely event that the demand exceeds the capacity of one pump. An epoxy sealant has been applied to the floor and walls of the 105-K West Basin to further limit leakage. The asphalt membrane does not extend below the discharge chute and associated construction joint; therefore, leakage through the construction/expansion joint water stop would be released to the ground. See Figures 2-2 and 2-3.

Figure 2-2. 105-K Basins Elevation.

Figure 2-3. Fuel Storage Basin.

NOTE: Top view showing isolation of the discharge chute, which is the location of the construction joint, from the main basin. Isolation at this location would limit leakage in the event an earthquake opens the construction joint. The water level in the main basin would remain at normal levels, ensuring continued coverage of irradiated fuel in the basin.

2.1.1.4 Basin Superstructure. Each storage basin is enclosed by a building with steel-framed, asbestos-cement panel walls on three sides and a flat roof. The fourth wall consists of the 1.5-m- (5-ft-) thick, reinforced-concrete, outlet (discharge side of reactor) shielding wall to the reactor building. The storage basin building is 4.5 m (15 ft) high supported partially by the basin walls and partially by the foundation of the reactor building.

2.1.2 Basin Storage Support Systems

Several systems are used to withdraw, treat, and store Columbia River water for use in the storage basins. Installed systems recirculate, clean, and chill the water. Additional systems monitor basin water level and temperature, monitor area radiation levels, and ventilate the facilities.

2.1.2.1 Basin Water Makeup System. Makeup water for the fuel storage basins is drawn from the Columbia River using vertical turbine-driven pumps rated at 121,000 L/min (32,000 gal/min) and is piped to the 183-K East Water Treatment Plant. The river water is treated with alum, which is used as a coagulant to help remove solids, and a filter aid. The water is then filtered through a sand filter system.

A large underground storage structure, commonly called the clear well, can supply 5,680 L/min (1,500 gal/min) of filtered emergency makeup water to the cooling pool to recover from an abnormal reduction in water level. The clear well can contain approximately 3.4×10^7 L (9.0×10^6 gal) of treated water. If electrical power is lost to the area (only one source of power is available), portable pumps or fire trucks can be used to supply water from the clear well. Alternatively, if the clear well is unavailable, water can be drawn directly from the Columbia River.

A commercial water demineralizer consisting of an anion and cation ion exchange package is used to treat service water for use as makeup water for the fuel storage basins. Each basin has an identical unit. The rating of each system is 45 L/min (12 gal/min). When the cation and anion material is saturated, the tanks are collected by the vendor and replaced with fresh tanks.

Another source of demineralized water is available when larger amounts of water are needed. Water can be delivered to the 105-K Basins at 190 L/min (50 gal/min) through existing piping from a 90,850-L (24,000-gal) storage reservoir in the 1706K Building.

2.1.2.2 Basin Water Recirculation System. Three 15-cm- (6-in-) diameter suction lines (one in each bay) are located along the north side of the basin. The suction lines reach to approximately 3.8 m (12.5 ft) above the basin floor. Each line has independent valving so the desired amount of water for recirculation can be selected from each bay. These lines manifold into a 20-cm (8-in.) line that goes to the suction sides of two self-priming 190-L/min ($\approx$ 50-gal/min) recirculation pumps. One pump is used during normal operation, resulting in a recirculation flow of $\approx$ 1,900 L/min ($\approx$ 500 gal/min).

Water is discharged from the pumps through a 5- μ m, cartridge-type disposable filter, then flows through the chillers, discussed in Section 2.1.2.3, and back to the south side of the basin. Each of the three bays has a discharge line equipped with a valve and a flow meter. The recirculation system also can provide water to the ion exchange columns to discharge into the east bay. Three ion exchange columns, each with 190-L/min ($\approx$ 50-gal/min) capacity, operate in parallel for a total flow capacity of 570 L/min ($\approx$ 150 gal/min). A schematic of the water recirculation system is shown in Figure 2-4.

2.1.2.3 Basin Cooling System. Each basin has a forced-circulation cooling system to remove the decay heat generated by the irradiated fuel stored in the basin. The cooled water also reduces leaching of radioisotopes from the fuel elements and results in decreased airborne activity in the basin area.

A 60-hour quiescent cooling period (no forced circulation or addition of coolant) for a basin full of 150-day-old fuel and a maximum water temperature of 54.5 °C (130 °F) was established in 1977 as the design basis for cooling requirements. The 60-hour period was chosen because it provided sufficient time to bring in emergency engine-driven pumps for makeup of level loss caused by evaporative cooling. A reanalysis of cooling system failure was performed in 1994 (Meichle 1995). Failure of the cooling system under worst-case conditions was calculated to result in a peak temperature of 69.1 °C (156.4 °F) for 105-K East and 85.7 °C (186.3 °F) for 105-K West (Barrington 1994). The minimum times to go from 10 °C (50 °F) to 37.8 °C (100 °F) were 60 days for 105-K East and 47 days for 105-K West. The required time for each basin temperature to rise from 37.8 °C (100 °F) to 54.4 °C (130 °F) was 88 days for 105-K East and 47 days for 105-K West. These calculations assumed the pumps were running, there was no evaporation, the basin buildings had no ventilation, and the highest outside temperatures were used. If evaporation from the basin was assumed, neither pool reached 37.8 °C (100 °F) (Barrington 1994).

The water cooling system is equipped with an air-cooled chiller, which is used during normal operation of the water cooling system. The basin water temperature is maintained at between 7 °C and 10 °C (45 °F and 50 °F).

2.1.2.4 Basin Water Cleanup System. Each basin water cleanup system is designed to maintain water quality and clarity and to control the concentration of soluble and particulate radioactive material in the water. The system consists of the following components:

- Skimmer pump
- Sand filter
- Ion exchange modules (IXMs)
- Ion exchange columns (IXCs).

Controlling the radionuclide concentration helps to reduce personnel radiation exposure.

Figure 2-4. 105-K Basins Water Recirculation Systems.

These components, except the IXC, are supplied by water from the basin skimmer pump. The skimmer pump, sand filter, and IXMs are referred to as the IXM or skimmer pump system. These systems treat the water that is drawn from the surface of the fuel storage basin.

Basin water enters the IXM system through three adjustable, screened weirs located along the north wall of the basin. The single skimmer pump, with a capacity of 1,500 L/min (≈ 400 gal/min) draws water from the weirs and pumps it to the sand filter. After passing through the sand filter, a portion of the water is routed through one or two of the IXMs (600-L/min [≈ 160 -gal/min] capacity each) before being returned to the basin. The IXMs are filled with ion exchange resin and are separately contained in a concrete block. The IXMs are configured to operate in parallel and remain in service until their resins no longer effectively remove radionuclides from the basin water. At that time they are replaced. A spent IXM with its concrete shielding block serves as its own container and is disposed of as a single unit.

2.1.2.4 Radiation Protection. The basin water provides both shielding of the work area from direct radiation and scrubbing of particulates, which significantly reduces the release of radioactive materials to the environment. Assuming 2.4 m (8 ft) of water covering the irradiated fuel, the radiation dose rate from the fuel was calculated to be <1.0 mrem/h ($<1.0 \times 10^{-5}$ Sv/h). However, the irradiated N Reactor fuel is stored in the 105-K East and 105-K West Basins with a nominal 4.9 m (16 ft) of water, which is twice the calculated coverage.

The radiation dose rates above the basins from all sources are approximately 20 mrem/h (2.0×10^{-4} Sv/h) at 105-K East Basin and <1.0 mrem/h ($<1.0 \times 10^{-5}$ Sv/h) at 105-K West Basin.

2.1.2.5 Basin Water Level, Temperature, and pH Monitoring. Redundant electrode-operated alarms are provided to monitor each cooling pool water level (Meichle 1995). Any high- or low-level trip provides annunciation in the affected basin's control room and initiates a system trouble annunciation in the other basin's control room. The alarm system is functionally tested routinely.

Pool temperature is monitored and recorded in the facility operations offices. A basin temperature above the set point causes an alarm in the facility operations office and initiates a trouble annunciation in the other basin's control room (Meichle 1995).

The basin water sampling system is used to take samples of basin water at various locations throughout the basin. The samples are used to test for different forms of radiation and to determine the water chemistry. The samples are taken weekly and put in a test bottle to be sent to the 200 Areas laboratory. Basin water pH is monitored periodically and recorded in the facility operations office. Basin water pH is maintained between 5.0 and 10.0 to limit material degradation (Meichle 1995).

2.1.2.6 Basin Ventilation System. The basin ventilation system consists of four roof-mounted centrifugal exhaust fans and two evaporative coolers. The exhaust fans are controlled from a switch station located on the east side of

the south wall of the basin. The nominal rating of the exhaust fans varies from 200 m³/min to 285 m³/min ($\approx 7,000$ ft³/min to 10,000 ft³/min). The two roof-mounted evaporative coolers have a nominal rating of 285 m³/min ($\approx 10,000$ ft³/min) each. Two exhaust fans and two coolers are located above the basin, and two exhaust fans are located above the fuel transfer area.

2.2 ISOLATION BARRIERS

Isolation barriers are installed to isolate the discharge chute and associated construction joint water seal from the fuel storage basin. Two isolation barriers are installed in each basin: one in each opening between the discharge chute and the basin (see Figure 2-3). These isolation barriers are modeled after the cofferdams that were used successfully to isolate the basin when the 105-K East construction joint was repaired. The isolation barriers are part of the basin water boundary and are considered permanent installations. The isolation barriers are Safety Class 1 because they will mitigate potential post-earthquake leakage to within the level currently documented in the SAR (Meichle 1995). Limiting the leak rate and manual use of makeup water will maintain the basin water level within the limits established in the SAR. The isolation barriers are designed to survive the DBE shocks and aftershocks.

One installed barrier is shown conceptually in Figure 2-5; the fuel storage basin is in the foreground, and the discharge chute is in the upper left of the figure. The isolation barrier has a flat sealing surface with a flexible seal installed on the sealing surfaces. After installation, the sealing area is compressed against the walls of the island and the basin floor to seal in the water.

A top view of the isolation barrier installation is shown in Figure 2-6. Nineteen tie rods and their associated brackets compress the flexible seal against the face of the opening. The nuts on these threaded rods are tightened to achieve uniform pressure and seal compression over the sealing face of the barrier. Figure 2-7 shows the isolation barrier before installation. Figure 2-8 illustrates the seal, barrier, and wall installation, showing the seal after it is compressed by tightening the tie rods nuts.

During normal operation, the water level in both the discharge chute and the basin will be approximately the same. Figure 2-7 shows a slot (weir) through the isolation barrier. The weir, at a water level of approximately 4.95 m (16.25 ft), provides for normal water inventory control during operation without interference from the presence of the isolation barriers. This slot provides for water transfer between the two areas under normal conditions when the water level in the chute is slightly higher than the weir. In the unlikely event of a leak in the discharge chute that is large enough to reduce the water level, the level in the basin will not continue to drop once it reaches the weir level. Below that level passive isolation begins.

Figure 2-5. General Layout of Basin.
(Not to Scale)

NOTE: The basin, where fuel is stored, is shown in the foreground. The discharge chute, where the construction joint is located, is in the upper left of the figure. The isolation barrier, which has a gasket facing the concrete basin wall and island, is drawn snug by tie rods. The tie rods, 2.5 cm (1 in.) steel rods, pass through the bracket assembly. They are threaded on the discharge chute side, allowing the 19 tie rods to draw the barrier up uniformly. This configuration allows isolation of the water covering the fuel from the construction joint in the discharge chute.

Figure 2-6. Top View of Basin Discharge Chute Isolation Barrier.

H8409021.4

Figure 2-7. Side Views of Basin and Discharge Chute Isolation Barriers.

H9409021.5

Figure 2-8. Isolation Barrier with Seal Compressed.

H9409021.6

The operation of the basin auxiliary systems described in Section 2.1 is slightly changed by the installation of the isolation barriers. The demineralized makeup water and the recirculated water from the basin water cleanup skimmer system IXMs are delivered to the discharge chute rather than to the basin. With this method, the discharge chute water level increases above the elevation of the weirs and flows through them into the basin. This provides a clean-water purging effect for the discharge chute. Purging the discharge chute is prudent and a good engineering practice, although not a requirement. In the unlikely event of a major leak in the chute, the water released is anticipated to be cleaner than the bulk water in the basin, and would therefore reduce the environmental consequences.

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3.0 APPROACH TO ISSUE RESOLUTION

This section discusses the evaluations leading to definitions of objectives and design requirements and to the selection of the design modification to resolve the USQ.

3.1 REQUIREMENTS GOVERNING BASIN DESIGN MODIFICATION

The DOE-established safety requirements for nuclear facilities are generally set forth in regulations and orders, with supplemental detail contained in DOE directives, notices, standards, and instructions. These safety requirements are made applicable to contractors, in whole or in part, through the contract negotiated with the DOE. Documentation of compliance with established requirements is, in turn, reflected in facility-specific safety documentation including SARs, interim safety bases, and interim operational safety requirements. These documents, when approved by the DOE, constitute the facility safety authorization basis.

The USQ process is used to evaluate changes against the facility safety authorization basis and determine whether a proposed change to the facility falls within the boundary analyzed in the safety documentation. The process is also used when a condition is discovered that may be outside of the analyzed safety envelope. The discovery that a seismic event could result in a leak rate greater than that previously analyzed was evaluated and determined to be a USQ.

As an initial step in resolving the USQ, WHC reviewed the requirements that make up its authorization basis to identify those that would be directly affected by installing isolation barriers in the 105-K Basins. These requirements are documented in Appendix A. Safety analyses (including design analysis) were performed to support USQ resolution. Following final approval by the DOE, WHC will make appropriate revisions to the affected authorization basis documents.

3.2 DEVELOPMENT OF DESIGN CRITERIA

In their installed configuration, the isolation barriers are considered to be Safety Class 1 structures and will form a continuation of the fuel storage basin walls. Therefore, the design criteria appropriate to apply to the isolation barriers are those that apply to the storage basin. The storage basin design criteria are contained in WHC-SD-SNF-SAR-001, *K Basins Safety Analysis Report* (Meichle 1995).

As part of the design process, the storage basin design criteria were reviewed to identify those criteria that apply to the isolation barriers. The following two design criteria are of greatest importance to the design of the isolation barriers:

- The ability to limit leaks from the basin if the construction joint seal leaks in the discharge chute.

- The ability of the barriers to maintain their integrity during and following a DBE.

The DBE is defined as a seismic event producing a maximum horizontal ground acceleration of 1.96 m/s^2 (0.20g) simultaneously with a maximum vertical ground acceleration of 1.27 m/s^2 (0.13g) at zero period. In addition to the initial seismic shock, and with equal levels of water in the basin and discharge pool, the isolation barriers must be capable of withstanding an aftershock when all water on the discharge chute side of the barrier has leaked out through the construction joint.

During normal operation, the water on both sides of the isolation barrier will be at approximately the same level. Following a DBE, it is assumed that all the water on the discharge chute side drains out through the open construction joint.

3.3 COMPLIANCE WITH DESIGN CRITERIA

The isolation barriers have been designed to comply with the design criteria applied to the fuel storage basins themselves. Analyses were performed to ensure that the isolation barrier structure, as a complete unit, will withstand the design-basis loads and that each individual component will withstand the loads placed on it (Winkel 1995). The analyses confirm that the design criteria are met. The isolation barriers will maintain basin integrity under the worst-case accident scenario (a seismic aftershock and no water on the discharge chute side of the barrier).

3.4 EVALUATION OF OPTIONAL APPROACHES FOR BASIN ISOLATION

A value engineering study was performed in June 1994 to develop design options for isolating the discharge chute in 105-K East and 105-K West from the main storage basins (KCM 1994). More than 100 options, including filling the discharge chute with concrete, were developed for consideration. These options were screened to identify final candidate solutions for the potential post-seismic leakage at the construction joint.

Three major options (with two variations on one option) for isolation or limited isolation were identified for further evaluation. These options are discussed in Table 3-1.

3.5 SELECTION OF FINAL ISOLATION BARRIER DESIGN

This section discusses the selection of the preferred design of full-height isolation barriers with weir slots from the options discussed in Section 3.4.

Table 3-1. Candidate Solutions for Potential Post-Seismic Leakage at Construction Joint.

Options	Discussion
1	FULL-HEIGHT ISOLATION BARRIERS. Barriers would extend above the current basin water level, effectively isolating the 2.3×10^5 L (60,000 gal) of discharge chute water from the main basin volume (assuming approximately 3.8×10^6 L [1,000,000 gal]).
2	FULL-HEIGHT ISOLATION BARRIER WITH WEIR SLOT. Same as Option 1, except that a weir slot would be cut in the barrier. Two possibilities were evaluated with this option: 2a. LOW SLOT. In this option, the slot would be cut below the water level at a height that ensures that the island would not be overstressed, even during an aftershock equal to the full initial shock and total drain of the discharge chute. The slot would be open normally, but would be closable for leak-rate testing. A variation of this approach that was considered (but is functionally identical) is use of a normally open isolation valve at this level. 2b. HIGH SLOT. In this option, the slot would be cut above the current water level at a height that ensures that the island would not be overstressed, even in the event of an aftershock equal to the existing facility criterion (1.18 m/s^2 [0.12g]), and total drain of the discharge chute. The slot would be open normally, but would be closable for leak-rate testing. It would be sized to allow raising of the basin water level to allow for radiation field reduction if judged necessary as part of the dose reduction program.
3	HALF-HEIGHT BARRIERS. Partial barriers would extend 1.8 m (6 ft) above the basin floor, but not above the water surface, allowing continued passage of fuel or material above them into the discharge chute. Fuel coverage would be maintained with reduced shielding above the fuel elements. The 2.3×10^5 L (60,000 gal) of discharge chute water and the water in the main basin from a 4.9-m (16-ft) elevation to a 1.8-m (6-ft) elevation might overflow the top of the partial barrier (approximately 2.6×10^5 L [685,000 gal]) in the event of a severe leak in the discharge chute. Although a number of height variations were considered, they are represented well by this option.

3.5.1 Factors Considered in Selection of the Final Design

Factors considered in reaching the final selection included public safety protection from airborne release, reversibility of change, cost, constructability, environmental protection capability, worker safety, seismic performance, and operational impacts. Although important in eliminating other options, the first four factors (public safety, reversibility of change, cost, and constructability) were essentially identical for the final options.

The last four factors (environmental protection capability, worker safety, seismic performance, and operational impacts) have subelements that allow for differentiation between options.

Environmental protection capability, or limiting environmental impact potential, has three subelements:

- Limiting release of contaminated water to the surroundings
- Optimizing the quality of any water that might be released
- Providing the ability to detect, characterize, and isolate operational leaks from the construction joint.

Worker safety has two subelements:

- Support for dose reduction measures
- Accessibility of the basin following a quake if additional emergency operator intervention is necessary.

Seismic performance has two subelements:

- Ability to survive the main shock
- Ability to survive aftershocks.

Operational impacts has three subelements:

- Capability to verify leak integrity of isolation barrier
- Vulnerability to errors from routine operation
- Impacts on other facility operations.

3.5.2 Evaluation of Options

As with all engineering problems, a number of approaches are viable. Once the top-level design requirements are met, selecting the optimum solution is often a tradeoff between various factors. A summarization of the tradeoff analyses is contained in Table 3-2.

Option 1, Full-Height Isolation Barriers, was originally selected, and Option 2a, Full-Height Barrier with Low Weir Slot, was identified as the backup if adequate seismic capability of the full-height barriers in Option 1 could not be demonstrated. Moving from Option 1 to Option 2b is simple; structural analyses show that cutting a slot in the full-height barrier reduces structural capability insignificantly. Option 3, Half-Height Barriers, was not chosen for several reasons, as follow.

- The barrier is vulnerable to damage by movement of materials over it.
- Operators would be unable to perform tests that demonstrate leak integrity.
- The barrier provides relatively poor reduction of environmental impacts.

Option 2b, Full-Height Barrier with High Weir Slot, was selected when it became obvious that the barrier could survive the primary quake and, using reduced criteria for existing facilities, the aftershock. The primary problem identified with this solution was the restriction on use of the discharge chute for most future basin activities. The evaluation based on the factors discussed in Section 3.5.1 concluded that although Option 2b would preclude use of the discharge chute and upset the current baseline for facility operations, it would be unlikely to dramatically delay final fuel removal from the basin.

Table 3-2. Comparison of Options.^(a) (2 sheets)

Options	1	2a	2b	3
Title	Full-Height Isolation Barriers	Full-Height Barrier With Low Weir Slot	Full-Height Barrier With High Weir Slot	Half-Height Barriers
ENVIRONMENTAL IMPACT				
a. Limiting release of contaminated water to the surroundings	Limited to 2.3×10^5 L (60 kgal)	Depends on weir slot height. For 4.3 m (14 ft) above bottom, 7.0×10^5 L (185 kgal)	Limited to 2.3×10^5 L (60 kgal), unless water height raised to support dose reduction	For 1.8 m (6 ft) above bottom, 2.3×10^5 L (60 kgal)
b. Optimizing the quality of any water that might be released	Water quality improvement possible, minimizing impact	Water quality improvement not possible	Water quality improvement possible even for raised height, minimizing impact	Water quality improvement not possible
c. Providing ability to detect, characterize, and isolate operational leaks from the construction joint	Leak detection easy Isolation easy	Leak detection and isolation possible if weir slot designed to be closed with plate or valve	Leak detection easy Isolation easy	Leak characterization not possible without adding top half of barrier
WORKER SAFETY				
a. Dose reduction	Poor ability to support dose reduction by raising water level	Good ability to support dose reduction by raising water level	Good ability to support dose reduction by raising water level	Good ability to support dose reduction by raising water level
b. Accessibility following a quake	Limited field increase expected	Dose increase expected from exposed walls, contamination, and from reduced fuel shielding	Limited field increase expected	Dose increase expected from exposed wall contamination and reduced fuel shielding
SEISMIC PERFORMANCE				
a. Main shock	Will survive same quake as basin	Will survive same quake as basin	Will survive same quake as basin	Will survive same quake as basin
b. Aftershocks	Will survive reduced aftershock quake magnitude ^(b)	Will survive same quake magnitude as primary shock	Will survive reduced aftershock quake magnitude ^(b)	Will survive same quake magnitude as primary shock

Table 3-2. Comparison of Options.^(a) (2 sheets)

Options	1	2a	2b	3
Title	Full-Height Isolation Barriers	Full-Height Barrier With Low Weir Slot	Full-Height Barrier With High Weir Slot	Half-Height Barriers
OPERATIONAL IMPACTS				
a. Leak integrity verification	Straightforward	Requires sealing of weir slot, then straightforward	Straightforward	No obvious method to verify seal performs as designed
b. Vulnerability to errors	Impact on the basin side is limited by normal crane routing	Impact on the basin side is limited by normal crane routing	Impact on the basin side is limited by normal crane routing	Barrier protection from fuel canisters and components bump or drop must be added
c. Impacts on facility operations	Precludes use of discharge chute for handling contaminated equipment and fuel ^(c)	Precludes use of discharge chute for handling contaminated equipment and fuel ^(c)	Precludes use of discharge chute for handling contaminated equipment and fuel ^(c)	Would allow continued use of discharge chute for fuel and equipment handling if special tooling designed to allow raising canisters above the barriers

^(a) Shaded blocks denote negative consequences associated with the option.

^(b) In the 105-K East Basin, the center island is attached to one of two basin dividing walls by a steel beam that was provided in the early 1970's to reduce island stresses with the discharge chute water level drawn down. Preliminary assessments indicate that this beam is adequate to ensure that the island can survive an aftershock equal to the main shock, even with the discharge chute drained. Efforts to retrieve the design and analysis from storage have not been successful. Efforts currently focus on characterizing the installed beam and anchors and incorporating the beam in the basin analysis.

^(c) The discharge chute has two primary advantages as a work location in 105-K East: (1) low radiation fields and (2) relatively large open work area. The first is important because general radiation fields are high throughout 105-K East, but are relatively low in the discharge chute. Although use of the discharge chute provides for lower exposures than working elsewhere in the basin, the optimum solution providing for the largest projected dose commitment reduction, comes from dose reduction measures such as raising the water level. Once dose reduction measures have been taken, a number of other areas become available for operational use; for example, large areas of empty racks can be consolidated and work platforms can be installed by setting them on pedestal mounts that match the existing rack canister configurations.

Options 2a and 3, although seriously considered, were judged to allow for excessive post-earthquake environmental impact from water leakage. Of the two, Option 2a allowed for less water leakage; the "low slot" could be placed as high as 4.3 m (14 ft) and provide full seismic survival without invoking existing facility seismic criteria¹. This would limit water leakage to 7.0×10^5 L (185,000 gal). Option 3 has the worst environmental impact from water leakage and poses additional difficulty because periodic leak testing could not be performed to confirm that the seal would perform adequately if needed.

Design and fabrication of Option 2b are underway. Two issues were resolved with implementation of Option 2b. The first was the acceptability of using existing-facility reduced criteria for aftershocks. This became a moot point. In the 105-K East Basin, the center island is attached to one of two basin dividing walls by a beam that was provided in the early 1970's to reduce island seismic stresses with the discharge chute level drawn down. Assessments indicate that the effect of the beam is adequate to ensure that the island will not be overstressed by an aftershock equal to the main shock, even with the discharge chute drained (Winkel 1995). Further, the 105-K West Basin has been shown to survive; although with a reduced margin.

