EHS&L Document

SNM-1227 - Chapter 1
General Information

Nature of Changes

Item	Paragraph	Description	Justification
1.	New Document	New Document	License Renewal
2.			
3.			
4.			
5.			
6.			
7.			
8.			
9.			
10.			
List Below any Documents, including Forms & Operator Aids which must be issued concurrently with this document revision:			
Delete E10-01-000 through E10-01-016		Do Not make document effective until Loren Maas releases after NRC approval.	

This Document contains a total of 19 pages excluding the signature page generated by Documentum, the document control application software.

DOCUMENT REVIEW/APPROVAL/DELETION CHECKLIST

All new and/or revised procedures shall be approved by the change author, cognizant manager(s) of areas affected by the changes, and by applicable manager(s) of any function that approved the previous revision of the document unless responsibility for such approval has been transferred to another organization. Also, the procedure shall be approved by manager(s) of functional organizations that provide technical reviews with the exception of the Training Department. Finally, Document Control shall verify that the required approvals have been properly obtained and that any documents that must be issued concurrently are ready to be issued.

Minor Changes: If the proposed changes are limited to editorial and/or administrative changes check the box at the right. The document will be routed directly for review by EHS&L without technical review. All applicable approvals must still be obtained.	☐

Document Reviews			**Document Approvals**	
Purpose/Function of Review	Specify Reviewer(s) (Optional except for change author)	(Check all that apply)	Title of Approver	(Check all that Apply)
Document Control (Automatic)		☒	Document Control (Automatic)	☒
Change Author	LJ Maas	☒	Author	☒
Independent Technical Review	RE Link	☒		
Operability Review(s)			Mgr, Richland Operations[1]	☐
Conversion		☐	Mgr, Uranium Conversion & Recovery Operations[1]	☐
Recovery		☐		
Ceramics		☐	Mgr, Ceramic Operations[1]	☐
Rods		☐	Mgr, Rods & Bundles[1]	☐
Bundles		☐		
Transportation		☐		
Components		☐	Mgr, Component Fabrication[1]	☐
Maintenance Review		☐	Mgr, Maintenance[1]	☐
Lab Review		☐	Mgr, Analytical Services[1]	☐
EHS&L Review(s)			Mgr, EHS&L[2]	☒
Criticality		☐	Mgr, Criticality Safety[2]	☐
Radiation Protection		☐	Mgr, Safety, Security & Emergency Preparedness[2]	☐
Safety/Security		☐		
Emergency Preparedness		☐		
MC&A	LJ Maas	☒	Mgr, Licensing & Compliance[2]	☐
Transportation		☐		
Environmental		☐		
BWR Product Eng. Review		☐	Mgr, BWR Product Engineering	☐
Quality Review		☐	Mgr, Quality	☐
Ops. Projects & Planning Review		☐	Mgr, Ops. Projects & Planning	☐
Others:		☐	Mgr, Richland Site/Other	☐
Training & Employee Dev.:[3]		☐	Training & Employee Dev.	☐

[1]Note: If approvals include 2 or more product center managers, the Operations manager can be substituted for the applicable product center managers.

[2]Note: If approvals include 2 or more EHS&L functional managers, the EHS&L manager can be substituted for the applicable EHS&L functional managers.

[3]Note: Training department review is required for all procedures that require or affect a Learning Plan and if additional training materials or curriculum must be revised before issuing procedure.

EHS&L Change Impact Evaluation Form		
Document / ECN No*.: E10-08-001	Change Evaluator: RE Link	
Does the change potentially impact Criticality Alarm System (CAS) coverage?	☐ Yes ☒ No	If yes, explain:
NRC Pre-Approval Evaluation:		
Is NRC Pre-approval (License Amendment) Needed? (Based on "Yes" answer to any of five questions below). (Based on "No" answer to all five questions below).	☒ Yes ☐ No	This is part of our NRC license renewal.
1. Does the change create new types of accident sequences that, unless mitigated or prevented, would exceed the performance requirements of 10 CFR 70.61 (create high or intermediate consequence events) and that have not previously been described in AREVA NP Inc's ISA Summary?	☐ Yes ☒ No	If yes, explain:
2. Does the change use new processes, technologies, or control systems for which AREVA NP Inc. has no prior experience?	☐ Yes ☒ No	If yes, explain:
3. Does the change remove, without at least an equivalent replacement of the safety function, an item relied on for safety that is listed in the ISA Summary?	☐ Yes ☒ No	If yes, explain:
4. Does the change alter any item relied on for safety, listed in the ISA Summary, that is the sole item preventing or mitigating an accident sequence of high or intermediate consequences?	☐ Yes ☒ No	If yes, explain:
5. Does the change qualify as a change specifically prohibited by NRC regulation, order or license condition?	☐ Yes ☒ No	If yes, explain:
Actions Required Prior to or Concurrent with Change Implementation Evaluation:		
Action		**Explanation**
6. Modification / Addition to CAS system or system coverage documentation	☐ Yes ☒ No	If yes, explain:
7. Acquire NRC pre-approval (license amendment)	☒ Yes ☐ No	If yes, explain: License renewal
8. Conduct/modify ISA	☐ Yes ☒ No	If yes, explain:
9. ISA Database Modification	☐ Yes ☒ No	If yes, explain:
10. Modification of other safety program information / underlying analyses (PHA, RHA, FHA, NCSA, etc.)	☐ Yes ☒ No	If yes, explain:
Actions required subsequent to Change Implementation Evaluation:		
11. Update safety program information (PHA,RHA,FHA,NCSA, P&ID)	☐ Yes ☒ No	If yes, explain:

* **If this form exists as a part of a document, the document number is not required.**

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Application Date:	NRC Docket No. 70-1257
October 27, 2006	

1.0 General Information

1.1 *Facility and Process Information*

1.1.1 General Facility Description

The AREVA NP Inc. (AREVA NP) Richland fuel fabrication facility is located at 2101 Horn Rapids Road, just within the northern limits of the City of Richland. The fenced exclusion area of approximately 50 acres is located within 320 acres of AREVA NP-owned land, the remainder of which is either devoted to vehicle parking areas, is undeveloped, or is leased (currently) for agricultural purposes. The developed industrial site (fenced exclusion area plus surrounding parking areas) in relation to the overall AREVA NP site boundary is depicted in Figure 1-1. Additional information describing the AREVA NP Richland facility, including its location with respect to geographic features, roadways, population centers, industrial facilities, and public facilities, is provided in Section 1.3, "Site Description".

1.1.2 Facility Buildings and Structures

The AREVA NP Richland facility contains numerous buildings plus various outside facilities/structures (tank farms, storage pads, etc.). The buildings and structures are confined within the secured fenced area and include the major SNM-processing production facilities, a number of SNM-handling production support facilities (product storage warehouses, waste treatment facilities, etc.), and a large number of non-SNM-handling production and administrative support facilities (materials warehouses, craft shops, office buildings, etc.). A map of the developed industrial site (fenced exclusion area plus surrounding parking areas) is provided as Figure 1-2.

The major site features, their closest locational coordinates on Figure 1-2, and a statement as to their current primary function(s) are provided below. The current facility functions are provided for informational purposes and are not intended to be restrictive of future potential activities in those facilities.

UF_6 Cylinder Storage Facility (F-7) Receipt, handling and storage of full, empty, and heel-quantity uranium hexafluoride (UF_6) cylinders, including weighing and assaying of cylinder contents.

Dry Conversion Facility (E-6) Chemical conversion of UF_6 to uranium dioxide (UO_2) powder and mechanical processing of the powder (powder preparation) for subsequent pellet pressing.

UO_2 Building (D-6) Pressing of UO_2 powder into pellets and subsequent pellet sintering and grinding. Loading of finished pellets into fuel rods and assembly of fuel rods and associated hardware into fuel bundles. Loading of products (powder, pellets, fuel rods, assemblies) for shipment. Recovery of uranium via the ammonium diuranate (ADU) process. Bulk UO_2 storage. Analytical laboratory and UF_6 cylinder washing activities.

Specialty Fuels (SF) Building (C-6) Production of UO_2 fuel pellets (blending, pressing, sintering, grinding) containing neutron absorber additive. Fuel rod fabrication activities. Housing of the Solid Waste Uranium Recovery (SWUR) incinerator.

Engineering Laboratory Operations (ELO) Building (D-7) Dissolution and solvent extraction processing of uranium fuel scrap for removal of contaminants. Laboratory facilities for research and development activities in support of fuel fabrication and related functions.

UF_6 Cylinder Recertification Facility (F-7) Testing and inspection for the recertification of UF_6 cylinders.

UNH Drum Storage Warehouse (E-8) Storage of drums of uranyl nitrate solution for eventual uranium recovery processing.

Warehouse 1, 2, 3, Facility (C-5) Materials receipt and storage. Loading of containers of powder/pellet product into shipping containers; loading of containers into trucks. Mechanical component operations.

Fuel Storage Warehouse (Warehouse 4) (C-3) Storage of uranium-bearing product or scrap. Miscellaneous production support activities.

Warehouse 5 (D-3) Shipping container storage and set-up. Miscellaneous product support and general storage activities.

Uranium Storage Warehouse (Warehouse 6) (E-5) Storage of uranium powder and pellet product material and uranium fuel scrap in closed containers. Miscellaneous production support activities.

Operations Scrap Warehouse (Warehouse 7) (G-7) Storage of containers of uranium fuel feed stock, product, and scrap.

Waste Storage Facility (F-3) Storage of containers (drums/boxes) of radioactively contaminated wastes awaiting off-site disposal.

Lagoon Uranium Recovery (LUR)/Solids Processing Facility (SPF) (E-4) Processing of waste liquids and sludges/solids. Powder blending operations. Miscellaneous production support activities.

Ammonia Recovery Facility (ARF) (E-7) Recovery of ammonium hydroxide and uranium from liquid process effluents. Temporary tank accumulation of liquid process effluents.

Modular Extraction Recovery Facility (MERF) (E-4) Sorting and recovery of uranium from contaminated solid wastes.

Fuel Services Building (Building 9) (B-4) Miscellaneous production support activities, including computer operations. Fuel bundle defabrication activities.

Shipping Container Refurbishment Facility (D-2) Maintenance, cleaning and painting of product shipping containers; mechanical fabrication activities.

Product Development Test Facility (PDTF) (D-4) Hydraulic, heat transfer, and mechanical/ seismic testing of fuel assemblies.

Machine Shop (C-4) Mechanical component operations.

Maintenance Shop (C-5) Maintenance craft shops and offices.

North Tank Farm (E/F-7) Tank storage of liquid chemical feed and product materials (hydrofluoric acid, anhydrous and aqua ammonia, sodium hydroxide, nitric acid, nitrogen)

Carpenter Shop (D-4) Carpentry/Painting activities.

Chemical Storage Warehouse (D-5) Storage of containerized chemicals.

Acid/Oxidizer Storage Warehouse (D-5) Storage of containerized chemicals.

Office Buildings 1 through 6 (C-7), 7 (C-6), and 8 (D-8) Office/Administrative functions.

Central Guard Station/Emergency Operations Center (B-6) Security and emergency response operations.

1.1.3 General Process Description

The primary operation of the AREVA NP Richland facility is the manufacture of fuel assemblies and intermediate fuel components for commercial light-water power reactors. The primary uranium feed material to the plant is UF_6, received in 30-inch diameter steel cylinders. After receipt, weighing, and assaying at the UF_6 Cylinder Storage Facility, the cylinders are transferred to the Dry Conversion Facility (DCF) where the UF_6 undergoes chemical conversion to UO_2 powder. Also in the DCF the powder undergoes physical conditioning (powder preparation) to put it in a physical form most amenable to pellet pressing.

For pellet pressing, the UO_2 powder is transferred in containers to the UO_2 Building where it is pressed into pellets. After sintering, the pellets are normally ground to size, inspected, and placed into interim storage. As needed, the pellets are brought to the rod loading area of the UO_2 Building for loading into fuel rod cladding. The loaded and end-capped rods are then available for combining with appropriate fuel bundle hardware to produce finished fuel assemblies. This work takes place in the bundle assembly area within the UO_2 Building. Finished bundles are placed in interim storage or loaded into bundle shipping containers for ultimate transport to utility customers.

A number of noteworthy process flow variants exist relative to the typical process flow outlined above, most notably:

- The final product is not always finished fuel assemblies, but instead may be either uranium oxide powder or pellets, or finished fuel rods. These products are removed from the process at the appropriate point and loaded into licensed shipping containers for shipment to other licensed facilities.
- Fuel pellets containing gadolinia (neutron absorber fuel) are produced in the Specialty Fuels (SF) Building. UO_2 powder produced on-site in the UO_2 Building or DCF, as previously discussed, is blended with gadolinia. The resulting blended powder is pressed into pellets; the pellets are sintered, normally ground to size, and inspected. The finished pellets are transferred to rod loading or, in some cases, may be packaged for shipment off-site to other fuel fabrication facilities.
- Blended low-enriched uranium (BLEU) is received from off-site as uranium oxide powder in licensed shipping containers. The powder is downloaded into drums for interim storage. As needed, this BLEU powder is removed from storage and pressed into pellets; the pellets are sintered, ground to size, inspected, and placed in interim storage. These steps occur primarily within the BLEU addition to the UO_2 Building. Subsequent fuel production steps (rod loading, bundle assembly) occur within the traditional non-BLEU portions of the UO_2 Building.
- Scrap processing utilizes the ADU conversion process in the UO_2 Building. Uranium-bearing scrap (powder, pellets, or other uranium residues) may be generated on-site or received from off-site facilities. The scrap is dissolved in dissolvers located in the UO_2 or ELO Buildings. The resultant uranyl nitrate solutions serve as feed to the ADU conversion process. In some cases the uranyl nitrate may have been processed as an intermediate step through the solvent extraction process in the ELO Building to remove gadolinium or

other contaminants. UO_2 powder produced in the ADU process is placed into drums and transferred to the DCF for powder preparation and returned to drums. This drummed ADU-produced powder proceeds through the subsequent fuel fabrication steps in the same manner as UF_6-derived powder produced in the DCF, as previously discussed.

1.1.4 Raw Materials, Products, By-Products and Wastes

As discussed in Section 1.1.3, the primary uranium feed material for the plant is UF_6. Secondary feeds include BLEU powder, powder or pellets from other fuel cycle facilities, and various uranium-bearing scrap materials. The production, production-support, and waste processing activities are supported by a number of non-radiological chemical materials, most notably bulk quantities of anhydrous and aqua ammonia, nitric acid, nitrogen, and sodium hydroxide. A significant number of chemicals are utilized on-site in lesser quantities.

Finished products of the plant containing licensed material include fuel assemblies, fuel rods, uranium oxide pellets, and uranium oxide powder.

Byproducts produced at the Richland plant include hydrofluoric acid recovered from the dry conversion process and ammonium hydroxide (aqua ammonia) recovered from the ADU process. Recovered hydrofluoric acid is sold as a commercial chemical product; recovered aqua ammonia is recycled into the ADU process but may also be sold as a commercial chemical product. License authorizations exist for the release of these materials (see Section 1.2.5).

The site processes produce liquid, solid, and airborne wastes. Liquid process wastes are collected within the plant's wastewater treatment system. The system provides processes for the treatment/removal of certain constituents and characteristics (ammonia, uranium, particulates, pH) prior to the treated effluent being combined with domestic sewage and other non-hazardous liquid effluents. The combined liquid effluent, after having been sampled for applicable radioactive and non-radioactive chemical constituents, is discharged to the City of Richland sewer at a lift station located immediately to the south of the plant site. Small volumes of certain liquid wastes are containerized for treatment/disposal at appropriate off-site facilities.

The site produces a variety of regulated solid wastes (obsolete equipment and hardware, used ventilation filters, used personal protective equipment, waste treatment residues/filter cakes, demolition debris, miscellaneous combustible waste, etc.). These wastes are typically containerized for shipment off-site to an appropriate low-level waste disposal site. Certain combustible wastes are burned in the on-site SWUR incinerator.

The site facilities discharge airborne effluents to atmosphere via a number of process stacks. All process stacks exhausting air that may contain concentrations of radioactive materials that are significant with respect to the site's compliance with 10 CFR 20 are provided with high-efficiency particulate absolute (HEPA) filtration and continuous sampling for radioactive particulates. A subset of those stacks also emits certain chemical contaminants (oxides of nitrogen, hydrogen fluoride) and is provided with appropriate liquid scrubbers; emissions are quantified via stack monitoring/testing.

Levels of trace radioactive impurities or contaminants (fission products, transuranic elements) in products, by-products, and wastes produced at the Richland plant are a function of the composition of the feed materials to the plant and the processes applied to these materials. Transuranics in feed materials are limited by license condition (see Section 1.2.5) to 50 Bq/gU. Although not a license condition, most feed UF_6 meets the radiological limits for commercial grade UF_6, as specified in ASTM Standard C-996 (2004). Similarly, most non-UF_6 uranium-bearing scrap feeds meet commercial-grade radiological limits specified in ASTM Standard

C-1334 (2005); BLEU-bearing powder feed meets the reprocessed uranium specifications of ASTM C-1334 (2005). Uranium fuel products must meet radiological criteria, including isotopic purity limits, imposed by AREVA NP customers. Processing of all feed materials is managed such that worker exposures meet 10 CFR 20 occupational exposure limits and wastes meet the effluent criteria of 10 CFR 20.

1.2 *Institutional Information*

1.2.1 Corporate Identity

The full name and address of the applicant and the facility is as follows:

AREVA NP Inc.

2101 Horn Rapids Road

Richland, Washington 99354

The U.S. Nuclear Regulatory Commission (NRC) license number for this facility is SNM-1227 (Docket No. 70-1257).

The facility is located within the City of Richland, Benton County, State of Washington.

The applicant is incorporated in the State of Delaware, with its principal corporate offices located at:

3315 Old Forrest Road

Lynchburg, Virginia 24506-0935

AREVA NP Inc. is owned by AREVA NP USA, a Delaware corporation with headquarters in Bethesda, Maryland. AREVA NP USA is, in turn, wholly-owned by AREVA NP, SAS, headquartered in Paris, France. AREVA NP, SAS is jointly owned by Societe des Participations du Commissariat a l'Energie Atomique," a French corporation headquartered in Paris, France and Doing Business As AREVA SA (66%); and Siemens AG, a German corporation headquartered in Munich, Germany (34%).

1.2.2 Financial Qualifications

AREVA NP provides financial information to the NRC as required to reaffirm passage of the financial test underlying its parent company guarantee for decommissioning financial assurance. That information includes audited annual financial statements and supporting independent auditors' reports and demonstrates the company's current and continuing access to the financial resources necessary to safely conduct its licensed activities, including decommissioning. If, in the future, AREVA NP no longer utilizes a parent company guarantee for financial assurance, similar underlying financial qualification information will be provided at a frequency acceptable to the NRC.

1.2.3 Type, Quantity, and Form of Licensed Material

1.2.3.1 Uranium-235

1. Three hundred fifty (350) grams, in addition to the limits listed below, of any enrichment or chemical/physical form for analytical/testing purposes and for sources.
2. Seventy-five thousand (75,000) kilograms contained in uranium compounds in any chemical/physical form enriched to a maximum of 5 wt. % in the U-235 isotope.

The uranium compounds containing the U-235 isotope may include enriched reprocessed uranium materials and their associated radioisotopes, including plutonium and other transuranic isotopes (see Section 1.2.5.1).

1.2.3.2 Plutonium

In addition to the plutonium entering the plant as transuranic contamination in uranium feedstock (see Section 1.2.5.1), plutonium possession is authorized as follows:

1. One milligram and not more than 1.5 millicuries as contained in sealed sources and standards.
2. Less than 500 grams as PuO_2 or PuO_2-UO_2 as stored waste.

1.2.4 Authorized Uses

This license authorizes the use of special nuclear material for the production of uranium fuel products for ultimate usage in nuclear power reactors. This also includes all support activities related to the production of these products, including but not limited to the receipt and storage of raw materials; the storage of finished products and the preparation/offering of these products for transportation off-site; SNM recycling/recovery operations; the processing/disposal of SNM-containing waste materials, excluding on-site burial; process and product development activities; laboratory operations; and maintenance/repair of contaminated equipment and facilities.

This license application requests renewal of License No. SNM-1227 (expiration date November 30, 2006) and meets the 30-day timely renewal criterion of 10 CFR 70.38. The requested renewal period is forty (40) years.