The second issue was use of the discharge chute for other activities. The discharge chute has two primary advantages as a work location in 105-K East:

- Low radiation fields
- A relatively large open work area.

The first is important because general radiation fields are high throughout the 105-K East Basin area but relatively low in the discharge chute. Although use of the discharge chute results in lower radiation exposures than working elsewhere in the basin, the optimum solution, which provides for the largest projected committed dose reduction, comes from dose reduction measures such as raising the water level. Once dose reduction measures have been taken, a number of other areas become available for operational use. For example, large areas of empty racks could be consolidated and work platforms could be installed by setting them on pedestal mounts that match the existing rack canister configurations. The dose reduction measures taken during barrier installation were effective, and a full basin dose reduction program is underway.

¹This position, developed during the decision process, results in subsequent reanalysis. Further structural analysis (Winkel 1995) showed that "low slot" structural integrity could be achieved at 5.2 m (16 ft 8 in.). This has the effect of combining options 2a and 2b, i.e., they become equivalent.

3.5.3 Design Analyses

Analyses were performed to demonstrate that the isolation barrier design approach would be satisfactory. These analyses include earthquake survivability, as described previously, brittle fracture analysis, falling barrier analysis, and impact of sliding hardware in the discharge chute (Winkel 1995).

3.5.3.1 Brittle Fracture Analysis. The carbon steels used in construction of the isolation barriers are representative of a class of alloys that could be subject to brittle fracture. The analysis presented in Strickland (1995) demonstrates that brittle fracture of the carbon steels used in fabricating the isolation barriers under the basin environmental conditions will not occur.

3.5.3.2 Falling Barrier Analysis. WHC-SD-SNF-SARR-004, *Results of Falling Barrier Analyses* (Fox 1994), evaluated the consequences of an accident in which the isolation barrier is dropped and falls over onto the fuel stored in the 105-K East Basin. In this analysis, the plate, which is 5.5 m (18.04 ft) high, 1.1 m (3.6 ft) wide, and 1.3 cm (0.51 in.) thick, is assumed to fall by rotating about its bottom edge as it rests on the basin floor. The falling velocity is shown to become quasi-steady when the gravity forces are in equilibrium with those from water drag. The analysis shows that the presence of water in the basin reduces the plate kinetic energy by a factor of 20. The plate first impacts the fuel canisters while at an angle of 27° with the basin floor, losing 20% of its kinetic energy in the collision. The energy dissipation is expected to occur by flattening the outside edge of the canisters and bending the wall inward. The fuel is expected to be pushed forward, but few elements, if any, would be crushed. The plate rotational speed is increased 50% by the collision because of the new pivot point, although the kinetic energy is decreased. The plate's second impact is flat across the top of the canisters but does not cause crushing because the loadings are distributed.

An edgewise falling event is considered to be incredible but was analyzed as a worst-case accident. Water drag does not slow the plate, which is assumed to fall in a diagonal direction and impact some canisters with its outer edge. Fuel crushing could occur in as many as four canisters, but this would not affect basin criticality calculations or result in any significant release to the environment.

3.5.3.3 Sliding Discharge Chute Hardware Analysis. The 105-K East discharge chute currently contains hardware that was to be used in encapsulating stored N Reactor fuel and sludges, and two old cofferdam doors that were used in the late 1970's. Although the encapsulation hardware is to be removed from the discharge chute before installation of the isolation barriers, WHC analyzed the case in which the hardware remains in place (Winkel 1995).

This analysis shows that the encapsulation hardware could slide as much as 10 cm (4 in.) in a Hanford Site DBE of 0.2g but would not overturn and thus would not cause any damage. However, the two old cofferdam doors could "pendulum out" because they are attached to the discharge chute by cable stays at the top. If the door(s) swung like a pendulum out a distance of over 1.2 m (4 ft) they could strike the isolation door support brackets. The analysis shows that this unlikely pendulum action should not exceed a sliding

prediction of 0.6 m (2 ft). The analysis also includes a recommendation that a hardware-free zone of 1.2 m (4 ft) be established around the isolation door support brackets to allow for analytical uncertainties and ensure that no earthquake-induced damage could occur. This recommendation has been implemented.

3.5.4 Additional Analyses

Additional relevant analyses and reports were prepared to demonstrate the suitability of installing isolation barriers. The analyses include evaluation of the impact of low pH on the concrete (Strickland 1996), selection and testing of the epoxy sealant used on the isolation barrier sealing surfaces (Graves 1994), evaluation of the impact of a seiche (Strickland 1996), and development of calculations for leak rate testing (Irwin 1995).

The pH analysis evaluated the impact of basin water having a pH ranging from 5.5 to 9 on the concrete structure of the basin. The conclusion is that the concrete corrosion rate is estimated to be on the order of 1.27 cm (0.5 in.) in 100 years. In addition, if any carbon steel rebar were exposed, the corrosion rate would be less than 1 mil per year because of the control of basin water conductivity (Brehm 1995).

Graves (1994) evaluated the selection and testing of the epoxy sealant to be used on the isolation barrier sealing surfaces. The material selected and evaluated is a two-part component consisting of an epoxy resin (a condensation product of bisphenol A and epichlorohydrin) and a curing agent (a proprietary cycloaliphatic polyamine) solvent-free material capable of withstanding the basin radiation fields over an estimated 15-year service life. The material demonstrated tensile strengths greater than 862 kPa (125 lbf/in²) and suitability for underwater application using vendor-supplied equipment and operators.

The impact of a seismically induced seiche was evaluated to respond to a DOE Richland Operations Office readiness assessment finding. The review concluded that the maximum slosh height considering only the fundamental harmonic of a slow moving long-period Love or Rayleigh wave would be 0.27 m (0.9 ft) and 0.46 m (1.5 ft) in the north-south or east-west direction, respectively (Strickland 1996). These do not exceed the 1.5-m (4.9-ft) basin freeboard and thus would not result in any release to the environment.

Irwin (1995) is a compilation of analytical fluid flow developments describing the leak rate equations to support WHC-SD-SNF-TP-009, *105-KE/KW Isolation Barrier Leak Test Specification/Test Plan* (McCracken 1994), and WHC-SD-SNF-ATP-005, *K East Basin Isolation Barrier Leak Rate Test* (Whitehurst 1994a).

3.5.5 Conclusion

Installation of full-height isolation barriers with weir slots above the current water level, i.e., a nominal 4.95 m (16 ft 3 in.) above the basin floor is the best option for satisfying all constraints.

3.6 PROCUREMENT OF MATERIAL USED TO FABRICATE THE ISOLATION BARRIERS

The detailed design for the isolation barriers was developed by modifying an existing design for cofferdams intended for use as temporary isolation devices at 105-K Basins. A detailed review of the design of these cofferdams was performed, and significant design changes were made to satisfy requirements for installation as long-term isolation barriers.

Two isolation barriers are required for each basin to complete isolation of the discharge chute area from the fuel storage pool area. Therefore, four isolation barriers were fabricated.

Two cofferdams were fabricated in 1993 to ensure inventory should an event require temporary use, but they had never been installed. A review was performed to determine the feasibility of modifying these cofferdams to meet the requirements of the final design for the long-term isolation barriers. It was determined that this approach was feasible and offered cost and schedule advantages over complete fabrication of four new isolation barriers. These two cofferdams were modified to meet the requirements of the final design and are referred to as the modified isolation barriers.

Two additional isolation barriers were constructed to meet the requirement of four barriers. These two barriers are referred to as the new isolation barriers.

WHC procedures identify two acceptable methods for procuring material for use in Safety Class 1 applications at the Hanford Site (WHC-CM-4-2). The first method is procurement of material as Safety Class 1 from a supplier approved for the specific scope of supply on the Evaluated Suppliers List. The second method is procurement of material as "commercial grade" from a supplier not on the Evaluated Suppliers List, followed by dedication of this material for use in its intended Safety Class 1 application. The dedication process includes application of quality assurance and quality control activities that provide a level of assurance that the material is suitable and equivalent to that obtained through procurement of the material as Safety Class 1 from an approved supplier.

These two methods for procuring material intended for use in DOE facility Safety Class 1 applications and the requirements of DOE Order 5700.6C, *Quality Assurance* (superseded by 10 CFR 830.120, which is essentially identical to the order). These methods are essentially identical to and, in large part, based on practices developed in the commercial nuclear power industry. Properly implemented, both methods meet applicable regulatory requirements of the U.S. Nuclear Regulatory Commission and DOE. A detailed discussion of the commercial-grade procurement and dedication process and justification for its use is provided in Appendix B to this report.

Both methods of procuring material for Safety Class 1 applications were used in the fabrication of the 105-K Basins isolation barriers. Both sets of isolation barriers contain parts procured using each method. More of the materials used in the modified cofferdams were qualified using the commercial-grade dedication process because all of the materials used to construct the preexisting cofferdams were procured as commercial-grade materials.

A detailed discussion of material procurement for each set of isolation barriers is provided in Section 6.3. Detailed technical evaluations and documentation of the material quality for each set of isolation barriers are provided in PNL-10384, *K-Basin Isolation Barrier Commercial Grade Dedication*, (Sorenson et al. 1995).

3.7 FABRICATION OF THE ISOLATION BARRIERS

The isolation barrier design is an evolution of basin isolation doors used since the earliest days of 105-K East and 105-K West Reactor operation. Rubber pads, commonly called "mattress pads," were employed on the slopes of the discharge chutes to absorb some of the kinetic energy of the fuel elements ("slugs") discharged from the reactor. These mattress pads had to be replaced periodically. This replacement required that the discharge chutes be drained to allow maintenance personnel to remove the worn pads and replace them with new ones. Isolation doors were used to isolate the main basin from the discharge/pickup chutes so that the chutes could be drained to the point that work forces could gain access to the mattress pads. Leak tightness of the doors was an issue only if the pumps/drains could not keep up with the seal leak-by. Once the mattress pad work was done the discharge chutes were refilled and the isolation doors were removed from the two passways between the main basin and the discharge chute basin.

The isolation doors became unusable following 20 years of service. Subsequently, when leakage was observed in the construction joint at 105-K East cofferdams were fabricated to allow emptying the discharge chute to facilitate repairs. These contaminated cofferdams were stored in the discharge chute. Additional cofferdams were fabricated and placed in storage as a contingency against further construction joint leakage.

The isolation barriers are based on the design of the cofferdams but differ primarily because following an earthquake and aftershocks, the barriers must form leak-tight seals between the discharge chute and the main basins rather than simply controlling leakage. Final configuration of both sets of isolation barriers is identical and meets the same design requirements; however, each set has a different fabrication and material procurement history. Each set of isolation barriers is discussed separately in the following subsections.

3.7.1 Fabrication/Modification of Existing Isolation Barriers

Two cofferdam doors were fabricated in the summer of 1993 by ICF Kaiser Hanford Company engineers at the 2704 West shops. The design of the 1993 cofferdams was an update of a design of a pair of doors built in the late 1970's. (The old doors still reside in the pool at 105-K East Basin; they were last used in 1979 to isolate the 105-K East discharge chute from the main basin to allow repair of the leaking construction joint). Fabrication of the new cofferdams was needed in order to have a pair of doors in inventory that could be used to isolate either of the 105-K Basins from their respective discharge chutes in the event that repair of the construction joint was needed.

The cofferdams fabricated in 1993 were essentially identical to the design of the 1994 isolation barriers with the following exceptions.

- The material was procured in accordance with Safety Class 2 requirements.
- Corrosion was not a design consideration. Consequently, no steps were taken to prevent water from filling the interior of the square tubing nor to exclude water from intruding into the space between overlapping steel parts having unprotected surfaces. Thus, the square tube ends were open, and where the square tube was joined to other parts as a stack only stitch welding was specified.
- There was only one lifting bail made of 2.54-cm- (1-in.-) diameter round bar, bent into an inverted U, and welded to the plate at the top of the assembly.
- The seal was a 0.95-cm- (0.38-in.-) thick, natural rubber strip, glued to the plate in the same relative position as the J seal in the isolation barrier design.
- The tie rods were all 2.54-cm (1-in.) round bar with a C hook formed at the end connected to the pad eye of the cofferdam.
- There was no provision in the design for dealing with east-west accelerations during a seismic event.

Because the original design and fabrication were controlled to Safety Class 2 requirements, and conversion of the cofferdams to isolation barriers for permanent installation required their designation be changed to Safety Class 1, there was a question as to whether the fabrication methods and records would support reclassification. A thorough review of the fabrication records and procurement records was performed to determine whether it was feasible to modify and use the cofferdams (Sorensen et al. 1995). The review is discussed in detail in Appendix B and summarized in Section 6.3 of this report. That review concluded that the fabrication package provided adequate documentation of the design, materials procurement, and fabrication to support the decision to modify the cofferdams and designate them as Safety Class 1 barriers. This reclassification was determined to be subject only to the completion of certain additional quality control activities.

The conversion of the existing two cofferdams to isolation barriers and the fabrication of the two new isolation barriers was done simultaneously in the 2704 West shops. Conversion of the cofferdams was accomplished by the following activities.

1. All materials used in the conversion and/or as replacement parts were procured as Safety Class 1 items or were dedicated for Safety Class 1 use.
2. All the paint was removed by sandblasting. The glued on, natural rubber sealing strip was removed.

3. The frame was 100% fillet welded to the plate, establishing a seal to keep water out of the interface layer between plate and tube. The interface layer between mating, welded assemblies was likewise 100% seal welded.
4. The open ends of all square tubing were plugged and seal welded.
5. The top, round stock lifting bail was removed and a lifting bale identical to the isolation barrier design was attached.
6. The bottom most pad eye was converted to a lifting bail/pad eye to match the isolation barrier design.
7. The C hook portion of each tie rod was cut off and the remaining threaded end was welded to a hook cut from plate stock to match the isolation barrier design.
8. Four support brackets, for the purpose of restraining east-west accelerations, were added to each cofferdam in accordance with the isolation barrier design.
9. Studs were welded to the plate to retain the J seal in accordance with the isolation barrier design.
10. Steel surfaces were painted with epoxy to isolation barrier specifications.
11. A J seal was installed to the isolation barrier design.

At the conclusion of this effort, each converted cofferdam met the material and dimensional requirements of the isolation barrier design drawing. The dedication of the converted cofferdams to Safety Class 1 isolation barriers is discussed in Appendix B and in Sorenson et al. (1995).

3.7.2 Fabrication of New Isolation Barriers

Two additional isolation barriers were fabricated by ICF Kaiser Hanford Company. Materials for these barriers were purchased in accordance with WHC requirements for Safety Class 1 application. Fabrication was controlled as Safety Class 1 work throughout.

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4.0 INSTALLATION, OPERATION, AND MAINTENANCE

This section contains a discussion concerning isolation barrier installation and test procedures, procedures for periodic barrier leak testing, and barrier maintenance.

4.1 INSTALLATION OF ISOLATION BARRIERS

The isolation barriers have been installed in each discharge chute opening in the 105-K East and 105-K West Basins. The barriers have been fabricated to Safety Class 1 standards from an upgrade of the design originally used to isolate the 105-K East discharge chute in 1979. The installation process included the following:

- Measuring the chute opening dimension
- Relocating 25 canisters of fuel at 105-K West
- Relocating empty canisters and debris at 105-K East
- Pumping sludge at 105-K East
- Cutting fuel rack segments at each discharge chute opening
- Preparing sealing surface at each discharge chute opening
- Installing the isolation barriers at both basins
- Leak testing to verify acceptable installation at each basin.

4.1.1 Chute Opening Measurement

The chute openings were measured at the locations of the Z brackets to verify that the barriers would fit when installed. The measurements were obtained using an underwater camera and a video micrometer. While the measurements were performed, a video was made of the embedded angle at each opening.

4.1.2 Clearing Seal Surface Tooling Work Space

A 1.5-m x 1.5-m (5-ft x 5-ft) working area was cleared in front of each chute opening in both basins to provide sufficient space for the seal preparation tools. This involved relocating the loaded fuel canisters in 105-K West and the empty fuel canisters in 105-K East. This space also provided clearance to limit risk of fuel canister damage during actual installation of the barriers.

Loaded fuel canisters were relocated, using approved procedures, from the 105-K West east and west discharge chute openings. Thirteen canisters were relocated from the 105-K West east chute opening and 11 canisters from the 105-K West west chute opening.

At 105-K East, jackstrawed empty fuel canisters and debris were relocated, using approved procedures, from the front of the discharge chute openings as needed to provide the desired working space for the seal strip installation tool.

4.1.3 Pumping Sludge

After the area in front of the 105-K East openings was cleared of debris, the sludge, which is approximately 1 cm to 7 cm (0.4 in to 2.75 in.) deep, was pumped to the weasel pit in accordance with approved procedures. Air pumps and new suction headers developed for the long-term sludge removal program were used. During this stage the remaining sludge inside the 105-K East discharge chute was also pumped to the weasel pit.

4.1.4 Cutting Fuel Racks

A segment of the fuel racks was removed from in front of the chute opening to make room for the seal preparation tool. The racks were then cut by a remotely operated abrasive wheel saw. The rack saw was precisely placed for each cut and clamped to the rack to remain stationary. The actual cut was controlled remotely and monitored using an underwater camera.

4.1.5 Preparation of Seal Surfaces

The sealing surface at each opening was prepared with a remotely operated tool. Preparation of the surface consisted of the following steps:

- Cutting away the existing inflatable rubber seal
- Removing the existing steel strips bordering the inflatable seal by shearing the cap screws holding the strip
- Cleaning the surface of the embedded angle at the corner of each opening with a paint/rust stripper and a grinding wheel
- Spreading a bead of epoxy along the cleaned angle surface
- Positioning a prefabricated stainless steel sealing strip that is held in place by the epoxy.

The individual tools for performing the separate operations were mounted on a chain-drive carrier that follows a designed path on the seal preparation tool. The tool itself was positioned by hydraulic jacks that wedged the tool into the discharge chute opening.

The seal preparation tool was also equipped with a suction pump/filtration system that contained the debris generated during the cutting operation both above and under the water.

Following leak testing, areas of incomplete epoxy penetration were repaired by injecting epoxy into the voids while maintaining a differential pressure to draw epoxy into the gap.

4.1.6 Isolation Barrier Installation

The isolation barriers were installed by ICF Kaiser Hanford Company. The 1,270-kg (2,810-lb) barriers were rigged from the plant rail system using a 1,800-kg (2-ton) chain fall in accordance with a rigging plan developed by ICF Kaiser Hanford Company. The brackets that hold the barrier from the back are attached with stainless steel tie rods tightened by 2.5-cm (1-in.) lock nuts. Final compression of the J seal was verified by feeler gauge and by using an underwater camera from the discharge chute side.

4.1.7 Leak Testing

The barrier installation was verified for each basin by a leak test involving a differential water level between the basin and the discharge chute and measurement of the change in water level between the discharge chute and the basin. The leak test was performed in accordance with an approved procedure and is capable of detecting a leak rate of 96 L/min (25 gal/min) with minimal uncertainty (Whitehurst 1994a and 1994b). The preliminary test results were satisfactory.

4.2 FACILITY OPERATION

Facility operations with the barriers installed are straightforward and consist of maintaining water level control in the discharge chute and maintaining clearances between the barriers and equipment installed in the basins.

The minimum water level in the discharge chute will be controlled to routinely remain above the top of the opening between the reactor rear face shield wall opening to the chute. The basis for this minimum level is limiting the cross-communication of air spaces. The water level may be lowered below the shield wall opening occasionally for operational exigencies, e.g., repair of construction joint leakage. Lowering of the water level would be procedurally controlled to manage the potential for cross-communication of air spaces.

High water level in the discharge chute is automatically controlled by the weir slot height at 4.95-m (16-ft 3-in) unless the storage basin level is higher. The maximum water height for which the basins have been analyzed is 6.3 m (20 ft 9 in.) (Winkel 1996). Operation of the basins at that height would be structurally acceptable.

The performance of the isolation barriers could be jeopardized by impact from moving objects during a DBE. A 10.2-cm (4-in.) clearance will be maintained between isolation barriers and equipment resting on the basin floor or storage racks. Further, a cone of influence should be maintained around the barriers such that objects that may tip are either restrained or will not impact the barriers. This operational requirement does not preclude movement of fuel or equipment suspended from the monorails past the barriers. Normal management of this is by work control systems, e.g., the Job Control System, and operating procedures. Routine in-service inspections are used to verify that adequate clearances are maintained.

management of this is by work control systems, e.g., the Job Control System, and operating procedures. Routine in-service inspections are used to verify that adequate clearances are maintained.

4.3 IN-SERVICE INSPECTION AND MAINTENANCE OF ISOLATION BARRIERS

This section provides a discussion of the inspection and maintenance procedures that will be used to ensure continued satisfactory performance of the isolation barriers. In-service inspection will be provided for isolation barrier subsystem components that may degrade from exposure to basin environmental conditions. The inspection is one element in ensuring acceptable performance of the barriers over their 15-year design life.

Corrosion has been identified as the only significant age-dependent phenomenon that could reduce the structural margins for the barrier (Strickland 1995). Measures have been taken to preclude corrosion, including seal welding to isolate interior surfaces, painting to protect exterior surfaces, provision of adequate material to accommodate corrosion at 2 to 3 mil/yr for key areas, and use of stainless steel in applications such as the hooks and threaded tie rods that cannot be protected by paint.

4.3.1 Inspection Criteria and Basis

Inspection will be performed either by use of underwater video camera or comparable methods. Dimensions can be verified either by comparison with the dimensions of structural members of the isolation barriers or by use of video micrometry against a standard. Uncertainties of plus 100% are acceptable; therefore, relatively simple inspection methods may be used. Final documentation may be by detailed inspection report, inspection summary and video tape, inspection summary and still photos, or other equivalent methods of documenting the results. Analysis need only be performed if acceptance criteria are exceeded. Table 4-1 contains a listing of inspection criteria and the technical bases.

4.3.2 Frequency

A baseline inspection was performed immediately following installation of the barriers. The inspection was performed in accordance with procedures developed and approved by WHC and included leak tests to verify satisfactory performance of the barrier system. The first followup corrosion inspection will be held within 2 years (not to exceed 2.5 years). The basis for this inspection frequency is that maximum corrosion credible within this period with total paint failure would not significantly degrade any component. Isolation barrier performance, i.e., leakage, will be verified within 1 year of installation and at least annually thereafter.

Further inspections for corrosion may be relaxed to every 4 years if epoxy paint performs as expected.

Table 4-1. Inspection Criteria and Bases.

Component	Inspection criterion	Technical basis
Frame		
Tubing - sides	Bare or corrosion covered spots less than 100 cm ² (15.5 in ²)	Limit corrosion on structural member
Tubing - cross pieces	Bare or corrosion covered spots less than 1 cm ² (0.15 in ²)	Limit corrosion on member seeing maximum stress during empty discharge seismic event
Welds - structural	Bare or corrosion covered spots less than 1 cm ² (0.15 in ²)	Limit corrosion on member seeing maximum stress during empty discharge seismic event
Welds - seal	Bare or corrosion covered spots less than 100 cm ² (15.5 in ²)	Limit seal weld corrosion to prevent water incursion to interior of structure
End plates	Unlimited corrosion	No structural requirements
Side stops	Bare or corrosion covered spots less than 100 cm ² (15.5 in ²)	Limit corrosion structural material
Plate	Unlimited corrosion	Adequate margin to take total corrosion allowance on both surfaces of plate
Weir opening	Unlimited corrosion	No structural requirements
Z Brackets		
Tubing	Bare or corrosion covered spots less than 1 cm ² (0.15 in ²)	Limit corrosion on member seeing maximum stress during empty discharge seismic event
Welds - structural	Bare or corrosion covered spots less than 1 cm ²	Limit corrosion on weld seeing maximum stress during empty discharge seismic event
Welds - seal	Bare or corrosion covered spots less than 100 cm ² (15.5 in ²)	Limit seal weld corrosion to prevent water incursion to interior of structure
End plates	Unlimited corrosion	No structural requirements
Interior guide tubes	Unlimited corrosion	No structural requirements
Interior tubing (painted segment)	Unlimited corrosion	Limited structural requirements - adequate margin to take total corrosion allowance on interior of tube
Tie rod-related structures		
Hook	No inspection required	Stainless
Rod	No inspection required	Stainless
Seal	Inspect for signs of general degradation	Inspection is precautionary only; seal life exceeds 15 years

4.3.3 Acceptance of Corrosion in Excess of Criteria

Allowable corrosion rate assumptions and acceptance criteria for the barrier design were conservatively set during design. Should corrosion in excess of current acceptance criteria be detected at some point, further evaluation against the stress report to identify critical portions of members and critical thicknesses may be used to determine continued acceptability.

4.3.4 Repair Program

If acceptance criteria are exceeded and the acceptance analysis identifies a need for protection, any compatible underwater covering may be used. The in-service inspection program frequency will be reevaluated based on the material used.

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5.0 CONCLUSIONS

The identification of a USQ related to the 105-K Basins seismic design prompted WHC to perform the appropriate reviews and evaluations in accordance with DOE requirements. In responding to the USQ, WHC developed a plan that included both immediate mitigating actions as well as long-term resolution. In developing both the immediate and long-term actions, WHC used accepted engineering methods (value engineering study) to arrive at appropriate and cost-effective solutions (KCM 1994).

The use of isolation barriers as a long-term solution to the USQ has been proven as an effective means of isolating the discharge chute area from the main basin. The design for the isolation barriers has been improved and strengthened from the cofferdam door design as a result of the detailed analysis performed to ensure survival during a DBE and other design loads. Application of techniques such as commercial-grade dedication allowed the process to move forward quickly, without having to scrap work that had already been accomplished, thus saving time and money.

Following installation, testing, and demonstrated satisfactory performance of the isolation barriers, the operation of the 105-K Basins will continue to be within the design basis as defined in the safety authorization basis (Meichle 1995; Besser 1995; Brehm 1995). Until the final installation and testing were complete, interim actions were taken to provide reasonable assurance that mitigating actions could be taken to protect the safety and welfare of the public in the event of a DBE and opening of the construction joint.

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6.0 SUMMARY OF REFERENCE DOCUMENTS

Numerous reference documents listed in Section 7.0 are cited throughout this summary report. Most of these documents were prepared to support resolution of the 105-K Basins seismic leak USQ. This section summarizes information contained in key reference documents. These documents may be retrieved from WHC Central Files or the Hanford Technical Library for detailed review if desired.

6.1 OVERVIEW OF REQUIREMENTS RELEVANT TO UNREVIEWED SAFETY QUESTION RESOLUTION

Section 107(a) of the *Energy Reorganization Act of 1974* authorized the newly created Energy Research and Development Administration (ERDA) to perform certain functions previously set forth in the *Atomic Energy Act of 1954* (AEA). Under the *Department of Energy Organization Act of 1977*, all functions of the ERDA were transferred to the newly formed DOE. In particular, Section 161(b) of the AEA authorized ERDA, and now DOE, to control the use of special nuclear material through "rule(s), regulation(s), or order(s)." The authority given to the DOE under Section 161 is broad and includes all aspects of ensuring the safe management of nuclear facilities authorized by the AEA. Refer to 59 FR 15843 (final rule for 10 CFR 830).

Under this broad authority, the DOE (and its predecessors) has established safety requirements for its nuclear facilities. These requirements are generally contained in DOE orders. The orders provide minimum requirements to be followed in the design and operation of the facilities. The requirements in the various orders are supplemented by other DOE documents, such as directives, notices, and instructions, which contain detailed safety requirements. Augmenting these requirements are safety and implementation guides (which are similar to U.S. Nuclear Regulatory Commission regulatory guides) and technical standards (the DOE's implementing documents for industry codes and standards).

Based on these generally applicable requirements, each contractor negotiates facility-specific safety requirements that are imposed through contract conditions. These requirements include such things as technical specifications and operational safety requirements and are relied on by the DOE to ensure safe management of its nuclear facilities. Together, all of the negotiated design and operational requirements are considered as the facility safety authorization basis. The specific details of the safety authorization basis are typically described in documents such as the interim safety basis document, the facility SAR, the interim operational safety requirements, DOE-issued safety evaluation reports, and other documents containing facility-specific commitments made to comply with DOE orders and policies.

In the case of the 105-K Basins, WHC performed a seismic analysis and leak rate assessment that indicated the projected leakage following an earthquake would likely exceed the current authorization basis established in

the SAR (Meichle 1995). Applying DOE Order 5480.21, *Unreviewed Safety Questions*, WHC performed a safety evaluation to determine whether a USQ existed.

As the initial step, WHC reviewed its authorization basis to determine what requirements might be affected. Under its contract with the DOE, the following orders are specifically applicable (see Appendix A for more detailed discussion):

- DOE Order 5400.5, *Radiation Protection of the Public and the Environment*
- DOE Order 5480.11, *Radiation Protection for Occupational Workers*
- DOE Order 5480.21, *Unreviewed Safety Questions*
- DOE Order 5480.22, *Technical Safety Requirements*
- DOE Order 5480.23, *Nuclear Safety Analysis Reports*
- DOE Order 5480.24, *Nuclear Criticality Safety*
- DOE Order 5480.31, *Startup and Restart of Nuclear Facilities Operational Readiness Review and Readiness Assessments*
- DOE Order 5700.6C, *Quality Assurance*¹
- DOE Order 5700.7C, *Work Authorization System*
- DOE Order 6430.1A, *General Design Criteria*.

In addition, WHC reviewed the facility-specific documents that are part of its authorization basis.² The following documents, and their specific sections, are affected by the USQ resolution:

- WHC-SD-SNF-SAR-001, Rev. 0, *K Basins Safety Analysis Report* (Meichle 1995)
 - Section 2.3 - "Radiological Control"
 - Section 2.5 - "Design Basis Earthquake"
 - Section 5.2.4 - "Discharge Chute Isolation Doors"
 - Section 5.4.1 - "Demineralized Water Supply System"
 - Section 6.8 - "Effects of Normal Operation"
 - Section 8.4.3 - "Loss of Pool Coolant"
 - Section 8.6.2 - "Design Basis Earthquake"

¹This DOE order has been superseded by 10 CFR 830.120; however, the basic requirements of the two documents are identical so there is no substantive change. See 59 FR 15843 and 15847.

²Appendix A contains the specific language of the relevant authorization basis sections cited.

- WHC-SD-NR-TA-020, Rev. 0, *Safety Evaluation of Fuel Encapsulation in the 105-KE Storage Basin* (Schmidt 1992)
 - Section 3.3.3 - "Design Basis Earthquake"
 - Section 3.6.7 - "Clearwell Integrity During DBE"
 - Section 3.6.9 - "Unsupported Basin Construction Joint"
- WHC-SD-WM-OSR-006, Rev. 0, *Operations Safety Requirements 100-KE and 100-KW Fuel Storage Basins* (Bennett 1992)
 - Section 5.1 - "Facility Cooling Pool"
 - Section 3/4.2 - "Bases"
 - Section 5.8 - "Seismology"
- WHC-SD-NR-HC-002, Rev. 0, *Facility Hazard Classification for KE/KW Fuel Storage Basins* (Morton 1991).

Based on this set of affected authorization basis documents, the next step by WHC was to perform a safety evaluation³ as specified in DOE Order 5480.21. This evaluation called for answering the seven questions provided below. If the answer to any one of the questions is "yes," then the proposed activity is considered a USQ, and steps must be taken to bring the facility back into conformance with its authorization basis.

1. Could the proposed activity increase the probability of occurrence of an accident previously evaluated in the safety analysis?
2. Could the proposed activity increase the consequences of an accident previously evaluated in the safety analysis?
3. Could the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety analysis?
4. Could the proposed activity increase the consequence of a malfunction of equipment important to safety previously evaluated in the safety analysis?
5. Could the proposed activity create the possibility of an accident of a different type than any previously evaluated in the safety analysis?
6. Could the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the safety analysis?
7. Does the proposed activity reduce the margin of safety as defined in the basis for any technical safety requirement?

³A safety evaluation is somewhat more limited than a safety analysis (see DOE Order 5480.5). Its purpose is to serve as a benchmark for whether the authorization basis is being preserved. A safety analysis establishes the set of "limiting analyses" important for safe operation.

WHC's safety evaluation demonstrated that a USQ existed. The USQ resolution approach called for the design, fabrication, and installation of isolation barriers to ensure that the authorization basis of the 105-K Basins was maintained. These activities are controlled under the DOE orders previously listed, in particular those dealing with quality assurance, work control, and design criteria. On final approval by the DOE, appropriate revisions to the affected authorization basis documents will be implemented.

6.2 DISCUSSION OF TECHNICAL SUPPORTING DOCUMENTS

Technical aspects of the design modification were evaluated and documented in the engineering process leading to the barrier installation. The documents are summarized in the following subsections.

6.2.1 Functions and Requirements Report

Section 3 of this report discusses the WHC decision to install 105-K Basins isolation barriers. The installation of the barriers involves various tasks such as providing working space in front of the barrier location, measurement and preparation of the sealing surfaces, barrier installation, and leak testing.

Detailed discussion of functions and requirements developed for the performance of the barriers and for the implementation of these tasks is provided in Appendix C.

6.2.2 Stress Analysis Reports

The 105-K East and 105-K West facilities were constructed between 1953 and 1955. Because of the age of these facilities, little historical documentation, such as design calculations, is available to demonstrate their structural adequacy. Also, criteria (especially seismic) have changed over the years such that additional analyses were warranted.

In recent years several seismic evaluations of the 105-K East and 105-K West Basin facilities have been performed to establish seismic resistance of the storage basin boundaries and the reinforced concrete fuel storage basins for an additional 20 years of service (Kanjilal 1994, Winkel 1994, Orbeta 1994, Frier 1994, Winkel 1995).

One analysis, discussed in Section 6.2.2.1, was performed to determine the adequacy of the existing structures and to identify modifications that would bring the facilities into compliance with the appropriate requirements if the analysis did not demonstrate adequate seismic resistance (Winkel 1994).

A second objective of the evaluation was to evaluate the seismic adequacy of the steel-frame superstructures immediately above the storage basins. The steel-frame superstructures are classified as Safety Class 3 facilities, but an evaluation for the Safety Class 1 DBE loading is necessary because of a

three-over-one concern. That is, if a Safety Class 3 superstructure collapses because of a Safety Class 1 DBE, significant damage to the Safety Class 1 basins and fuel canisters might occur.

An additional need for structural qualification arose with the requirement for the addition of the isolation barrier doors to both basins. These doors have also had structural evaluation.

Figure 6-1 summarizes the documentation for the structural qualification of the basin and the barrier isolation doors. The following paragraphs summarize the contents of the noted documents.

6.2.2.1 Analysis of Basin Concrete Substructure. The seismic evaluation documented in WHC-SD-NR-SA-024, *105KE/105KW Irradiated Fuel Storage Basins Seismic Qualifications* (Winkel 1994) evaluated both the Safety Class 1 belowground concrete basins and the Safety Class 3 aboveground superstructures for the 105-K East and 105-K West Basins. A summary of the substructure analysis follows, and a summary of the superstructure analysis is provided in Section 6.2.2.2.

The seismic member loads for the basin substructure were developed with a three-dimensional, soil-structural interaction model using the SASSI computer code. These dynamic member loads were combined with static loads developed with the ANSYS computer code to obtain the total seismic demand. Demand/capacity comparisons for the critical sections of the basins were developed. The demand predictions are less than the capacities at all locations except for a local overstress at the northwest corner of the 105-K West Basin. This local capacity exceedance was not predicted for the 105-K East Basin because the walls in 105-K East are thicker.

The corner bending moments are deformation-limited and are therefore not associated with collapse. Conservatism in the code capacity requirements makes it likely that no actual damage would result from the predicted seismic demand. If structural damage did occur, it would be in the form of local corner cracking in the upper third of the corner, which could result in relatively slow leakage of the cover water. The leakage rate would likely be within the makeup flow capacity. Even without makeup water, the potential water level drop is predicted to remain above the 2.4-m (8-ft) cover requirement stated in the storage basin SAR (Meichle 1995). Recommendations for resolving this corner-cracking issue are provided.

An independent third-party review of this document was provided by Advent Engineering Services, Inc. and issued as WHC document WHC-SD-NR-SA-025, *Independent Review of 105KE/105KW Irradiated Fuel Storage Basin Seismic Qualification Report* (Dyrness and Wang 1992).

Revision 0A to WHC-SD-NR-SA-024 was issued via engineering change notice 191290 to make a minor correction to the calculated building response spectra, and presented calculations to qualify the reactor building north wall against collapse during a seismic event (Winkel 1994).

Figure 6-1. K Basins Structural Analysis.

An independent third-party review of this document was provided by Advent Engineering Services, Inc. and issued as WHC document WHC-SD-N031-SA-004, *Independent Review of 105KE/105KW Irradiated Fuel Storage Basins, Phase II and III, Dynamic Structural Analysis* (Baliga 1994).

A final format summary of the seismic and structural analyses was prepared to document the work described above in an easily accessible fashion (Winkel 1995).

6.2.2.2 Analysis of Basin Superstructure. The superstructure interfaces with the basin substructure through support column attachments at the top of the basin concrete walls. The structure is also coupled to the reactor building through attachments to the reactor building north wall. This structural coupling of the superstructure resulted in the need for a coupled seismic model including the superstructure, concrete basin, and reactor building. Kanjilal (1994) analyzed the design through the development of three separate computer models using the ANSYS computer code: concrete-basin substructure model, steel-frame superstructure model, and reactor/shielding model. Because the focus of the evaluation was on the superstructure, the coupled model size was reduced by creating ANSYS superelements from the basin substructure and reactor building models.

The superstructure seismic loads were generated from the Safety Class 1, 0.2g DBE. The seismic member loads were combined with the static loads to obtain the total seismic demand. Demand/capacity margins of safety were calculated and provided in tabular form. Several local regions of overstress (mainly in the roof support structure) were indicated, but a general collapse of the superstructure is not expected. Recommendations for resolving the local overstress concerns were provided.

The steel superstructure analyses of Winkel (1994) included only the steel superstructure over the basin and none of the adjoining abovegrade steel structure. Therefore, some conservative assumptions were made regarding fixity in the east-west direction. The results indicated a number of overstressed members. Additional analyses were subsequently performed and documented in Kajilal (1994). That analysis included a more detailed model of the entire superstructure, i.e., added the office and transfer bay as a part of the model. The results indicated a significant reduction in stresses in the superstructure.

The primary cause for the reduced stresses was that the structure is more stable because of the addition of east- and west-side structures. The aspect ratio (length over breadth) is therefore reduced, causing a redistribution of stiffness.

In spite of the reduction of overall stresses, some local areas are still overstressed. On the office side, some north-south, long-span beams supporting the distributing east-west beams are overstressed. The allowable stresses in flexure for these beams are small because of the long unsupported length and the unrestrained compression flange. The unrestrained compression flange results in an overstress condition for the dead weight and snow-load combination. This localized overstressing can be overcome by adding local stiffeners at suitable locations and by stiffening one of the bays with

inclined horizontal braces at roof level, connecting the long span beams with the columns and thus restraining the compression flange of the beams and reducing the unsupported length.

Also, four welded connections between the roof girders and the columns on column line 7.1, on the storage-basin side, are overstressed. These joints need additional welds to sustain the seismic loads.

An independent third-party review of this document was provided by Advent Engineering Services, Inc. and issued as WHC document WHC-SD-N031-SA-004 (Baliga 1994).

6.2.2.3 Analyses for the Isolation Doors. Three analyses were performed to support the design and installation of the isolation doors. These analyses are contained in Strickland (1996) and are summarized below.

Positive safety margins exist with the barrier door installed, the discharge chute drained and the basin water depth of 4.95m (16.25 ft) and at an earthquake level of 0.2g. (Winkel 1995).

Strickland (1996) presents the structural analyses performed on the isolation doors. All members are structurally adequate provided that modifications are made to the tie-rod hook, mid-lifting pad, lower lifting pad, and top lifting ring. It should also be noted that the maximum load in the tie rods that were assessed was 692.2 kg (1,526 lb).

Strickland (1996) also contains a report that summarizes the analyses performed to qualify the design changes to the isolation doors. Major change is a redesign of the tie-rod hook. All components of the door are shown to have positive margins of safety. The margins of safety resulting from the structural analysis of the isolation barrier assembly components are shown in Table 6-1. These margins of safety are for barrier components that are affected by the change in J seal compression load addressed in these calculations.

6.2.3 Isolation Barrier Corrosion Analysis

Potential corrosion of the barriers over their lifetime was evaluated to ensure that corrosion would not interfere with their function. The 105-K Basins have an ongoing corrosion monitoring program (Strickland 1995). The coupons used in this program are 1018 low-carbon steel. The corrosion behavior of this material is typical or perhaps more pronounced than the behavior of the carbon steel used in the isolation barriers. Consequently, coupon material corrosion can be used to predict corrosion of the barriers.

Table 6-1. Structural Margin of Safety for 105-K Basins Isolation Barriers
(Tie Rod Load = 1,360.8 kg [3,000 lb]).

Component drawing H-1-80104	Material	Allowable stress (Pa [kip/in ²])	Actual stress (Pa [kip/in ²])	Margin of safety normal loading
Components (no corrosion allowance)				
Door assembly (Item 4) tube steel frame	Carbon steel A500, GR. B	2.1 E+08 (30.4)	1.4 E+08 (20.3)	0.50
Bracket assembly (Items 5 through 14)	Carbon steel A500, GR. B	*	*	0.46
Pad eye (Item 60)	Carbon steel A36	9.9 E+07 (14.4)	2.0 E+07 (2.9)	>2.0
Tie-rod (Items 69 through 87)	Stainless steel 304L or 316L	*	*	0.68
Welds (No corrosion allowance)				
Door assembly tube steel connections	--	1.4 E+08 (21.0)	6.8 E+07 (9.8)	1.14
Bracket assembly welds	--	1.4 E+08 (21.0)	8.7 E+07 (12.6)	0.66

*Margin of safety is based on the interaction of axial load and bending moment or combined bending, and is equal to (allowable stress/actual stress) - 1.

Coupon examinations show uniform corrosion with minimal pitting. With the chemistry control and demineralization currently practiced at the 105-K Basins, the evaluation concluded that a 2- or 3-mil/yr corrosion rate of uncoated carbon steel would be appropriate to consider for the design. This predicted corrosion rate is conservative because the isolation barriers will have a protective coating. Strickland (1996) contains an analysis that indicates that the corrosion rate of unprotected steel rebar in the concrete would be <1 mil/yr.

6.2.4 Isolation Barrier Weir Sizing Analysis

The isolation barriers passively isolate the fuel storage basin water from the location of a potential leak in the discharge chute. The barriers are also designed to provide for water circulation from the chute to the basin and to maintain a common water level for normal operation. This is accomplished by a weir through the barrier to allow normal water inventory control during operation without interference from the presence of the isolation barriers. The weir will provide for water transfer between the two areas for normal conditions when the water level in the chute and the basin is higher than the weir. In the unlikely event of a leak in the discharge chute large enough to reduce the water level, the level in the basin will not drop below the weir. Below that level the passive isolation function becomes operable.

The weir must be sized to allow for future operation at higher water levels such that, in the unlikely event of a major leak in the chute area, the basin water level reaches the weir elevation of 4.95 m (16 ft 3 in.) above the basin floor before the chute drains. This requirement supports the seismic aftershock analysis. A higher water level in the basin with the chute empty would increase the stress on the separating wall (island) between the basin and the chute.

The analysis contained in WHC-SD-NR-CN-001, *Calculations for Overflow from Basin to Discharge Chute* (Smith 1994) examined the water volume transfers and time for different opening sizes. The conclusion was that a flow area size of 146 cm² (22.6 in²) would provide sufficient water transfer to meet the requirement. This flow area is the total for both basin barriers. The weirs were designed conservatively and provide 205 cm² (32 in²) of flow area for each basin by using a single weir, 41 cm by 5 cm (16 in. by 2 in.). One weir is located in a barrier in each basin.

6.2.5 Seal Performance Analysis

The isolation barrier seal design was established after the design objectives were considered and a number of alternatives were evaluated. Details of the seal performance analysis are provided in WHC-SD-SNF-ER-002, *K-Basin Isolation Barrier Seal* (Ruff 1994), and included the following performance objectives:

- Compressibility to conform to the surface of the basin wall
- Material suitable for a 15-year service life suitable to the water chemistry and the radiation field
- A configuration that fit the design characteristics of the basin wall and the discharge chute opening
- A seal design that could be tested after the barriers were installed and periodically during its design lifetime
- A seal with previous applications with similar environments
- A seal that would retain sufficient compression after a period of service following initial installation to minimize the need for later recompression.

The seal design selected is a compressible J cross-section seal. This is a hollow bulb seal made of ethylene propylene, with ≈ 2.5 cm (≈ 1 in.) diameter of internal air space. The uncompressed external diameter is ≈ 4.4 cm (≈ 1.75 in.).

The J seal is mounted to the barrier by a stainless steel clamping strip and bolts. The clamping strip holds the J seal in the proper position against the barrier to ensure that, when the barrier is positioned and the tie rods are tightened, the seal is in the proper position to be compressed against the sealing strip. The basin sealing surface is a 12.7-cm- (5-in-) wide strip that begins 5.5 m (18 ft) above the basin floor, drops to the bottom, crosses the bottom of the opening, and extends up 5.5 m (18 ft) to the other side of the opening. The seal attached to the barrier fits against that sealing area. (see Figures 6-2 and 6-3). The seal compression will be limited by the thickness of the seal clamping strip. With the barrier tightened to the point where the clamping strip is close to the wall surface, the seal will be compressed to $\approx 41\%$.

Figure 6-2. Isolation Barrier with J Seals in Place.

Figure 6-3. Isolation Barrier Showing Detail of J Seal.

Subjects that were investigated to decide on a seal that satisfied the listed criteria were as follows:

- Various alternate seal concepts
- Consideration for lateral motion
- Compressibility characteristics and forces
- Relaxation and fatigue
- Resistance to abrasion and side forces
- Determination of the physical dimension of the openings.

Alternative seal designs considered were a flat seal and a double J seal. The flat seal has less capability to conform to variations in the seal surface. The double J seal is easier to test for leaks but is too wide for the sealing surface.

Lateral motion during a seismic event was considered undesirable because the movement could alter the relationship between the seal and the sealing strip or cause abrasion of the seal, either of which could lessen the integrity of the seal. The decision was made to allow for small initial movement but to limit the movement using mechanical stops. The movement was limited to that which would allow the sides of the flexible seal to flex laterally but would not move the seal relative to the seal surface. This was accomplished by using a lateral motion limiter (see Figure 6-4).

Nineteen rods provide the compressive force on the seal. They are attached to the seal face of the barrier and when tightened by the nuts, hold the barrier seal against the sealing strip and compress the J seal. Compression tests were conducted and the tie rods were designed to accommodate the forces resulting from 41% seal compression. That is enough to allow the J seal to relax later, while accomplishing its function without being retightened during its design life.

To ensure that the barriers fit into the current openings, the deviation from plumb of the two vertical surfaces of each opening was determined. The openings were found to be uniform and to provide a suitable fit-up within the ability of the flexible seal.

6.2.6 Isolation Barrier Leak Rate Test Methodology and Results

A post-installation leak test has been performed to verify that the isolation barriers control leakage from the basin to the chute within the limit specified in the SAR. The test was also used to generate baseline data (Whitehurst 1994a, 1994b). The test equipment was designed to provide rapid data reduction for realtime decision making; the field results are documented in the test procedure and summarized in Table 6-2.

Note that the seal leak rate safety analysis-based acceptance criteria were met for the isolation barriers as installed. Continued action is underway to further reduce the isolation barrier leak rates. This action is judged appropriate to further reduce potential environmental impacts of both post DBE and potential operational leakage. When the final modifications are made, the final leakrate will be documented in the test reports and validated by a formal acceptance test review.

Figure 6-4. Isolation Barrier Showing Lateral Motion Limiter.

Detail Showing Lower Side Stops Which Restrain Sideways Motion During Seismic Events

H9409021.11

Table 6-2. Post-Installation Leak Test Results.

		Acceptance criterion	105-K West actual	105-K East actual
Turbulent	Maximum flow at 27.94 cm (11 in.)	3.78 E-01 L/s (6 gal/min) at 27.94 cm (11 in.)	2.71 E-01 L/s (4.3 gal/min)	3.09 E-01 L/s (4.9 gal/min)
	Maximum flow at 4.88 m (16 ft)	(1,500 gal/h) at 4.88 m (16 ft)	3,948 $\pm$ 95 L/h (1,043 $\pm$ 25 gal/h)	4,259 $\pm$ 95 L/h (1,125 $\pm$ 25 gal/h)
	Coefficient	Less than 0.55 within two standard deviations	0.491	0.524
Laminar component correction	Maximum flow range	3.78 E-01 to 7.57 E-02 L/s (6.0 to 1.2 gal/min) (adjust for coefficient)	Not required	Not required

During the leak test, the water level in the chute was lowered relative to the water level in the basin, which generated a hydraulic head across the barriers. The basin water level was recorded over a period of time. The basin water leak rate is calculated by the water volume change over time. Because the water level can be influenced by evaporation, condensation, and temperature changes, these effects have been determined and compensated for in determining the leak rate.

The differential head established between the discharge chute and the basin for the test is limited by operational considerations. Water level measurements have been made over a small range of testable differential heights. However, in the postulated seismic leak, the differential height could increase to its maximum ($\approx$ 4.95 m [16.25 ft]) if the chute were to empty. To determine a leak rate for this hypothetical condition, a curve-fitting technique based on the available data has been used to apply the test results to leakage anticipated with full height differential across the barriers. The methodology is presented in detail in McCracken (1994).

6.3 MATERIAL PROCUREMENT

Ensuring the suitability of material procured and used in fabrication of the 105-K Basins isolation barriers was an important task in the USQ resolution. This section describes the approach used to ensure suitability of the materials for use in the Safety Class 1 application of the 105-K Basin isolation barriers. Detailed documentation that the materials used are of adequate quality is provided in Sorenson et al. (1995).