1.2.5 Special Exemptions and Special Authorizations

1.2.5.1 Plutonium and Other Transuranic Contamination in Feedstock

AREVA NP may receive, process, store, and ship reprocessed uranium containing plutonium and other transuranic isotopes.

1. The concentration of transuranic isotopes (i.e., the alpha activity of plutonium and neptunium) in such uranium shall be limited to less than 50 Bq/gU.
2. When it expects to receive such uranium, AREVA NP shall obtain certification from the shipper that the uranium is within the limits for transuranics as specified in Item 1, above.
3. AREVA NP may receive, store, analyze and ship up to 200 gU of samples of reprocessed uranium whose transuranic activity exceeds 50 Bq/gU for purposes of confirming transuranic activity. If it is confirmed that the sample exceeds this limit, AREVA NP will either blend the remaining sample material to comply with the limit or return it to the shipper.

1.2.5.2 Plutonium Contaminated Waste Storage

AREVA NP is presently in possession of plutonium as PuO_2-UO_2 and PuO_2 (see Section 1.2.3.2) left over from past decommissioning of an on-site mixed oxide facility. This material is in the form of contaminated solid waste stored in drums.

These materials shall be stored in accordance with the following controls.

1. The materials shall be confined within closed and appropriately sealed containers in a location where the containers are not exposed to the elements.

2. Assurance of containment shall be verified biannually by visual inspection and smear surveys of the stored waste containers. The inspection and surveys shall be documented. All indications of container leakage shall be investigated and appropriate action taken. Alpha contamination in excess of procedural limits discovered by smear surveys shall be examined for plutonium content.

In order to provide continued containment, AREVA NP may find it necessary to repackage the containers. These activities shall be performed using a special Radiation Work Procedure (RWP).

1.2.5.3 Labeling Exemption

Pursuant to 10 CFR 20.1904(a) requirements, a sign bearing the legend, "Every container or vessel in this area, unless otherwise identified, may contain radioactive material," may be posted at entrances to each building in which radioactive materials are used, stored or handled, in lieu of the requirement to have a "Caution, Radioactive Material" or "Danger, Radioactive Material" label affixed to each container of licensed material.

1.2.5.4 Waste Disposal

Pursuant to 10 CFR 20.2002, disposal of solid waste material containing uranium at 30 pCi/ gram or less to other than a licensed waste disposal facility is authorized. The low enriched uranium shall not exceed 30 pCi/gram of dry solid waste material. The uranium shall be distributed throughout the waste material.

1.2.5.5 Authorization at Reactor Sites

AREVA NP is authorized to possess fuel assemblies or fuel rods at reactor sites for the purpose of loading them into shipping containers and delivering them to a carrier for transport.

For such operations, AREVA NP shall be exempted from conditions of 10 CFR 70.24, "Criticality Accident Requirements," provided:

- As finished fuel assemblies are removed from their storage facilities, they shall be constrained in an arrangement that is no more reactive than that which they will assume in the shipping package.
- The total number of fuel assemblies in process at any one time shall not exceed the maximum authorized contents of the package being loaded.
- If two fuel assemblies are in movement at the same time, a 12-inch-minimum edge-to-edge separation shall be maintained between them, and only one fuel assembly at a time shall be loaded into the shipping package.
- Loaded packages will be stored in an approved shipping array pending delivery to a carrier.

1.2.5.6 Notification

Notifications to the NRC shall be made as required by regulations, with the exception of 10 CFR 20.2202(a)(2) and (b)(2) as they apply to restricted areas. Reports to the NRC shall be made as required by regulations, with the exception of those paragraphs in 10 CFR 20.2203 that refer to 10 CFR 20.2202(a)(2) and (b)(2) as they apply to restricted areas.

1.2.5.7 Authorized Release Guidelines for Hydrofluoric Acid

AREVA NP is authorized to release hydrofluoric acid manufactured by the dry conversion process for unrestricted commercial use, provided the following conditions are met:

1. A representative sample of each batch of hydrofluoric acid product shall be obtained and analyzed for uranium.
2. A batch shall be no larger than the capacity of the applicable storage tank.
3. The uranium activity of any batch released for unrestricted use shall be $\leq$ 3 pCi/ml.

1.2.5.8 Authorized Release Guidelines for Ammonium Hydroxide

AREVA NP is authorized to release ammonium hydroxide produced at the Ammonia Recovery Facility for unrestricted commercial use, provided the following conditions are met:

1. A representative sample of each batch of ammonium hydroxide product shall be obtained and analyzed for uranium.
2. A batch shall be no larger than the capacity of the applicable storage tank.
3. The uranium concentration in the ammonium hydroxide shall not exceed 0.05 ppm.

1.2.5.9 Release from Prior Commitments

All commitments made to NRC staff prior to the approval date of this license application shall no longer be binding on AREVA NP following approval of this license application, unless re-imposed as license conditions.

1.2.6 Terminology

The following definitions apply to terms as used in this license:

Term	Definitions
Weekly	Within each calendar week
Monthly	Within each calendar month
Quarterly	Within each calendar quarter
Biannually or Semi-Annually	Twice per year with an interval not to exceed 8 months
Annually	Once per calendar year with an interval not to exceed 15 months
Biennially	Every second calendar year with an interval not to exceed 27 months
Triennially	Every third calendar year with an interval not to exceed 40 months

1.3 *Site Description*

1.3.1 Site Geography

The AREVA NP Richland fuel fabrication facility is located at 2101 Horn Rapids Road, just within the northern limits of the City of Richland in Benton County, Washington. The fenced exclusion area of approximately 50 acres lies within 320 acres of land owned by AREVA NP within the Horn Rapids Industrial Park. The property is situated at approximately latitude N46°21'003" and longitude W119°18'020" in Sections 15 and 16 of Township 10N, Range 28E, Willamette Meridian. The facility itself is located in the southwest quarter of Section 15 (15-SW/4).

The facility is bounded on the north by Horn Rapids Road, an asphalt roadway providing access to the plant and located approximately 300 feet north of the fenced exclusion area; on the south by Battelle Boulevard, approximately 450 feet south of the fenced exclusion area; on the west by Kingsgate Boulevard, approximately 2,500 feet to the west of the fenced exclusion area, and on the east by Kelly Avenue, approximately 750 feet to the east of the fenced exclusion area. Stevens Drive, the primary route south into Richland, is approximately 4,000 feet to the east. There are no major highways in the immediate vicinity of the plant.

There are no bodies of surface water adjacent to or in the immediate vicinity of the plant. The Columbia River is located approximately 1.5 miles to the east and the Yakima River, a tributary to the Columbia, passes approximately 2 miles to the west. The Columbia River, the much larger of the two, is regulated by multiple dams upstream of Richland. At its closest point, the site lies approximately 25 feet above the normal level of the Columbia.

The immediate area surrounding the site is a relatively flat and essentially featureless plain. There are no significant geographic features that may impact accident analyses within one mile of the site.

1.3.2 Demographics

The AREVA NP Richland plant is located within the Horn Rapids Industrial Park and therefore there are no residences adjacent to or in the immediate vicinity of the plant. The nearest residential areas are about 1.5 miles to the southwest in the City of Richland. The City of Richland, with a current approximate population of 43,520, is a part of the Tri-Cities metropolitan area. Other major population centers within the Tri-Cities metropolitan area include Kennewick, also located within Benton County, at a distance of approximately ten miles southeast of the plant (approximate population 60,410); Pasco, located in adjacent Franklin County, at a distance of approximately seven miles southeast of the plant (approximate population 44,190); and West Richland, in Benton County, at a distance of approximately five miles southwest of the plant (approximate population 10,210).

There are no public facilities (schools, hospitals, parks) in the immediate vicinity of the plant site. The nearest schools, WSU at Tri-Cities and the Hanford Middle School/High School, are approximately two miles southeast of the plant and the northernmost portion of Leslie Groves Park along the Columbia River is about three miles southeast of the site. The West Richland Public Golf Course is approximately 3.5 miles southwest of the plant. The nearest hospital, Kadlec Hospital, is located approximately five miles south of the plant in Richland.

Land use within the one mile zone around the plant is a mixture of agricultural activities with a number of rather widely interspersed industrial facilities. The industrial facilities, including Pacific Eco Solutions, a low-level radioactive waste processing facility located approximately 0.5 miles from the plant, do not manage hazardous materials in quantities that pose hazards to the AREVA NP facility under normal or off-normal conditions. The land north of the AREVA NP site, across Horn Rapids Road, is part of the approximately 550-square-mile U.S. Department of Energy (DOE) Hanford Site. The nearest major Hanford operational area, the 300 Area, is located approximately three miles north of the plant. It has no remaining significant industrial activities and is being progressively shut down. Similarly, the U.S. DOE Fast Flux Text Facility, located seven miles north of the plant, is shut down and being decommissioned. The Energy Northwest Columbia Nuclear Generating Station is located eight miles north of AREVA NP; as such, AREVA NP is within the ten-mile emergency planning zone for that facility.

The Columbia and Yakima Rivers, located 1.5 and 2 miles, respectively, from the plant, are used primarily for recreational purposes (boating, fishing, etc.) and serve as a source of

irrigation water for agricultural activities. The Columbia River serves as a source of drinking water for Richland and the immediately downstream communities of Kennewick and Pasco. The Columbia River is also utilized by a limited number of cruise boats visiting the Hanford Reach portion of the river and very infrequently by barges delivering large components to the Hanford Site.

1.3.3 Meteorology

The prevailing wind at the AREVA NP site is from the southwest along the Yakima River corridor, which enters the Columbia Basin near the site. Secondary direction frequency maxima are from the northwest and the southeast along the axis of the Columbia River valley, and the lowest frequencies are from the east and northeast. Based on long-term wind speed data from the adjacent DOE Hanford Site, wind speeds are between 1 to 12 mph approximately 88% of the time (36.8% at 4 to 7 mph, 32.5% at 1 to 3 mph, and 18.6% at 8 to 12 mph).

Benton County is subject to high winds and blowing dust. Wind speeds can reach 60+ miles per hour. The AREVA NP Richland facility, located on the southern border of the Hanford region, experiences high wind speeds due to squall lines, frontal passages, strong pressure gradients, and thunderstorms. The Hanford historical wind speed-direction data show that daily peak gusts of at least 40 mph have occurred from all but 4 of the 16 compass points.

Based on historical Hanford area wind data, the peak gust of wind at 50 feet of elevation expected to occur once in 100 years is 86 mph. This gust speed translates to a fastest mile wind speed of 66 to 78 mph. The fastest mile of wind is generally defined as either the fastest speed associated with one mile of passing wind or the fastest observed one minute wind speed. The facilities at the AREVA NP Richland facility are designed and constructed in accordance with the Uniform Building Code to withstand sustained winds of 80 mph without appreciable damage.

The Tri-Cities region has a very dry climate with rather mild winters and warm sunny summers. The average annual precipitation in the Richland area is approximately 6 to 6.5 inches, with nearly half of the precipitation occurring during November, December and January. Most of the precipitation occurs as rain.

Based on historic data, the Hanford area can expect two inches of rain in 24 hours once per 100 years. The probable maximum precipitation (PMP) event calculated for the Columbia Generating Station of Energy Northwest (located approximately 8 miles northeast of AREVA NP) is 9.2 inches of rain in five hours from a thunderstorm. Building roofs at the AREVA NP Richland facility are designed for a rain load of 20 lb/ft^2 (approximately 4 inches of water) and have adequate drainage.

The average annual snowfall in the lower Columbia Basin is approximately 14 inches. Four to six inches is the average depth of snow that stays on the ground for two to four weeks. Building roofs at the AREVA NP Richland facility site are designed for a snow load of 20 lb/ft^2 (approximately 24 inches of fresh snow) and have adequate drainage.

No tornados have been sighted in Benton County since 1956; however, between 1950 and 1980, two sightings were recorded in adjacent Grant County and three events were recorded in adjacent Walla Walla County. Due to the low frequency of tornados in this area, no specific design criteria relative to tornados are required in the Uniform Building Code (UBC).

Because the lightning risk to the AREVA NP Richland facility site is low, the design bases for the facility buildings do not include protection against lightning. The most probable consequence of a lightning strike on the AREVA NP Richland facility site is a loss of normal

power. Electronic components may be damaged and, in certain highly unlikely circumstances, emergency back-up power could be lost to some equipment. It is also possible, but highly unlikely, that a fire at the site could be initiated by lightning.

1.3.4 Hydrology

There are no bodies of surface water adjacent to or in the immediate vicinity of the plant. The Columbia River, approximately 1.5 miles to the east, is historically subject to flooding but is now highly regulated by the many dams upstream of Richland within the State of Washington and province of British Columbia. The Yakima River, approximately 2 miles to the southwest, is a source of recharge for the shallow, unconfined groundwater aquifer below the site. Recharge to that aquifer also occurs from infiltration of precipitation runoff from ridges surrounding the Columbia Basin as well as from the infiltration of irrigation water applied to farmland directly upgradient of the plant.

Depth to water table in the vicinity of the plant ranges from approximately 10 to 50 feet below land surface but is typically only approximately 15 to 20 feet below land surface in the fenced portion of the facility. The unconfined aquifer is typically 20 feet thick and is separated from the lower confined aquifer by a 30- to 35-foot-thick silt aquitard. The groundwater beneath the site exhibits a very flat gradient and flows from south-southwest to north-northeast at an estimated average groundwater flow velocity range of from 2 to 8 feet/day. A potentiometric surface map for the groundwater beneath the site is provided as Figure 1-3.

The AREVA NP Richland facility site lies between the Yakima and Columbia Rivers. The flows of both rivers are regulated by multiple dams upstream of the site. Floods have historically been a common hazard in Benton County, but on the DOE Hanford Reservation adjacent to the AREVA NP Richland facility, the probable maximum flood, as determined by the Corps of Engineers, would have little effect on the area as a whole. Historical flood frequency data for the Hanford area shows that a 500-year flood from rainfall or snowmelt will not reach the AREVA NP Richland facility with or without the presence of the flood control dams on the Yakima, Columbia, and Snake Rivers. Therefore, flood-related hazards are not considered a viable risk at the site.

1.3.5 Geology

The Columbia Basin is underlain by thick sequences of basaltic lava flows more than 10,000 feet thick. Within the area of the basaltic lava flows are a number of structural basins containing layers of unconsolidated sands and gravels tens to hundreds of feet thick over the basaltic bedrock. The AREVA NP site lies near the southwestern margin of the largest of such structural basins, known as the Pasco Basin. Underlying the site itself are poorly and well-graded sands and gravels that, in turn, are underlain by a silt aquitard layer occurring from 30 to 50 feet below land surface with a thickness of 30 to 35 feet. The silt aquitard separates the unconfined groundwater aquifer in the sands and gravels above it from the confined aquifer in the sands and gravels below it.

The UBC Seismic Risk Map places the AREVA NP site within Seismic Zone 2B. The UBC for this seismic zone requires structures to be able to withstand a peak ground acceleration (PGA) of 0.20g. The AREVA NP Richland facility plant structures are designed to withstand this level of earthquake acceleration with no significant structural damage. As such, the buildings will withstand acceleration associated with a Seismic Zone 2B event without experiencing a loss of geometry control provided by the facility design.

The U.S. DOE Hanford Site, which is adjacent to AREVA NP, has been extensively investigated for earthquake potential. The records of eastern Washington show only infrequent, low-

intensity, deep earthquakes. Historical Hanford area seismic information shows that the horizontal PGA return frequencies are as follows:

Return Period (Years)	100	500	1,000	2,500	5,000	10,000	100,000
PGA (g)	0.040	0.097	0.139	0.216	0.295	0.398	0.884

During the past 100 years, there have been three earthquakes of intensity large enough to cause moderate damage to structures within 30 to 60 miles of the site, though no damage has been reported at the AREVA NP Richland site itself.

Most of the landslide areas in Benton County are 500 to 2,000 years old and are now stable. The Rattlesnake Hills, running along the southwest boundary of the Hanford Reservation and extending to the west of Kennewick, have steep slopes (over 15%), with some slopes greater than 50%, but these areas are also stable at present. There are no steep or unstable slopes on site or adjacent to the AREVA NP Richland facility.

The most recent volcanic activity affecting the plant site via minor ash fall was the eruption of Mt. St. Helens in 1980. The lower-activity 2004 eruption did not affect the plant site. Relatively recent eruptions of other volcanoes that affected Washington State include eruptions of Mt. Baker (1975), Mt. Hood (1800 to 1804), Mt. Rainier (1820 to 1894), and Mt. Adams (1000 to 2000 years ago). An eruption of one of these volcanoes, as well as Mt. St. Helens, could possibly lead to ash fall at the AREVA NP Richland facility site. No other serious effects are likely.