6.3.1 Procurement Methods

Two methods of material procurement for use in the Safety Class 1 applications are allowed under WHC procedures:

1. Safety Class 1 procurement from a supplier currently on the Evaluated Suppliers List for the materials being procured

2. Procurement of commercial-grade material, and dedication of that material for use in its specific Safety Class 1 application.

These two methods of procuring material for safety-related applications meet the requirements of DOE Order 5700.6C (or 10 CFR 830.120). The requirements and procedures are equivalent to those used in the commercial nuclear power industry.

As part of the USQ resolution process, a review was performed to evaluate the feasibility of using the commercial-grade dedication method at the Hanford Site generally, and for procurement of the material for the 105-K Basins isolation barriers specifically. The results of this review indicated that use of the commercial-grade procurement and dedication option was feasible under DOE regulatory requirements and implementing WHC procedures (Sorenson et al. 1995). A detailed summary of this review is provided in Appendix B.

As noted in Section 3.6, two sets of isolation barriers were fabricated for use at the 105-K Basins. The modified set consists of two existing cofferdams, originally fabricated under Safety Class 2 controls, which were evaluated and then modified under Safety Class 1 controls to meet the requirements of the isolation barrier final design. The new set of isolation barriers was fabricated entirely under Safety Class 1 controls.

Both the Safety Class 1 method and the commercial-grade method of material procurement were used on both sets of isolation barriers. The procurement histories for the modified and new sets of barriers are discussed separately.

6.3.2 Procurement of Material for the Modified Set of Isolation Barriers

The modified set of isolation barriers include materials procured as commercial grade during fabrication of the cofferdams, materials procured as commercial grade to support modification of the cofferdams, and materials procured as Safety Class 1 to support modification of the cofferdams. Each of these materials is discussed in the following subsections.

6.3.2.1 Material Procured as Commercial Grade During Fabrication of the Existing Cofferdams. Before reaching a decision to modify the cofferdams, a thorough review of their fabrication and material records was performed to determine the feasibility of such an approach. The fabrication of the cofferdams was controlled to Safety Class 2 standards. The review was to determine whether adequate controls had been applied such that, when augmented with additional quality controls, adequate verification would exist that the materials used were of suitable quality for use in the isolation barriers. The key questions were whether material control and traceability to sources of supply had been maintained, and whether fabrication practices were commensurate with Safety Class 1 requirements. The review of the fabrication records for the cofferdams indicated the following.

- The fabrication effort had been controlled using the same procedures that would have applied if the work had been controlled as Safety Class 1.

- All steps in the fabrication effort were fully documented in the same manner as if the work had been Safety Class 1.
- Welder qualification and weld records were maintained in the same manner as if the work were Safety Class 1.
- Traceability was maintained on all materials used in fabrication, with the specific purchase orders used to obtain the material identified for each part.
- Quality control during fabrication generally met practices used for Safety Class 1 work, with the only significant exception being in the weld inspections performed.

Based on the results of the record review, a decision was made to evaluate the specific materials used and establish dedication requirements for use of these materials in the 105-K Basins isolation barriers. The main steps in this evaluation were as follows:

- Determining which materials were used in parts that perform safety functions in the final isolation barrier design, and which materials were only used in non-safety related parts
- Performing commercial-grade item dedication evaluations, which identified the critical characteristics that, when verified, ensure that the materials are of adequate quality for their intended functions
- Obtaining and reviewing available quality documentation that provided assurance that the materials used were acceptable
- Identifying additional tests and inspections required to augment the available documentation to provide adequate assurance that the materials were acceptable.

In general, the critical characteristics that apply to the isolation barriers are of two basic types: fabrication related, such as material of proper dimensions (including angular dimensions where specified), and material related, such as chemical composition and physical strength of material. Dimensional measurements were specified in the dedication evaluations as documented in Sorenson et al. (1995) and were verified during fabrication of the isolation barriers.

Verification of chemical and physical composition of the materials can be obtained either by obtaining mill test reports for the material used, or by performing testing and analysis of material samples from the as-built structure. Use of mill test reports (when available) is preferred, as it avoids the need to select sample locations and evaluate the impact of removing such samples from the as-built structure. For this reason, a review of procurement records both in WHC files and from the material suppliers was performed to determine whether mill test reports traceable to the materials used to fabricate the cofferdams were available. As is often the case, these records were available in the supplier's files, even though they had not been required in the WHC purchase orders. Results of the physical and chemical

test reports obtained were compared to the requirements of the applicable American Society of Testing and Materials specifications identified in the isolation barrier design documents, and found to be within acceptable ranges in all cases. Copies of these mill test reports are included in Sorenson et al. (1995).

6.3.2.2 Material Procured as Commercial Grade During Modification of the Existing Cofferdams. In several cases, materials required to modify the existing cofferdams to conform to the final isolation barrier design could not be procured as Safety Class 1. In these cases, dedication evaluations were prepared, and necessary tests and inspections were conducted to verify that the items were of adequate quality for their intended Safety Class 1 applications. Dedication evaluations for these items and associated procurement records are included in Sorenson et al. (1995).

6.3.2.3 Material Procured as Safety Class 1 During Modification of the Existing Cofferdams. Wherever possible, materials used to modify the existing cofferdams were procured as Safety Class 1 from approved suppliers. Items procured as Safety Class 1 for use in the modified isolation barriers are identified in Sorenson et al. (1995). Related procurement and receipt inspection records are also included.

6.3.3 Procurement of Material for the New Set of Isolation Barriers

Materials for the new set of isolation barriers were procured as Safety Class 1 wherever possible. All structural steels that form the isolation barrier and the Z brackets that secure it in place were procured as Safety Class 1. Items procured as Safety Class 1 and the associated procurement, receipt inspection, and fabrication records are included in Sorenson et al. (1995).

Some of the safety-related parts of the new isolation barriers were procured as commercial-grade items. Commercial-grade dedication evaluations were prepared for these items and identify tests and inspections required to ensure adequate quality. Procurement records, receipt inspection records, and commercial-grade dedication evaluations for these items are provided in Sorenson et al. (1995).

Materials used in non-safety related applications on the isolation barriers were procured as commercial items, and do not require completion of dedication evaluations. Justification for classification of these items as non-safety related are included in Sorenson et al. (1995).

6.3.4 Conclusion

Evaluation of the materials used in fabrication of both the modified and new isolation barriers resulted in the determination that all materials are of adequate quality for use in the 105-K Basins isolation barriers.

6.4 REPORT ON FABRICATION OF ISOLATION BARRIERS

This section discusses the fabrication process for the isolation barriers.

6.4.1 Summary of Fabrication Effort

The isolation barriers were fabricated at the 2704 West shops at the 200 West Area by ICF Kaiser Hanford Company.

6.4.1.1 Material Procurement. The material specifications were translated into purchase requisitions, which were reviewed by the cognizant engineering and quality assurance organizations as part of the procurement process.

On receipt, the material was inspected for compliance with the purchase order by Procurement Quality Support. Records of receipt inspection are retained by Procurement Quality Support and in the work package records set. Material staged for fabrication was verified as that specified before being released to the process.

6.4.1.2 Isolation Barrier Assembly Hierarchy. Each basin isolation barrier consists of the barrier, the Z brackets and tie rods, and the fasteners that hold them together. When installed, the barrier assemblies are interlocked with and press against the concrete structure and reliably isolate the discharge chutes from the main storage basin.

The components of a barrier assembly are as follows:

- Frame: made from welded sections of 8.9-cm x 8.9-cm x 0.64-cm (3.5-in. x 3.5-in. x 0.25-in.) wall square, A36 square tubing. The interior of the tubing is closed to water. The frame provides rigidity to the plate.
- Plate: made from 1.3-cm (0.5-in.) A36 flat plate. The plate fills, and is sealed to, the concrete structure that forms a passway between the storage basin and the discharge/pickup chutes.
- Vertical lifting bails: Made of 2.5-cm (1-in.) flat plate welded and gusseted to the frame.
- Pad eyes: made of 1.3-cm (0.5-in.) A36 steel, they are welded to the plate and function as the attachment point for the tie rod hooks.
- Lateral motion limiters: made from 5.1-cm- (2-in.-) thick 304 stainless steel and welded to the plate, they limit the possible side-to-side movement of the barrier within the passway during a seismic event.
- J seal and mounting hardware: consist of the rubber J seal assembly, the threaded, 1.3-cm (0.5-in.) stainless steel studs welded to the plate that will hold the seal in place, and the stainless steel clamping strips and retaining nuts.

Each barrier assembly has 10 Z brackets. Each bracket is a welded assembly of 8.9-cm x 8.9-cm x 0.64-cm- (3.5-in. x 3.5-in. x 0.25-in.) wall square steel tubing, with the interior of the tube closed to water entry.

Each pad eye level of a barrier has one bracket. The brackets stack on top of each other and bear on the opposite side of the concrete wall from the barrier. Each bracket has tie rod connection points that match the pad eyes on the barrier at that level. The bottommost bracket has one tie rod connection to adapt to the horizontal run of the J seal at the bottom of the barrier.

Tie rods secure the barrier assembly to the concrete passway structure by connecting the eye pad with a corresponding Z bracket. Each tie rod is loaded in tension, and it is the pulling force of 19 tie rods that secures the isolation barrier to the concrete.

A tie rod consists of 2.5-cm- (1-in.-) diameter stainless steel round rod welded to 2.5-cm- (1-in.-) thick stainless steel plate stock with a hook shape. The hook shape engages the opening of a barrier pad eye. The overall length of a tie rod varies from 1.8 m to 0.46 m (5.83 ft to 1.5 ft), the length being dictated by location of the tie rod in the Z bracket. The outside 14 cm (5.5 in.) of each rod is threaded.

Spacers, stainless steel washers, and a self-locking nut make up the remainder of a finished tie rod subassembly.

6.4.2 Barrier Assembly Fabrication

The following isolation barrier fabrication sequence was used to construct the barriers:

1. Build the frame: Square tubing was cut to length to make up the pieces for the frame. The pieces are clamped to the welding table and tack welded together. Upon verification of dimensions, the joints were finish welded, inspected, and the surfaces to butt against the plate were ground flush.
2. Install lifting bails: Tubing cut-outs per drawing were made, assembled and tack welded. The dimensions were verified, finish welded, and inspected when completed.
3. Cut out and attach plate: The plate was cut from 5.5-m x 110-cm (18-ft x 43-in.), 1.3-cm- (0.5-in.-) thick plate and straightened as required to fit flush to the frame. The frame on the cover was aligned and clamped together on the weld table. The components were then tack welded together, the dimensions were verified, and the assembly finish welding was completed.
4. Attach pad eyes and lateral motion limiters: The eye parts were cut out in accordance with drawing specifications, the barrier turned over to place the plate surface facing up. The pad eye locations

were identified and each eye tack welded. The dimensions were verified and finish welded. The assembly was inspected and then deburred.

5. Weir: The location of the weir was identified, cut out, and deburred.
6. Seal weld/isolate tube interiors: End caps were cut out and welded to the open tube ends per drawing specifications.
7. Attach seal retaining studs: The location of the seal retaining studs was identified and the studs located, verified, welded in place, and inspected.
8. Clean and paint: All surfaces were then sand blasted and painted using epoxy to the thickness specified on the drawing.
9. Attach J seal: Holes on the J seal were then cut out to match stud layout, the seal clamping strips were installed with washers and lock nuts, and the nuts were torqued to the specified amount.

6.4.3 Fabrication of the Z Bracket Assemblies

The Z brackets were fabricated as follows:

1. Cut out components in accordance with drawing: Structural components were cut from square tube stock and pad eyes were cut from flat stock to the dimensions specified on the drawing and the dimensions were verified.
2. Assemble and weld tube parts: The tubing parts were assembled in a tubing alignment fixture and tack welded together. The dimensions were verified, the alignment checked, and the assembly finish welded and inspected.
3. Locate and weld on pad eyes: The location of the pad eye on the bracket was identified, the eye tack welded into position, the dimensions verified, and the eye finish welded and inspected.
4. Seal weld/isolate tube interiors: Square tubing end caps were fabricated and welded to the open tube ends as required by the drawing.
5. Balance bracket assemblies: The balance plate hidden surfaces were then seal welded as required.
6. Clean and paint: All accessible surfaces were then sand blasted, cleaned, and painted in accordance with drawing requirements. The Z bracket assemblies were then inspected.

6.4.4 Fabrication of Tie Rod Hooks

The tie rods were fabricated in the following sequence:

1. Cut rounds to preliminary length, thread one end: 2.5-cm (1-in.) round stock was cut to length, threads machined on one end, and inspected.
2. Cut out hooks: The hook periphery was cut using a torch while leaving adequate material for finish to machined dimensions.
3. Machine hook: The hook was then machined to drawing dimensions and inspected.
4. Final cut: The unthreaded end of the round was then cut so that the rod/hook assembly matched the length specified on the drawing.
5. Weld hook to rod: The hook/rod pieces were assembled in an alignment fixture and tack welded into position, the dimensions were verified, and the pieces finish welded together and inspected.

6.5 RECORDS

The finished work packages were reviewed by the shop and then by Quality Assurance for completeness and compliance to requirements. The reviewed packages were then sent to Records Retention at the 712 Building. Records of procurement and fabrication are also retained in the 712 Building and are retrievable by work package identification number.

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- DOE Order 5480.31, *Startup and Restart of Nuclear Facilities Operational Readiness Review and Readiness Assessments*, U.S. Department of Energy, Washington, D.C.
- DOE Order 5700.6C, *Quality Assurance*, U.S. Department of Energy, Washington, D.C.
- DOE Order 5700.7C, *Work Authorization System*, U.S. Department of Energy, Washington, D.C.
- DOE Order 6430.1A, *General Design Criteria*, U.S. Department of Energy, Washington, D.C.

APPENDIX A
REVIEW OF UPPER TIER REQUIREMENTS

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LIST OF TERMS

AEA	<i>Atomic Energy Act of 1954</i>
DOE	U.S. Department of Energy
ERDA	Energy Research and Development Administration
SAR	safety analysis report
WHC	Westinghouse Hanford Company

APPENDIX A

REVIEW OF UPPER TIER REQUIREMENTS

A1.0 INTRODUCTION

On August 4, 1977, pursuant to the *Department of Energy Organization Act of 1977* (Public Law 95-91), all functions of the Energy Research and Development Administration (ERDA) as well as the functions of the Federal Energy Administration and Federal Power Commission were transferred to the U.S. Department of Energy (DOE). Under Section 107(a) of the *Energy Reorganization Act of 1974* (Public Law 93-438), the ERDA had been authorized to perform certain functions specified in the *Atomic Energy Act of 1954* (AEA). In particular, Section 161(b) of the AEA authorized ERDA, and now DOE, to control the use of special nuclear material through "rule(s), regulation(s), or order(s)." The authority given to the DOE under Section 161 is broad and includes all aspects of ensuring the safe management of nuclear facilities authorized by the AEA. Refer to FR 15843 (final rule for 10 CFR 830).

Based on its authority under the AEA, the DOE has promulgated regulations and orders that establish the requirements under which contractors carry out their obligations.

A1.1 FORMATION OF DOE FROM ERDA

On August 4, 1977, pursuant to the *Department of Energy Organization Act of 1977*, all functions of ERDA as well as the functions of the Federal Energy Administration and Federal Power Commission were transferred to the DOE. Selected responsibilities of the Interstate Commerce Commission and U.S. Departments of Navy, Interior, Commerce, and Housing and Urban Development were also transferred to the DOE. Section 107(a) of the *Energy Reorganization Act of 1974* authorized ERDA to perform functions specified in the AEA (e.g., Section 161). Section 161(b) of the AEA allows for control of the use of special nuclear material to be accomplished through "rule(s), regulation(s), or order(s)."

A1.2 CODE OF FEDERAL REGULATIONS

Section 161 of the AEA provides DOE with broad authority to carry out its responsibilities under the AEA, including the safe management of facilities authorized by the AEA. This authority is the basis for all the rules concerning the safe management of nuclear facilities that DOE intends to adopt in 10 CFR 830 (59 FR 15843).

- Section 830.3 — Definitions

Implementation Plan — a document prepared by a contractor that sets forth:

1. When and how the actions appropriate to comply with the requirements of a section of this Part, including the requirements of a plan or program required by the section, shall be taken, and
2. What relief will be sought if a contractor cannot attain full compliance with a requirement in a reasonable manner.

NOTE: An implementation plan would be, for example, the quality assurance program.

- Section 830.120 — Quality Assurance Requirements: includes procurement, design, records, etc.
- Subparts B, Design, and C, Operations, are reserved for promulgation later.

A1.3 DOE ORDERS

Two DOE orders are pertinent to this summary report for design and installation of isolation barriers: DOE Order 5480.21, *Unreviewed Safety Questions*, and DOE Order 5480.22, *Technical Safety Requirements*.

A1.3.1 DOE Order 5480.21, *Unreviewed Safety Questions*

Section 6 of the Preamble contains several definitions, reproduced in the following list, which have a bearing on events leading to preparation of this report.

1. **Authorization Basis.** Those aspects of the facility design basis and operational requirements relied upon by the DOE to authorize operation. These requirements are considered to be important to the safety of facility operations. The authorization basis is described in documents such as the facility safety analysis report (SAR) and other safety analyses; hazard classification documents, the technical safety requirements, DOE-issued safety evaluation reports, and facility-specific commitments made in order to comply with DOE orders or policies.
2. **Nonreactor Nuclear Facility.** Those activities or operations that involve radioactive and/or fissionable materials in such form and quantity that a nuclear hazard potentially exists to the employees or the general public. Included are activities or operations that: (1) produce, process, or store radioactive liquid or solid waste, fissionable materials, or tritium; (2) conduct separations operations; (3) conduct irradiated materials inspection, fuel fabrication, decontamination, or recovery operations; (4) conduct

fuel enrichment operations; or (5) perform environmental remediation or waste management activities involving radioactive materials. Incidental use and generation of radioactive materials in a facility operation (e.g., check and calibration sources, use of radioactive sources in research and experimental and analytical laboratory activities, electron microscopes, and X-ray machines) would not ordinarily require the facility to be included in this definition. Accelerators and reactors and their operations are not included. The application of any rule/order to a nonreactor nuclear facility shall be applied using a graded approach.

3. **Safety Analysis.** See DOE Order 5480.5, *Safety of Nuclear Facilities*.
4. **Safety Analysis Report.** See DOE Order 5480.6, *Safety of Department of Energy-Owned Nuclear Reactors*.
5. **Technical Safety Requirements.** Those requirements that define the bounding conditions for safe operation, and bases thereof, and the management or administrative controls required to ensure the safe operation of a nuclear facility. (Westinghouse Hanford Company [WHC] is currently in transition from compliance with the safety analysis requirements established in DOE Order 5480.5 to compliance with DOE Order 5480.22 and DOE Order 5480.23. This change is not complete and 105-K Basins fuel storage activities are currently [January 1995] in accordance with operational safety requirements derived from WHC-SD-SNF-SAR-001 [Meichle 1995]. An interim safety basis, which includes updated interim operational safety requirements [based on requirements established in DOE Order 5480.22] has been submitted to DOE for review and approval.)

Section 10 of DOE Order 5480.21, "USQ Program Requirements," establishes the fundamental elements required of an unreviewed safety question program. The program consists of (1) development of facility safety evaluations based on facility design configuration and activities performed in the facility and (2) evaluation of configuration and activity changes on the analyzed risk. An evaluation to determine if an unreviewed safety question exists is defined by the DOE as follows:

When a contractor identifies a situation that indicates a potential inadequacy of previous safety analyses or a possible reduction in the margin of safety, the following basic actions, among others, are required:

- Conduct a safety evaluation to determine if a USQ exists
- If a USQ exists, take action to place facility in safe condition pending resolution
- Prepare a resolution plan
- Obtain approval for resolution from Program Office

- Implement resolution plan
- Revise authorization basis document(s) to reflect resolution.

The body of the order contains implementation guidance in Chapter III. Section 3 of that chapter discusses "Understanding the Authorization Basis". Essential aspects are as follows.

1. Safety analyses are intended to define those aspects of design and operations that are important to safety and therefore those aspects that DOE relies upon to allow initial and continued operation.
2. The authorization basis of a facility may not be reflected in total in the current facility SAR. This basis, depending upon the facility, may reside in several different type of documents. These may include not only the facility SAR, but historical commitments made by contractors to support modifications and the imposition of new DOE requirements or administrative changes. These may also include DOE safety evaluation reports that modify contractor-proposed changes or analyses.
3. The contractor must define that population of documents comprising the various elements of the authorization basis and must use this defined population of documents as the basis for performing safety evaluations under the requirements of this order.

A1.3.2 DOE Order 5480.22, *Technical Safety Requirements*

Section 6 of DOE Order 5480.22 contains definitions, some of which have particular importance to 105-K Basins activities, including installation of the isolation barriers. As discussed earlier, 105-K Basins activities are currently controlled by operational safety requirements prepared in accordance with DOE Order 5480.5. Interim operational safety requirements, which are part of the interim safety bases, have been prepared and forwarded to DOE for review and approval. They were developed using DOE Order 5480.22 as the basis and are essentially identical to technical safety requirements in both form and content. Some of the definitions from the DOE Order 5480.22 are as follows:

- A. **Safety Analysis.** Documented processes: (1) to provide systematic identification of hazards within a given DOE operation, (2) to describe and analyze the adequacy of the measures taken to eliminate, control, or mitigate identified hazards; and (3) to analyze and evaluate potential accidents and their associated risks.
- B. **Safety Analysis Report (SAR).** A report which documents the adequacy of safety analysis for a nuclear facility to ensure that the facility can be constructed, operated, maintained, shut down, and decommissioned safely and in compliance with applicable laws and regulations.

- C. **Technical Safety Requirements.** Those requirements that define the conditions, safe boundaries, and the management or administrative controls necessary to ensure the safe operation of a nuclear facility and to reduce the potential risk to the public and facility workers from uncontrolled releases of radioactive materials or from radiation exposures due to inadvertent criticality. A technical safety requirement consists of safety limits, operating limits, surveillance requirements, administrative controls, use and application instructions, and the basis thereof.

A2.0 REFERENCES

A2.1 ACTS, CODES, AND STANDARDS

Atomic Energy Act of 1954, 42 USC 2011, et seq.

Department of Energy Organization Act of 1977, 42 USC 7101 et seq.

Energy Reorganization Act of 1974, Public Law 93-438.

10 CFR 830.120, "Quality Assurance Requirements," *Code of Federal Regulations*, as amended.

59 FR 15843, 1994, "Nuclear Safety Management," *Federal Register*, Vol. 59, p. 15843 (April 5) (to be codified as 10 CFR 830).

A2.2 DOCUMENTS

Meichle, R. H., 1995, *K Basins Safety Analysis Report*, WHC-SD-SNF-SAR-001, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

A2.3 U.S. DEPARTMENT OF ENERGY ORDERS

DOE Order 5480.5, *Safety of Nuclear Facilities*, U.S. Department of Energy, Washington, D.C.

DOE Order 5480.6, *Safety of Department of Energy-Owned Nuclear Reactors*, U.S. Department of Energy, Washington, D.C.

DOE Order 5480.21, *Unreviewed Safety Questions*, U.S. Department of Energy, Washington D.C.

DOE Order 5480.22, *Technical Safety Requirements*, U.S. Department of Energy, Washington, D.C.

DOE Order 5480.23, *Nuclear Safety Analysis Reports*, U.S. Department of Energy, Washington, D.C.

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APPENDIX B
DEDICATION OF EXISTING ISOLATION BARRIERS

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5	ELECTRIC POWER RESEARCH INSTITUTE REPORTS	B-59

LIST OF TERMS

DBE	design basis earthquake
DOE	U.S. Department of Energy
EPG	Engineering Practice Guide
EPRI	Electric Power Research Institute
ESL	evaluated suppliers list
GL	Generic Letter
MTR	mill test record
NRC	U.S. Nuclear Regulatory Commission
NUMARC	Nuclear Management and Resources Council
PNL	Pacific Northwest Laboratory
QA	quality assurance
QAP	quality assurance program
QI	Quality Instruction
SSC	systems, structures, and components
USQ	unreviewed safety question
WHC	Westinghouse Hanford Company

APPENDIX B

DEDICATION OF EXISTING ISOLATION BARRIERS

B1.0 INTRODUCTION

B1.1 BACKGROUND

Following the evaluation of the dynamic seismic analysis results and leak rate assessment for the spent fuel storage basin at the 105-K Basins, Westinghouse Hanford Company (WHC) determined that previous conclusions about the integrity of the 105-K Basins following a design basis seismic event did not fully envelope all potential accident scenarios.¹ Specifically, the safety analysis report for the 105-K Basins establishes an upper limit leak rate at 5,680 L/h (1,500 gal/h) (Meichle 1995). The recent seismic analysis indicates that during a design basis earthquake (DBE) the construction joint located in the discharge chute, an area adjacent to the main spent fuel basin, could fail and cause leakage estimated at between 18,930 L/h and 94,635 L/h (5,000 gal/h and 25,000 gal/h). Even with makeup water capability, as assumed under the safety analysis report accident analysis, this leakage rate significantly challenges the capability of maintaining fuel coverage.

Consistent with U.S. Department of Energy (DOE) requirements, WHC determined that the seismic analysis raised an unreviewed safety question (USQ). Subsequently, WHC developed an action plan for resolution of the USQ. Among the possible solutions, WHC chose to use isolation barriers (also referred to as cofferdams) to seal the two openings between the main spent fuel basin and the discharge chute. This resolution method involved fabrication of two sets of barriers (one set for 105-K East and one for 105-K West), each capable of surviving a DBE.

One set of isolation barriers had previously been fabricated for the purpose of facilitating repair of operational leaks. The barriers were purchased and built to Safety Class 2 standards, which means they were not designed to survive the Safety Class 1 DBE, i.e., an earthquake having a horizontal acceleration of 0.2g. Because the basin structure has been defined as Safety Class 1, the permanent isolation barriers must also be designed and fabricated to Safety Class 1 standards.² WHC performed an extensive design review of using isolation barriers and determined this approach to be an appropriate resolution of the USQ. Starting from this design position, the question is whether the quality of the materials and the fabrication/installation work is considered acceptable.

¹WHC considers a seismic event of low probability, but the damage of such an event of potentially high consequence.

²Safety class is defined in WHC-CM-1-3, *Management Requirements and Procedures*, MRP 5.46, "Safety Classification of Systems, Components, and Structures," Westinghouse Hanford Company, Richland, Washington.

WHC contracted with Pacific Northwest Laboratory (PNL), to prepare a document that provided reasonable assurance of the qualification of the existing isolation barriers for use in a Safety Class 1 application. Assisting PNL in the effort was a team that included representatives from Winston & Strawn, for legal and regulatory matters, and Sequoia Consulting Group, Incorporated, for technical matters.

Based on technical and financial considerations, as well as timeliness, WHC wanted to utilize a strategy that had a high potential for success when submitted to DOE for review. Therefore, the initial stage of the work involved determining whether the commercial-grade dedication methods used in the commercial nuclear reactor industry would be applicable to the USQ resolution plan. If not applicable, the alternative was to purchase and fabricate new isolation barriers to Safety Class 1 standards. Alternately, if applicable, WHC believed that use of the reactor industry standards had a high probability of acceptance by DOE. Furthermore, application of the commercial-grade dedication methods would result in cost savings and timely resolution of the USQ.

A review of the dedication methods used in the commercial reactor industry, as specifically endorsed by the U.S. Nuclear Regulatory Commission (NRC), showed them to be state-of-the-art. Based on this position, the PNL team concluded that these methods would be applicable to the USQ resolution plan. This conclusion was confirmed during a preliminary analysis of the design and fabrication requirements for the isolation barriers.

WHC agreed with the PNL team's recommendation to proceed with commercial-grade dedication of the isolation barriers consistent with NRC standards. The following text discusses the methodology used to dedicate the isolation barriers.

B2.0 DISCUSSION

To properly dedicate the isolation barriers for use in a Safety Class 1 application, particularly in a noncommercial nuclear reactor setting, several basic questions must be addressed.

The first question is: What are the governing regulatory requirements? Because WHC is a management and operations contractor for the DOE, the procurement methods set forth in the applicable DOE regulations, orders, and WHC implementing procedures must be identified.³

The second question is: Are the applicable DOE regulatory requirements and WHC implementing procedures consistent with commercial-grade dedication standards routinely employed in the commercial reactor industry and endorsed by the NRC? Past focus by the federal government on procurement activities in the commercial aviation and nuclear power fields has resulted in significantly

³There may be other independent requirements that relate to this matter, such as through the WHC contract with DOE. However, an evaluation of these requirements is not within the scope of this task.

enhanced commercial-grade dedication practices in these industries. Because of the clear similarities between the commercial reactor and the DOE weapons reactor industries, the techniques and standards endorsed by the NRC provide the best template for dedicating the isolation barriers.

The final question is: On applying the NRC "standards," do the findings support dedication of the isolation barriers? The answer to this question depends on the amount of objective evidence available concerning the quality of the materials and fabrication activities involved, e.g., vendor certifications or test results.

Each of these questions is addressed in order in the following subsections.

B2.1 APPLICABLE REQUIREMENTS

The control of purchased items and services is a basic element of quality assurance (QA) programs for the design, construction, operation and decommissioning of nuclear facilities.⁴ As such, the design and material procurement processes are regulated under the substantive QA requirements contained in 10 CFR Part 830. Sections 830.120(a) and (b) explicitly require that DOE contractors responsible for nuclear facilities develop and submit for approval to DOE a quality assurance program (QAP) which contains, among other things, provisions for ensuring the proper procurement of items and services used at DOE nuclear facilities.

Specifically, Section 830.120(c)(2)(iii) states that:

"Procured items and services shall meet established requirements and perform as specified. Prospective suppliers shall be evaluated and selected on the basis of specified criteria. Processes to ensure that approved suppliers continue to provide acceptable items and services shall be established and implemented."

This provision includes commercial-grade procurement activities⁵ and, along with several other provisions (i.e., Sections 830.120(c)(1)(iv) and 830.120(c)(2)(iv) involving documentation and acceptance testing, respectively), forms the base requirements for control of the overall material procurement process.

In the final rule promulgating 10 CFR Part 830 (59 FR 15843), DOE contractors were required to submit their QAPs by November 1, 1994. The QAPs should be limited to a description of the proposed QA activities, with the details to be contained in the contractors' implementing procedures and

⁴See ASME NQA-1-1989, *Quality Assurance Program Requirements for Nuclear Facilities* (ASME 1989). This industry standard is endorsed as an acceptable means for complying with DOE QA requirements. Refer to 59 FR 15843 and 15849 (April 5, 1994); see also DOE implementation guide for 10 CFR Part 830.120, "Quality Assurance," G-830.120, Rev. 0 (April 15, 1994).

⁵59 FR 15848.

policies. Until a contractor submits a new QAP, however, the existing QAP remains in effect. These existing QAPs are implemented pursuant to DOE Order 5700.6C, *Quality Assurance*. Although 10 CFR 830.120 is intended to supersede DOE Order 5700.6C, there should be no substantive change in contractors' existing QAPs because the basic requirements are identical.⁶

In this regard, WHC has implemented WHC-CM-4-2, *Quality Assurance Manual* (effective date May 31, 1991), to meet the requirements of DOE Order 5700.6C. Although the QA manual contains a number of sections that address various aspects of the overall material procurement process, Quality Instruction (QI) 7.5, "Control of Commercial Grade Items," specifically addresses commercial-grade procurement. In addition, WHC-IP-1026, *Engineering Practice Guidelines*, in Engineering Practice Guide (EPG) 5.3 (May 27, 1994), supplements the guidance in the WHC QA manual. These two documents combine to form the WHC implementing requirements for the commercial-grade dedication of the isolation barriers.

B2.2 CONSISTENCY OF WHC IMPLEMENTING REQUIREMENTS WITH NRC REGULATIONS AND INDUSTRY GUIDANCE

Appendix B to 10 CFR Part 50 establishes the basic QA requirements for safety related systems, structures, and components (SSCs) used in commercial nuclear power plants regulated by NRC. Appendix B sets forth these requirements in the form of criteria. Criteria III, IV, VII, VIII and XV of Appendix B are applicable to the control of purchased SSCs.⁷ However, the process of dedication, i.e., assuring that an item meets its design requirements and can therefore perform its intended safety function, is principally controlled by the first four criteria listed.

From the standpoint of commercial-grade dedication, Criterion III concerns design control and involves the process of determining the safety function and classification of an item. Criterion IV concerns document control and involves the process of assuring that the design specifications are properly translated into procurement documents, such as purchase orders. Criterion VII is largely directed at assuring the traceability of an item from the original equipment manufacturer, through a distributor if applicable, to the ultimate end-user. And finally, Criterion VIII involves the control of the item while being warehoused and during the processes of fabrication and installation. (Attachment 1 contains the text of the applicable Appendix B criteria.)

To satisfy the requirements of Appendix B, QA activities such as commercial-grade dedication must be shown to provide "adequate confidence" that an SSC will perform satisfactorily while in service. As recognized by the Atomic Safety and Licensing Appeal Board, this language means that proper Appendix B implementation requires "simply a finding of reasonable assurance that, as built, the facility can and will be operated without endangering the

⁶59 FR 15847.

⁷54 FR 9229 (March 6, 1989).

public health and safety."⁸ The "reasonable assurance" standard implements the overarching statutory mandate to "provide adequate protection to the health and safety of the public" as required pursuant to Section 182(a) of the AEA (42 USC Section 2232(a)).⁹

Of course, what constitutes "reasonable assurance" is within the discretion of the NRC to decide depending on the particular circumstances at the time. In the context of commercial-grade procurement, however, NRC Inspection Procedure 38703 (Attachment 2) provides some insight. Specifically, the procedure states that "reasonable assurance" consists of the "purchaser controlling or verifying the activities affecting the item's quality to an extent consistent with the item's importance to safety or ensuring that these activities are adequately controlled by the supplier."

In the context of commercial-grade dedication, the critical activities affecting an item's quality include:

- Design process (includes evaluation of safety function)
- Translating design requirements to procurement specifications
- Specifying critical characteristics
- Acceptance process, i.e., utilizing methods to reasonably ensure that the item specified is the item received
- Receipt inspection
- Fabrication
- Installation and post-installation testing.

NRC has provided specific regulatory guidance concerning the control of these critical activities for SSCs used in safety related applications. This guidance is made available in a variety of ways, e.g., regulatory guides, generic communications, and technical reports. While a number of these guidance documents relate to the commercial-grade dedication process, the principal guidance on applying the above referenced Appendix B criteria in this context is contained in NRC Generic Letters 89-02 (Attachment 3) and 91-05 (Attachment 4), as well as in NRC Inspection Procedure 38703. Although not regulatory requirements themselves, these guidance documents provide the NRC's view of what is an acceptable means of complying with the applicable Appendix B requirements.

Generic Letter (GL) 89-02 in turn discusses relevant industry guidance established by the Electric Power Research Institute (EPRI). The GL conditionally endorses EPRI Report NP-5652, *Guidelines for the Utilization of*

⁸See, e.g., Wisconsin Electric Power Co. (Point Beach Nuclear Plant, Units 1 and 2), ALAB-739, 18 NRC 335, 346 (1983).

⁹Carstens v. Nuclear Regulatory Commission, 742 F.2d 1546, 1557 (1984).

*Commercial Grade Items in Nuclear Safety Related Applications (NCIG-07).*¹⁰ Specifically, GL 89-02 states that those who "use methods similar to those described in EPRI NP-5652 . . . to verify the critical characteristics of commercial-grade items intended for safety-related applications have the basis for effective dedication programs. . . . Properly implemented, the EPRI guidelines . . . [as modified by GL 89-02] establish methods which satisfy existing requirements of Appendix B." Therefore, EPRI NP-5652 forms the backbone of the NRC's regulatory guidance for commercial-grade dedication.

NRC GL 91-05 builds on GL 89-02 by providing further guidance on several key elements of the dedication process. This guidance is furthered by NRC Inspection Procedure 38703. The procedure explicitly indicates that it is "consistent" with the nuclear industry's comprehensive procurement initiative adopted in July 1990. This initiative was developed in concert with the nuclear utilities by the Nuclear Management and Resources Council (NUMARC).

The NUMARC initiative, documented in NUMARC 90-13, references several other EPRI reports: (1) EPRI NP-6629, *Guidelines for the Procurement and Receipt of Items for Nuclear Power Plants (NCIG-15)* and (2) EPRI NP-6630, *Guidelines for Performance Based Audits*. While not specifically referenced by NUMARC, EPRI has developed additional guidance as a supplement to those reports already mentioned. This guidance is principally contained in EPRI Reports NP-6895, NP-6406, NP-5652 and TR-102260. Each of these EPRI guidance documents apply to the overall commercial-grade dedication process. For the task at hand (determining whether the isolation barriers meet their technical requirements), only the guidance included in Reports NP-6895, NP-6406, NP-5652 and TR-102260 apply.

EPRI Report NP-6895, *Guidelines for the Safety Classification of Systems, Components and Parts Used in Nuclear Power Plant Applications (NCIG-17)*, contains guidance for determining which portions of a nuclear power facility's equipment are to be considered safety related, and therefore subject to QAP requirements applicable to SSCs. EPRI Report NP-6406, *Guidelines for the Technical Evaluation of Replacement Items in Nuclear Power Plants (NCIG-11)*, provides guidance for establishing technical and quality requirements for replacement items. EPRI Reports NP-5652 and TR-102260, *Supplemental Guidance for the Application of EPRI Report NP-5652 on the Utilization of Commercial Grade Items*, contain specific guidance on the process by which items normally available in the commercial market can be procured and approved for use in nuclear safety related applications.

The EPRI guidance was compared with QI 7.5 and EPG 5.3. In general, the details of the guidance are not incorporated into these documents. However, the documents do follow the basic guidance for commercial-grade dedication embodied in EPRI NP-5652. Therefore, the existing WHC procedures provide an adequate basis for the dedication of commercial-grade items. (A more detailed discussion of the EPRI guidance documents is provided in Attachment 5.)

¹⁰The GL makes only minor modifications to the guidance concerning two of the four acceptance methods (i.e., Methods 2 and 4) discussed in EPRI NP-5652.

B2.3 DEDICATION OF 105-K BASINS ISOLATION BARRIERS

The discussion below provides a summary of the dedication activities for the isolation barriers.

The first major step in the dedication process was to gather all available technical and procurement information concerning the isolation barriers. Consistent with the PNL team's industry experience, and the guidance in NRC Inspection Procedure 38703, various procurement documents (such as drawings, bills of material, and fabrication records) applicable to the 105-K Basins isolation barriers were obtained through a review of the WHC records.

This gathered information was used in the second step, preparing preliminary dedication evaluations for each of the unique structural parts used in constructing the barriers. In preparing these preliminary evaluations, the safety function of the isolation barriers was compared with the procurement information gathered. The safety function of the isolation barriers is to prevent or limit leakage from the main spent fuel basin in case of a DBE. To provide this function, the isolation barriers must maintain structural integrity and maintain an adequate seal at the surface between the barriers and the main fuel basin wall following a DBE and aftershocks.

The third step in the dedication process was to identify the critical characteristics for each unique part. Basically, the isolation barriers and their associated support brackets are fabricated from plate and tubular carbon steel materials, with stainless steel tie rods serving as fasteners to secure the overall structure in place.

Where possible, similar parts were grouped into a single evaluation package based on part function and material construction. Since EPG 5.3 did not include any forms for use with the procedure, the PNL team developed appropriate forms to document the dedication evaluations.

The critical characteristics which apply to the isolation barrier parts are of two basic types: (1) fabrication-related, such as material of proper dimensions (including angular dimensions where specified); and (2) material-related, such as chemical composition and physical strength of material.

In principle, all of the required critical characteristics for each part of the isolation barriers could be verified through test and inspection activities. However, exclusive use of testing would be extensive, time-consuming and costly, in particular since a significant number of samples of the actual materials would be required. Further, the testing would require substantial engineering input and review to ensure that the removal of samples from the affected parts allowed the isolation barriers to remain within design limits.

As an alternative that is consistent with the guidance contained in NP-5652 and commercial nuclear industry practice, a more detailed review of the fabrication and procurement records was undertaken to determine the extent to which existing documentation could be relied on to provide the required objective evidence of material acceptability. The following provides a

summary of the PNL team's findings for this final step, that is, determining whether the acceptance criteria are met for the two categories of critical characteristics.

B2.3.1 Fabrication-Related Critical Characteristics

A review of the fabrication records for the isolation barriers indicated that this activity was, with the exceptions noted below, performed under quality controls which were essentially identical to those used for fabrication of Safety Class 1 SSCs at WHC. The records document the (1) materials used (including traceability to the procurement documents), (2) dimensional requirements and tolerances, (3) weld locations, (4) type of welds, (5) weld materials used, and (6) the in-process and/or post-work quality control inspections performed.

The specific exceptions are as follows:

- Verification of threading machined into the tie rods during fabrication was waived. Each tie rod was subsequently inspected to verify threading is as specified in the design.
- Verification of dimensions on the barrier assembly pad eyes was documented at an insufficient level of detail. Additional dimensional checks of the as fabricated pad eyes on the barrier assembly was performed.
- Weld inspections performed during fabrication of the barriers consisted of visual inspection only. All final welds on the barriers and associated brackets were inspected using penetration testing, ultrasonic testing, or another test method approved for use on Safety Class 1 welds of this type.

B2.3.2 Material-Related Critical Characteristics

A review of the fabrication records for the 105-K Basins isolation barriers indicated that all of the safety related structural steel materials were procured on two requisitions to Pacific Steel. Pacific Steel was contacted and they were able to provide mill test records (MTRs) for all of the material provided on the two requisitions. The MTRs were reviewed and determined to be applicable to the material supplied. In addition, the results of the physical and chemical tests performed by the mills who produced the materials were compared to the applicable American Society of Testing and Materials specifications identified in the isolation barrier design documents. All of the results were found to be within acceptable ranges.

Pacific Steel is on the WHC evaluated suppliers list (ESL) for structural steels of the type provided, and has been continuously since 1989. Pacific Steel verified in writing that the two subject requisitions were procured and controlled under procedures used for Safety Class 1 purchase orders. These procedures are the same ones reviewed and approved by WHC as part of the process to place Pacific Steel on the ESL.

Based on the documentation gathered from Pacific Steel, there appears to be adequate objective evidence to demonstrate that the materials provided by Pacific Steel are of acceptable quality and meet the specified design requirements. Therefore, no further testing of these materials was required.

B3.0 CONCLUSION

A thorough review of the existing documentation indicates that the parts comprising the isolation barriers are of adequate quality to be used in the proposed Safety Class 1 application. This quality provides reasonable assurance that the isolation barriers can perform the intended safety function under a DBE.

B4.0 REFERENCES

B4.1 ACTS, CODES, ORDERS, AND REQUIREMENTS

ASME, 1989, *Quality Assurance Program Requirements for Nuclear Facilities*, ASME NQA-1, American Society of Mechanical Engineers, New York, New York.

Atomic Energy Act of 1954, 42 USC 2011, et seq.

DOE Order 5700.6C, *Quality Assurance*, U.S. Department of Energy, Washington, D. C.

10 CFR 830.120, "Quality Assurance Requirements," *Code of Federal Regulations*, as amended.

54 FR 9229, 1989, "Acceptance of Products Purchased for Use in Nuclear Power Plant Structures, Systems, and Components," *Federal Register*, Vol. 54, p. 15843 (March 6) (10 CFR 50).

59 FR 15843, 1994, "Nuclear Safety Management," *Federal Register*, Vol. 59, p. 15843 (April 5) (to be codified as 10 CFR 830).

59 FR 15849, 1994, "Nuclear Safety Management," *Federal Register*, Vol. 59, p. 15849 (April 5) (to be codified as 10 CFR 830).

B4.2 CONTROLLED MANUALS

WHC-CM-1-3, *Management Requirements and Procedures*, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-4-2, *Quality Assurance*, Westinghouse Hanford Company, Richland, Washington.

B4.3 DOCUMENTS

Meichle, R. H., 1995, *K Basins Safety Analysis Report*, WHC-SD-SNF-SAR-001, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

WHC-IP-1026, *Engineering Practice Guidelines*, Westinghouse Hanford Company, Richland, Washington.

ATTACHMENT 1

QUALITY ASSURANCE CRITERIA

Appendix B to 10 CFR Part 50 establishes the basic quality assurance requirements for safety related structures, systems, and components (SSCs) used in commercial nuclear power plants regulated by NRC. Appendix B sets forth these requirements in the form of criteria. Criteria III, IV, VII, VIII and XV of Appendix B are applicable to the control of purchased SSCs.¹ However, the process of dedication, i.e., assuring that an item meets its design requirements and can therefore perform its intended safety function, is principally controlled by the first four criteria listed.

From the standpoint of commercial-grade dedication, Criterion III concerns design control and involves the process of determining the safety function and classification of an item. The text of Criterion III is provided below.

III. DESIGN CONTROL

Measures shall be established to ensure that applicable regulatory requirements and the design basis, as defined in Section 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to ensure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Measures shall also be established for the selection of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components.

Measures shall be established for the identification and control of design interfaces and for coordination among participating design organizations. These measures shall include the establishment of procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces.

The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. The verifying or checking process shall be performed by individuals or groups other than those who performed the original design, but who may be from the same organization. Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, it shall include suitable qualifications testing of a prototype unit under the most adverse design conditions. Design control measures shall be applied to items such as the following: reactor

¹54 Fed. Reg. 9229 (March 6, 1989).

physics, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for in-service inspection, maintenance, and repair; and delineation of acceptance criteria for inspections and tests.

Design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design and be approved by the organization that performed the original design unless the applicant designates another responsible organization.

IV. DOCUMENT CONTROL

Criterion IV concerns document control and involves the process of assuring that the design specifications are properly translated into procurement documents, such as purchase orders. And finally, Criterion VIII involves the control of the item while being warehoused and during the processes of fabrication and installation.

IV. PROCUREMENT DOCUMENT CONTROL

Measures shall be established to ensure that applicable regulatory requirements, design bases, and other requirements which are necessary to ensure adequate quality are suitably included or referenced in the documents for procurement of material, equipment, and services, whether purchased by the applicant or by its contractors or subcontractors. To the extent necessary, procurement documents shall require contractors or subcontractors to provide a quality assurance program consistent with the pertinent provisions of this appendix.

VII. CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES

Measures shall be established to ensure that purchased material, equipment, and services, whether purchased directly or through contractors and subcontractors, conform to the procurement documents. These measures shall include provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished by the contractor or subcontractor, inspection at the contractor or subcontractor source, and examination of products on delivery. Documentary evidence that material and equipment conform to the procurement requirements shall be available at the nuclear power plant or fuel reprocessing plant site prior to installation or use of such material or equipment. This documentary evidence shall be retained at the nuclear power plant or fuel reprocessing plant site and shall be sufficient to identify the specific requirements, such as codes, standards, or specifications, met by the purchased material and equipment. The effectiveness of the contractors shall be assessed by the applicant or designee at intervals consistent with the importance, complexity, and quantity of the product or service.

VIII. IDENTIFICATION AND CONTROL OF MATERIALS, PARTS, AND COMPONENTS

Measures shall be established for the identification and control of materials, parts, and components, including partially fabricated assemblies. These measures shall ensure that identification of the item is maintained by heat number, part number, serial number, or other appropriate means, either on the item or on records traceable to the item, as required throughout fabrication, erection, installation, and use of the item. These identification and control measures shall be designed to prevent the use of incorrect or defective material, parts, and components.

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ATTACHMENT 2

**NRC INSPECTION PROCEDURE 38703,
COMMERCIAL GRADE DEDICATION,
11/08/93**

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NRC INSPECTION MANUAL

RV18

INSPECTION PROCEDURE 38703

COMMERCIAL GRADE DEDICATION

PROGRAM APPLICABILITY: 2515

SALP FUNCTIONAL AREA: ENGINEERING (SOETS-0)

38703-01 INSPECTION OBJECTIVES

01.01 To determine whether the failure of a safety-related system, structure, component (SSC), or part to perform its intended safety function was the result of a deficient commercial grade item (CGI) dedication process.

01.02 To verify that the licensee's process for dedicating CGIs, as implemented, meets the applicable portions of Appendix B to 10 CFR Part 50 and provides reasonable assurance that CGIs will perform their intended safety function.

38703-02 INSPECTION REQUIREMENTS

02.01 Reactive Inspection Requirements

- a. Initial Evaluation. After reviewing the licensee's evaluation of the failed item, determine if the failed item was procured as a CGI and dedicated for safety-related applications. If the failed item was dedicated, review the complete procurement and dedication records to determine if the commercial grade dedication process was sufficiently thorough.
- b. Further Assessments. If it is determined that the dedicated item failed as the result of certain critical characteristics not being identified and/or properly accepted, the inspector should perform the following assessments:
 1. Determine if other CGIs from the same accepted lot or batch as the failed dedicated CGI have been similarly dedicated and installed in other safety-related applications. If yes, determine if the licensee has evaluated the operability of the systems or components where these CGIs are installed. The inspector also should review licensee-provided data, if available, for some CGIs (non-dedicated) that failed in applications that were not safety-related. Explore the possibility that the same CGIs also may have been used (following dedication) in a safety-related application and may have the potential to affect the safe operation of a SSC.

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2. If possible select and evaluate, as in step 1 above, at least three other dedicated CGIs having similar applications and critical characteristics as the CGI(s) that resulted in the identified failures.
3. If, after performing step 2 above, it is determined that there were weaknesses in the commercial grade dedication process, the inspector should perform a more comprehensive inspection of the licensee's dedication process in accordance with the inspection requirements in Section 02.02 below.