Figure 1-1 Fuels Manufacturing Plant Arrangement

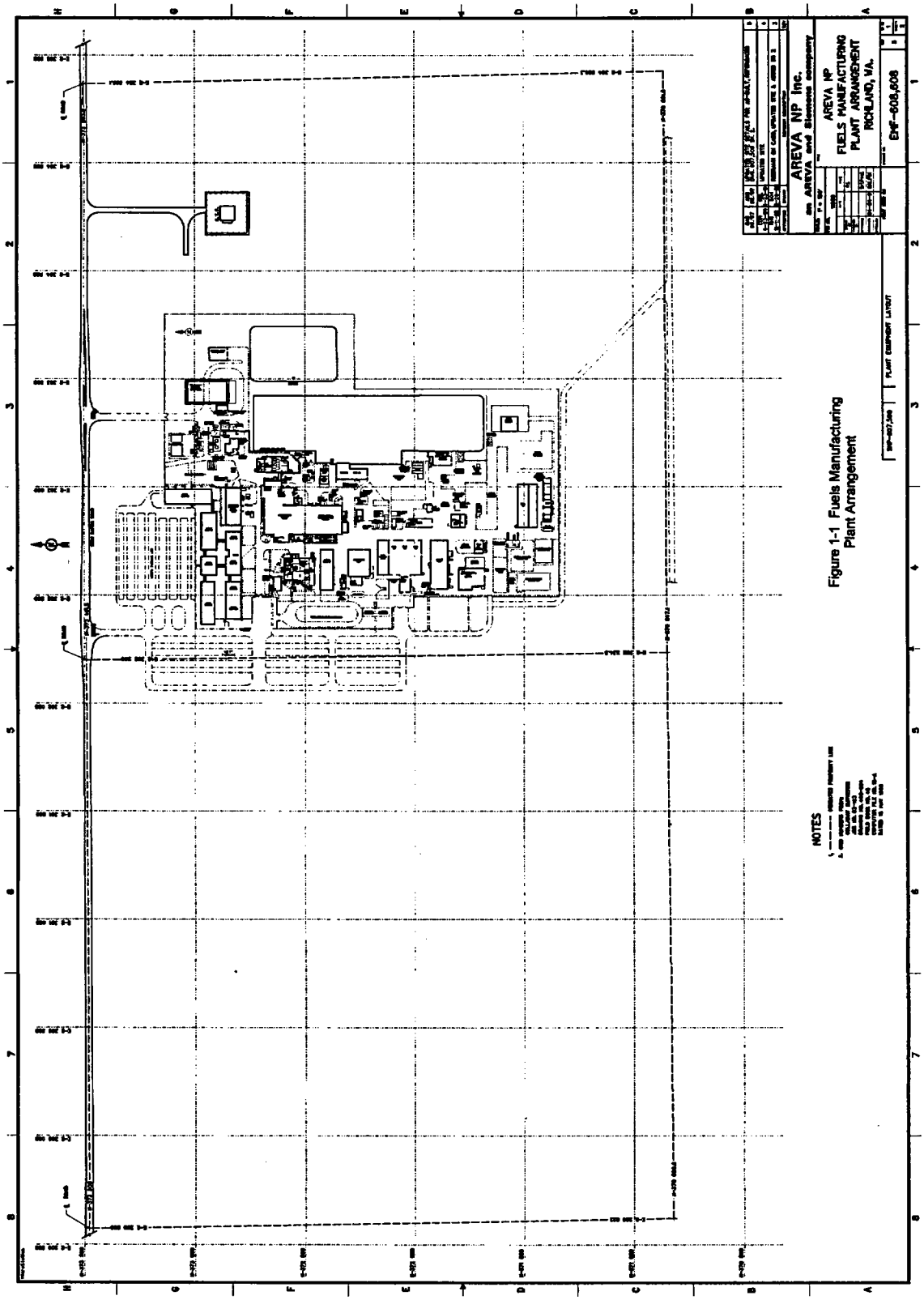


Figure 1-2 Site General Arrangement

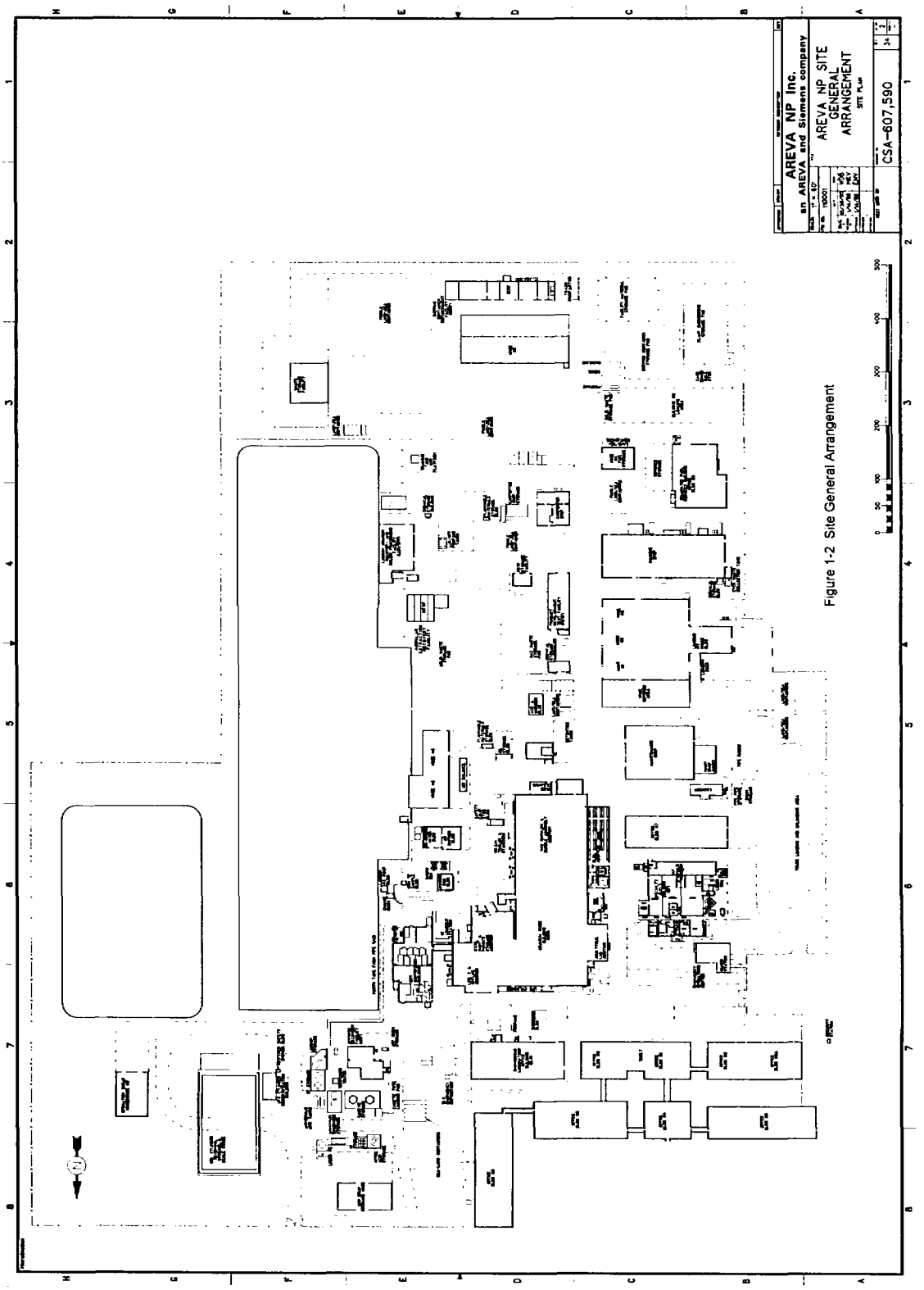
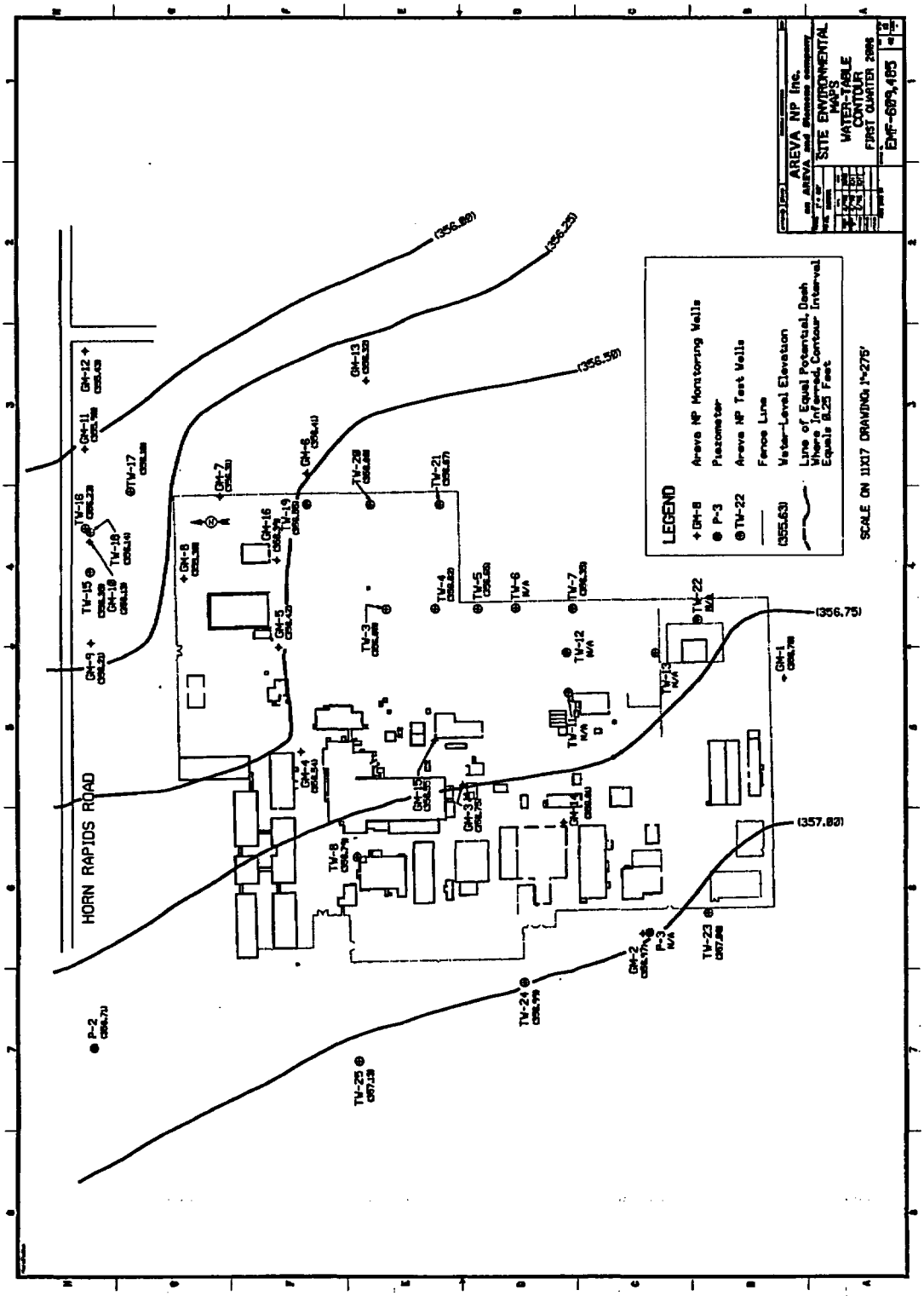


Figure 1-3 Groundwater Potentiometric Surface Map



PROJECTS

MicroStation

NOTES

1. ——— INDICATES PROPERTY LINE
2. GRID NUMBERS FROM:
   COLLADAY SURVEYING
   JOB NO. 82-123
   DRAWING NO. 400-264
   FIELD BOOK NO. 45
   COMPUTER FILE NO. 16-A
   DATED 18 MAY 1982

Figure 1-1 Fuels Manufacturing Plant Arrangement

REV	DESCRIPTION	APPROVED	DRAWN
5	UPDATED SITE DETAILS PER AS-BUILT, REFERENCED DWG. 607,508 SH. 2.	RAB 08/07	ABB 08/07
4	UPDATED SITE	CDN 6-25-94	RDN 6-25-94
3	REDRAWN ON CADD, UPDATED SITE & ADDED SH 2	TPD 11-11-92	BDH 5-28-92

AREVA NP Inc.
an AREVA and Siemens company

SCALE: 1" = 150'
PW NO. 11200

	DATE	INIT
DRAWN BY		RS
CHECKED BY		
APPROVED		BSPING
APPROVED	04-04-8	GALPIN
APPROVED		

NEXT USED ON

AREVA NP
FUELS MANUFACTURING
PLANT ARRANGEMENT
RICHLAND, WA.

DRAWING NO. EMF-608,608 REV 5 SH NO. 1 SHEETS 2

SNP-607,590 | PLANT EQUIPMENT LAYOUT

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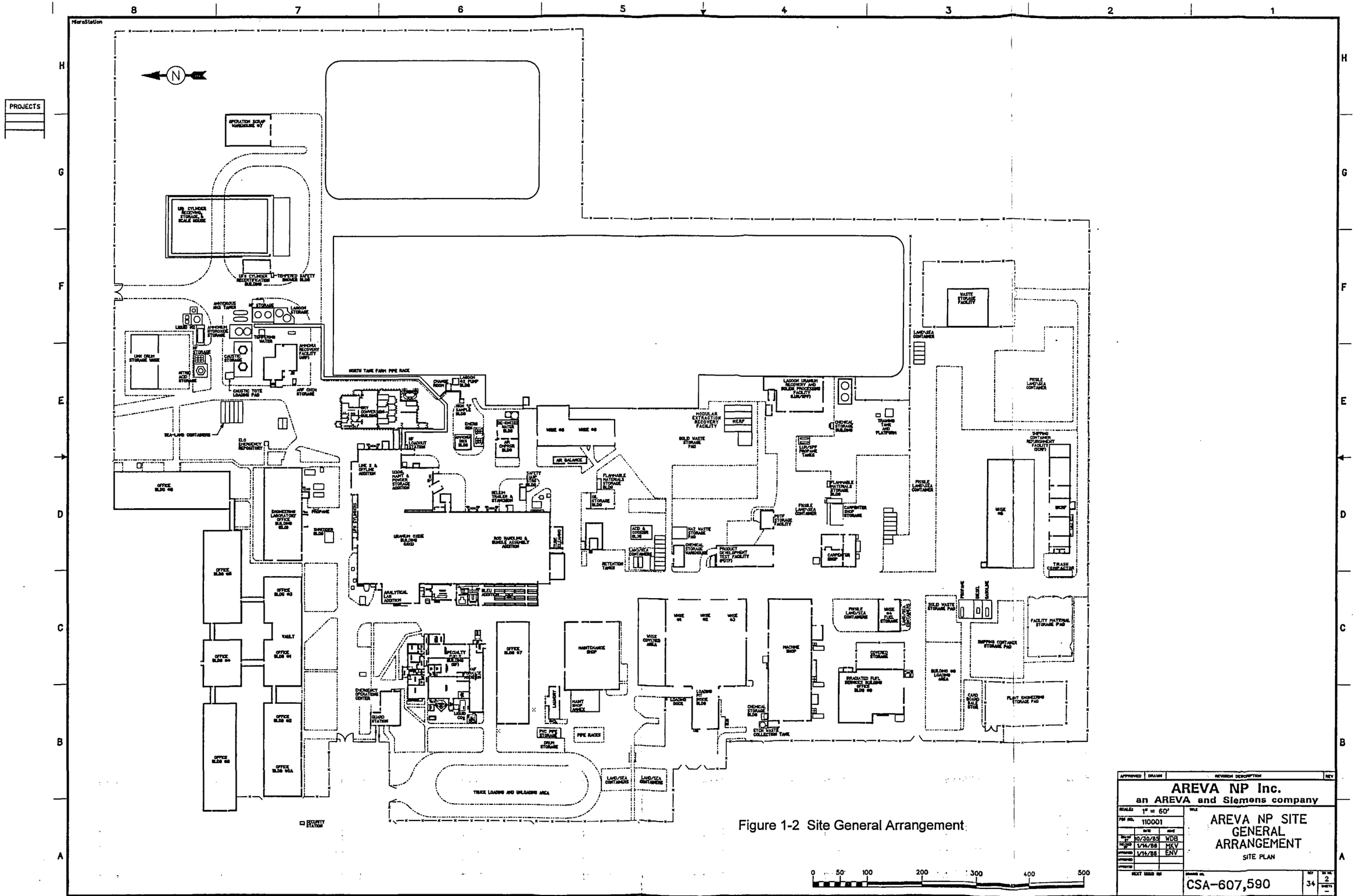

Figure 1-2 Site General Arrangement

APPROVED	DRAWN	REVISION DESCRIPTION	REV

AREVA NP Inc.
an AREVA and Siemens company

SCALE:	1" = 60'
PDM NO.	110001
DRAWN BY	10/30/85 WDB
RECORD BY	1/14/86 MKV
APPROVED	1/14/86 ENV
APPROVED	
APPROVED	
NEXT ISSUED ON	

TITLE: AREVA NP SITE GENERAL ARRANGEMENT
SITE PLAN

DRAWING NO. CSA-607,590 REV 34 SH. NO. 2 SHEETS —

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EHS&L Document

SNM-1227 - Chapter 2
Organization and Administration

Nature of Changes

Item	Paragraph	Description	Justification
1.	New Document	New Document	License Renewal
2.			
3.			
4.			
5.			
6.			
7.			
8.			
9.			
10.			
List Below any Documents, including Forms & Operator Aids which must be issued concurrently with this document revision:			
Do Not make document effective until Loren Maas releases after NRC approval.			

This Document contains a total of 9 pages excluding the signature page generated by Documentum, the document control application software.

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Minor Changes: If the proposed changes are limited to editorial and/or administrative changes check the box at the right. The document will be routed directly for review by EHS&L without technical review. All applicable approvals must still be obtained.	☐

Document Reviews			**Document Approvals**	
Purpose/Function of Review	Specify Reviewer(s) (Optional except for change author)	(Check all that apply)	Title of Approver	(Check all that Apply)
Document Control (Automatic)		☒	Document Control (Automatic)	☒
Change Author	LJ Maas	☒	Author	☒
Independent Technical Review	RE Link	☒		
Operability Review(s)			Mgr, Richland Operations[1]	☐
Conversion		☐	Mgr, Uranium Conversion & Recovery Operations[1]	☐
Recovery		☐		
Ceramics		☐	Mgr, Ceramic Operations[1]	☐
Rods		☐	Mgr, Rods & Bundles[1]	☐
Bundles		☐		
Transportation		☐		
Components		☐	Mgr, Component Fabrication[1]	☐
Maintenance Review		☐	Mgr, Maintenance[1]	☐
Lab Review		☐	Mgr, Analytical Services[1]	☐
EHS&L Review(s)			Mgr, EHS&L[2]	☒
Criticality		☐	Mgr, Criticality Safety[2]	☐
Radiation Protection		☐	Mgr, Safety, Security & Emergency Preparedness[2]	☐
Safety/Security		☐		
Emergency Preparedness		☐		
MC&A	LJ Maas	☒	Mgr, Licensing & Compliance[2]	☐
Transportation		☐		
Environmental		☐		
BWR Product Eng. Review		☐	Mgr, BWR Product Engineering	☐
Quality Review		☐	Mgr, Quality	☐
Ops. Projects & Planning Review		☐	Mgr, Ops. Projects & Planning	☐
Others:		☐	Mgr, Richland Site/Other	☐
Training & Employee Dev.:[3]		☐	Training & Employee Dev.	☐

[1]Note: If approvals include 2 or more product center managers, the Operations manager can be substituted for the applicable product center managers.

[2]Note: If approvals include 2 or more EHS&L functional managers, the EHS&L manager can be substituted for the applicable EHS&L functional managers.

[3]Note: Training department review is required for all procedures that require or affect a Learning Plan and if additional training materials or curriculum must be revised before issuing procedure.

EHS&L Change Impact Evaluation Form		
Document / ECN No*.: E10-08-002	Change Evaluator: RE Link	
Does the change potentially impact Criticality Alarm System (CAS) coverage?	☐ Yes ☒ No	If yes, explain:
NRC Pre-Approval Evaluation:		
Is NRC Pre-approval (License Amendment) Needed? (Based on "Yes" answer to any of five questions below). (Based on "No" answer to all five questions below).	☒ Yes ☐ No	This is part of our NRC license renewal.
1. Does the change create new types of accident sequences that, unless mitigated or prevented, would exceed the performance requirements of 10 CFR 70.61 (create high or intermediate consequence events) and that have not previously been described in AREVA NP Inc's ISA Summary?	☐ Yes ☒ No	If yes, explain:
2. Does the change use new processes, technologies, or control systems for which AREVA NP Inc. has no prior experience?	☐ Yes ☒ No	If yes, explain:
3. Does the change remove, without at least an equivalent replacement of the safety function, an item relied on for safety that is listed in the ISA Summary?	☐ Yes ☒ No	If yes, explain:
4. Does the change alter any item relied on for safety, listed in the ISA Summary, that is the sole item preventing or mitigating an accident sequence of high or intermediate consequences?	☐ Yes ☒ No	If yes, explain:
5. Does the change qualify as a change specifically prohibited by NRC regulation, order or license condition?	☐ Yes ☒ No	If yes, explain:
Actions Required Prior to or Concurrent with Change Implementation Evaluation:		
Action		**Explanation**
6. Modification / Addition to CAS system or system coverage documentation	☐ Yes ☒ No	If yes, explain:
7. Acquire NRC pre-approval (license amendment)	☒ Yes ☐ No	If yes, explain: License renewal.
8. Conduct/modify ISA	☐ Yes ☒ No	If yes, explain:
9. ISA Database Modification	☐ Yes ☒ No	If yes, explain:
10. Modification of other safety program information / underlying analyses (PHA, RHA, FHA, NCSA, etc.)	☐ Yes ☒ No	If yes, explain:
Actions required subsequent to Change Implementation Evaluation:		
11. Update safety program information (PHA,RHA,FHA,NCSA, P&ID)	☐ Yes ☒ No	If yes, explain:

* **If this form exists as a part of a document, the document number is not required.**

Table of Contents

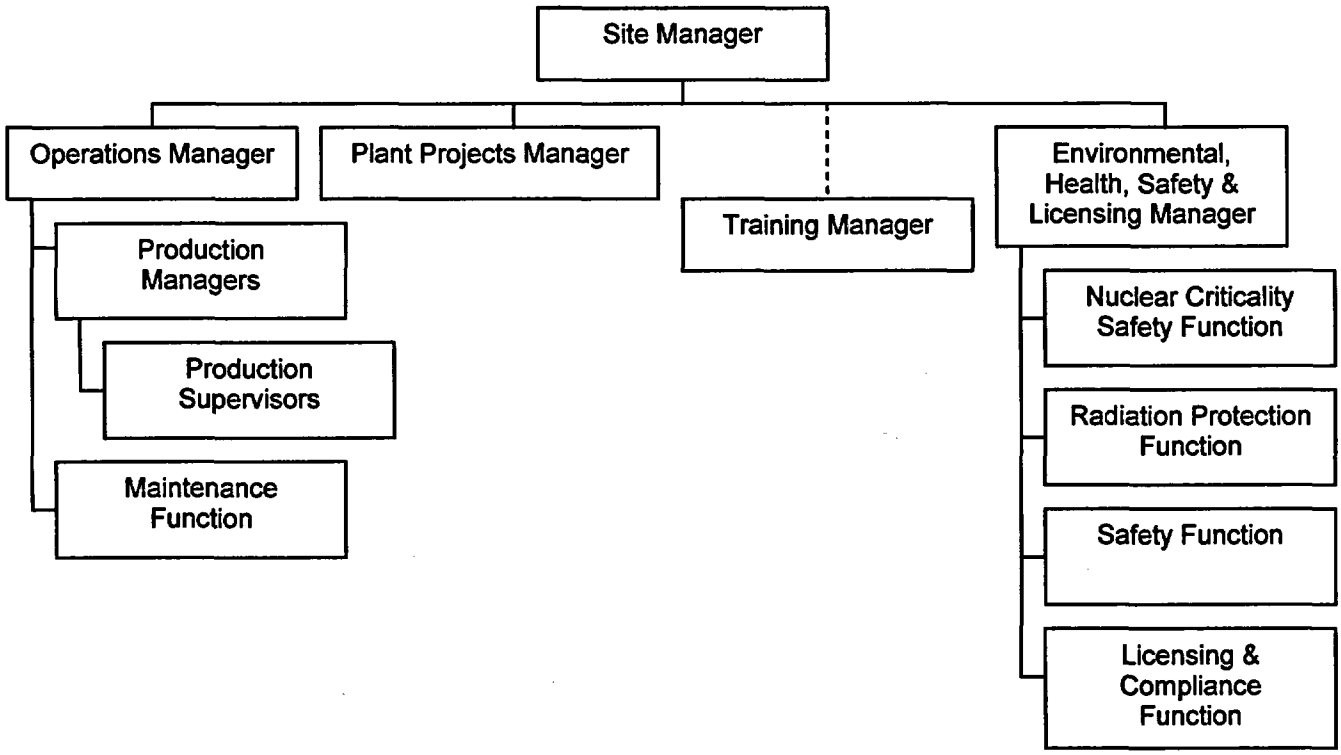
Application Date:	NRC Docket No. 70-1257
October 27, 2006	

2.0 Organization and Administration

2.1 *Site Organization*

The AREVA NP Inc. (AREVA NP) Fuel America (FA) Vice President, Manufacturing, has the ultimate responsibility for ensuring that Richland site operations utilizing special nuclear material (SNM) are conducted in a manner that is protective of its workers, the public, and the surrounding environment, and remain in compliance with applicable Federal, State, and local regulations, licenses, and permits. This is accomplished by putting in place an on-site organization with defined accountabilities and assuring that the organization is given the authority and resources to meet its objectives. The primary components of that organization relevant to plant safety, their accountabilities and the key administrative measures utilized to assure safe plant operations are described below. The organization is depicted on a functional basis in Figure 2.1.

Figure 2.1 Site Management Organization



2.2 *Organizational Responsibilities and Authority*

2.2.1 Site Manager

The Site Manager has the overall responsibility for the nuclear fuel manufacturing activities on the Richland site. This includes responsibility for production activities, as well as the responsibility for assuring that those activities are conducted in a manner that is protective of workers, the public, and the environment. These responsibilities shall be discharged by:

- The designation of defined responsibilities to qualified personnel.
- The establishment of mechanisms for the review of program effectiveness.

The Site Manager shall have a degree in science or engineering, a minimum of two years' experience in the nuclear industry, and at least five years' experience in management, or a combination of education and experience judged appropriate by the Vice President, Manufacturing.

2.2.2 Operations Manager

The Operations Manager has overall responsibility for fuel manufacturing activities and, as such, for operations involving the receipt, processing, storage, and shipment of SNM. Inherent in that responsibility is assurance that the operations are conducted safely and in compliance with license conditions. Control shall be established by:

- The designation of defined responsibilities to qualified personnel.
- The assurance that operating personnel are provided adequate work instructions and have been properly trained.
- The prompt correction of non-conforming conditions.

The Operations Manager is also responsible for the plant maintenance function, which includes activities to assure that Items Relied On For Safety (IROFS) are available and reliable when needed.

The Operations Manager shall have a degree in science or engineering, a minimum of two years' experience in the nuclear industry, and at least five years' experience in management, or a combination of education and experience judged appropriate by the Site Manager.

2.2.3 Production Managers

Production managers have responsibility for nuclear fuel manufacturing activities involving SNM. Those activities entail the safe use and control of SNM from initial receipt, through stages of processing, to ultimate shipment of product or process-related wastes. This authority, with regard to direct production activities, is conducted via a network of production supervisors overseeing trained workers who proceed in accordance with formal operating procedures.

Each production manager shall have a degree in science or engineering and at least two years' experience in the nuclear industry, or a combination of education and experience judged appropriate by the Operations Manager.

2.2.4 Production Supervisors

Production supervisors are directly responsible for the control of materials, personnel, equipment and activities in specific areas. These responsibilities include assuring that formal approved procedures are available and adhered to by operators and other applicable personnel.

Minimum qualifications for production supervisors shall include a high school education and two years' experience in the nuclear industry. Experience shall include practical application of criticality control and radiological safety techniques, and familiarity with specific applicable limitations imposed on production operations.

2.2.5 Environmental, Health, Safety and Licensing (EHS&L) Function

The EHS&L function has overall responsibility for the development and implementation of programs addressing worker health and safety; environmental protection; and licensing/permitting, including monitoring compliance with those licenses and permits. Technical EHS&L areas addressed within this overall function include nuclear criticality safety; radiation protection; environmental protection; integrated safety analysis; nuclear materials safeguards; industrial hygiene and safety; emergency preparedness; fire protection; and security. This function's responsibility with respect to manufacturing operations is only to confirm the safety of those operations, but it has authority to order shutdown and approve re-start of operations that are judged to be unsafe for continued operation or non-compliant with applicable regulatory requirements.

The individual responsible for the EHS&L function shall have a degree in science or engineering, with at least five years of experience that would develop an understanding of the health, safety, and environmental aspects of SNM processing activities.

2.2.5.1 Nuclear Criticality Safety Function

The nuclear criticality safety function has responsibility for the development and implementation of a comprehensive nuclear criticality safety program, as defined in Chapter 5.0, "Nuclear Criticality Safety." Key responsibilities include the performance of nuclear criticality evaluations of applicable SNM operations and changes to those operations; establishing limits and controls based on those evaluations; assuring the proper incorporation of limits and controls into applicable work instructions; and monitoring plant compliance with the criticality safety requirements.

The individual responsible for the nuclear criticality safety function shall have a degree in science or engineering, with at least three years' experience in nuclear criticality safety analysis. The criticality analysts working in the nuclear criticality safety function shall have a degree in science or engineering and are subject to successfully completing a formal internal training and qualification program.

2.2.5.2 Radiation Protection Function

The radiation protection function has responsibility for the development and implementation of a comprehensive program to limit radiological personnel exposures and environmental impacts associated with manufacturing and manufacturing support activities. The radiation protection function includes the Health and Safety Technicians (HSTs) who perform the day-to-day radiological surveillance activities required in the plant (e.g., workplace air sampling, effluent sampling, and contamination surveys). The radiation protection function is also responsible for the plant ALARA program, which is aimed at maintaining occupational exposures to radiation and

radioactivity in environmental effluents as low as reasonably achievable.

The individual(s) responsible for the radiation protection function shall have a degree in science or engineering, with at least three years' experience in radiation protection programs. Applicable work experience providing an understanding of radiation protection principles and programs may be substituted for the post-secondary educational requirements on the basis of two years' experience per one year of academic study. Assignment of an individual with no post-secondary education will require a minimum of ten years of applicable work experience. The HSTs shall have a high school diploma or GED equivalent and are subject to successfully completing a formal internal training and qualification program.

2.2.5.3 Safety Function

The safety function has responsibility for industrial safety/hygiene, fire protection, emergency preparedness, and security.

The individual(s) responsible for the safety function shall have a degree in a technical field, with at least two years' experience in one or more of the safety disciplines included in this function. Applicable work experience providing an understanding of one or more of the pertinent safety disciplines may be substituted for the post-secondary educational requirements on the basis of two years' experience per one year of academic study. Assignment of an individual with no post-secondary education will require a minimum of ten years of applicable work experience.

2.2.5.4 Licensing and Compliance Function

The licensing and compliance function has overall responsibility for acquiring and maintaining environmental, health, and safety-related licenses and permits as required to operate the Richland facility. In this regard, the licensing and compliance function has broad responsibility for interface with regulatory agencies relative to manufacturing-related activities. In addition to this role, this function has technical responsibility for the plant nuclear material accountability and environmental programs. Responsibility relative to radiological environmental programs is shared with the Radiation Protection function.

The individual responsible for the licensing and compliance function shall have a degree in a technical field, with at least two year' experience in the nuclear or general environmental, safety and health field, or a combination of education and experience judged appropriate by the manager of the EHS&L function.

2.2.6 Plant Projects Function

The plant projects function provides engineering services and support for the facilities, equipment, and peripheral support systems involved in product manufacturing, process development, and research and development. This involves support for existing equipment and systems, as well as engineering services for modifications and/or additions to plant equipment and facilities. This includes ownership of the plant's configuration management system for equipment, facilities, and systems.

The individual responsible for the plant projects function shall have a degree in engineering and at least two years' experience in the nuclear industry, or a combination of education and experience judged appropriate by the Site Manager.

2.2.7 Training Function

The training function is responsible for the development, implementation, and administration of

plant training programs, including maintenance of the plant training database. The training programs provided and/or coordinated by the training function address qualification of workers to perform work activities involving SNM (work station training) as well as required safety training.

The individual responsible for the training function shall have a degree and at least two years' experience in technical training or adult education, or a combination of education and experience judged appropriate by the Site Manager.

2.3 *Administration*

2.3.1 Management Measures

AREVA NP has established management measures to ensure that engineered and administrative controls and control systems that are identified as items relied on for safety pursuant to 10 CFR 70.61 (e) are designed, implemented, and maintained to ensure they are available and reliable to perform their function as needed to comply with the performance requirements of 10 CFR 70.61. Those management measures include: 1) configuration management, 2) maintenance, 3) training and qualification, 4) procedures development and implementation, 5) audits and assessments, 6) incident investigation and corrective action, 7) records management, and 8) quality assurance for IROFS.

AREVA NP's programs for provision of these management measures are detailed in Chapter 11, "Management Measures."

2.3.2 Reporting of Unsafe Conditions or Activities

AREVA NP provides to employees a uniform mechanism for the reporting of unsafe conditions or activities to the EHS&L function via the FA Corrective Action Program. The concern is captured via a Condition Report (CR) and entered/managed as an EHS&L Condition. The CR is processed through a screening team with EHS&L representation. The team assigns an issue owner and an importance level that, in turn, defines follow-up investigation/evaluation requirements. Corrective actions are assigned and tracked to completion via the Corrective Action Program.

The reporting of unsafe conditions with immediate emergency implications is addressed in the site emergency plan described in Chapter 8, "Emergency Management."

2.3.3 Off-Site Emergency Response Resources

AREVA NP maintains written agreements with appropriate off-site organizations for the provision of emergency fire, police, ambulance/rescue, and medical services. These agreements are also addressed in Chapter 7, "Fire Safety," and Chapter 8, "Emergency Management."

EHS&L Document

SNM-1227 - Chapter 3
Integrated Safety Analysis (ISA) and ISA Summary

Nature of Changes

Item	Paragraph	Description	Justification
1.	New Document	New Document	License Renewal
2.			
3.			
4.			
5.			
6.			
7.			
8.			
9.			
10.			
List Below any Documents, including Forms & Operator Aids which must be issued concurrently with this document revision:			
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Document Control (Automatic)		☒	Document Control (Automatic)	☒
Change Author	LJ Maas	☒	Author	☒
Independent Technical Review	CD Manning	☒		
Operability Review(s)			Mgr, Richland Operations[1]	☐
Conversion		☐	Mgr, Uranium Conversion & Recovery Operations[1]	☐
Recovery		☐		
Ceramics		☐	Mgr, Ceramic Operations[1]	☐
Rods		☐	Mgr, Rods & Bundles[1]	☐
Bundles		☐		
Transportation		☐		
Components		☐	Mgr, Component Fabrication[1]	☐
Maintenance Review		☐	Mgr, Maintenance[1]	☐
Lab Review		☐	Mgr, Analytical Services[1]	☐
EHS&L Review(s)			Mgr, EHS&L[2]	☒
Criticality		☐	Mgr, Criticality Safety[2]	☐
Radiation Protection		☐	Mgr, Safety, Security & Emergency Preparedness[2]	☐
Safety/Security		☐		
Emergency Preparedness		☐		
MC&A	LJ Maas	☒	Mgr, Licensing & Compliance[2]	☐
Transportation		☐		
Environmental		☐		
BWR Product Eng. Review		☐	Mgr, BWR Product Engineering	☐
Quality Review		☐	Mgr, Quality	☐
Ops. Projects & Planning Review		☐	Mgr, Ops. Projects & Planning	☐
Others:		☐	Mgr, Richland Site/Other	☐
Training & Employee Dev.:[3]		☐	Training & Employee Dev.	☐

[1]Note: If approvals include 2 or more product center managers, the Operations manager can be substituted for the applicable product center managers.

[2]Note: If approvals include 2 or more EHS&L functional managers, the EHS&L manager can be substituted for the applicable EHS&L functional managers.

[3]Note: Training department review is required for all procedures that require or affect a Learning Plan and if additional training materials or curriculum must be revised before issuing procedure.

EHS&L Change Impact Evaluation Form		
Document / ECN No*.: E10-08-003		Change Evaluator: CD Manning
Does the change potentially impact Criticality Alarm System (CAS) coverage?	☐ Yes ☒ No	If yes, explain:
NRC Pre-Approval Evaluation:		
Is NRC Pre-approval (License Amendment) Needed? (Based on "Yes" answer to any of five questions below). (Based on "No" answer to all five questions below).	☒ Yes ☐ No	This is part of our NRC license renewal.
1. Does the change create new types of accident sequences that, unless mitigated or prevented, would exceed the performance requirements of 10 CFR 70.61 (create high or intermediate consequence events) and that have not previously been described in AREVA NP Inc's ISA Summary?	☐ Yes ☒ No	If yes, explain:
2. Does the change use new processes, technologies, or control systems for which AREVA NP Inc. has no prior experience?	☐ Yes ☒ No	If yes, explain:
3. Does the change remove, without at least an equivalent replacement of the safety function, an item relied on for safety that is listed in the ISA Summary?	☐ Yes ☒ No	If yes, explain:
4. Does the change alter any item relied on for safety, listed in the ISA Summary, that is the sole item preventing or mitigating an accident sequence of high or intermediate consequences?	☐ Yes ☒ No	If yes, explain:
5. Does the change qualify as a change specifically prohibited by NRC regulation, order or license condition?	☐ Yes ☒ No	If yes, explain:
Actions Required Prior to or Concurrent with Change Implementation Evaluation:		
Action		**Explanation**
6. Modification / Addition to CAS system or system coverage documentation	☐ Yes ☒ No	If yes, explain:
7. Acquire NRC pre-approval (license amendment)	☒ Yes ☐ No	If yes, explain: License renewal.
8. Conduct/modify ISA	☐ Yes ☒ No	If yes, explain:
9. ISA Database Modification	☐ Yes ☒ No	If yes, explain:
10. Modification of other safety program information / underlying analyses (PHA, RHA, FHA, NCSA, etc.)	☐ Yes ☒ No	If yes, explain:
Actions required subsequent to Change Implementation Evaluation:		
11. Update safety program information (PHA,RHA,FHA,NCSA, P&ID)	☐ Yes ☒ No	If yes, explain:

* **If this form exists as a part of a document, the document number is not required.**

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3.0 Integrated Safety Analysis (ISA) and ISA Summary

3.1 *ISA Program*

AREVA NP Inc. (AREVA NP) maintains an ISA program for the areas of the Richland Horn Rapids Road (HRR) facility that involve or could impact the safe handling of SNM in quantities greater than 1400 g ^{235}U. The ISA program consists of the following elements: 1) an ISA Summary, 2) ISA program commitments described in this chapter, and 3) ISA supporting information maintained at the facility. The supporting information includes items such as the nuclear criticality safety analyses (NCSAs), radiological hazards analyses (RHAs), fire hazards analyses (FHAs), process hazards analyses (PHAs), and chemical consequence determinations.

3.1.1 Process Safety Information

The uranium operations at the HRR facility consist of nuclear fuel fabrication, including chemical conversion, pellet fabrication, component assembly, and various uranium recovery operations. The complexity of some operations and the potential consequences of equipment failure are such that some operations require the use of piping and instrumentation diagrams (P&IDs), flow diagrams, and other aids to adequately assess these operations. The facility will maintain process safety information in sufficient detail to support the safety analysis, including process descriptions, P&IDs, equipment drawings, and hazardous material inventories.

3.1.2 ISA Change Management

Changes to the facility that impact the ISA are accomplished using the configuration management system(s) described in Chapter 11. The Manager of the Environmental, Health, Safety and Licensing (EHS&L) function or his delegate determines whether an ISA team will convene and, if so, which disciplines are required to perform hazards evaluations for these changes.

3.1.3 ISA Team (PHA)

PHAs will be performed by a team comprised of a team leader, individuals knowledgeable of the process, and individuals within appropriate safety functions, commensurate with the process being reviewed.

Prior to starting the hazards analysis, the team will be instructed in the methodology to be used. Team members may represent more than one functional area being evaluated. Functions that are not affected by the proposed change being evaluated do not require representation.

3.1.4 Identification of Potential Accident Sequences

Credible accident sequences will be identified using any of the methodologies listed in NUREG 1513, "Integrated Safety Analysis Guidance Document" (e.g., What If, Check List, FMEA, HAZOP, fault trees).

3.1.5 Identification of Consequences and Likelihood of Potential Accident Sequences

For each identified credible accident sequence, the consequences will be classified as "high," "intermediate," or "low." This determination is based on information included in existing safety

analyses (PHAs, NCSAs, RHAs and FHAs). If insufficient information for this determination is included in these analyses, a subject matter expert is assigned to document a new consequence determination. The scope of the ISA does not include initiating events caused by acts of war or sabotage.

A qualitative risk assessment will be performed for credible accident sequences with high or intermediate consequences. Nuclear criticality accident sequences will be assumed to have high consequences. The ISA Summary provides a definition of the qualitative risk assessment methodology.

The ISA Summary will provide a description of each credible accident or bounding accident condition with intermediate or high consequences of concern.

3.1.6 Identification of Items Relied On For Safety (IROFS)

After determination of potential accident sequences that result in consequences of concern, the IROFS will be identified for prevention or mitigation of these accidents. The reliability characteristics of the system of controls will be evaluated to ensure that high-consequence events are highly unlikely and that intermediate consequence events are at least unlikely as defined in the ISA Summary. Management measures for ensuring the availability and reliability of IROFS are described in Chapter 11.

3.2 *Chemical Hazards*

An evaluation of chemical use at this facility shall be performed against the criteria of 10 CFR 70.61, i.e. chemical hazards of licensed materials and hazardous chemicals produced from licensed material. This evaluation need not include hazardous chemicals prior to co-mingling with SNM or after they have been separated from SNM (except as potential initiating events that may adversely impact safety of SNM activities in other areas). The ISA Summary includes the quantitative chemical exposure criteria that define high or intermediate consequences.

3.3 *ISA Summary and ISA Documentation*

The ISA Summary is a stand-alone document and includes the following elements:

- General description of the site.
- General description of the facility.
- Description of facility processes, hazards, and types of accident sequences.
- Demonstration of compliance with 10 CFR 70.61 performance requirements.
- Description of the ISA team qualifications and ISA methods.
- Descriptive list of IROFS highlighting sole IROFS, if any.
- Definitions of the terms "credible", "unlikely," and "highly unlikely."

The ISA Summary will be updated at least annually by January 30 when changes affecting the summary have been made that did not require pre-approval in accordance with 10 CFR 70.72. A copy of this summary, along with its annual updates, will be submitted to the NRC.

EHS&L Document

SNM-1227 - Chapter 4
Radiation Protection

Nature of Changes

Item	Paragraph	Description	Justification
1.	New Document	New Document	License Renewal
2.			
3.			
4.			
5.			
6.			
7.			
8.			
9.			
10.			
List Below any Documents, including Forms & Operator Aids which must be issued concurrently with this document revision:			
Do Not make document effective until Loren Maas releases after NRC approval.			

This Document contains a total of 13 pages excluding the signature page generated by Documentum, the document control application software.

DOCUMENT REVIEW/APPROVAL/DELETION CHECKLIST

All new and/or revised procedures shall be approved by the change author, cognizant manager(s) of areas affected by the changes, and by applicable manager(s) of any function that approved the previous revision of the document unless responsibility for such approval has been transferred to another organization. Also, the procedure shall be approved by manager(s) of functional organizations that provide technical reviews with the exception of the Training Department. Finally, Document Control shall verify that the required approvals have been properly obtained and that any documents that must be issued concurrently are ready to be issued.