02.02 Programmatic Inspection Requirements

- a. Review of Program and Procedures. Using the inspection guidance contained in Section 03.02 and Appendix A to this procedure, review the licensee's program and procedures for the procurement and dedication of CGIs in order to understand the basic operation of the licensee's program.
- b. Selection of Dedication Packages. Select approximately 20 dedication packages for evaluation from a list of commercially dedicated items provided by the licensee. Request that the licensee provide (or make available for review) a complete package of the pertinent procurement and dedication records for each item.
- c. Evaluation of Dedication Packages. Using the inspection guidance contained in Section 03.01 of this procedure, perform a detailed evaluation of the dedication packages selected in item b above.
- d. Evaluation of Training Effectiveness. If the inspector's evaluation of commercial grade dedication activities indicates there are weaknesses in the way these activities are being performed, the inspector should investigate further to determine if weaknesses within the licensee's training program may have contributed to the cause. The inspector should determine if the licensee is implementing an effective training program.

38703-03 INSPECTION GUIDANCE

GENERAL GUIDANCE

Background. Licensees are required to ensure the quality of items purchased and installed in safety-related applications. In the past, licensees procured major assemblies from approved vendors who maintained quality assurance (QA) programs pursuant to Appendix B to 10 CFR Part 50. Because of the decrease in the number of qualified nuclear-grade vendors, licensees are increasing the numbers of commercial grade replacement parts that they procure and dedicate for use in safety-related applications.

Since commercial grade dedications have increased in number, the Nuclear Regulatory Commission (NRC) has developed this inspection procedure to provide guidance to assist the inspector in assessing the effectiveness of the implementation of the licensee's commercial grade procurement practices and provide for early identification of any adverse trends or emerging problems.

The Vendor Inspection Branch, of the NRC's Office of Nuclear Reactor Regulation, is available to assist with specific questions that arise during the performance of this procedure.

Scheduling the Inspection. This inspection procedure should be considered for implementation when there is reason to believe that the failure of a SSC or part to perform its intended safety function was the result of weaknesses in CGI dedication. This inspection procedure may be implemented independently or it may be used as a supplement to other major team inspections. Such inspections may include maintenance, modification, or system-specific inspections where review of failed SSCs or parts is appropriate, or an augmented inspection team investigating failures.

The NRC should contact the appropriate NRC and licensee personnel to schedule the inspection. When practical, inform the licensee of the objectives of the inspection 4-6 weeks before the inspection is to begin and advise them of information that will be needed, such as a list of items that the licensee purchased as commercial grade after July 1990 and subsequently dedicated for use in safety-related applications. Before the beginning of the onsite inspection, the inspector should request and review the licensee's program and procedures to become familiar with the licensee's procurement and dedication process. Also explore with the licensee the possibility of obtaining a list of recent component failures. Request this list only if the licensee states this type of information would be easily retrievable. The list of component failures can be used by the inspector during the selection of dedication packages for review described in Section 02.02 of this inspection procedure.

This inspection procedure is consistent with the Nuclear Management and Resources Council (NUMARC) initiative for improving the utilization of CGIs in nuclear safety-related applications that was implemented in July 1990. The methods used to commercially dedicate items procured by licensees before that date will not necessarily meet the guidance contained in this inspection procedure. If the inspector encounters a significant failure of a commercially dedicated item, which was dedicated before July 1990, the inspector may review the dedication of that item with the understanding that the licensee was not expected to meet the current guidelines.

SPECIFIC GUIDANCE

03.01 Reactive Inspection

- a. Initial Evaluation. A failure resulting from general weaknesses in the commercial grade dedication program may occur when the important design, material, and performance characteristics that are necessary to provide reasonable assurance that the dedicated CGI will perform its intended safety function are not addressed during dedication. For example, failures of safety-related bolting have occurred when the dedication process did not verify that the material composition and/or mechanical properties met the specified requirements and nonconforming material was supplied.

Review and discuss with licensee personnel the failure/root-cause analysis when required or applicable for the failed CGI. The inspector should attempt to determine if the failure was due to a design deficiency, failure unrelated to the item's safety function, or normal wear, and eliminate these from further review. The inspector should focus on the inspection of failures that appear to be due to weaknesses in the commercial grade dedication process. If none of the failures are due to weaknesses in the commercial grade dedication process, then the inspector should not continue using this inspection procedure. If the inspector decides to change the focus of the inspection to examine other issues related to the failures, such as the adequacy of corrective

actions, other procedures should be used, such as NRC Inspection Procedure 92720, "Corrective Action." Once the failure mode and cause of failure have been postulated or determined, review the dedication package as described in Section 03.01a(1) to determine if appropriate critical characteristics had been identified by the licensee. Appendix A to this inspection procedure should not be interpreted as inspection requirements but only as a discussion of dedication issues including guidance on selection and verification of critical characteristics. Appendix A, if properly implemented, represents an acceptable means of complying with regulatory requirements. Individual licensee's may develop alternate methods of achieving Appendix B compliance. Appendix B provides definitions of terms used for commercial grade dedication activities, and Appendix C provides the typical contents of a dedication package.

The goal of the review of the dedication packages is to provide reasonable assurance that the CGIs dedicated for safety-related applications will perform their intended safety functions. Inspection effort should be directed towards the identification of weaknesses in the dedication process that could potentially render SSCs or parts inoperable. When reviewing licensee's operability determinations for dedication of CGIs, the inspector should refer to the "Technical Guidance" section of NRC Inspection Manual, Part 9900, for further guidance.

a(1) Review of Dedication Packages. After becoming familiar with the licensee's procurement and dedication program and procedures, perform a detailed review of the dedication package as described below.

- Determine if the safety function of the item for its intended use has been identified by reviewing the documents associated with the technical evaluation including, as applicable:
 - classification of the item
 - consideration of credible failure modes
 - item equivalency/substitution evaluations
- Determine if the important design, material, and performance characteristics relevant to the safety function have been identified. Determine whether the licensee verified the characteristics necessary to provide reasonable assurance that the item will perform its intended safety function. If appropriate, take into account post-installation testing and periodic surveillance testing and inspection. Review the basis for engineering judgment when it is used as the basis for selection or verification of critical characteristics.
- Determine whether the item is an equivalent replacement or a new item replacement of an obsolete item.
- Determine if the item is or may be used in a different safety-related application than previously evaluated in which different design, material, and performance characteristics may be applicable. This is especially applicable for generic dedications of bulk items and stock material. Determine if the dedication ensures those design, material, and performance characteristics relevant to the safety function.

- Determine why the item is being replaced. Have there been repeated failures? Is the degraded performance a result of adverse environment? Did it fail because it was a refurbished or fraudulent item? General information on similar activities subject to Appendix 8 to 10 CFR Part 50 is provided in American National Standards Institute (ANSI) ANSI N45.2-1977, "Quality Assurance Program Requirements for Nuclear Power Plants," Section 17, "Corrective Action."
- Determine how the identity of the item is controlled from the time it is receipt inspected until the time it is installed. General information on similar activities subject to Appendix 8 to 10 CFR Part 50 is provided in ANSI N45.2-1977, Section 9, "Control of Parts and Components."
- Determine if information learned during the dedication process is fed back to the appropriate persons to evaluate existing stock items, or installed items, and for future use in surveys and source verifications. This information could include positive and adverse findings obtained during surveys and source verifications. General information on similar activities subject to Appendix 8 to 10 CFR Part 50 is provided in ANSI N45.2.13-1976, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants," Section 9, "Corrective Action."

Refer to the discussion of significant dedication issues in Appendix A for guidance during the review of dedication packages. Also refer to the specific guidance for each of the four dedication methods provided below.

Focus should be on those activities that are likely to affect the performance of the items being dedicated. It is not necessary to review the licensee's programmatic compliance to the 18 criteria of Appendix 8 to 10 CFR Part 50 as they may not apply to the activities reviewed. Appendix 8 to 10 CFR 50 does not apply to commercial grade activities which occur prior to dedication for use in a safety-related system. It also should be recognized that this appendix provides for the application of QA to safety-related systems and components consistent with their importance to safety (graded quality approach).

Although guidance concerning the application of graded quality assurance is discussed in the first paragraph of Appendix A to this inspection procedure, it is expected that the inspector will need to exercise considerable judgment in determining the adequacy of controls applied to a specific activity.

The following are the four acceptance methods that can be used to accept CGIs. These methods provide, either individually or in combination, a means to reasonably ensure that a CGI that is received meets the requirements of the item specified. The results of employing each method should be documented.

Method 1 - Special Tests and Inspections

General information on similar activities subject to Appendix 8 to 10 CFR Part 50 is provided in ANSI N45.2.13-1976, Section 10, "Acceptance of Item or Service." Use the following approach to review packages that were dedicated using this method:

- To the extent practicable, attempt to witness receipt inspections and tests of in-process dedication of CGIs that are similar to that of the failed item to verify the identified critical characteristics.
- Review receiving records and associated tests and inspections.
- Review post-installation test records.
- Verify that the tests and inspections specified for acceptance adequately verify the identified critical characteristics.
- Verify that sampling plans are controlled and have adequate technical basis, considering lot traceability and homogeneity, complexity of the item, and adequacy of supplier controls.
- Verify that CGI receiving inspection activities are adequately controlled under a quality program regardless of whether they are being performed in conjunction with other plant receipt inspection activities.
- Verify that receipt inspection activities establish and maintain traceability of CGIs by capturing and appropriately relating traceability documents through identification and monitoring of CGIs.
- Verify that measuring and test equipment was properly calibrated, that approved vendors were used to perform tests, and that personnel were qualified to perform the tests.

Method 2 - Commercial Grade Survey

Use the following guidance to review packages that were dedicated using this method:

- Determine if the guidance of Generic Letter 89-02, or an appropriate alternate, is included in the appropriate procedures. Specifically, confirm that (1) the documented commercial quality program was effectively implemented and (2) the surveys were conducted at the location necessary to verify that adequate controls were exercised on distributors as well as manufacturers.
- Through interview, determine if the persons who perform vendor surveys are knowledgeable in the following:
 - the use of performance-based surveys
 - screening third-party surveys
 - processing and evaluating adverse findings resulting from the review of third-party surveys to ascertain if those findings affect CGIs already installed or stored in the warehouse awaiting future installation

General information on similar activities subject to Appendix B to 10 CFR Part 50 is provided in ANSI N45.2.12-1977, "Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants," and ANSI N45.2.23-1978, "Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants."

- Verify that the supplier's commercial quality controls are imposed in the procurement documents.
- Determine if the critical characteristics that are to be verified by the survey team are accurately and completely incorporated in the survey plans.
- Determine if the validity of supplier documentation, relied on in the dedication of the item, is verified during the survey.
- Determine if surveys of commercial grade suppliers are performance based as opposed to programmatic. Specifically, verify that the critical characteristics for the CGIs being surveyed are controlled by the supplier's quality activities.
- Determine if survey teams include technical and quality personnel, as appropriate, that are knowledgeable in the operation of the item(s) and the associated critical characteristics to be verified, including any special processes such as welding and heat treatment that are specific to the critical characteristics.
- Determine if surveys are conducted at appropriate times relative to the procurement. Are surveys required to be updated on a regular basis to support dedication?
- Determine if the control of subvendors is adequately addressed by the surveys so that the supplier has an adequate basis to accept test results and certifications from the subvendor.
- Determine if pertinent information about a supplier or its products is used to plan, conduct, and report results of surveys and source verifications. Such information could have been available from source verifications, receiving inspections, the dedication process, supplier/product performance history, or outside sources such as NRC information notices and bulletins, nuclear plant reliability data system reports, or Nuclear Utility Procurement Issues Committee (NUPIC) commercial grade survey reports.

Method 3 - Source Verification

General information on similar activities subject to Appendix 8 to 10 CFR Part 50 is provided in ANSI N45.2.13-1976, Section 10.3.2, "Acceptance by Source Verification." Use the following approach to review packages that were dedicated using this method:

- Determine if source verifications involve witnessing the supplier performing quality activities on the actual items being procured and adequately verify the item's critical characteristics.
- Determine if personnel who participated in the source verification surveys were qualified for their specific assignment.
- Determine if appropriate hold points are imposed in the purchase orders. This would include a hold point to verify design, material, and performance characteristics relevant to the safety function that cannot be verified after the item has been completely manufactured.

- Determine if the results of the source verifications were adequately documented.

Method 4 - Acceptable Supplier/Item Performance Record

Use the following guidance to review packages that were dedicated using this method:

- Determine if the guidance of Generic Letter 89-02, or an appropriate alternate, has been incorporated. Specifically, (1) the established historical record is based on industry-wide performance data that is directly applicable to the item's critical characteristics and the intended safety-related application and (2) the manufacturer's measures for the control of design, process, and material changes have been adequately implemented as verified by survey (multi-licensee team surveys are acceptable).
- Determine if information pertinent to the CGI's quality of performance, obtained from outside sources (e.g., operational event reports, NRC, vendor equipment and technical information program, and Institute of Nuclear Power Operations) and from commercial grade surveys, source verifications, receipt inspections, previous dedication or qualification and operational history, is factored into the dedication process.
- Determine if the item or manufacturer is included in the licensee's performance trending program.

b. Further Assessments

1. No inspection guidance.
2. From the list of dedicated items provided by the licensee, the inspector should select for review approximately three other dedication packages having similar applications and critical characteristics as the CGI(s) that resulted in the identified failures. After the selections have been made, the inspector should request that the licensee compile a complete package of all the procurement and dedication records for each item. Typical contents of a dedication package are described in Appendix C of this inspection procedure. The inspector should review the dedication packages as described in Section 03.01a(1) of this inspection procedure.
3. No inspection guidance.

03.02 Programmatic Inspection

- a. Review of Program and Procedures. The review of the program and procedures should be performed to familiarize the inspector with the licensee's CGI dedication process. For cases in which problems are identified with the licensee's CGI dedication process, the inspector may decide to perform a more extensive review of the program and procedures to determine if these problems are the result of inadequate procedures.

The inspector's review should include procedures that control: procurement activities; material control; the dedication of CGIs, including receipt inspection and acceptance testing; surveys of

commercial grade suppliers; classification of components; training of personnel; trending of supplier performance; and equipment failures. Attempt to identify any apparent weak areas to concentrate on during the evaluation of the program implementation.

After arriving onsite, the inspector should request that the licensee explain its commercial grade dedication process and conduct a walkthrough of areas associated with it. Areas in the walkthrough could include the engineering, receipt inspection, component testing, and warehouse. The inspector should become familiar with key licensee personnel involved in the commercial grade dedication process. These key personnel should include the responsible engineer(s) who developed the dedication package(s) and systems engineers, procurement engineers, receipt inspectors, quality assurance engineers and inspectors, and warehouse personnel. The inspector should discuss the commercial grade dedication process with these key personnel to gain a better understanding of the process, including:

- How processing of CGI procurement documents is controlled under the quality program and how they receive review and approval. General information on similar activities subject to Appendix 8 to 10 CFR Part 50 is provided in ANSI N45.2.13-1976, Section 3, "Procurement Document Preparation, Review, and Change Control."
 - How technical personnel participate in the preparation, review, and approval process of procurement documents.
 - How consistency and coordination is maintained between corporate level, engineering/support level, and site level programs and implementing procedures.
- b. Selection of Dedication Packages. As discussed in the general guidance section above, the NUMARC initiative for the utilization of CGIs in nuclear safety-related applications was not implemented until July 1990. Therefore, the methods used to perform commercial grade dedication of items procured or dedicated by licensees before that date will not necessarily meet the guidance contained in this inspection procedure.

The selection process should be performance oriented (e.g., weighted toward the review of dedication packages for equipment, components, or parts that have experienced failures). To accomplish this, the inspector should request from the licensee approximately 20 packages for review using the two-step approach described below. The licensee should be given sufficient lead time to prepare the 20 packages and make them available for the first day of onsite inspection.

Step 1: Review the licensee's records available at the plant site to identify recent failures (approximately the last 2 years) of equipment, components, or parts. Review these failures to determine if any were CGIs dedicated for use in safety-related applications. If available, select approximately 75 percent of the total sample from CGI failures.

Step 2: From the list of dedication packages supplied by the licensee, under the "Inspection Guidance" section of this procedure, select the remainder of packages for review. The total sample size including packages from steps 1 and 2 should be approximately 20 packages. However, the inspector can select a larger or smaller sample depending on the complexity of the packages and the time available. The inspector

should select these packages on the basis of the following considerations:

- The inspector should select packages for items whose failure would have the most effect on the ability of the plant to safely operate, safely shutdown from an adverse condition, or maintain a safe shutdown condition. If time permits, review the plant-specific probabilistic risk assessment, individual plant examination, and risk-based inspection guides that provide information on the risk significance of safety-related plant equipment.
 - The inspector should take a performance oriented approach to the selection process by including in the sample packages those items that have been problems in the past. Review available sources of information to identify any known failures of CGIs that were used in safety-related applications. These sources of information could include:
 - component failure lists or lists of items requiring frequent maintenance or replacement as provided by the licensee
 - misrepresented or fraudulent items reported in NRC information notices
 - licensee trending of equipment and supplier performance
 - previous history of component failures or malfunctions as reported in licensee event reports or plant nonconformance reports
 - The inspector should include both simple and complex packages in the sample as well as packages that include a variety of dedication methods (e.g., Methods 1 through 4) described in Section 03.01a(1) above.
 - In addition to selecting packages based on the above considerations (safety significance, complexity, and failures), the inspector should attempt to select samples from each of the following areas: electrical, instrumentation and control, mechanical equipment, and materials.
- c. Evaluation of Dedication Packages. Perform a detailed review of the dedication packages as described above in Section 03.01a(1).
- d. Evaluation of Training Effectiveness. Experience gained during the procurement assessments and pilot inspections suggested that training of personnel involved in CGI dedication activities was a very important factor in the development of a good CGI dedication program. The CGI dedication process generally requires more highly qualified/trained personnel than specified in Appendix 8 to 10 CFR Part 50 procurement. Personnel involved in this process need to be familiar with current industry and NRC guidance and have a strong interface with the licensee's design/engineering organizations. The training expectations, however, should not exceed what is required by the existing licensee's QA program.

As applicable to their job function, select and review the training records for individuals involved in the following areas:

- Determining the safety classification of an item. Training in this area is appropriate when the job function includes reclassification

of items or establishing safety classification of piece parts of safety-related components.

- Specifying design, material, and performance characteristics relevant to the safety function and establishing the acceptance criteria for these characteristics.
- Specifying or performing commercial grade surveys, source verifications, and tests and inspections, including enhanced post-receipt verification testing or inspection.
- The preparation and review of procurement documents.

Through observation, interviews, and a review of records of work performed by the individuals:

- Determine if the individuals selected have adequate knowledge to perform the specific tasks assigned to them. Attend a training course, if available, or review the lesson plans for selected training courses.
- Determine if training inadequacies contributed to any of the deficiencies that may be identified during the inspection.
- Determine if the personnel are familiar with the program requirements and procedures and if they have been properly trained in the dedication process.

It should be noted that alternatives to a formal training program may be adequate to ensure satisfactory program implementation (e.g., on the job training). Additional information in this area is provided in NRC Inspection Procedure 41500, "Training and Qualification Effectiveness."

38703-04 INSPECTION RESOURCE ESTIMATE

The estimated number of onsite inspection hours required to complete all inspection requirements is 144 hours when both the reactive and programmatic options are implemented. This estimate is for broad resource planning and is not intended as a quota or standard for judging inspector or regional performance. The on-site hours can be expected to vary significantly depending on the specific circumstance and scope of each inspection.

38703-05 REFERENCES

The following documents are listed for the inspector's information only and are not considered regulatory requirements unless the licensee has formally committed to implementing any of these documents for application to safety-related activities. The inspector may wish to review these documents to become familiar with commercial grade dedication issues.

ANSI N45.2-1977, "Quality Assurance Program Requirements for Nuclear Power Plants," as endorsed by NRC Regulatory Guide 1.28, Revision 2.

ANSI N45.2.13-1976, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants," as endorsed by NRC Regulatory Guide 1.123, Revision 1.

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Electric Power Research Institute (EPRI) NP-5652, "Guidelines for the Utilization of Commercial-grade Items in Nuclear Safety Related Applications (NCIG-07)," as conditionally endorsed in NRC Generic Letter 89-02.

Generic Letter 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products" (microfiche 48960-001).

Generic Letter 91-05, "Licensee Commercial Grade Procurement and Dedication Programs" (microfiche 57468-264).

NRC Inspection Procedure 41500, "Training and Qualification Effectiveness."

SECY-90-304, "NUMARC Initiatives on Procurement" (microfiche 55277-049).

SECY-91-291, "Status of NRC's Procurement Assessments and Resumption of Programmatic Inspection Activity" (microfiche 59490-079).

END

Appendices:

- A. Dedication Issues
- B. Definitions
- C. Contents of Dedication Packages

APPENDIX A

DEDICATION ISSUES

BASIS FOR THE SELECTION AND VERIFICATION OF CRITICAL CHARACTERISTICS

1. Consideration of Item's Safety Function

Critical characteristics should be based on the item's safety function. The licensee is responsible for (a) identifying the important design, material, and performance characteristics that have a direct effect on the item's ability to accomplish its intended safety function and (b) selecting from these characteristics a set of critical (or acceptance) characteristics that, once verified, will provide reasonable assurance that the item will perform its intended safety function. Criterion II of Appendix 8 to 10 CFR Part 50 provides for the application of quality assurance over activities affecting the quality of structures, systems, and components to an extent consistent with their importance to safety. This graded quality approach can be used in the selection and verification of critical characteristics for commercial grade items (CGIs). When an existing equipment specification is available that contains adequate technical requirements for the item being purchased, that specification can be used to select the critical characteristics for this item.

2. Graded Quality Assurance

The application of graded quality assurance to the CGI dedication process should include consideration of the item's importance to safety and other factors specific to the item being procured. Certain items and services may require extensive controls throughout all stages of development while others may require only a limited quality assurance involvement in selected phases of development. The following factors should be considered in determining the extent of quality assurance to be applied: (a) The importance of malfunction or failure of the item to plant safety, (b) the complexity or uniqueness of the item, (c) the need for special controls and surveillance over process and equipment, (d) the degree to which functional compliance can be demonstrated by inspection and test, and (e) the quality history and degree of standardization of the item. Additional guidance on the use of graded quality assurance can be found in the nonmandatory appendix to ANSI N45.2.13-1976.

3. Consideration of Failure Modes

An evaluation of credible failure modes of an item in its operating environment and the effects of these failure modes on the item's safety function may be used in the safety classification of an item and as a basis for the selection of critical characteristics.

4. Reasonable Assurance

The dedication process represents an acceptable method of achieving compliance with Appendix 8 to 10 CFR Part 50 with the purchaser assuming many of the responsibilities for ensuring quality and functionality of an item that had previously been the responsibility of the vendor. In this context, reasonable assurance consists of the purchaser controlling or

verifying the activities affecting the item's quality to an extent consistent with the item's importance to safety or ensuring that these activities are adequately controlled by the supplier. For more complex items, dialogue with the original equipment manufacturer may be necessary to identify the design and functional parameters of specific piece parts. Once the dedication process is completed, the quality assurance and/or other measures applied to those aspects of the item that directly affect its safety function should result in the same level of performance as for a like item manufactured or purchased under a quality assurance program of Appendix B to 10 CFR Part 50.

5. Engineering Judgment

Engineering judgment can be used in selecting those important design, material, and performance characteristics that are identified as the item's critical characteristics. The bases for engineering judgment for this application should be documented.

TRACEABILITY

Material/Items Purchased From Distributors

Traceability can be defined as the ability to verify the history, location, or application of an item by means of recorded identification. Where the item's acceptance is based entirely or partially on a certification by the manufacturer, the traceability must extend to the manufacturer. The purchaser should ensure by survey or by other means that the manufacturer has established adequate traceability controls and that these controls are effectively implemented. For situations in which intermediaries (distributors) are included in the supply chain, the activities of these organizations may need to be surveyed to ensure that traceability and proper storage conditions are maintained. A survey of the distributor may not be necessary if the distributor acts only as a broker and does not warehouse or repackage the items or in cases where traceability can be established by other means such as verification of the manufacturer's markings or shipping records.

SAMPLING

1. Established Heat Traceability (Materials)

When heat traceability of metallic material has been established and each piece of the material is identified with the material heat number, chemical analysis and destructive testing required for the acceptance of this material may be performed on one piece of the material. The same rationale may be used for the acceptance of containers of nonmetallic materials such as lubricants providing that traceability has been established and each container is identified with a unique mix or batch number.

2. Established Lot/Batch Control (Items)

When lot/batch (defined as units of product of a single type, grade, class, size, and composition, manufactured under essentially the same conditions and at essentially the same time) control is established through a commercial grade survey, the party performing dedication of such items can use sampling prescribed by standard statistical methods

that are based on homogeneous product lots. Such sample plans should be identified and should provide for the verification of the critical characteristics with confidence level consistent with the item's importance to safety. Other means of demonstrating adequate lot/batch control may include satisfactory performance history and the results of receipt inspection/testing. When such methods are used as a basis for developing product sampling strategy, they should be supported by documented objective evidence.

3. Material and Items With No Lot/Batch Control

When lot/batch control cannot be established, sampling plans need to be considered on individual, item-specific basis and ensure that they are capable of providing a high level of assurance of the item's suitability for service. There may be situations where each item needs to be tested.