Minor Changes: If the proposed changes are limited to editorial and/or administrative changes check the box at the right. The document will be routed directly for review by EHS&L without technical review. All applicable approvals must still be obtained.	☐

Document Reviews			**Document Approvals**	
Purpose/Function of Review	Specify Reviewer(s) (Optional except for change author)	(Check all that apply)	Title of Approver	(Check all that Apply)
Document Control (Automatic)		☒	Document Control (Automatic)	☒
Change Author	LJ Maas	☒	Author	☒
Independent Technical Review	RK Burklin	☒		
Operability Review(s)			Mgr, Richland Operations[1]	☐
Conversion		☐	Mgr, Uranium Conversion & Recovery Operations[1]	☐
Recovery		☐		
Ceramics		☐	Mgr, Ceramic Operations[1]	☐
Rods		☐	Mgr, Rods & Bundles[1]	☐
Bundles		☐		
Transportation		☐		
Components		☐	Mgr, Component Fabrication[1]	☐
Maintenance Review		☐	Mgr, Maintenance[1]	☐
Lab Review		☐	Mgr, Analytical Services[1]	☐
EHS&L Review(s)			Mgr, EHS&L[2]	☒
Criticality		☐	Mgr, Criticality Safety[2]	☐
Radiation Protection		☐	Mgr, Safety, Security & Emergency Preparedness[2]	☐
Safety/Security		☐		
Emergency Preparedness		☐		
MC&A	LJ Maas	☒	Mgr, Licensing & Compliance[2]	☐
Transportation		☐		
Environmental		☐		
BWR Product Eng. Review		☐	Mgr, BWR Product Engineering	☐
Quality Review		☐	Mgr, Quality	☐
Ops. Projects & Planning Review		☐	Mgr, Ops. Projects & Planning	☐
Others:		☐	Mgr, Richland Site/Other	☐
Training & Employee Dev.:[3]		☐	Training & Employee Dev.	☐

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[3]Note: Training department review is required for all procedures that require or affect a Learning Plan and if additional training materials or curriculum must be revised before issuing procedure.

EHS&L Change Impact Evaluation Form		
Document / ECN No*.: E10-08-004		Change Evaluator: RK Burklin
Does the change potentially impact Criticality Alarm System (CAS) coverage?	☐ Yes ☒ No	If yes, explain:
NRC Pre-Approval Evaluation:		
Is NRC Pre-approval (License Amendment) Needed? (Based on "Yes" answer to any of five questions below). (Based on "No" answer to all five questions below).	☒ Yes ☐ No	This is part of our NRC license renewal.
1. Does the change create new types of accident sequences that, unless mitigated or prevented, would exceed the performance requirements of 10 CFR 70.61 (create high or intermediate consequence events) and that have not previously been described in AREVA NP Inc's ISA Summary?	☐ Yes ☒ No	If yes, explain:
2. Does the change use new processes, technologies, or control systems for which AREVA NP Inc. has no prior experience?	☐ Yes ☒ No	If yes, explain:
3. Does the change remove, without at least an equivalent replacement of the safety function, an item relied on for safety that is listed in the ISA Summary?	☐ Yes ☒ No	If yes, explain:
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Action		**Explanation**
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7. Acquire NRC pre-approval (license amendment)	☒ Yes ☐ No	If yes, explain: License renewal.
8. Conduct/modify ISA	☐ Yes ☒ No	If yes, explain:
9. ISA Database Modification	☐ Yes ☒ No	If yes, explain:
10. Modification of other safety program information / underlying analyses (PHA, RHA, FHA, NCSA, etc.)	☐ Yes ☒ No	If yes, explain:
Actions required subsequent to Change Implementation Evaluation:		
11. Update safety program information (PHA,RHA,FHA,NCSA, P&ID)	☐ Yes ☒ No	If yes, explain:

* **If this form exists as a part of a document, the document number is not required.**

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4.0 Radiation Protection

4.1 *Radiation Protection Program Implementation*

AREVA NP Inc. (AREVA NP) shall develop, document and implement a Radiation Protection Program to protect the radiological health and safety of its workers in accordance with the regulatory requirements of 10 CFR Parts 19, 20 and 70. Programs to protect the environment and the health and safety of the public from radiological releases are presented in Chapter 9, *Environmental Protection.*

The Radiation Protection Program shall control the receipt, possession, use and transfer of radioactive materials such that the occupational radiation exposure dose limits of 10 CFR Part 20 are not exceeded under normal operations. Additionally, the Radiation Protection Program incorporates the ALARA philosophy which provides a systematic approach to reducing occupational exposures through engineering, administrative, or other applicable controls.

Radiological safety analyses of individual facility processes are conducted as part of the facility's Integrated Safety Analysis (ISA) to systematically evaluate radiological hazards and credible upset conditions (accident scenarios) in which the exclusive loss of radiological controls could lead to personnel exposures exceeding the radiation exposure performance criteria of 10 CFR 70.61(b) and (c). Such analyses identify the administrative and engineered controls to ensure that radiation safety-related parameters remain within their specified limits and prevent, or mitigate the radiological consequences of, upset (accident) conditions. Radiological hazards and safety controls that are designated to be Items Relied On For Safety (IROFS) with respect to these hazards are presented in the ISA Summary.

4.2 *ALARA Program*

The ALARA Program is an integral component of the Radiation Protection Program. The ALARA Program shall be consistent with the regulatory requirements of 10 CFR 20.1101 and shall be implemented through written policies and procedures.

The Richland Site Manager shall be responsible for ensuring adherence to the ALARA philosophy. To implement this philosophy, key positions in the radiation protection organization and in organizations involved with the use or handling of licensed material shall be filled by persons whose qualifications are discussed in Chapter 2, *Organization and Administration.*

The ALARA Committee shall oversee the implementation of the ALARA Program. The ALARA Committee, chaired by a key member of the Radiation Protection group, or a designee, and including key managers (or designees) from operations and engineering functions, shall maintain an awareness of the radiological trends in employee dose and radiological environmental releases. The Committee shall meet at least annually and shall prepare an annual ALARA Report that documents radiological trends in employee dose and environmental

releases, identifies potential areas for improvement and tracks the recent (normally the previous calendar year's) ALARA projects.

AREVA NP shall review and revise, when it deems appropriate, the ALARA program goals and objectives. New ALARA approaches, technologies, and operating procedures that could reduce radiation exposures may be incorporated when suitable.

4.3 *Organization and Personnel Qualifications*

Personnel trained in radiation protection practices and who are well-versed in applicable regulations and the facility's ISA shall implement the Radiation Protection Program. The functional organization responsible for radiation protection, including the minimum qualifications and responsibilities of key individuals, is presented in Chapter 2, *Organization and Administration.*

4.4 *Written Procedures*

Written and approved procedures shall govern activities involving the handing of radioactive materials. Radiation protection requirements shall be incorporated into operating procedures, equipment maintenance procedures, Radiation Work Permits (RWPs), and/or Radiation Job Permits (RJPs) to alert workers to special hazards or controls necessary for their protection.

Radiation protection procedures shall be authorized, approved and distributed in accordance with site procedures.

RJPs shall be issued for non-routine activities involving licensed materials that are not covered by written and approved radiation protection procedures, where doses or dose rates are deemed by the radiation protection function to require monitoring. These activities include access to high radiation areas, or situations that may subject workers to radiation doses that could potentially exceed regulatory exposure limits.

4.5 *Radiation Protection Training*

The radiation protection training program shall comply with the regulatory requirements of 10 CFR Parts 19 and 20. Training that serves as a management measure to ensure that an administrative control IROFS is available and reliable when required is addressed in Chapter 11, *Management Measures.*

Unescorted individuals entering areas of the facility that contain NRC licensed material shall be appropriately trained. The following topics may be included:

- dose limits and the ALARA principle
- precautions and procedures for the safe handling of radioactive material
- access and egress controls, personal surveys and escort procedures
- radiation safety principles, policies and procedures, including health problems associated with exposure to radiation
- contamination control, including protective clothing and equipment for the minimization or control of exposure to radioactive material or radiation

- purposes and functions of protective devices (including IROFS) used to control exposure to radiation and radioactive material
- emergency response

The level of radiation protection training shall be commensurate with a worker's risk of exposure to radioactive materials. Only personnel who have successfully demonstrated their understanding of radiological protection principles through oral evaluations, written evaluations or observation of their performance will be authorized to work with radioactive materials and to have unescorted access to facility areas where the likelihood exists to receive radiation doses greater than 100 mrem in a calendar year. Facility visitors will either be provided with appropriate training (commensurate with the scope of their visit) or be escorted by trained employees.

Workers shall receive radiological refresher training at least every three years .

The adequacy of the radiation protection training program shall be reviewed on at least a triennial basis.

4.6 *Ventilation and Respiratory Protection*

Ventilation systems and respiratory protection programs are designed to reduce worker exposure to radioactive airborne contamination.

4.6.1 Ventilation

Ventilation systems are designed to limit the spread of airborne contamination by maintaining contaminated areas at slightly negative pressures compared to adjacent clean areas and to maintain exposures to airborne radioactivity to below that permitted by 10 CFR Part 20. Hoods, hybrid-glove boxes, downdraft tables and other localized ventilation designs may also be used to limit the potential for intake by inhalation. Ventilation of a contaminated area of the facility may be suspended when no processing of radioactive materials is underway or when, for example, the plant is shut down. In these circumstances appropriate precautions will be taken to contain contamination and protect individuals.

Ventilation system performance shall be demonstrated at frequencies that are established, controlled and documented by the ventilation engineering function and shall encompass the following:

- techniques to ascertain flow direction (from clean to contaminated areas) such as smoke tests and/or differential pressure readings
- measurement of the differential pressure across HEPA filters installed for general re-circulation or exhaust air. A differential pressure reading of greater than 5.0 inches of water across a final HEPA filter will precipitate a scheduled shutdown of the ventilation system to allow for a change-out of the HEPA filter. Differential pressures shall be recorded periodically.
- measurement of average air velocity through openings in uranium handling hoods and laboratory hoods containing readily dispersible uranium. The requirements for the minimum average linear velocities shall be established by procedure. Flow rates

below these minimum values shall result in the temporary suspension of hood work or require the use of respiratory protection. Airflow velocities shall be monitored periodically.

- measurement of HEPA filter efficiency. Criteria for the use and testing of HEPA filters are presented in Chapter 9, *Environmental Protection.*

4.6.2 Respiratory Protection

Workers' intake of airborne radioactive material is primarily controlled through use of process or other engineered controls. In the absence of such controls or when enhanced controls are desired, use of respiratory protection equipment may be appropriate.

The respiratory protection program shall meet the requirements of 10 CFR Part 20, Subpart H and be governed by written procedures. Respiratory protection devices for radioactive materials shall be approved by the National Institute for Occupational Safety and Health (NIOSH). When respiratory protection devices are used, appropriate protection factors (no greater than those listed in 10 CFR Part 20) shall be applied when calculating intake or the Committed Effective Dose Equivalent (CEDE).

Personnel who wear respiratory protection devices shall be quantitatively fit tested and trained in proper methods for donning and doffing respiratory protection devices and in limitations for their usage. A physician shall medically qualify individuals prior to respirator fit-testing. Such individuals shall be re-qualified either annually or as a physician may thereafter determine.

Initial training in respiratory protection shall be provided prior to respirator use and retraining shall be provided at least every three years thereafter. Individuals must successfully pass a test to demonstrate an adequate understanding of respirator use.

4.7 ***Radiation Survey and Monitoring Programs***

A radiation survey and monitoring program that is consistent with the regulatory requirements of 10 CFR Part 20, Subpart F shall be implemented. The program will monitor airborne contamination, workers' external and internal occupational radiation doses and help prevent the spread of contamination from radiologically contaminated areas. This program is designed to ensure that occupational radiation exposures remain below the limits of 10 CFR Part 20.1201(a).

Written procedures shall specify frequencies and action levels above which specified steps shall be taken. Action levels may be established in accordance with consensus industry standards recognized by national and international radiation protection experts and may use facility-specific data, such as historical exposure and operating experience.

4.7.1 Radiation Surveys

Routine radiation surveys shall be performed in areas of the facility in which radioactive materials are stored or processed and to which personnel have access. New facility operations, or operations that have been modified in a manner that could substantially increase the potential for worker external dose, shall be surveyed during process commissioning, i.e. during the

introduction of radioactive materials. The frequency and scope of future radiation surveys on such new or modified processes shall be based on potential dose rates.

Routine surveys shall be performed for the controlling type or types of radiation of concern at frequencies that will ensure that annual permissible exposure limits for workers are not exceeded.

Radiation detection instruments in sufficient numbers and with detection ranges and lower detection limits adequate for contamination at levels of concern shall be used to perform the radiation surveys. GM meters, alpha scintillation detectors and counters, proportional counters, ionization chambers and solid state detectors may all be used. Unless taken out of service, radiation detection instruments shall be calibrated following maintenance that could adversely affect their current calibration and, at a minimum, annually. Calibration sources shall be traceable to a National Institute of Standards and Technology (NIST) standard or other recognized reference material or standard. Portable survey instruments shall be source-checked each day of use. Documentation of the daily source check is not required and use of a calibrated standard is not necessary.

4.7.2 Personnel Monitoring Program -- External Radiation Exposure

Workers who require external exposure monitoring per 10 CFR Part 20.1502(a) shall wear beta-gamma sensitive dosimeters in areas of the facility in which radioactive materials are processed or stored or where the potential exists to receive significant radiation doses. The beta-gamma dosimeters may be supplemented, as appropriate, by other types of dosimeters (e.g. direct-reading dosimeters, neutron dosimeters) and/or by radiation measurements made with radiation survey instruments. Thermoluminesecent dosimeter readings, with the possible exception of extremity dosimeters, shall be analyzed and evaluated by a processor holding current accreditation from NIST's National Voluntary Laboratory Accreditation Program (NVLAP). Personnel dosimeters shall normally be exchanged on a quarterly basis. Dosimetry data shall be included in the facility's exposure summation report as specified in 10 CFR Part 20.1202.

4.7.3 Personnel Monitoring Program -- Internal Radiation Exposure

The internal dose for workers who require radiation exposure monitoring per 10 CFR Part 20.1502(b) shall be established by means of a dose tracking system that determines intake based upon air sample results and stay times, modified by respiratory protection factors when respirators are used. Results from diagnostic bioassay studies (urine analysis, lung counting, fecal analyses, nasal swipes) may supplement or substitute for determinations from the dose tracking system. Bioassay results may also be used to determine or adjust the CEDE values.

Written bioassay procedures will identify those workers to be surveyed, frequencies of measurement, action levels at which the cause(s) of the elevated readings will be investigated, and corrective actions including procedure modifications, work restrictions or other measures.

Bioassay results shall be interpreted using internationally-accepted consensus models and may make use of incident-specific data and/or an exposed individual's personal characteristics.

In accordance with the requirements of 10 CFR Part 20.1202, for each person who requires both internal dose monitoring and external dose monitoring, the TEDE will be calculated by adding the Deep Dose Equivalent (DDE) to the CEDE. The TEDE will not exceed the 10 CFR Part 20 dose limits.

4.7.4 Airborne Radiation Sampling Program

Measurements of airborne radioactive materials shall be made in areas of the facility where non-encapsulated radioactive licensed materials are handled or processed and where the concentrations of airborne radioactive materials are likely to exceed 10% of DAC. DAC is based upon ICRP 66 and 68 assuming an Activity Median Aerodynamic Diameter (AMAD) of 5 micrometers or, at the discretion of AREVA, on future ICRP models.

AREVA NP may also elect to take credit for measured particle size distributions, based upon the ICRP 66 model (or a future ICRP model), that is, further adjust ALI and DAC when particle size data indicate AMADs are in excess of 5 micrometers. Such adjustments would increase ALI and DAC by the same ratio as the ratio of the committed effective dose equivalent assuming the AMAD of 5 micrometers to the committed effective dose equivalent derived from the measured AMADs.

If AREVA NP chooses to adjust DACs and ALIs by particle size, a particle size measurement and analysis will be performed at least semi-annually in each group of locations for which particle size credit is taken. After one year, the Health Physics Component may relax the frequency to once per calendar year if DAC determined by new measurement(s) for a group of locations does not differ significantly from that established from previous measurements.

Specialized air sampling and monitoring equipment, such as continuous air monitors, portable high volume and/or lapel air samplers, may supplement the facility's fixed air sampling network. The frequency of air sampling in contaminated areas shall be based on historical experience for each sampling area. Air sample rotameters or critical orifice devices will either be calibrated or checked annually with a secondary standard. For airflow devices that operate across a narrow range, the annual calibration may be made at one point.

The results of fixed air sampling shall be periodically evaluated to ensure they are reasonably representative of worker intakes. Re-evaluation of their representativeness shall be conducted annually for those workstations that averaged 25% or greater of DAC for the previous calendar year, and biennially for workstations whose average concentrations exceeded 10% of DAC for the previous calendar year.

Fixed air sampling results, lapel or other special air sampling results may be used to determine worker intake and to calculate CEDE in areas for which internal dose assessment is required. Correction factors may be applied to airborne concentrations determined from fixed air samplers or to intake concentrations to take into account the representativeness of fixed air samples.

4.7.5 Surface Contamination Control

Contamination surveys shall be conducted to minimize the spread of contamination from radiologically contaminated areas to clean areas and to help maintain worker doses ALARA. Except as noted in this chapter, contamination surveys shall be performed on all persons

leaving contaminated areas, on materials and equipment to be released from radiation protection requirements and on incoming and outgoing shipments of radioactive materials.

The Radiation Protection Program may require use of protective clothing to minimize opportunities for personnel and their clothing to become contaminated. Protective clothing requirements shall be commensurate with anticipated work conditions. Persons shall be surveyed for contamination with a survey instrument or monitor located at respective step-off areas, generally following removal of protective clothing. Exceptions shall be made for emergencies and emergency drills. If contamination is detected in excess of procedural limits, the individual shall be decontaminated, but if such efforts are not sufficiently effective, a member of the radiation protection group may release the individual. Follow-up surveys will be performed as deemed appropriate and practical.

Materials, equipment, and facilities shall be free-released from contaminated areas of the facility in accordance with NRC Branch Technical Position entitled "*Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source or Special Nuclear Material*" (April, 1993). Volumetrically contaminated materials may be free-released if the concentration of uranium is less than 30 pCi/g. Contaminated materials may also be free-released if analysis by one of the RESRAD family of codes demonstrates that the calculated TEDE is less than 1 mrem/yr to an average member of the critical group.

Equipment and materials may be transferred from one contaminated area through a non-contaminated area into a second contaminated area if the exterior surfaces of the item or its container have smearable contamination levels less than those permitted by facility procedures. The level of smearable contamination is the activity transferred to a swab or smear from a surface being rubbed with moderate pressure. Such transfers may be made without a survey if the material has only fixed contamination or is bagged or wrapped to prevent the spread of contamination. Transfers of radioactive material in overpacks, from one contaminated area to another, on site, when the overpacks have not entered contaminated areas, shall be permitted without performance of a survey.

Rods and/or other items that can be shown to have a greater than 95% probability of not being contaminated at levels exceeding the release criteria may be removed from contaminated areas without a radiological survey. These rods and/or other items that are candidates for release without surveys shall have significant historical evidence that such releases are below release levels. Note: The items referred to in this paragraph are not released to the general public, but rather to licensees, such as reactors or other fuel fabrication facilities.

4.8 *Additional Program Commitments*

4.8.1 Self Assessments

AREVA NP has committed to conduct audits and assessments as part of its Radiological Protection Program. These activities will assess compliance of the program with applicable radiological regulatory requirements and license commitments, achievement of corporate radiological protection objectives and facility-wide application of ALARA principles.

Instances in which radiation exposure action levels are exceeded shall be investigated, as needed, to assess the need for any corrective action(s). The results of these investigations (or self-assessments) are documented in "*Condition Reports*" that are graded, and acted upon, in accordance with their radiological safety significance. Significant abnormal radiation events that result in a worker dose exceeding the 10 CFR Part 20 limits or emissions exceeding the modified Appendix B concentration limits (over the period permitted) shall be investigated by qualified and independent personnel to determine causes. Additionally, events causing notifications according to 10 CFR 70.50 shall be investigated by qualified personnel to determine cause. Any corrective actions shall be tracked through to completion to minimize the probability of recurrence of the abnormal event. Abnormal events assessed to have minor radiological safety significance may be dispositioned without any corrective action.