COMMERCIAL GRADE SURVEYS

1. Verification of Vendor's Control of Specific Characteristics

A commercial grade survey should be specific to the scope of the CGI(s) being purchased. The vendor's controls of specific critical characteristics to be verified during the survey should be identified in the survey plan. The verification should be accomplished by reviewing the vendor's program/procedures controlling these characteristics and observing the actual implementation of these controls in the manufacture of items identical or similar to the items being purchased.

2. Identification of Applicable Program/Procedures

The vendor must have a documented program and/or procedures to control the critical characteristics of the item or items being procured that are to be verified during the survey. When many items are being purchased, a survey of a representative group of similar items may be sufficient to demonstrate that adequate controls exist. If the vendor's controls are determined to be satisfactory, purchase orders for these items should invoke these controls as contract requirements by referencing the applicable program/procedure(s) and revision. If multiple working level procedures are applicable to the vendor's activities, which affect the item's critical characteristics and these procedures, in turn, are controlled by a higher level document, it may be appropriate to reference that document in the purchase order. It is important to ensure that the specific controls reviewed and accepted during the survey be applied during the manufacturing process. Upon completion of the work, the vendor should certify compliance with the purchase order requirements.

3. Documentation of Survey Results

Commercial grade survey documentation should include the identification of the item or items for which the vendor is being surveyed, identification of the critical characteristics of these items that the vendor is expected to control, identification of the controls to be applied (program/procedure and revision), and a description of the verification activities performed and results obtained. Critical characteristics that are not adequately controlled should be addressed by contractually requiring the vendor to institute additional controls or by utilizing other verification and acceptance methods.

4. Survey Frequency

Commercial grade surveys should be conducted at sufficient frequency to ensure that the process controls applicable to the critical characteristics of the item procured continue to be effectively implemented. Factors to be considered in determining the frequency of commercial grade surveys include the complexity of the item, frequency of procurement, receipt inspection, item performance history, and knowledge of changes in the vendor's controls. The survey frequency should not exceed the audit frequency established for 10 CFR Part 50, Appendix B, suppliers.

ACCEPTANCE OF CERTIFIED MATERIAL TEST REPORTS (CMTRs) AND CERTIFICATES OF COMPLIANCE (CoCs)

Validity Verified Through Vendor/Supplier Audit or Testing

When the verification of critical characteristics is based on vendor CMTRs or CoCs, the validity of these documents should be ensured. This can be accomplished through a commercial grade survey or, for simple items, periodic testing of the product on receipt. Such verifications should be conducted at intervals commensurate with the vendor's past performance. If the item's supply chain includes a distributor, a survey of the distributor's activities may be necessary (see "Traceability").

USE OF INDUSTRY GUIDANCE

The Electric Power Research Institute (EPRI) NP-5652, "Guideline for the Utilization of Commercial Grade Items in Nuclear Safety Related Applications (NCLG-07)," defines critical characteristics as "identifiable and measurable attributes/variables of a CGI, which once selected to be verified, provide reasonable assurance that the item received is the item specified." NRC's conditional endorsement of EPRI NP-5652 by Generic Letter 89-02 was based on interpreting that in the EPRI definition of critical characteristics the "item specified" encompassed those attributes that are essential for the performance of the item's safety function. This interpretation is consistent with the definition of "critical characteristics for acceptance" found in EPRI NP-6406, "Guidelines for the Technical Evaluation of Replacement Items in Nuclear Power Plants," which notes that critical characteristics for acceptance are a subset of "critical characteristics for design." The EPRI NP-6406 definition of "critical characteristics for design" includes those attributes that ensure the performance of the item's design function.

Published NRC guidance does not differentiate between design and acceptance critical characteristics and the CGI dedication guidance provided in Generic Letters 89-02 and 91-05 does not suggest that all design requirements of an item need to be verified during the dedication process. Rather, the licensee is expected to identify the item's design, material, and performance characteristics that have a direct effect on the item's ability to accomplish its intended safety function and select from these characteristics a set of critical (or acceptance) characteristics that, once verified, will provide reasonable assurance that the item will perform that function. Consistency in the definition of critical characteristics can be improved by equating the NRC's definition of critical characteristics to the EPRI definition of "critical characteristics for acceptance."

END

38703, Appendix A

A-4

Issue Date: 11/08/93

APPENDIX B

DEFINITIONS

The following terms are listed to provide the inspectors with working definitions of important terms used during the procurement and dedication of commercial grade items (CGIs). These terms are defined only in the context of the CGI dedication process and are solely to aid the inspector in the inspection process.

Basic Component - A plant structure, system, component, or part thereof necessary to ensure one of the following:

- the integrity of the reactor coolant pressure boundary
- capability to shut down the reactor and maintain it in a safe shutdown condition
- the capability to prevent or mitigate the consequences of accidents that could result in offsite radiation exposures comparable to those referred to in 10 CFR Part 100.11

Certificate of Compliance - A written statement attesting that the materials are in accordance with specified requirements.

Certified Material Test Report - A document attesting that the material is in accordance with specified requirements, including the actual results of all required chemical analyses, tests, and examinations.

Commercial Grade Item - An item satisfying all the following criteria:

- not subject to design or specification requirements that are unique to nuclear facilities
- used in applications other than nuclear facilities
- ordered from the manufacturer/supplier on the basis of specifications set forth in the manufacturer's published product description (e.g., catalog)

Commercial Grade Survey - Activities conducted by the purchaser or its agent to verify that a supplier of CGIs controls, through quality activities, the critical characteristics of specifically designated CGIs, as a method to accept those items for safety-related use.

Critical Characteristics - Those important design, material, and performance characteristics that, once verified, will provide reasonable assurance that the item will perform its intended safety function.

Dedication - The process by which a CGI is designated for use as a basic component. This process includes the identification and verification of critical characteristics. (Also refer to definition in 10 CFR Part 21.3(4)(c-1))

Engineering Judgment - A process of logical reasoning that leads from stated premises to a conclusion. This process should be supported by sufficient documentation to permit verification by a qualified individual.

Source Verification - Activities witnessed at the suppliers' facilities by the purchaser or its agent for specific items to verify that a supplier of CGIs controls the critical characteristics of that item as a method to accept the item for safety-related use.

Traceability - Is the ability to verify the history, location, or application of an item by means of recorded identification.

END

ATTACHMENT 3

NRC GENERIC LETTER GL 89-02,
ACTIONS TO IMPROVE THE DETECTION OF COUNTERFEIT
AND FRAUDULENTLY MARKETING PRODUCTS,
MARCH 21, 1989

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555-0001
J. SHOP, COOK, PURCELL & REYNOLDS

March 21, 1989

TO: ALL HOLDERS OF OPERATING LICENSES AND CONSTRUCTION PERMITS FOR
NUCLEAR POWER REACTORS

SUBJECT: ACTIONS TO IMPROVE THE DETECTION OF COUNTERFEIT AND FRAUDULENTLY
MARKETED PRODUCTS (GENERIC LETTER 89-02)

Recent instances of counterfeit and fraudulently marketed vendor products have heightened the NRC's concerns for licensees' capability to assure the quality of procured products and to reduce the likelihood of the use of counterfeit or fraudulent products in nuclear power plants. During recent NRC inspections of licensees and vendors, the NRC has observed a wide variety of practices and programs for procurement, receipt inspection, testing and dedication of equipment and material for safety-related applications. The purpose of this generic letter is to share with all licensees some of the elements of programs that appear to be effective in providing the capability to detect counterfeit or fraudulently marketed products and in assuring the quality of vendor products. The staff is aware of and encourages the industry working group efforts to develop guidance in these areas.

Three characteristics of effective procurement and dedication programs have been identified during these NRC inspections. These characteristics are (1) the involvement of engineering staff in the procurement and product acceptance process, (2) effective source inspection, receipt inspection, and testing programs, and (3) thorough, engineering based, programs for review, testing, and dedication of commercial-grade products for suitability for use in safety-related applications. NRC has found that programs that embodied the above three elements were generally effective in providing enhanced capability to detect counterfeit or fraudulently marketed products and in assuring the quality of procured products, both in safety-related and other plant systems.

Licensees may want to consider the applicability of these characteristics to their programs to reduce the likelihood of the introduction of counterfeit or fraudulent products into their plants and to assure the quality of procured vendor products.

It should be noted that the NRC staff conditionally endorses the guidelines contained in EPRI NP-5652, "Guideline for the Utilization of Commercial-Grade Items in Nuclear Safety-Related Applications (NCIG-07)," that was issued by EPRI in June 1988 for evaluating commercial-grade products for suitability for use in safety-related applications.

Background:

Numerous instances have been identified by the NRC during the past 2 years in which the nuclear industry received, accepted, and installed items of hardware

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that were not of the quality purported by the manufacturer or supplier due to apparent misrepresentation. Significant deficiencies have also been identified in the programs for dedicating commercial-grade products for use in safety-related applications.

The use in nuclear facilities of products which are counterfeit or fraudulently marketed increases the likelihood that some plant equipment may not perform as expected. (See the enclosed list of NRC Information Notices and Bulletins regarding this matter.)

Discussion:

Procurement quality assurance (QA) controls for products to be used in safety-related applications are established in Appendix 8 to 10 CFR Part 50, and in Regulatory Guides 1.28, 1.33, and 1.123. It is recognized that Appendix 8 provides criteria for QA programs and does not specifically address fraudulent activities; however, an effectively implemented licensee QA program would increase the likelihood of detecting fraudulently marketed vendor products. Although a properly implemented QA program may more readily detect substandard products than will the commercial-grade component upgrade process, a licensee's commercial-grade dedication process, as described in paragraph C., will greatly enhance the effectiveness of current upgrade practices. The actions described in paragraphs A. and B. have also proved useful in detecting substandard, counterfeit or fraudulently marketed products intended for use in systems counted for the safe operation of the facility.

A. Engineering Involvement in the Procurement Process

Appropriate engineering involvement is warranted during the procurement and product acceptance processes, including testing, for products used in nuclear power plants. Inadequate engineering involvement has been a common weakness in licensees' procurement programs, particularly when commercial-grade procurements were involved. Involvement of a licensee's engineering staff in an effective procurement process would normally include (1) development of specifications to be used for the procurement of products to be used in the plant, (2) determination of the critical characteristics of the selected products that are to be verified during product acceptance, (3) determination of specific testing requirements applicable to the selected products, and (4) evaluation of test results. The extent of necessary engineering involvement is dependent on the nature and use of the products involved.

B. Product Acceptance Programs

Experience indicates that reliance on part number verification and certification documentation is insufficient to ensure the quality of procured products. Licensees with effective product acceptance programs have included receipt/source inspection and appropriate testing criteria, effective vendor audits, special tests and inspections and post-installation tests in their programs. These licensees have applied the inspection

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and testing criteria to products procured for use in safety-related systems and for all commercial-grade products being evaluated for suitability for use in safety-related systems. The inspection and testing criteria also have required identification and verification of the products' critical characteristics. In selecting the critical characteristics to be verified, consideration may be given to the safety significance, complexity, and application of the various products. For suppliers with acceptable QA programs, as confirmed by licensee audits, sampling plans are often utilized to perform the required inspections and tests. In addition to these receipt/source inspections and tests, effective licensee programs normally verify the traceability to the original manufacturers of procured materials, equipment, and components in those cases where original manufacturer's certifications are elements of the safety-related procurement or commercial-grade dedication program. Effective audits have included consideration of audit approach, depth of audit, and audit team composition and have included appropriate engineering/technical representatives. Comprehensive multi-licensee audit teams have also been found to be effective.

C. Dedication Programs

It is each licensee's responsibility to provide reasonable assurance that nonconforming products are not introduced into their plants. Dedication programs that ensure the adequacy of critical parameters of products used in safety-related applications can also contribute to the identification of counterfeit or fraudulently marketed vendor products.

The NRC staff believes that licensees who use methods similar to those described in EPRI NP-5652 "Guideline for the Utilization of Commercial-Grade Items in Nuclear Safety-Related Applications (NCIG-07)," to verify the critical characteristics of commercial-grade items intended for safety-related applications have the basis for effective dedication programs.

Properly implemented, the EPRI guidelines, as modified below, establish methods which satisfy existing requirements of Appendix B to 10 CFR Part 50 as they apply to the dedication process of commercial-grade items.

1. Acceptance Method 2, "Commercial-Grade Survey of Supplier," should not be employed as the basis for accepting items from suppliers with undocumented commercial quality control programs or with programs that do not effectively implement their own necessary controls. Likewise, Method 2 should not be employed as the basis for accepting items from distributors unless the survey includes the part manufacturer(s) and the survey confirms adequate controls by both the distributor and the part manufacturer(s).
2. Acceptance Method 4, "Acceptable Supplier/Item Performance Record," should not be employed alone unless:

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- a. The established historical record is based on industry-wide performance data that is directly applicable to the item's critical characteristics and the intended safety-related application; and
- b. The manufacturer's measures for the control of design, process, and material changes have been adequately implemented as verified by audit (multi-licensee team audits are acceptable).

The NRC staff believes that if licensees' procurement programs effectively implement the elements discussed in paragraphs, A., B., and C., they will reduce the likelihood of the introduction of counterfeit or fraudulent products into their plants.

Although no response to this letter is required, if you have any questions regarding this matter, please contact the technical contact listed below.

Sincerely,

Steven A. Varga
Acting Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:

1. List of Bulletins and Information Notices
2. List of Recently Issued Generic Letters

CONTACT:

E. William Brach, NRR
(301) 492-0961

Enclosure 1

BULLETINS AND INFORMATION NOTICES CONCERNING NONCONFORMING
MATERIALS AND EQUIPMENT AND INSTANCES OF INADEQUATE
DEDICATION OF EQUIPMENT FOR SAFETY-RELATED APPLICATIONS

<u>Bulletin No.</u>	<u>Title</u>	<u>Date</u>
87-02*	Fastener Testing to Determine Conformance with Applicable Material Specifications	11/06/87
87-02, Supplement 1*		04/22/88
87-02, Supplement 2*		06/10/88
88-05*	Nonconforming Materials Supplied by Piping Supplies, Inc., at Folsum, New Jersey, and West Jersey Manufacturing Company at Williamstown, New Jersey	05/06/88
88-05, Supplement 1*		06/15/88
88-05, Supplement 2*		08/03/88
88-10*	Nonconforming Molded-Case Circuit Breakers	11/22/88
<u>Information Notice No.</u>		
87-66	Inappropriate Application of Commercial-Grade Components	12/31/87
88-19*	Questionable Certification of Class 1E Components	04/26/88
88-35	Inadequate Licensee Performed Vendor Audits	06/03/88
88-46*	Licensee Report of Defective Refurbished Circuit Breakers	07/08/88
88-46, Supplement 1*		07/21/88
88-46, Supplement 2*		12/30/88
88-48*	Licensee Report of Defective Refurbished Valves	07/12/88
88-48, Supplement 1*		08/24/88
88-97	Potentially Substandard Valve Replacement Parts	12/16/88

*These items reflect instances in which suppliers and manufacturers of safety-related material may have intentionally eluded QA requirements to misrepresent the quality of their products. In the instances marked by an asterisk, the problem was brought to NRC's attention by either a licensee or a nuclear supplier.

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ATTACHMENT 4

**NRC GENERIC LETTER GL 91-05,
LICENSEE COMMERCIAL-GRADE PROCUREMENT
AND DEDICATION PROGRAMS,
APRIL 9, 1991**

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 9, 1991

LIBRARY

TO: ALL HOLDERS OF OPERATING LICENSES AND CONSTRUCTION PERMITS FOR
NUCLEAR POWER REACTORS

SUBJECT: LICENSEE COMMERCIAL-GRADE PROCUREMENT AND DEDICATION PROGRAMS
(GENERIC LETTER 91-05)

This generic letter notifies the industry of the staff's pause in conducting certain procurement inspection and enforcement activities and identifies a number of failures in licensees' commercial-grade dedication programs identified during recent team inspections performed by the U.S. Nuclear Regulatory Commission (NRC). The pause, which began in March of 1990, will end in late summer of 1991. The purpose of the pause is to allow licensees sufficient time to fully understand and implement guidance developed by industry to improve procurement and commercial-grade dedication programs. This generic letter expresses staff positions regarding certain aspects of licensee commercial-grade procurement and dedication programs which would provide acceptable methods to meet regulatory requirements.

During the period from 1986 to 1989, the NRC conducted 13 team inspections of the licensees' procurement and commercial-grade dedication programs. During these inspections, the NRC staff identified a common, programmatic deficiency in the licensees' control of the procurement and dedication process of commercial-grade items for safety-related applications. In a number of cases, the staff found that licensees had failed to adequately maintain programs as required by 10 CFR Part 50, Appendix 8, to assure the suitability of commercially procured and dedicated equipment for its intended safety-related applications. In addition, the staff identified equipment of indeterminate quality installed in the licensees' facilities.

Because of a decrease in the number of qualified nuclear-grade vendors, the NRC staff is aware that there has been a change in the industry's procurement practices. Ten years ago, licensees procured major assemblies from approved vendors who maintained quality assurance programs pursuant to Appendix 8 of Part 50 of Title 10 of the Code of Federal Regulations (10 CFR). Currently, due to the reduction in the number of qualified nuclear-grade vendors, licensees are increasing the numbers of commercial-grade replacement parts that they procure and dedicate for use in safety-related applications. This is a substantial change from the environment in which 10 CFR Part 50, Appendix 8 was promulgated. This has necessitated an increased emphasis by licensees and the NRC staff to maintain procurement and dedication programs that adhere to the requirements of 10 CFR Part 50, Appendix 8, and thus assure the quality of items purchased and installed in safety-related applications. Therefore, dedication processes for commercial-grade parts have increased in importance and NRC inspections have determined that a number of licensees have not satisfactorily performed this procurement and dedication process.

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The industry has been made fully aware of the NRC's concerns in this program area. In the past, escalated enforcement cases have provided notice to the affected licensees and to the industry of NRC's findings, concerns, and expectations in the implementation of procurement and dedication programs.

Further, the NRC staff continues to participate in numerous industry meetings and conferences at which the NRC's positions in this area have been presented. The Nuclear Utility Management and Resources Council (NUMARC) Board of Directors recently approved a comprehensive procurement initiative as described in NUMARC 90-13, "Nuclear Procurement Program Improvements," which commits licensees to assess their procurement programs and take specific action to enhance or upgrade the program if they are determined to be inadequate. The initiative on the dedication of commercial-grade items, which is part of NUMARC 90-13, was to be implemented by January 1, 1990. The staff is monitoring implementation of licensee program improvements by conducting assessments of their procurement and commercial-grade dedication programs and maintaining close interaction with the nuclear industry through participation in conferences, panels, and meetings.

The staff will continue to perform reactive inspections relating to plant specific operational events or to defective equipment and, as required, will continue to initiate resultant enforcement actions. In addition, the staff will continue to perform inspections of vendors. The staff expects to resume procurement and dedication inspection activities in the late summer of 1991. These resumed inspections will be conducted using 10 CFR Part 50, Appendix 3 (not the NUMARC initiatives) as the applicable regulatory requirement. Licensee programs must assure the suitability of commercially procured and dedicated equipment for its intended safety-related application.

The staff position is that the staff will not initiate enforcement action in cases of past programmatic violations that have been adequately corrected. In addition, the staff does not expect licensees to review all past procurements. However, if during current procurement activities, licensees identify shortcomings in the form, fit, or function of specific vendor products, or if failure experience or current information on supplier adequacy indicates that a component may not be suitable for service, corrective actions are required for all such installed and stored items in accordance with Criterion XVI of 10 CFR Part 50, Appendix 8. Also in accordance with Criterion XVI, licensees must determine programmatic causes when actual deficiencies in several products from different vendors are identified during current procurement activities and these deficiencies lead to the replacement of installed items as part of the corrective action. In such cases, a further sampling of previously procured commercial-grade items may be warranted.

In NRC Generic Letter (GL) 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products," the staff described its perspective on good practices in procurement and dedication and provided the NRC's conditional

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endorsement of an industry standard (EPRI NP-5652) on methods of commercial-grade procurement and dedication. A number of recent inspection findings, as discussed in Enclosure 1, indicate that licensees have failed to include certain key activities, as appropriate, in the implementation of the dedication process. The NRC staff's positions on the successful implementation of licensees' programs for commercial-grade dedication with respect to critical characteristics and like-for-like replacements are as follows. (These are also included in Enclosure 1.)

The term "critical characteristics" is not contained in Appendix B and has no special regulatory significance beyond its use and definition in various industry guides and standards. The NRC first used the term critical characteristics in GL 89-02 as constituting those characteristics which need to be identified and verified during product acceptance as part of the procurement process. The NRC has not taken the position that all design requirements must be considered to be critical characteristics as defined and used in EPRI NP-5652. Rather, as stated in Appendix B, Criterion III, licensees must assure the suitability of all parts, materials, and services for their intended safety-related applications (i.e., there needs to be assurance that the item will perform its intended safety function when required). The licensee is responsible for identifying the important design, material, and performance characteristics for each part, material, and service intended for safety-related applications, establishing acceptance criteria, and providing reasonable assurance of the conformance of items to these criteria.

A like-for-like replacement is defined as the replacement of an item with an item that is identical. For example, the replacement item would be identical if it was purchased at the same time from the same vendor as the item it is replacing, or if the user can verify that there have been no changes in the design, materials, or manufacturing process since procurement of the item being replaced. If differences from the original item are identified in the replacement item, then the item is not identical, but similar to the item being replaced, and an evaluation is necessary to determine if any changes in design, material, or the manufacturing process could impact the functional characteristics and ultimately the component's ability to perform its required safety function. If the licensee can demonstrate that the replacement item is identical, then the licensee need not identify the safety function or review and verify the design requirements and critical characteristics. Engineering involvement is necessary in the above activities. Reliance on part number verification and certification documentation is insufficient to ensure the quality of commercially procured products.

The other matters discussed in Enclosure 1 do not constitute NRC staff positions, but provide information on inspection findings and clarify the characterization of effective procurement and dedication programs previously described in GL 89-02.

BACKFIT DISCUSSION:

Based on past inspection findings and the resulting enforcement actions, the NRC staff has determined that licensee commercial-grade procurement and

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dedication programs needed to be improved to comply with the existing NRC requirements as described in 10 CFR Part 50, Appendix B, Criterion III (Design Control), IV (Procurement Document Control), VII (Control of Purchased Material, Equipment and Services), and XVIII (Audits). Specifically, licensees have failed to adequately maintain programs to assure the suitability of commercially procured and dedicated equipment for its intended safety-related application. Since the generic letter presents staff positions regarding implementation of existing regulatory requirements, as contained in Appendix B to 10 CFR Part 50, the staff has concluded, that this is a compliance backfit and has prepared the generic letter in accordance with 10 CFR 50.109 (a)(4)(i). In light of the inadequacies identified in the procurement and dedication programs of a large number of licensees, the issuance of this generic letter is necessary to express the staff's position on the key element that licensees must include as part of the dedication process, specifically that commercial-grade procurement and dedication programs must assure the suitability of equipment for its intended safety-related application. This generic letter is also intended to clarify the elements of effective procurement and commercial-grade dedication programs that were previously provided to licensees in GL 89-02. Since licensees' procurement and dedication programs may contain programmatic deficiencies, the staff has included in the generic letter the necessary licensee corrective action to address shortcomings identified in specific vendor products or components that directly lead to the component not being suitable for safety-related service.

Although no response to this letter is required, if you have any questions regarding this matter, please contact the persons listed below.

Sincerely,

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:

1. Characteristics of Effective Commercial-Grade Procurement and Dedication Programs
2. List of Recently Issued Generic Letters

Technical Contacts: Richard P. McIntyre, NRR
(301) 492-3215

Uldis Potapovs, NRR
(301) 492-0959

CHARACTERISTICS OF EFFECTIVE COMMERCIAL-GRADE
PROCUREMENT AND DEDICATION PROGRAMS

Background

Appendix 8 to 10 CFR Part 50 contains the NRC's regulations for procurement quality assurance (QA) and quality control (QC) for products to be used in safety-related applications. In addition, the NRC has provided further guidance in Regulatory Guides 1.28, 1.33, and 1.123. These requirements and guides, if properly implemented, provide a measure of assurance for the suitability of equipment, including commercial-grade items for use in safety-related systems. Criterion III of Appendix 8 requires licensees to select and review for suitability of application materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems, and components. Criterion IV requires that procurement documents specify the applicable requirements necessary to ensure functional performance. Criterion VII requires licensees to assure that the following are sufficient to identify whether specification requirements for the purchased material and equipment have been met: source evaluation and selection, objective evidence of quality, inspection of the source, and examination of products upon delivery. The process used to satisfy these requirements when upgrading commercial-grade items for safety-related applications is commonly called "dedication." The process of ensuring compliance with 10 CFR Part 50, Appendix 8, must include all those activities necessary to establish and confirm the quality and suitability of commercially procured and dedicated equipment for its intended safety-related application. Some of the dedication activities may occur early in the procurement cycle, before the item is accepted from the manufacturer. Generic Letter (GL) 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products," discussed commercial-grade dedication in terms of engineering involvement in the procurement process, product acceptance, and the dedication process as identified in the EPRI NP-5652 guidelines. This enclosure further discusses the characteristics of effective procurement and dedication programs previously discussed in GL 89-02 and provides examples of specific failures by licensees to effectively implement these characteristics for dedicating and ensuring the suitability of commercial-grade products for safety-related applications. Appropriate implementation of these characteristics would have avoided many of the failures to meet 10 CFR Part 50, Appendix 8 requirements in licensee procurement and commercial-grade dedication programs which were identified during past NRC inspections.