In accordance with the requirements of 10 CFR 20.1101(c), the content and implementation of the Radiation Protection Program shall be reviewed at least annually to ensure that its objectives, survey approaches and methodologies continue to protect workers from radiation and radioactive materials. Additional discussion of the audit and assessment and incident investigation functions as they apply to IROFS is presented in Chapter 11, *Management Measures.*

4.8.2 Records Management

AREVA NP shall maintain records of the Radiation Protection Program, including bioassay data, dose records, radiation protection (and contamination control) records, radiation training and re-training records, RWPs and RJPs as required by regulation. Information on corrective actions that were implemented as a result of abnormal event investigations shall also be retained. Changes to the Radiation Protection Program resulting from changes to the facility ISA will also be maintained. Additional discussion of records management is presented in Chapter 11, *Management Measures.*

4.8.3 NRC Reporting Program

Written procedures will govern reporting to the NRC within the timeframes specified in 10 CFR Parts 70.74 and 20.2202 for events that result in occupational radiation exposures that exceed regulatory limits. The annual report of worker radiation monitoring required by 10 CFR 20.2206(b) shall be prepared and submitted in accordance with regulatory requirements.

4.8.4 Signage

Appropriate radiation warning signs shall be posted for areas that meet the 10 CFR 20 definitions of radiation areas, high radiation areas, airborne radioactivity areas and radioactive material areas.

4.8.5 Shipments of Radioactive Materials

Surveys shall be performed on incoming and outgoing shipments of radioactive materials in accordance with facility shipping procedures that are consistent with applicable U.S. Department of Transportation (49 CFR Part 173) and NRC (10 CFR Part 20.1906) regulations.

For clarification purposes:

- The three hour clock referenced in 10 CFR 20.1906, as it applies to the contents of a van or similar closed conveyance, begins when the tamper indicating seal to the outer conveyance is broken.

EHS&L Document

SNM-1227 - Chapter 5
Nuclear Criticality Safety

Nature of Changes

Item	Paragraph	Description	Justification
1.	New Document	New Document	License Renewal
2.			
3.			
4.			
5.			
6.			
7.			
8.			
9.			
10.			
List Below any Documents, including Forms & Operator Aids which must be issued concurrently with this document revision:			
Do Not make document effective until Loren Maas releases after NRC approval.			

This Document contains a total of 19 pages excluding the signature page generated by Documentum, the document control application software.

DOCUMENT REVIEW/APPROVAL/DELETION CHECKLIST

All new and/or revised procedures shall be approved by the change author, cognizant manager(s) of areas affected by the changes, and by applicable manager(s) of any function that approved the previous revision of the document unless responsibility for such approval has been transferred to another organization. Also, the procedure shall be approved by manager(s) of functional organizations that provide technical reviews with the exception of the Training Department. Finally, Document Control shall verify that the required approvals have been properly obtained and that any documents that must be issued concurrently are ready to be issued.

Minor Changes: If the proposed changes are limited to editorial and/or administrative changes check the box at the right. The document will be routed directly for review by EHS&L without technical review. All applicable approvals must still be obtained.	☐

Document Reviews			**Document Approvals**	
Purpose/Function of Review	Specify Reviewer(s) (Optional except for change author)	(Check all that apply)	Title of Approver	(Check all that Apply)
Document Control (Automatic)		☒	Document Control (Automatic)	☒
Change Author	LJ Maas	☒	Author	☒
Independent Technical Review	CD Manning	☒		
Operability Review(s)			Mgr, Richland Operations[1]	☐
Conversion		☐	Mgr, Uranium Conversion & Recovery Operations[1]	☐
Recovery		☐		
Ceramics		☐	Mgr, Ceramic Operations[1]	☐
Rods		☐	Mgr, Rods & Bundles[1]	☐
Bundles		☐		
Transportation		☐		
Components		☐	Mgr, Component Fabrication[1]	☐
Maintenance Review		☐	Mgr, Maintenance[1]	☐
Lab Review		☐	Mgr, Analytical Services[1]	☐
EHS&L Review(s)			Mgr, EHS&L[2]	☒
Criticality		☐	Mgr, Criticality Safety[2]	☐
Radiation Protection		☐	Mgr, Safety, Security & Emergency Preparedness[2]	☐
Safety/Security		☐		
Emergency Preparedness		☐		
MC&A	LJ Maas	☒	Mgr, Licensing & Compliance[2]	☐
Transportation		☐		
Environmental		☐		
BWR Product Eng. Review		☐	Mgr, BWR Product Engineering	☐
Quality Review		☐	Mgr, Quality	☐
Ops. Projects & Planning Review		☐	Mgr, Ops. Projects & Planning	☐
Others:		☐	Mgr, Richland Site/Other	☐
Training & Employee Dev.:[3]		☐	Training & Employee Dev.	☐

[1]Note: If approvals include 2 or more product center managers, the Operations manager can be substituted for the applicable product center managers.

[2]Note: If approvals include 2 or more EHS&L functional managers, the EHS&L manager can be substituted for the applicable EHS&L functional managers.

[3]Note: Training department review is required for all procedures that require or affect a Learning Plan and if additional training materials or curriculum must be revised before issuing procedure.

EHS&L Change Impact Evaluation Form		
Document / ECN No*.: E10-08-005		Change Evaluator: CD Manning
Does the change potentially impact Criticality Alarm System (CAS) coverage?	☐ Yes ☒ No	If yes, explain:
NRC Pre-Approval Evaluation:		
Is NRC Pre-approval (License Amendment) Needed? (Based on "Yes" answer to any of five questions below). (Based on "No" answer to all five questions below).	☒ Yes ☐ No	This is part of our NRC license renewal.
1. Does the change create new types of accident sequences that, unless mitigated or prevented, would exceed the performance requirements of 10 CFR 70.61 (create high or intermediate consequence events) and that have not previously been described in AREVA NP Inc's ISA Summary?	☐ Yes ☒ No	If yes, explain:
2. Does the change use new processes, technologies, or control systems for which AREVA NP Inc. has no prior experience?	☐ Yes ☒ No	If yes, explain:
3. Does the change remove, without at least an equivalent replacement of the safety function, an item relied on for safety that is listed in the ISA Summary?	☐ Yes ☒ No	If yes, explain:
4. Does the change alter any item relied on for safety, listed in the ISA Summary, that is the sole item preventing or mitigating an accident sequence of high or intermediate consequences?	☐ Yes ☒ No	If yes, explain:
5. Does the change qualify as a change specifically prohibited by NRC regulation, order or license condition?	☐ Yes ☒ No	If yes, explain:
Actions Required Prior to or Concurrent with Change Implementation Evaluation:		
Action		**Explanation**
6. Modification / Addition to CAS system or system coverage documentation	☐ Yes ☒ No	If yes, explain:
7. Acquire NRC pre-approval (license amendment)	☒ Yes ☐ No	If yes, explain: License renewal.
8. Conduct/modify ISA	☐ Yes ☒ No	If yes, explain:
9. ISA Database Modification	☐ Yes ☒ No	If yes, explain:
10. Modification of other safety program information / underlying analyses (PHA, RHA, FHA, NCSA, etc.)	☐ Yes ☒ No	If yes, explain:
Actions required subsequent to Change Implementation Evaluation:		
11. Update safety program information (PHA,RHA,FHA,NCSA, P&ID)	☐ Yes ☒ No	If yes, explain:

* **If this form exists as a part of a document, the document number is not required.**

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Application Date:	NRC Docket No. 70-1257
October 27, 2006	

5.0 Nuclear Criticality Safety

5.1 *Program Applicability, Policy and Objectives*

AREVA NP Inc. (AREVA NP) maintains a nuclear criticality safety program that applies to special nuclear material (SNM) activities at the Richland Horn Rapids Road site, except where:

- SNM quantity is less than or equal to 1400 g ^{235}U; or
- The areal density is less than or equal to 45% of a critical areal density and double batching the material is not credible; or
- The fissile concentration or density does not exceed the amount listed in 49CFR§173.453 for fissile exempt quantities; or
- The SNM is contained within NRC/DOT-approved shipping packages arranged in an array(s) with a total Criticality Safety Index (CSI) of less than 100 and spaced at least 12 feet from other SNM-bearing packages or material.

References to SNM in this chapter refer to SNM not excluded by the above criteria.

It is AREVA NP's policy that the double contingency principle [(ANSI/ANS-8.11983 (R 1988)] will be the basis for design and operation of the SNM processes within the Richland fuel fabrication facility. Where practicable, process designs will incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

Nuclear criticality safety shall be assured to meet the requirements of 10CFR70 through both administrative and technical practices.

The objective of the NCS program is preventing an inadvertent nuclear criticality by:

- Conducting nuclear criticality safety analyses and implementing required Items Relied On For Safety (IROFS) to protect against accident sequences that could lead to an inadvertent nuclear criticality.
- Ensuring that sufficient IROFS are in place to render accidental nuclear criticality highly unlikely.
- Establishing and maintaining NCS safety parameters and procedures.
- Establishing and maintaining limits and controls on IROFS to ensure that accidental nuclear criticality remains highly unlikely.
- Providing NCS postings in appropriate areas where SNM is processed or stored.
- Ensuring that personnel understand the policy that they are to report all suspect NCS conditions to the NCS function, only perform actions in accordance with approved

procedures, and take no other action until the situation has been evaluated by the NCS staff and recovery instructions are provided.

5.2 *Organizational Structure*

Minimum qualifications of the NCS staff and the organizational relationship to other organizations with NCS responsibilities are described in Chapter 2.0.

The Manager of the EHS&L function is responsible for ensuring adequate staff, skilled in the interpretation of data pertinent to NCS and familiar with the operation of the facility.

The Manager of the NCS function is responsible for the implementation of the NCS program.

Qualified NCS personnel who are independent from the operations organization determine the basis for safety of processes involving the handling and storage of SNM. They shall assess both normal and credible abnormal conditions. They shall also specify functional requirements for criticality safety controls commensurate with design criteria and shall assist in assessment of control reliability.

Audits or assessments associated with the criticality safety program are described in Chapter 11.

5.3 *NCS Program Administration*

5.3.1 Management Measures

The management measures programs are described in Chapter 11. Key programs include training, procedures development and implementation, records management, audits and assessments, configuration management, maintenance, and incident investigation/corrective action. The configuration management program includes a requirement that prior to use in a process, nuclear criticality safety controls selected and installed are verified to fulfill the requirements identified in the criticality safety analyses. Processes are examined in the "as-built" condition to validate the safety design and to verify the installation.

The configuration management program also ensures nuclear criticality safety requirements are maintained and that changes that may impact them are evaluated prior to making the change.

5.3.2 Criticality Safety Control Philosophy

To ensure accidental nuclear criticality is highly unlikely, the double contingency principle as defined in the American National Standard ANSI/ANS-8.1 shall be followed in establishing nuclear criticality safety for equipment, systems and operations. Process designs shall incorporate sufficient factors of safety during normal and credible abnormal conditions to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

Process deviations for which there is a convincing argument given physical laws or experimental data that they are not possible are considered not credible.

Double contingency may be provided by either: (i) at least two-parameter control (the control of at least two independent process parameters) or (ii) single-parameter control (a system of multiple independent controls on a single process parameter) The first method is the preferred approach.

Where practical, reliance will be placed on equipment design in which dimensions are limited(i.e., favorable geometry) rather than on administrative controls. Where structural integrity is necessary to provide assurance for safety, the design and construction of the equipment will be made with due regard to abnormal loads, accidents and deterioration.

5.3.3 Preferences of NCS Controls

The relative effectiveness and reliability of controls are considered during the criticality safety analysis process. Passive engineered controls are preferred over other system controls and are utilized when practical and appropriate. Active engineered controls are the next preferred method of control followed by administrative controls.

To be considered a viable defense against accidental criticality, a criticality safety control or set of controls must be capable of preventing a criticality accident independent of the operation or failure of any other defense for a given credible initiating event.

Passive Engineered Controls are physical restraints or features that maintain criticality safety in a static manner (e.g., fixed geometry, fixed spacing, fixed size, fixed nuclear poisons, etc.). Beyond appropriate initial installation, passive engineered controls require no action or other response to be effective when called upon to ensure nuclear criticality safety. In addition to configuration management, assurance is maintained through specific periodic inspections or verification measurement(s) as appropriate.

Active Engineered Controls are a means of criticality control involving active hardware (e.g., electrical, mechanical, hydraulic) that protect against criticality. These devices act by sensing a process variable important to criticality safety and providing automatic action (i.e. no human intervention required) to secure the system to a safe condition. In addition to configuration management, assurance is maintained through specific periodic functional testing as appropriate. Active engineered controls that are provided with fail-safe conditions (i.e. the failure of the control results in a safe condition) are the preferred type of active engineered control.

Enhanced Administrative Controls involve human intervention augmented by warning devices or other automated prompts and are preferred over simple procedural requirements for a specific action.

Administrative Controls are controls that rely on actions, judgment, and responsible actions of people for their implementation. Their use is limited to situations where passive and active engineered controls are not practical. Assurance is maintained through selected management measures.

5.3.4 Nuclear Criticality Safety Analyses (NCSAs)

Operations or processes where the NCS program applies shall be determined by a documented and peer reviewed NCSA to be adequately subcritical under both normal and credible abnormal conditions.

NCSAs shall identify and document the basis of nuclear criticality safety for a particular system. The NCSA demonstrates that NCS criteria including meeting the double contingency principle are met.

The NCSA includes consideration of the potential accident scenarios or initiating events that the system may be subject to and the potential consequences associated with such conditions, and establishes the needed limits, controls, and management measures to ensure that an accidental nuclear criticality is highly unlikely. With the addition of the other safety discipline information, the NCSA is a major portion of the Integrated Safety Analysis (ISA) documentation required to demonstrate compliance with 10 CFR 70.62.

- NCSAs shall be performed and independently reviewed by personnel who meet the

requirements specified in Section 2.2.5.1. Such personnel may either be AREVA NP or contractor employees.

- Records of NCSAs and reviews shall be documented and retained in accordance with Chapter 11, Section 11.7.

5.3.5 Nuclear Criticality Safety Specifications (NCSSs)

NCSSs describe NCS requirements implemented by user organizations.

NCSSs established to document controls for SNM processing systems and storage areas shall be prepared based on limits established in nuclear criticality safety analyses. The NCSSs may be a section of the NCSA or a separate document. NCSSs shall contain, as appropriate, the following information: work location(s), equipment covered, SNM type allowed in the process or storage area, and the associated NCS limits and controls.

5.3.6 NCS Postings

Nuclear criticality safety postings (NCSPs) approved by the NCS function shall be posted at locations where SNM is handled, processed, transported, or stored as determined by the NCS function.

NCSPs contain selected criticality safety limits and controls controllable or observable by the user. The NCSPs normally include the type and form of material permitted, allowable quantity, restrictions on moderators, and required spacing from other special nuclear material. NCSPs are used in conjunction with operating procedures.

5.3.7 Operating Procedures

Operations in which nuclear criticality safety is pertinent shall be governed by written procedures. Personnel working with SNM shall follow these procedures. Persons supervising these operations shall be required to be familiar with such procedures.

The NCS function shall approve operating procedures and procedure changes involving fissile material processing and storage.

5.3.8 Fissile Content Determination

The fissile isotope content (enrichment) of incoming SNM shall be confirmed to meet NCS limits on enrichment prior to the conduct of any activity other than storage and sampling. Nondestructive analysis or laboratory analysis of material sampled on plant site or provided by the supplier may be used to make this determination.

5.3.9 Reporting Off-Standard NCS Conditions

Off-standard NCS conditions, regardless of how they are discovered, shall be reported to management and the NCS function and shall be resolved in accordance with the problem identification and corrective action follow up program described in Chapter 11 and, when necessary, in accordance with emergency response procedures.

When such conditions are observed, only recovery actions in accordance with approved procedures are allowed and no other action is permitted until the situation has been evaluated by the NCS staff and recovery instructions are provided.

5.3.10 Operational and Incident Reviews

A system of criticality safety audits and assessments shall be established (see Chapter 11, Section 11.5). The purpose of these reviews shall be to confirm that the criticality safety program meets the intended functions.

Criticality safety-related incidents are subject to the corrective action, incident investigation and causal analysis programs described in Chapter 11.

5.3.11 Emergency Procedures

Guidance on classifying, reporting, and responding to actual or potential criticality accidents shall be provided to personnel responsible for these activities.

Personnel other than escorted visitors shall be trained to the proper response to an activation of the Criticality Accident Alarm System.

5.4 *Methodologies and Technical Practices*

AREVA NP uses NCS methodologies for calculating k_{eff} that meet the requirements of ANSI/ANS-8.1-1983 as it relates to methodologies.

The authorized technical practices used to establish controls on controlled parameters that will ensure an adequate margin of subcriticality include the following:

- Using validated computer codes to calculate the effective multiplication factor (k_{eff}) (see section 5.4.1)
- Comparing critical values to peer reviewed handbooks and applying approved safety factors.
- Using single parameter subcritical limits listed in ANSI/ANS standards.
- Using empirical/semi-empirical techniques such as Buckling/Shape Conversion, Surface Density, Limiting Surface Density, Density Analog and Solid Angle Methods as described in Reference 20.

5.4.1 Methodologies for Calculating k_{eff}

If the methodology employs computer programs for calculating the effective multiplication factor (k_{eff}), the software and hardware associated with the program shall be controlled to ensure that changes that could impact results are evaluated prior to using them in criticality safety analysis calculations.

The documentation of the validation of methods used for calculating k_{eff} will include the following elements:

1. Justification for the safety margin applied to the methodology.
2. Justification that the methodology is applicable for the intended use.
3. A description of the theory of the methodology in sufficient detail to allow understanding of the methodology.
4. A description of how the methodology was verified to function correctly.
5. A description of the data used to validate the methodology and an explanation as to why they are applicable to plant conditions.

6. Any assumptions or boundaries that could limit the appropriate use of the methodology.
7. Justification of any extrapolations of its use beyond the area of applicability of the benchmark experiments.
8. A description of any calculation bias, and total methods uncertainty including uncertainty in the bias and benchmark experiments.
9. If the methodology uses computer calculations, a description of the software version and hardware configuration that has been validated.

The use of critical or near critical benchmarks shall be referenceable and peer reviewed.

5.4.2 Technical Practices

The technical criticality safety practices are summarized in this section. Section 5.4.2.1 discusses the limits placed on the maximum effective multiplication factor. Sections 5.4.2.2 through 5.4.2.15 discuss the parameters typically used to control the effective multiplication factor to acceptable values at normal and credible abnormal conditions and include acceptable alternatives to calculated k_{eff} limits.

5.4.2.1 Limits on Maximum Multiplication Factors

The maximum evaluated neutron multiplication factor at normal and credible abnormal conditions shall not exceed $k_{95/95} = 0.95$ for normal conditions or 0.97 for credible abnormal conditions, if justified by a sensitivity analysis. The definition of $k_{95/95}$ is as shown below.

$$k_{95/95} = k_{calc} + 2\sigma + \Delta k_b$$

where:

σ = The statistical or convergence uncertainties in the calculation of k_{calc}.

Δk_b = The calculational bias.

Alternatively, $k_{95/95}$ may be calculated from the expression:

$$k_{95/95} = k_{calc} + \Delta k_b + \Delta k_u$$

where:

Δk_b = The mean calculational method bias.

Δk_u = The uncertainty term defined by $M_{95/95} (\sigma_m^2 + \sigma_{keno}^2)^{1/2}$ where $M_{95/95}$ is the 95/95 multiplier appropriate to degrees of freedom for the number of validation analyses, σ_m^2 is the mean calculational variance deduced from the validation analyses, and σ_{keno} is the standard deviation appropriate to the KENO multiplication factor of interest.

The bias and its standard deviation are calculated using the methods described in Reference 17. These methods use standard analysis of variance principles.

Δk_b is calculated as $1-k_c$. The value for k_c is the weighted average (grand average) of the average k_{eff} values for a series of benchmark cases that are applicable to the system being modeled. Each individual benchmark case is weighted by the inverse of the k_{eff} variance (square of standard

deviation). A negative bias indicates conservative calculational results. A conservative bias will not be used unless the reason for the bias is well understood and justified.