Inspection Observations and Findings

From 1986 to 1989, headquarters and regional personnel conducted 13 team inspections of licensees' procurement and dedication programs. These inspections have identified a common, broad programmatic deficiency in licensees' control over the process of procurement and dedication of commercial-grade

items. In a number of cases, licensees have not maintained programs to ensure the suitability of equipment for use in safety-related applications as required by 10 CFR Part 50, Appendix B, Criterion III. These 13 inspections resulted in findings with significant safety implications. The staff identified eight findings that were considered to be Severity Level III violations and three findings that were Severity Level IV violations. At one plant, the staff did not assign a severity level to individual violations. Instead, the staff considered the entire group to be a Severity Level III problem and used enforcement discretion, as provided under the enforcement policy, based on the licensee's corrective actions (see 10 CFR Part 2, Appendix C, Section V.G.2). Only one of the plants that were inspected did not receive violations in this program area.

In GL 89-02, the NRC has conditionally endorsed the dedication methods described in EPRI NP-5652 guidelines. The staff believes that licensees who implement these dedication methods, in accordance with the NRC's endorsement, can establish a basis for satisfying the existing requirements of Appendix B to 10 CFR Part 50 as these requirements apply to the dedication process for commercial-grade items. An effective commercial-grade dedication program must include provisions to demonstrate that a dedicated item is suitable for safety-related applications. For a licensee to adequately establish suitability, certain key activities must be performed, as appropriate, as part of the dedication process. This generic letter is intended to clarify the dedication approaches described in GL 89-02.

During each of the 13 inspections, the staff identified a common element in each of the inspection findings. This element was the failure of the licensee to assure that a commercially procured and dedicated item was suitable for the intended safety-related application. A dedicated commercial-grade item must be equivalent in its ability to perform its intended safety function to the same item procured under a 10 CFR Part 50, Appendix B QA program. The following is a list of the 13 licensees inspected and the inspection report numbers. A summary of the general inspection findings and NRC observations on these findings follows the list of licensee inspections.

<u>LICENSEE and PLANT</u>	<u>INSPECTION REPORT NO.</u>
1. Tennessee Valley Authority (Sequoyah)	50-327/86-61 50-328/86-61
2. Southern California Edison (San Onofre)	50-206/87-02 50-361/87-03 50-362/87-04
3. Alabama Power (Farley)	50-348/87-11 50-364/87-11
4. Louisiana Power and Light (Waterford)	50-382/87-19

<u>LICENSEE and PLANT</u>	<u>INSPECTION REPORT NO.</u>
5. Sacramento Municipal Utility District (Rancho Seco)	50-312/88-02
6. Maine Yankee Atomic Power (Maine Yankee)	50-309/88-200
7. Northern States Power (Prairie Island)	50-282/88-201 50-306/88-201
8. Portland General Electric (Trojan)	50-344/88-39 50-344/88-46
9. Connecticut Yankee Atomic Power (Haddam Neck)	50-213/89-200
10. Washington Public Power Supply System (WNP-2)	50-397/89-21 50-397/89-28
11. Florida Power (Crystal River)	50-302/89-200
12. Gulf States Utilities (River Bend)	50-458/89-200
13. Commonwealth Edison (Zion)	50-295/89-200 50-304/89-200

1. Inspection Findings

- a. Failure to identify the methods and acceptance criteria for verifying the critical characteristics, such as during receipt inspection, dedication process, or post-installation testing.
- b. Failure to establish verifiable, documented traceability of complex commercial-grade items to their original equipment manufacturers in those cases where the dedication program cannot verify the critical characteristics.
- c. Failure to recognize that some commercial-grade items cannot be fully dedicated once received on site. Certain items are manufactured using special processes, such as welding and heat treating. Dedication testing of these items as finished products would destroy them. For these items, licensees may need to conduct vendor surveillances or to witness certain activities during the manufacturing process.

Discussion

The NRC staff has met on several occasions with NUMARC and licensee representatives to discuss "critical characteristics" as used in the context of commercial-grade procurement and dedication. The term "critical characteristics" is not contained in Appendix 8 and has no special regulatory significance beyond its use and definition in various industry

guides and standards. The NRC first used the term critical characteristics in GL 89-02 as constituting those characteristics which need to be identified and verified during product acceptance as part of the procurement process. The NRC has not taken the position that all design requirements must be considered to be critical characteristics as defined and used in EPRI NP-5652. Rather, as stated in Appendix B, Criterion III, licensees must assure the suitability of all parts, materials, and services for their intended safety-related applications (i.e., there needs to be assurance that the item will perform its intended safety function when required). The licensee is responsible for identifying the important design, material, and performance characteristics for each part, material, and service intended for safety-related applications, establishing acceptance criteria, and providing reasonable assurance of the conformance of items to these criteria. There is no minimum or maximum number of critical characteristics that need to be verified. Further, the critical characteristics for an item may vary from application to application depending on the design and performance requirements unique to each application.

A licensee may take different approaches for the verification of the critical characteristics, depending on the complexity of the item. In many cases, the licensee can verify the critical characteristics of each item during receipt inspection testing. However, for a complex item with internal parts which receive special processing during manufacturing, the licensee may need to conduct a source verification of the manufacturer during production to verify the critical characteristics identified as necessary for the item to perform its safety function. When these methods cannot verify the critical characteristics related to special processes and tests, certification by the original equipment manufacturer may be an acceptable alternative provided documented, verified traceability to the original equipment manufacturer has been established and the purchaser has verified by audit or survey that the original equipment manufacturer has implemented adequate quality controls for the activity being certified.

For items with critical characteristics that can be verified for the most severe or limiting plant application, the licensee might prefer to identify and verify the item's critical characteristics to qualify that item for all possible plant applications. For complex items that would be purchased for specific plant applications, it may be appropriate to address the acceptance criteria for each item individually. Engineering involvement is important in either method because the technical evaluation will identify the critical characteristics, acceptance criteria, and the methods to be used for verification.

2. Inspection Findings

- a. Failure to demonstrate that a like-for-like replacement item is identical in form, fit, and function to the item it is replacing. Part number verification is not sufficient because of the probability of undocumented changes in the design, material, or fabrication of commercial-grade items using the same part number.

-5-

- b. Failure to evaluate changes in the design, material, or manufacturing process for the effect of these changes on safety function performance (particularly under design basis event conditions) of replacement items that are similar as opposed to identical to the items being replaced.
- c. Failure to ensure that items will function under all design requirements. On some occasions, licensees only ensured that the commercial-grade item would function under normal operation conditions.
- d. Failure to verify the validity of certificates of conformance received from vendors not on the licensee's list of approved vendors/suppliers. An unverified certificate of conformance from a commercial-grade vendor is not sufficient.

Discussion

A like-for-like replacement is defined as the replacement of an item with an item that is identical. For example, the replacement item would be identical if it was purchased at the same time from the same vendor as the item it is replacing, or if the user can verify that there have been no changes in the design, materials, or manufacturing process since procurement of the item being replaced. If differences from the original item are identified in the replacement item, then the item is not identical, but similar to the item being replaced, and evaluation is necessary to determine if any changes in design, material, or the manufacturing process could impact the functional characteristics and ultimately the component's ability to perform its required safety function. If the licensee can demonstrate that the replacement item is identical, then the licensee need not identify the safety function or review and verify the design requirements and critical characteristics.

Engineering involvement is necessary in the above activities. The extent of this involvement is dependent on the nature, complexity, and use of the items to be dedicated. Participation of engineering personnel is appropriate in the procurement process, and product acceptance, to develop purchase specifications, determine specific testing requirements applicable to the products, and evaluate the test results. When engineering personnel specify design requirements for inclusion on the purchase documents for replacement components, they need not reconstruct and reverify design adequacy for procurement purposes, but need only ensure that the existing design requirements (which may reference the original design basis) are properly translated into the purchase order.

Reliance on part number verification and certification documentation is insufficient to ensure the quality of commercially procured products. Effective product acceptance programs have as elements, receipt and source inspection, appropriate testing criteria, effective vendor audits and surveillances (including witness/hold points as appropriate), special tests and inspections, and post-installation tests. Procedures and adequate qualifications and training for implementing personnel are also necessary factors in successful implementation.

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ATTACHMENT 5

ELECTRIC POWER RESEARCH INSTITUTE REPORTS

EPRI Report NP-6895

This report provides detailed guidance for determining the safety classification of SSCs used in nuclear power plants. It primarily deals with issues related to identification of safety functions; determining system boundaries through proper consideration of isolation and single failure criteria, and recommends an approach which is most useful when analyzing complex, interconnected mechanical and electrical systems.

Currently, there is no WHC procedure that implements the guidance included in this EPRI report. This is not considered an impediment to the current task because the safety functions of the individual structural components for the isolation barriers are straightforward and uncomplicated. Proper safety classification of these components does not require extensive use of the detailed guidance contained in NP-6895. The situation, of course, would be different if WHC were to undertake classification efforts involving complex electrical or mechanical plant systems and components. Under such circumstances, the development of a specific procedure based on NP-6895 would be strongly recommended.

EPRI Report NP-6406

This EPRI report provides guidance on a process that determines an item's safety function(s) and identifies the critical design characteristics required to ensure this safety function(s). Under the guidance, a replacement item which can provide the critical design characteristics of the original item is considered acceptable.

The guidance expressed in NP-6406 is not effectively incorporated into any current WHC procedure. This also is not a concern for the current task since the isolation barriers do not meet the definition of a "replacement item." A replacement item is one which is being procured to replace an original item already installed in the facility, such as an existing pump, valve or circuit breaker. The isolation barriers are not replacement items. The current design will provide a permanent barrier between the main fuel basin and the discharge chute — as such, it is designed and built to provide a function not previously included in the original design of the 105-K Basins fuel storage pools. Consequently, there is no need to perform a replacement item evaluation under the guidance in NP-6406 in this case. If, on the other hand, WHC intends to make greater use of the dedication process in the future for procuring commercial-grade replacements for installed safety related components or items, it is recommended that the current procedures be upgraded to better incorporate the guidance of NP-6406.

EPRI Reports NP-5652 and TR-102260

These two EPRI reports contain detailed guidance for performing commercial-grade item dedication activities, in particular as it relates to the process of providing reasonable assurance that acceptance criteria have been met. NP-5652 provides the basic procedural guidance, and TR-102260 (written over five years later) includes lessons learned through implementation of NP-5652 at commercial nuclear power plants.

In sum, these two guidance documents set forth the following basic elements for dedication of commercial-grade items:

- An item must meet the definition of a "commercial grade" item included in 10 CFR Part 21 in order to be procured commercial-grade and dedicated for use in safety related applications. Otherwise, the item must be procured as safety related from suppliers with an approved QA program.²
- "Dedication" is a process consisting of a technical evaluation and quality assurance control activities which, when properly implemented, provide reasonable assurance that the item received is the item ordered.
- A key aspect of the dedication process consists of selecting critical characteristics of the item. The critical characteristics are verified through various acceptance methods and support the reasonable assurance finding.
- Four acceptance methods are identified which can be used to verify that the item received is the item ordered. They are:
 - Special Tests and Inspections;
 - Source Verification;
 - Commercial-grade Supplier Surveys; and
 - Satisfactory Item and Supplier Performance History.

Procedures currently in place at WHC follow the general guidance included in EPRI Report NP-5652 for the dedication of commercial-grade items. Specifically, EPG-5.3 and QI 7.5 include provisions for dedication of commercial-grade items and qualification of commercial-grade vendors. While these procedures could be improved to enhance efficiency of the dedication process, they appear to provide an adequate procedural basis for dedication of the isolation barriers.

²A commercial grade item is one that is: (1) not subject to nuclear unique design or specification requirements; (2) used in applications other than nuclear facilities; and (3) ordered from the supplier based on specifications set forth in the supplier's published product description (e.g., product catalog). See 10 CFR, Section 21.3(a-1), (1994).

APPENDIX C

**105-K EAST AND 105-K WEST BASINS/DISCHARGE CHUTE
ISOLATION BARRIERS, FUNCTIONS, AND REQUIREMENTS**

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LIST OF TERMS

ALARA	as low as reasonably achievable
DBE	design basis earthquake
DOE	U.S. Department of Energy
HEPA	high-efficiency particulate air
QA	quality assurance
WHC	Westinghouse Hanford Company

APPENDIX C

105-K EAST AND 105-K WEST BASINS/DISCHARGE CHUTE ISOLATION BARRIERS FUNCTIONS AND REQUIREMENTS

C1.0 INTRODUCTION

C1.1 DOCUMENT PURPOSE

The purpose of this document is to identify functions and requirements for the installation of isolation barriers in the discharge chute openings of the 105-K East and 105-K West Basins. Once installed, the barriers will permanently isolate the discharge chute from the main basin. This activity is being performed to resolve the Unresolved Safety Question associated with the potential post-design basis earthquake leak postulated for the discharge chute construction joint.

C1.2 BACKGROUND

The 105-K Basins store spent fuel, and U.S. Department of Energy (DOE), recognizing that these facilities were not designed for long term storage, has committed to remove all the fuel and sludge by 2002. Because of the extended basin use, Westinghouse Hanford Company (WHC) has been systematically reevaluating analyses performed to earlier standards using modern techniques. These analyses have identified a potential vulnerability in 105-K Basins following a major earthquake.

This vulnerability is due to potential severe leakage caused by separation of the storage basin/dischARGE chute construction joint. Unlike other joints, it is not reinforced, therefore the resultant storage basin leakage is due to failed seals and other problems associated with the original cofferdams. The two new fabricated in 1993 were build as Safety Class 2, and thus cannot be used unless qualified to Safety Class 1 requirements.

Leakage through the construction joint following a design basis earthquake (DBE) could be of sufficient magnitude to uncover the fuel causing significant airborne radioactive contamination, a large incremental addition to the 105-K Basins soil contamination, and the introduction of a large tritium inventory to the groundwater. To counter this threat, the immediate compensatory actions of procuring Bentonite clay pellets for leak mitigation, and providing for temporary make-up water from the Columbia River, have been accomplished. The interim compensatory action of providing an isolation barrier between the storage basins and the discharge chutes will be accomplished next. The final corrective action will be to remove the fuel and sludge from the basins.

C1.2.1 Physical Description

The 105-K Basins are unlined, rectangular concrete pools, built in the early 1950's. They are 38.1 m long, 20.4 m wide, and 6.1 m deep (125 ft x 67 ft x 20 ft), with a nominal water depth of 4.9 m (16 ft). The basins are provided with an asphaltic membrane lined leakage collection sump under most of their footprints. Several structures extend beyond the collection sump, including the construction joint.

C1.2.2 Current Status

The basins were constructed to meet the requirements of the standards in effect during construction in the 1950's. A formal comparison between the existing construction and the DOE Order 6430.1A requirements has not been made; however, several areas of noncompliance have been identified. These areas of noncompliance include lack of high-efficiency particulate air (HEPA) filtration on exhaust; possible loss of storage pool integrity in the event of a cask drop accident; lack of water chemistry control; and the lack of a pool liner or leak detection.

The 105-K West Basin is coated with an epoxy sealant and the stored fuel elements are encapsulated in closed canisters. The 105-K East Basin contains open canisters of fuel elements. As a result, the radiation and contamination levels in the 105-K East Basin are higher than those of the 105-K West Basin. The 105-K West Basin also has a canister decapping station located in the transfer canal between the load out pit and the basin's west bay. The passways between the storage basins and the discharge chutes are open at both locations.

C2.0 SCOPE

The scope of this project is to install new isolation barriers, two at each basin, in the passways between the basin fuel storage pool and the discharge chutes at 105-K East and 105-K West. This project includes the following tasks.

105-K West:

- Measuring the chute opening dimensions.
- Moving fuel canisters as necessary to allow removing fuel racks. Fuel rack removal is required to place seal preparation and isolation barrier installation equipment in the storage pool.
- Removing racks.
- Preparing the seal surface, including moving tools and equipment into the area and removing grating and handrails.

- Installing the new isolation barriers, including moving the barriers into the area.
- Leak testing to verify acceptable installation.

105-K East:

- Measuring the chute opening dimensions.
- Moving empty canisters from fuel racks in the work area to allow removing racks.
- Moving the seal preparation tools and equipment from the west basin to the east basin, and moving the sludge pumping equipment into the area.
- Pumping sludge from the work area into a pit area.
- Relocating racks.
- Preparing the seal surface, including removing grating and handrails.
- Installing the new isolation barriers, including moving the barriers into the area.
- Leak testing to verify acceptable installation.

C3.0 FUNCTIONS AND REQUIREMENTS

C3.1 OVERALL FUNCTION

The overall function of this project is to provide an isolation barrier between the 105-K Basins storage pools and their associated discharge chutes to prevent draining the storage pools after a large seismic event due to leakage in the storage pool/discharge chute construction joint. The new isolation barriers, two in each basin, will limit leakage from each storage pool into its discharge chute to less than 96 L/min (25 gal/min) total per basin. This limit is the amount cited as part of the facility authorization basis in the Safety Analysis Report. This installation is semi-permanent (15-year design life) and the isolation barriers will become an integral part of the basin structure.

C3.1.1 Overall Assumptions

The following assumptions were used to develop the isolation barrier design.

1. The design life of the isolation barriers is based on the assumption that the fuel stored in the two 105-K Basins will be removed by the year 2002, or at the latest before 2010.
2. The isolation barriers are Safety Class 1. NOTE: Parallel analyses were initiated to evaluate whether Safety Class 1 was still necessary, given the age of the fuel. This analysis, as documented in the Interim Safety Basis (Brehm 1995), indicates that Safety Class 1 would not be necessary for the existing site boundary, but may be necessary if the DOE elects to move the site boundary to the nearest Columbia River bank. Hence, Safety Class 1 was elected as a design criterion.
3. The Safety Class 1 DBE criteria is 0.2g and the criteria for the aftershock can be lowered from 0.2g to 0.12g using UCRL 15910 (LLNL 1980).

C3.1.2 Overall Requirements

In addition to general requirements already in place, this project shall satisfy the following requirements:

1. Work shall be performed in accordance with approved work packages and their associated rigging plans, as low as reasonably achievable (ALARA) plans, radiation plans, installation procedures, operations procedures, and Safety plans.
2. Procedures shall be written and/or developed to control all fuel and canister handling operations.
3. The public and Hanford Site workers shall be protected from offsite and onsite release of airborne radioactive materials, and from radiological exposure. Practically, this means maintaining sufficient water in the basin to ensure that the fuel and sludge remain covered for all credible events, and at a level to provide personnel shielding.
4. All activities proposed for the basins should be responsive to stakeholder expressed concerns for limiting the potential impact of leakage to the soil and the Columbia River. Practically, this means limiting airborne and water releases to within limits currently established with state and federal regulators.
5. The safety classification shall be Safety Class 1. Material certifications, procedure approvals, quality assurance (QA) inspections, and other required activities and documentation shall be performed or provided as necessary.

6. Material selection and corrosion allowance shall be specified to provide a 15-year design life.
7. The installed isolation barriers, together with the center island shall be capable of withstanding a Design Basis Earthquake (DBE) of 0.2g with a full head of water 5.08 m (16 ft 8 in.) in both the discharge chute and the storage pool.
8. The isolation barriers and the center island shall be capable of withstanding an aftershock equal to 0.12g with a full head of water 5.08 m (16 ft 8 in.) in the storage pool and the discharge chute empty.
9. Installation of the isolation barriers shall be accomplished without divers.
10. The leak rate shall be less than 96 L/min (25 gal/min) total through both barriers, both routinely, and following a DBE.

C3.2 105-K WEST BARRIER ACTIVITIES FUNCTIONS AND REQUIREMENTS

C3.2.1 Chute Opening Measurement

The functions and requirements for measuring the discharge chute opening are provided below.

C3.2.1.1 Function. The function of measuring the chute opening locations is to ensure that the isolation barriers will fit properly when installed. Proper fit is necessary to achieve the required leak tightness.

C3.2.1.2 Requirements. The measurements shall verify the discharge opening widths to within ± 1.6 mm (1/16 in.). A video recording of the condition of the embedded angle shall also be made.

C3.2.2 Preparing Seal Surfaces

The functions and requirements for preparation of the sealing surfaces is discussed in the following subsections.

C3.2.2.1 Function. The function of preparing the seal surfaces is to provide the necessary surface conditions to ensure long-term seal performance.

C3.2.2.2 Requirements. The sealing surface shall be regular and smooth (step discontinuities of less than 3.2 mm [1/8 in.]). The sealing surface shall be capable of withstanding the force applied by the barrier during maximum seal compression. It shall not provide crevices or leak paths which bypass the seal. It shall remain in place with no significant change in performance during the DBE or aftershock.

C3.2.3 Install Barriers

The functions and requirements for installation of the isolation barriers are listed below.

C3.2.3.1 Function. The function of the installed isolation barriers is to isolate the discharge chute from the 105-K West main basin.

C3.2.3.2 Requirements. The new isolation barriers shall limit leakage into the discharge chute from the storage pool to less than 96 L/min (25 gal/min). The design and installation shall conform to Safety Class 1 requirements because the barriers become part of the boundary which prevents exceeding the authorization basis of the facility.

C3.2.4 Leak Testing

The function and requirement(s) for leak testing is listed below.

C3.2.4.1 Function. The function of leak testing is to verify that the isolation barriers will limit leakage between the basin storage pool and the discharge chute to less than 96 L/min (25 gal/min).

C3.2.4.2 Requirements. The leak test shall have sufficient accuracy to ensure that gross, including uncertainties, isolation barrier leakage can be established with 2 sigma confidence. The test shall be performable in less than 48 hours, and in the shortest time possible, consistent with obtaining satisfactory accuracy. The test shall provide for immediate field indications of unsatisfactory or satisfactory performance.

C3.3 105-K EAST BARRIER ACTIVITIES FUNCTIONS AND REQUIREMENTS

C3.3.1 Discharge Chute Opening Measurement

The functions and requirements for measurement of the discharge chute sealing surfaces is given below.

C3.3.1.1 Function. The function of measuring the chute opening locations is to ensure that the isolation barriers will fit properly when installed. Proper fit is necessary to achieve the required leak tightness.

C3.3.1.2 Requirements. The measurements shall verify discharge chute openings to within ± 1.6 mm (1/16 in.). A video recording of the condition of the embedded angle shall also be made.

C3.3.2 Relocate Equipment

The functions and requirements for relocating equipment currently located in the 105-K East discharge chute is provided below.

C3.3.2.1 Function. The function of moving equipment in the 105-K East discharge chute is to minimize the missile risk during an earthquake. Various equipment will be relocated in the discharge chute. The packager will be relocated to the storage pool to provide access to it for future use.

C3.3.2.2 Requirements. The equipment in the discharge chute shall be relocated in accordance with WHC-SD-WM-AP-030 (Hull 1994), latest revision.

C3.3.3 Pump Sludge and Remove Debris

The function and requirement for pumping sludge and removal of debris is listed below.

C3.3.3.1 Function. The function of pumping sludge and removing debris from the intended storage pool and discharge chute work areas is to minimize interference from these items during the seal preparation and barrier installation activities.

C3.3.3.2 Requirements. Sludge and debris from the affected storage pool and discharge chute work areas shall be removed and relocated. The determination of "what is sufficiently clean" will be made by Operations and Engineering based on visual inspection.

C3.3.4 Prepare Seal Surfaces

The functions and requirements for preparation of the sealing surfaces is discussed in the following subsections.

C3.3.4.1 Function. The function of preparing the seal surfaces is to provide the necessary surface conditions to ensure long-term seal performance.

C3.3.4.2 Requirements. The sealing surface shall be regular and smooth (step discontinuities of less than 3.2 mm [1/8 in.]). The sealing surface shall be capable of withstanding the force applied by the barrier during maximum seal compression. It shall not provide crevices or leak paths which bypass the seal. It shall remain in place with no significant change in performance during the DBE or aftershock.

C3.3.5 Install Barriers

The functions and requirements for installation of the isolation barriers is listed below.

C3.3.5.1 Function. The function of installing isolation barriers is to isolate the discharge chute from the 105-K East Basin.

C3.3.5.2 Requirements. The new isolation barriers shall limit leakage into the discharge chute from the storage pool to less than 96 L/min (25 gal/min).

C3.3.6 Leak Testing

The function and requirement(s) for leak testing is listed below.

C3.3.6.1 Function. The function of leak testing is to verify that the isolation barriers will limit leakage between the basin storage pool and the discharge chute to less than 96 L/min (25 gal/min).

C3.3.6.2 Requirements. The leak test shall have sufficient accuracy to ensure that gross, including uncertainties, isolation barrier leakage can be established with 2 sigma confidence. The test shall be performable in less than 48 hours, and in the shortest time possible, consistent with obtaining satisfactory accuracy. The test shall provide for immediate field indications of unsatisfactory or satisfactory performance.

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