The value for Δk_u is a statistical combination of several uncertainties. These uncertainties include σ_m^2 (the mean calculational method variance deduced by the validation analysis) and σ^2_{keno} (the square of the standard deviation appropriate to the KENO multiplication factor of interest). The term σ_m^2 is the sum of the variance about the mean and the average total uncertainty which includes consideration of the uncertainty of the benchmark experiments. The pooled uncertainty $[(\sigma_m^2 + \sigma_{keno}^2)^{1/2}]$ is multiplied by the 95/95 multiplier appropriate for the degrees of freedom for a one-sided 95% confidence limit. The 95% upper limit for k_{eff} of a process system is the sum of the calculated k_{eff} for the most reactive credible condition for the system, plus the bias, plus the pooled uncertainty that has been multiplied by the appropriate k factor.

The specific validation method used will be described in the appropriate validation documents, and may alternatively include nationally-recognized methods such as those documented in NUREG/CR-4604, "Statistical Methods for Nuclear Material Management"; NUREG/CR-6361, "Criticality Benchmark Guide for Light Water Reactor Fuel in Transportation and Storage Packages"; and NUREG/CR-6698 "Guide for Validation of Nuclear Criticality Safety Calculational Methodology".

5.4.2.2 Limits on Controlled Parameters

When establishing limits on controlled parameters for credible abnormal conditions, conditions that are not controlled or limited shall be considered to be in the optimum credible condition for causing criticality, unless historical data and/or sound engineering determinations can be applied to justify a lesser reactive condition. Data may be obtained by controlled experimentation.

5.4.2.3 Mass Limits

Mass Limits on Uranium

Mass limits on fissile material may be used for criticality control. Mass may be controlled such that the k_{eff} of the unit / process meet the guidelines given in Section 5.4.2.1.

Workstations that are controlled only by fissile material mass shall conform to the following requirements:

1. The work station shall be limited to no more than 0.45 of the minimum critical mass of the material in process;
2. A record shall be maintained of the SNM inventory at the mass-limited workstation; and
3. Where engineered controls prevent double batching, and mass is the only controlled parameter, a mass of 75% of a minimum critical mass shall not be exceeded.

Alternate mass controls may be used for operations where equipment / container geometry in reality constitutes a multi-parameter control.

Mass Limits on Moderators

Mass limits on moderators may be used in conjunction with fissile material mass limits or concentration limits on moderators.

5.4.2.4 Areal Density Limits

Areal density limits on fissile material may be used for criticality control. Workstations or processes that are controlled by areal density shall conform to the following requirements:

1. The work station or process shall be limited to no more than 0.45 of the minimum critical areal density.
2. The calculation of areal density shall assume the maximum credible mass is present and is distributed over the minimum credible area.
3. SNM shall be assured to not build up over time to the extent that the areal density limit is exceeded.

When areal density control is used and process variables can impact the areal density of the material produced, the process variables that are needed to assure a safe areal density shall be controlled.

5.4.2.5 Limiting Surface Density Limits

Limiting surface density limits on fissile material, as shown in Appendix D of Reference 20, may be used for criticality control.

5.4.2.6 Geometry

Safe dimensions and/or volumes may be established by using any one of the following safety factors:

1. The k_{eff} of the unit may be established by using the guidelines given in Section 5.4.2.1; or
2. Critical dimensions and/or volumes from ANSI standards or peer reviewed NCS handbooks multiplied by the applicable safety factors given in Tables 5-1 and 5-2. When cylinders and slabs are not infinite in extent, safe dimensions may be increased by means of standard buckling conversion methods or reactivity formula calculations which incorporate validated k-infinities, migration areas (M2), and extrapolation distances.

Where applicable, dimensional limitations shall include an allowance for fabrication tolerance and/or potential dimensional changes from corrosion or mechanical distortion.

When favorable geometry is not designed into equipment, geometry control may be achieved by altering the piece of equipment or by controlling the geometry of the fissile material administratively. An example of such alteration is the installation of drain holes in an otherwise non-safe vessel to control solution slab thickness. Geometry control may also be achieved by administrative action such as limiting the depth of fuel rods on a storage tray or by controlling the volume of a single container or the cumulative volume of multiple containers at a work station.

5.4.2.7 Density

Control of material density may be used to limit the k_{eff} of the unit to meet the guidelines given in Section 5.4.2.1.

5.4.2.8 Enrichment

Designs shall assume ^{235}U enrichment of 5.0 wt.% unless at least one of the following criticality safety controls on enrichment is established:

- Active engineered controls are established to verify enrichment and to prevent the introduction of uranium at unacceptable enrichment levels within the process system.
- Administrative controls are established to prevent the introduction of unacceptable enrichments within a defined subsystem within the same area. Considerations shall include minimizing human error such as using physical separation of areas of differing enrichment limits when practical.

5.4.2.9 Reflection

Critical parameters for units and arrays of units at abnormal conditions shall be based on full water reflection (30 cm), unless other reflectors in the immediate vicinity could result in higher reactivities, or controls on reflection are established to ensure that the k_{eff} meets the limits in Section 5.4.2.1. In such cases, the limitations on reflectors that are needed to assure criticality safety shall be controlled or it shall otherwise be shown that exceeding the amount of reflection assumed in establishing k_{eff} limits for credible abnormal conditions is highly unlikely.

5.4.2.10 Moderation

Critical parameters derived from nuclear criticality safety analyses shall be based on optimum credible moderation, unless controls on the amount of moderator are applied, or other controls on moderators are established to ensure that the k_{eff} meets the limits in Section 5.4.2.1.

The amount of hydrogenous material within the SNM may also be limited to a small percentage by weight of the SNM (moderation control) to prevent criticality, provided that each of the following is met:

1. The permitted concentration of hydrogenous material shall be equal to or less than 50 percent of the critical concentration for the system in question;
2. No credible accident conditions exist where localized concentration of the hydrogenous material could cause the k_{eff} guidelines given in Section; 5.4.2.1 to be exceeded.
3. The material shall be contained within a fire resistant barrier, or in a process area containing limited sources of hydrogenous material. In the absence of a fire resistant or noncombustible barrier, controls shall be used to prevent fires and to control the use of moderators in fire fighting in such process areas.

When moderation control is used and process variables can impact the moderator content of the material produced, the process variables needed to assure a safe amount of moderator shall be controlled.

Other mechanisms of controls on moderators may be used if established consistent with ANSI/ANS-8.22-1997.

5.4.2.11 Uranium Concentration Control

Uranium concentration control may be used for criticality control in areas where geometry control is not practical.

The concentration of uranium dispersed or dissolved in another medium may be limited to prevent criticality based on either of the following:

1. The concentration limit shall assure that the k_{eff} meets the limit in Section 5.4.2.1 at normal and credible abnormal conditions. The abnormal conditions evaluated shall include:
   a. credible sources of high concentrations of SNM unless analyzed safe at any concentration
   b. precipitation or other credible mechanisms concentrating the SNM to the most reactive credible extent;
   c. increasing the concentration of the SNM to the maximum credible extent due to effects such as evaporation; and
   d. for arrays of units on concentration control, additional abnormal conditions to include (as applicable) array size, unit spacing, and interspersed moderation effects.

or

2. The concentration limit at worst case credible conditions shall not exceed 50 percent of the minimum critical concentration in the system being evaluated.

When concentration control is used, the actual concentration shall be measured prior to using concentration control or the process variables that can impact the concentration of the material produced shall be controlled to ensure a safe concentration.

Precautions shall be taken to ensure that precipitating agents will not be inadvertently introduced, e.g. using a normally closed tank. The effect and limitations placed in precipitants may be determined by controlled experimentation.

5.4.2.12 Neutron Interaction

Neutron interaction (exchange between individually subcritical units) shall be considered. Equipment within 365 cm (nominally 12 feet) that contains fissile material shall be considered. Equipment and facilities may be considered to be neutronically isolated if they are separated by any of the following:

- A 30 cm thick slab of water
- A 25 cm thick slab of concrete
- 365 cm (nominally twelve feet) of air

The following items may be excluded from interaction considerations:

- transfer pipes two inches or less in diameter
- in-transit un-moderated material
- in-transit moderated material in nominal 5-gallon or less containers spaced at least 30 cm away.
- items separated / isolated as defined above (e.g. 365 cm of air)

5.4.2.13 Spacing and Multi-Unit Arrays

The required spacing between moderated units within an array shall be clearly delineated or limited by mechanical means such that at least one of the following requirements is met:

1. The k_{eff} of the array under the maximum credible accident conditions shall be limited by the guidelines given in Section 5.4.2.1;
2. For multi-unit arrays, the number of units in the array does not exceed 50 percent of the calculated critical number.
3. The system may be compared to and shown to meet accepted empirical criteria, (e.g., Solid Angle methods).

The mechanical design of equipment or storage arrays in which deformation or rearrangement could result in the loss of a controlled parameter shall be reviewed for adequacy and verified to be correctly installed.

In instances where physical spacing restrictions from the design of equipment are credited in the supporting analyses, dimensions and, if applicable, nuclear properties will be verified. The facility configuration management program will be used to maintain these parameters.

5.4.2.14 Neutron Absorbers

Fixed Neutron Absorbers

Criticality safety may be assured through the use of fixed neutron absorbers, for example cadmium or boron, provided that:

1. Neutron absorbers are designed and fabricated as an integral part of the equipment;
2. Inspections to verify the continued integrity of the equipment and neutron absorber structure shall be performed on established time frequencies sufficient to ensure their effectiveness.

Borosilicate glass Raschig rings may be used in solutions of fissile material in a manner consistent with ANSI/ANS Standard 8.5.

Neutron Absorber Additives

Credit in the calculation of minimum critical mass or other critical parameters may be taken for the presence of neutron absorbers added to SNM during processing provided that:

1. Its continued presence and uniform distribution in the SNM can be ensured; and
2. The quantity of additive was confirmed to meet the required amount.

5.4.2.15 Process Parameter Controls

The facility may rely on control of process parameters for criticality control, e.g. the control of process temperature and residence time to assure dry powder.

When process variables are used for NCS controls, the process variables that are needed to ensure safe conditions shall be controlled. The bounding conditions and operational limits shall be specifically identified in the criticality safety analysis.

5.5 *Criticality Accident Alarm System (CAAS)*

SNM covered by the NCS program (see section 5.1) shall be located such that at least two detectors in the alarm system are capable of detecting a criticality originating in the material. The

system shall be capable of detecting, with at least two detectors, a criticality that provides an absorbed dose in soft tissue of 20 rads of combined neutron and gamma radiation at an unshielded distance of two meters from the reacting material within one minute. This capability meets the requirements of 10CFR70.24(a)(1) and the requirements of ANSI/ANS-8.3 (1986).

The criticality accident alarm system initiates immediate evacuation of the facility. Employees are trained in recognizing and responding to the evacuation signal.

The alarm shall be clearly audible in areas that must be evacuated or alternate means of notification shall be provided to notify personnel that evacuation is necessary.

The nuclear criticality accident alarm system is maintained through routine calibration and scheduled functional tests conducted in accordance with internal procedures. In the event of loss of normal power, emergency power is supplied to the criticality accident alarm system.

Should the nuclear criticality accident alarm system or a portion of the system be out of service for a period of more than four hours, movement of SNM in the affected area will cease until the alarm service has been restored, or until compensatory monitoring, approved by the nuclear criticality safety component, has been implemented.

The CAAS system shall be uniform throughout the facility for the type of radiation detected, the mode of detection, and the sound generated by the alarm signal.

Routine testing, calibration and/or maintenance of the system are permitted with no suspension of SNM movements.

5.6 *Sources of Criticality Data and Analytical Techniques*

The sources of criticality data and analytical techniques currently authorized for use by AREVA NP in performing criticality safety analyses include those identified below:

1. J. H. Chalmers, G. Walker, and J. Pugh, "Handbook of Criticality Data," UKAEA Handbook AHSB (S), 1965.
2. H. C. Paxton, J. T. Thomas, D. Callihan, and E. B. Johnson, "Critical Dimensions of Systems Containing 235U, 239Pu, and 233U," TID-7028, Division of Technical Information Extension, USAEC (1964).
3. Subcommittee 8 of the American Standards Association Sectional Committee N6 and Project 8 of the American Nuclear Society Standards Committee, "Nuclear Safety Guide," TID-7016, Rev. 2, Division of Technical Information Extension, USAEC (1978).
4. H. K. Clark, "Critical and Safe Masses and Dimensions of Lattices of U and UO2 Rods in Water," DP-1014, Savannah River Laboratory (1966).
5. H. K. Clark, "Maximum Safe Limits for Slightly Enriched Uranium and Uranium Oxide," Criticality Control of Fissile Materials, International Atomic Energy Agency, Vienna (1966), pp. 35-49.
6. H. C. Paxton, "Criticality Control in Operations with Fissile Material," LA-3366, Los Alamos Scientific Laboratory (1972).
7. H. F. Henry, C. E. Newlon, and J. R. Knight, "Extensions of Neutron Interaction Criteria," K-1478, Union Carbide Corporation, Nuclear Division (1969).

8. C. E. Newlon, AEC Research and Development Report, "Minimum Critical Cylinder Diameters of Hydrogen Moderated U (4.9) Systems," K-1629, Union Carbide Corporation, ORGDP, (1965).
9. Standards Committee N16 of the American National Standards Institute, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," ANSI N16.1-1969, American National Standards Institute, New York, NY (1969).
10. R. D. Carter, G. R. Kiel, and K. R. Ridgway, "Criticality Handbook," Volumes I, II, and III, ARH-600, Atlantic Richfield Hanford Company, (1968).
11. W. Thomas, W. Weber, and W. Heinicke, "Handbuch zur Kritikalitat," (1970-73).
12. H. F. Henry, "Studies in Nuclear Safety," K-1380, (1957).
13. C. L. Brown, "Nuclear Criticality Safety Analysis, Uranium Fuels Plant," Jersey Nuclear Company, BNW/JN-29, (1971).
14. "Determination of H/U Ratios in UO2 Water and ADU-Water Mixtures," JN-71-2, (1971).
15. C. L. Brown, et. al., "Validation of Boundary Conditions for Assuming Nominal Reflection in Solid Angle Interaction Method (As Applied in Exxon Fuel Fabrication Plants)," BNW/XN-184, (1975).
16. "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," NUREG/CR-0200.
17. EMF-2670 PC-SCALE 4.4a Validation Revision 2, dated March 10, 2004.
18. American Nuclear Society, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," ANSI/ANS 8.1-1998.
19. US Nuclear Regulatory Commission Regulatory Guide 3.14, "Validation of Calculational Methods for Nuclear Criticality Safety."
20. Ronald Allen Knief, "Nuclear Criticality Safety Theory and Practice." American Nuclear Society, 1993.

Table 5-1 Safety Factors for Homogeneous Single Units

Safety Parameter	Critical Parameter	Safety Factor
Safe Mass SNM, M_s	Critical Mass SNM, M_k	0.45
Safe Mass SNM, M_{sL} (Exceeding of M_{sL} is excluded by engineering controls)	Critical Mass SNM, M_k	0.75
Safe Mass Moderator, M_{smod}	Critical Mass of Moderator, M_{kmod}	0.50
Safe Spherical Volume, V_s	Critical Spherical Volume,	0.80
Safe Infinite Cylinder, D_s	Critical Infinite Cylinder, D_k	0.90
Safe Infinite Slab, S_s	Critical Infinite Slab, S_k	0.90
Safe Concentration, C_s	Critical Concentration, C_k	0.50

Table 5-2 Safety Factors for Heterogeneous Single Units

Safety Parameter	Critical Parameter	Safety Factor
Safe Mass SNM, M_s	Critical Mass SNM, M_k	0.45
Safe Mass SNM, M_{sL} (Exceeding of M_{sL} is excluded by engineered controls)	Critical Mass SNM, M_k	0.75
Safe Mass Moderator, M_{smod}	Critical Mass of Moderator, M_{kmod}	0.50
Safe Spherical Volume, V_s	Critical Spherical Volume, V_k	0.80
Safe Infinite Cylinder, D_s	Critical Infinite Cylinder, D_k	0.90
Safe Infinite Slab, S_s	Critical Infinite Slab, S_k	0.90

EHS&L Document

SNM-1227 - Chapter 6
Chemical Process Safety

Nature of Changes

Item	Paragraph	Description	Justification
1.	New Document	New Document	License Renewal
2.			
3.			
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10.			
List Below any Documents, including Forms & Operator Aids which must be issued concurrently with this document revision:			
Do Not make document effective until Loren Maas releases after NRC approval.			

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All new and/or revised procedures shall be approved by the change author, cognizant manager(s) of areas affected by the changes, and by applicable manager(s) of any function that approved the previous revision of the document unless responsibility for such approval has been transferred to another organization. Also, the procedure shall be approved by manager(s) of functional organizations that provide technical reviews with the exception of the Training Department. Finally, Document Control shall verify that the required approvals have been properly obtained and that any documents that must be issued concurrently are ready to be issued.

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Document Reviews			Document Approvals	
Purpose/Function of Review	Specify Reviewer(s) (Optional except for change author)	(Check all that apply)	Title of Approver	(Check all that Apply)
Document Control (Automatic)		☒	Document Control (Automatic)	☒
Change Author	LJ Maas	☒	Author	☒
Independent Technical Review	TC Probasco	☒		
Operability Review(s)			Mgr, Richland Operations[1]	☐
Conversion		☐	Mgr, Uranium Conversion & Recovery Operations[1]	☐
Recovery		☐		
Ceramics		☐	Mgr, Ceramic Operations[1]	☐
Rods		☐	Mgr, Rods & Bundles[1]	☐
Bundles		☐		
Transportation		☐		
Components		☐	Mgr, Component Fabrication[1]	☐
Maintenance Review		☐	Mgr, Maintenance[1]	☐
Lab Review		☐	Mgr, Analytical Services[1]	☐
EHS&L Review(s)			Mgr, EHS&L[2]	☒
Criticality		☐	Mgr, Criticality Safety[2]	☐
Radiation Protection		☐	Mgr, Safety, Security & Emergency Preparedness[2]	☐
Safety/Security		☐		
Emergency Preparedness		☐		
MC&A	LJ Maas	☒	Mgr, Licensing & Compliance[2]	☐
Transportation		☐		
Environmental		☐		
BWR Product Eng. Review		☐	Mgr, BWR Product Engineering	☐
Quality Review		☐	Mgr, Quality	☐
Ops. Projects & Planning Review		☐	Mgr, Ops. Projects & Planning	☐
Others:		☐	Mgr, Richland Site/Other	☐
Training & Employee Dev.:[3]		☐	Training & Employee Dev.	☐

[1]Note: If approvals include 2 or more product center managers, the Operations manager can be substituted for the applicable product center managers.

[2]Note: If approvals include 2 or more EHS&L functional managers, the EHS&L manager can be substituted for the applicable EHS&L functional managers.

[3]Note: Training department review is required for all procedures that require or affect a Learning Plan and if additional training materials or curriculum must be revised before issuing procedure.

EHS&L Change Impact Evaluation Form		
Document / ECN No*.: E10-08-006		Change Evaluator: TC Probasco
Does the change potentially impact Criticality Alarm System (CAS) coverage?	☐ Yes ☒ No	If yes, explain:
NRC Pre-Approval Evaluation:		
Is NRC Pre-approval (License Amendment) Needed? (Based on "Yes" answer to any of five questions below). (Based on "No" answer to all five questions below).	☒ Yes ☐ No	This is part of our NRC license renewal.
1. Does the change create new types of accident sequences that, unless mitigated or prevented, would exceed the performance requirements of 10 CFR 70.61 (create high or intermediate consequence events) and that have not previously been described in AREVA NP Inc's ISA Summary?	☐ Yes ☒ No	If yes, explain:
2. Does the change use new processes, technologies, or control systems for which AREVA NP Inc. has no prior experience?	☐ Yes ☒ No	If yes, explain:
3. Does the change remove, without at least an equivalent replacement of the safety function, an item relied on for safety that is listed in the ISA Summary?	☐ Yes ☒ No	If yes, explain:
4. Does the change alter any item relied on for safety, listed in the ISA Summary, that is the sole item preventing or mitigating an accident sequence of high or intermediate consequences?	☐ Yes ☒ No	If yes, explain:
5. Does the change qualify as a change specifically prohibited by NRC regulation, order or license condition?	☐ Yes ☒ No	If yes, explain:
Actions Required Prior to or Concurrent with Change Implementation Evaluation:		
Action		**Explanation**
6. Modification / Addition to CAS system or system coverage documentation	☐ Yes ☒ No	If yes, explain:
7. Acquire NRC pre-approval (license amendment)	☒ Yes ☐ No	If yes, explain: License renewal.
8. Conduct/modify ISA	☐ Yes ☒ No	If yes, explain:
9. ISA Database Modification	☐ Yes ☒ No	If yes, explain:
10. Modification of other safety program information / underlying analyses (PHA, RHA, FHA, NCSA, etc.)	☐ Yes ☒ No	If yes, explain:
Actions required subsequent to Change Implementation Evaluation:		
11. Update safety program information (PHA,RHA,FHA,NCSA, P&ID)	☐ Yes ☒ No	If yes, explain:

* **If this form exists as a part of a document, the document number is not required.**

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Application Date:	NRC Docket No. 70-1257
October 27, 2006	

6.0 Chemical Process Safety

6.1 *Richland Facility Safety Program*

AREVA NP Inc. (AREVA NP) has established, and will maintain, a safety program for the Richland facility that includes chemical process safety. With respect to NRC oversight, the program will address chemical risks of licensed materials, risks of chemicals produced from licensed materials, and chemical risks of plant conditions that could affect the safety of licensed materials. Chemical safety is an element of the Integrated Safety Analysis (ISA) process set forth in Chapter 3, Integrated Safety Analysis (ISA) and ISA Summary.

AREVA NP has developed, and will maintain, approved written procedures to assure effective implementation of its facility safety program.

6.2 *Application of Chemical Safety Controls*

The relative effectiveness and reliability of controls are considered during the chemical safety analysis process. Passive engineered controls to ensure chemical containment are preferred over other system controls and are utilized when practical and appropriate. Active engineered controls are the next preferred method of control followed by administrative controls.

Passive Engineered Controls are physical restraints or features that maintain chemical safety in a static manner (e.g., pipe and tank integrity, barriers to prevent collisions with pipes and tanks, etc.). Beyond appropriate initial installation, passive engineered controls require no action or other response to be effective when called upon to ensure safety. Assurance is maintained through periodic inspections or verification measurement(s) as appropriate.

Active Engineered Controls are a means of control involving active hardware (e.g., electrical, mechanical, hydraulic) that protect against chemical release/exposure. These devices act by providing automatic action to secure the system to a safe condition or to remove chemicals before they enter the environment, e.g., shutting off tank input prior to overfilling, scrubbing chemicals prior to release. Assurance is maintained through specific periodic functional testing as appropriate. Active engineered controls that are provided with fail-safe conditions (e.g., the failure of the control results in a safe condition) are the preferred type of active engineered control.

Enhanced Administrative Controls involve human intervention augmented by warning devices or other automated prompts and are preferred over simple procedural requirements for a specific action.

Administrative Controls are controls, typically imposed via procedures, that rely on actions, judgment, and responsible actions of people for their implementation. Their use is primarily limited to situations where passive and active engineered controls are not practical. Assurance is maintained through training, experience, and audit.

6.3 *Process Chemical Risk and Accident Sequences*

6.3.1 Provision of Process Descriptions

A general process description for the primary production operations on the Richland site is provided in Section 1.1 of Chapter 1, General Information, of this license application. This is supplemented by more detailed process system-by-process system descriptions in the Richland facility's ISA Summary.

6.3.2 Identification and Evaluation of Chemical Accident Sequences

Potential accident sequences involving chemical hazards are addressed as part of the Richland site ISA. Accident sequence identification, consequence determination, likelihood determination, and risk evaluation are further described in the Richland site ISA Summary.

The Richland ISA considers chemical risks of licensed materials, risks of chemicals derived from licensed materials, and chemical risks imparted by plant conditions that in turn could potentially affect the safety of licensed materials. Assumptions consider maximum foreseeable inventories at specific use locations; cover routine, non-routine, and abnormal operational scenarios; and conservatively consider the physical properties of the pertinent chemicals. Results of the evaluations are compared to the performance criteria in 10 CFR 70.61 and documented in the site ISA Summary.

6.3.3 Vapor Dispersion Modeling

Vapor dispersion models may be utilized to assess the consequences of chemical release accident scenarios, within the applicability of the ISA, for comparison to the performance criteria in 10 CFR 70.61. Source term input shall be conservative and underlying assumptions shall be documented. Dispersion evaluation approach and capabilities of the model shall be appropriate for the chemical species under evaluation. Preference will be given to models specifically validated for the pertinent chemical species.

6.3.4 Chemical Exposure Standards

In support of the evaluation of chemical consequences under the ISA, AREVA NP will identify quantitative chemical exposure standards in accordance with 10 CFR 70.65(b). These exposure standards are documented in the Richland site ISA Summary.

6.4 *IROFS and Management Measures*

Chemical hazards pertinent to NRC oversight are addressed as part of the Richland site ISA. Consequences of unmitigated chemical accident sequences are compared to the performance criteria of 10 CFR 70.61.

For unmitigated chemical accident sequences not meeting 10 CFR 70.61 performance criteria, items relied on for safety (IROFS) are identified and implemented to establish compliance with those criteria. The accident sequences, their associated consequences and likelihoods, and IROFS, as applicable, are documented in the Richland site ISA Summary.

Management measures shall be identified and implemented as required to assure that engineered and administrative IROFS are available and reliable to perform their functions when called upon. AREVA NP's management measures programs are addressed in Chapter 11, Management Measures.

6.5 *Requirements for New Facilities/New Processes*

For planned new facilities and/or new processes at the Richland site meeting the 10 CFR 70.72 criteria for a license amendment, facility/process design shall address the baseline design criteria (BDC) of 10 CFR 70.64 as they apply to control of chemical hazards. Application of the BDC may be limited to those chemical safety controls that are identified as items relied on for safety per the applicable ISA.

EHS&L Document

SNM-1227 - Chapter 7
Fire Safety

Nature of Changes

Item	Paragraph	Description	Justification
1.	New Document	New Document	License Renewal
2.			
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Training & Employee Dev.:[3]		☐	Training & Employee Dev.	☐

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[3]Note: Training department review is required for all procedures that require or affect a Learning Plan and if additional training materials or curriculum must be revised before issuing procedure.

EHS&L Change Impact Evaluation Form		
Document / ECN No*.: E10-08-007		Change Evaluator: TC Probasco
Does the change potentially impact Criticality Alarm System (CAS) coverage?	☐ Yes ☒ No	If yes, explain:
NRC Pre-Approval Evaluation:		
Is NRC Pre-approval (License Amendment) Needed? (Based on "Yes" answer to any of five questions below). (Based on "No" answer to all five questions below).	☒ Yes ☐ No	This is part of our NRC license renewal.
1. Does the change create new types of accident sequences that, unless mitigated or prevented, would exceed the performance requirements of 10 CFR 70.61 (create high or intermediate consequence events) and that have not previously been described in AREVA NP Inc's ISA Summary?	☐ Yes ☐ No	If yes, explain:
2. Does the change use new processes, technologies, or control systems for which AREVA NP Inc. has no prior experience?	☐ Yes ☒ No	If yes, explain:
3. Does the change remove, without at least an equivalent replacement of the safety function, an item relied on for safety that is listed in the ISA Summary?	☐ Yes ☒ No	If yes, explain:
4. Does the change alter any item relied on for safety, listed in the ISA Summary, that is the sole item preventing or mitigating an accident sequence of high or intermediate consequences?	☐ Yes ☒ No	If yes, explain:
5. Does the change qualify as a change specifically prohibited by NRC regulation, order or license condition?	☐ Yes ☒ No	If yes, explain:
Actions Required Prior to or Concurrent with Change Implementation Evaluation:		
Action		**Explanation**
6. Modification / Addition to CAS system or system coverage documentation	☐ Yes ☒ No	If yes, explain:
7. Acquire NRC pre-approval (license amendment)	☒ Yes ☐ No	If yes, explain: License renewal.
8. Conduct/modify ISA	☐ Yes ☒ No	If yes, explain:
9. ISA Database Modification	☐ Yes ☒ No	If yes, explain:
10. Modification of other safety program information / underlying analyses (PHA, RHA, FHA, NCSA, etc.)	☐ Yes ☒ No	If yes, explain:
Actions required subsequent to Change Implementation Evaluation:		
11. Update safety program information (PHA,RHA,FHA,NCSA, P&ID)	☐ Yes ☒ No	If yes, explain:

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Application Date:	NRC Docket No. 70-1257
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7.0 Fire Safety

The AREVA NP Inc. (AREVA NP) fire safety program for the Richland Horn Rapids Road (HRR) facility includes design features, management measures, and operational controls to provide protection against fires and explosions that could affect the safety of licensed materials, thereby creating an increased radiological risk. The potential for, and consequences of, fires and explosions associated with ongoing Richland operations are analyzed in conjunction with the overall site integrated safety analysis as described in Chapter 3, Integrated Safety Analysis (ISA) and ISA Summary. The operational fire hazards analyses provided by the ISA program are complemented by the Richland site general fire safety program. Included in the general fire safety program are general fire safety management measures, facility design requirements, and general fire protection and emergency response measures.

7.1 *Fire Safety Management Measures*

7.1.1 Fire Safety Organization

The manager of the Environmental, Health, Safety and Licensing (EHS&L) function is the senior-level manager vested with the authority and staff to assure that fire safety receives appropriate priority. The EHS&L functional manager reports directly to the Richland Site Manager independently of Operations. Within the EHS&L function, responsibility for fire safety issues rests within the Safety Function. The organizational structure and qualification requirements pertinent to these management positions are set forth in Chapter 2, Organization and Administration.

The Operations Manager is responsible for the day-to-day maintenance of fire safety in the plant production areas, including the provision of trained operators cognizant of fire safety and applicable fire safety-related IROFS. Through the Maintenance Function, the Operations Manager is also responsible for the periodic testing and maintenance of installed fire detection and protection systems.

The Manager of the Plant Projects Function as defined in Chapter 2 is responsible for the engineering and installation of new/modified facilities and equipment and ensuring that these additions/modifications comply with codes, standards and regulations pertinent to fire safety. He manages the plant configuration management program, which assures that plant changes are properly evaluated with respect to fire safety impact and then properly documented as part of the plant safety basis.

7.1.2 Fire Prevention

7.1.2.1 Employee Training

General awareness training with respect to fire safety is provided to site employees as an element of the annual general employee safety training program.

7.1.2.2 Fire Prevention Procedures

The Richland site maintains procedures for:

- flammable and combustible liquids/solids storage and handling;
- control of hot work, including a permit system for cutting, welding, and grinding outside of designated areas.

7.1.2.3 Inspections

The Richland site maintains a program for periodic fire safety inspections of facilities containing licensed materials.

7.1.2.4 Non-Smoking Policy

The Richland site enforces a non-smoking policy for site facilities, irrespective of radiological or chemical inventories. Smoking on-site is restricted to designated outdoor smoking areas.

7.1.3 Inspection, Testing, and Maintenance of Fire Protection Systems

Preventive maintenance (PM) procedures are established for the inspection, testing, and maintenance of fire protection systems in accordance with applicable state and local (City of Richland) fire codes. These procedures are applied to fire detection and warning systems, fixed fire suppression systems, portable fire extinguishers, and emergency power sources. Records of these activities are maintained within the PM system.

7.1.4 Emergency Response Organization

AREVA NP maintains an emergency response organization/system commensurate with the potential emergencies at the Richland site and their potential adverse impacts to workers, the public, and the surrounding environment. Commitments to maintain an emergency plan in accordance with 10 CFR 70.22 as well as the procedures to implement the plan are set forth in Chapter 8, Emergency Management. The Emergency Plan outlines the Richland site's overall emergency response program, including but not limited to staffing, training, drills and exercises, response measures, and offsite agency coordination.

7.1.5 Pre-Fire Plan

AREVA NP maintains a pre-fire (pre-emergency) plan which is provided to, and meets the requirements of, the local fire jurisdiction (City of Richland).

7.2 *Fire Hazards Analysis*

A fire hazards analysis (FHA) is performed for facilities on the Richland site that contain SNM in sufficient quantities and in a form that, if released in a fire, could result in at least an intermediate consequence event as defined in 10 CFR 70.61. FHA's are conducted in general accordance with the NRC's "Guidance on Fire Protection for Fuel Cycle Facilities (August 1992)." The FHAs focus on bounding fire scenarios for discrete fire areas within buildings and address fire loading, consequences of an unmitigated fire (analysis and impacts), and mitigative controls. FHAs are one component of the overall Richland site ISA as described in Chapter 3.

Fire hazards at the operational/process level are analyzed with respect to potential accident sequences, likelihoods, consequences, and resultant risk by the ISA teams. Fire/explosion-related accident sequences with the potential to create high or intermediate consequences as defined in 10 CFR 70.61 are controlled via the application of items relied on for safety (IROFS) as required to meet the performance requirements of 10 CFR 70.61. IROFS for the Richland

site, including those related to fire/explosion hazards, are documented in the Richland site ISA Summary. The need for new and/or modified IROFS related to plant additions and modifications is assessed as part of the plant configuration management program. This and other management programs to assure that IROFS are available and reliable when needed are described in Chapter 11, Management Measures.

7.3 *Facility Design*

7.3.1 Facility Design Criteria

Richland site facilities and buildings have been designed and built to the applicable national, state, and local building, electrical, and fire codes as required by the City of Richland Fire Marshal and Building Department at their time of construction. In recognition of the fire-induced hazards associated with licensed materials and process chemicals, enhanced emphasis is placed on:

- minimizing combustible materials in the construction of facilities;
- provision and maintenance of effective intra-building fire barriers;
- segregating non-radiological and radiological operations to the extent feasible;
- use of physical layouts, configurations, and materials that promote ease of decontamination;
- provision of HEPA-filtered ventilation systems that maintain flow gradients from areas of lower contamination to areas of higher contamination and that provide effective smoke control in the event of a fire;
- provision of auxiliary electrical power to key plant systems, especially those important in the detection and mitigation of off-normal events, including fires; and
- utilization of non-combustible and segregated and/or remotely located storage facilities for large quantities of radioactive materials.

For planned new facilities and/or new processes at the Richland site meeting the 10 CFR 70.72 criteria for a license amendment, facility/process design shall address the baseline design criteria (BDC) and defense-in-depth requirements of 10 CFR 70.64 as they apply to fire protection.

7.4 *Process Fire Safety*

Process fire safety is appropriately considered in the planning, design, and construction of new facilities and processes. At the operational level, process fire safety as it relates to processes utilizing licensed material is evaluated and assured via the ISA process described in Chapter 3. The ISA evaluates the special fire risks associated with:

- combustibles and flammable process chemicals (solids, liquids, gases), in use and in storage;
- exothermic reactions of uranium oxides;
- high temperature and/or high pressure equipment, including but not limited to the incinerator, sintering furnaces, calciners, and boilers; and
- laboratory operations, including specialty laboratory equipment, hoods, and chemicals.

Process-related fire hazards are controlled with items relied on for safety (IROFS) to the extent necessary to meet the performance requirements of 10 CFR 70.61.

7.5 *Fire Protection and Emergency Response*

7.5.1 Plant Water Supply

Pressurized water for plant usage (process, domestic, sprinklers, hydrants) is supplied to the Richland plant via two mains which enter at opposite sides of the plant. The mains are fed from separate portions of the City of Richland water grid. Fire hydrants are strategically located throughout the plant site in accordance with local fire code requirements.

7.5.2 Fire Detection/Alarm Systems

The Richland site is covered by an electronically-supervised fire alarm system that alarms locally (fire bells) at the continuously manned Central Guard Station and at a remote alarm monitoring facility that provides notification to the City of Richland.

The alarm system may be activated via a variety of alarm modules, including:

- smoke detectors
- heat detectors
- manual alarm pull stations
- water flow switches within fire sprinkler lines
- monitor switches from the hydrogen detection systems serving certain H_2-utilizing process systems.

7.5.3 Fire Suppression

Fire suppression relies primarily on the fire alarm system and subsequent truck, hydrant, and hose coverage provided by the City of Richland Fire Department. A limited number of facilities are served by a piped fire sprinkler system, including the onsite combustible waste incinerator.

7.5.4 Portable Fire Extinguishers

Portable fire extinguishers are available across the site to provide incipient fire fighting capability. The following types of extinguishers are in use:

- CO_2
- Dry Chemical
- Class D (Metal X)

In addition the site has an AFFF (aqueous film-forming foam) trailer that can quickly be transported to the scene of a fire.

7.5.5 Emergency Power Generators

The site maintains a number of fixed-location, emergency power generators to provide auxiliary electrical power to designated equipment and emergency systems, including the fire alarm system. The generators are periodically tested in accordance with an established preventive maintenance procedure.

7.5.6 Fire Protection System/Equipment Standards

Fire protection systems and equipment on the Richland site are installed, maintained, and tested in accordance with International Building and Fire Codes as adopted/modified within state and local (City of Richland) fire codes.

7.5.7 Emergency Response Support

The Richland site's emergency response capabilities (internally and externally provided) are set forth in the Richland Emergency Plan (see Chapter 8, Emergency Management). Services provided by outside agencies are formalized in written agreements (Memoranda of Understanding) maintained on-file at the Richland site. These include agreements with the City of Richland which provides both firefighting and emergency medical response services. City of Richland emergency response personnel are included as appropriate in site emergency drills and exercises.

EHS&L Document

SNM-1227 - Chapter 8
Emergency Management

Nature of Changes

Item	Paragraph	Description	Justification
1.	New Document	New Document	License Renewal
2.			
3.			
4.			
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Document Reviews			**Document Approvals**	
Purpose/Function of Review	Specify Reviewer(s) (Optional except for change author)	(Check all that apply)	Title of Approver	(Check all that Apply)
Document Control (Automatic)		☒	Document Control (Automatic)	☒
Change Author	LJ Maas	☒	Author	☒
Independent Technical Review	TC Probasco	☒		
Operability Review(s)			Mgr, Richland Operations[1]	☐
Conversion		☐	Mgr, Uranium Conversion & Recovery Operations[1]	☐
Recovery		☐		
Ceramics		☐	Mgr, Ceramic Operations[1]	☐
Rods		☐	Mgr, Rods & Bundles[1]	☐
Bundles		☐		
Transportation		☐		
Components		☐	Mgr, Component Fabrication[1]	☐
Maintenance Review		☐	Mgr, Maintenance[1]	☐
Lab Review		☐	Mgr, Analytical Services[1]	☐
EHS&L Review(s)			Mgr, EHS&L[2]	☒
Criticality		☐	Mgr, Criticality Safety[2]	☐
Radiation Protection		☐	Mgr, Safety, Security & Emergency Preparedness[2]	☐
Safety/Security		☐		
Emergency Preparedness		☐		
MC&A	LJ Maas	☒	Mgr, Licensing & Compliance[2]	☐
Transportation		☐		
Environmental		☐		
BWR Product Eng. Review		☐	Mgr, BWR Product Engineering	☐
Quality Review		☐	Mgr, Quality	☐
Ops. Projects & Planning Review		☐	Mgr, Ops. Projects & Planning	☐
Others:		☐	Mgr, Richland Site/Other	☐
Training & Employee Dev.:[3]		☐	Training & Employee Dev.	☐

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[2]Note: If approvals include 2 or more EHS&L functional managers, the EHS&L manager can be substituted for the applicable EHS&L functional managers.

[3]Note: Training department review is required for all procedures that require or affect a Learning Plan and if additional training materials or curriculum must be revised before issuing procedure.

EHS&L Change Impact Evaluation Form		
Document / ECN No*.: E10-08-008		Change Evaluator: TC Probasco
Does the change potentially impact Criticality Alarm System (CAS) coverage?	☐ Yes ☒ No	If yes, explain:
NRC Pre-Approval Evaluation:		
Is NRC Pre-approval (License Amendment) Needed? (Based on "Yes" answer to any of five questions below). (Based on "No" answer to all five questions below).	☒ Yes ☐ No	This is part of our NRC license renewal.
1. Does the change create new types of accident sequences that, unless mitigated or prevented, would exceed the performance requirements of 10 CFR 70.61 (create high or intermediate consequence events) and that have not previously been described in AREVA NP Inc's ISA Summary?	☐ Yes ☒ No	If yes, explain:
2. Does the change use new processes, technologies, or control systems for which AREVA NP Inc. has no prior experience?	☐ Yes ☒ No	If yes, explain:
3. Does the change remove, without at least an equivalent replacement of the safety function, an item relied on for safety that is listed in the ISA Summary?	☐ Yes ☒ No	If yes, explain:
4. Does the change alter any item relied on for safety, listed in the ISA Summary, that is the sole item preventing or mitigating an accident sequence of high or intermediate consequences?	☐ Yes ☒ No	If yes, explain:
5. Does the change qualify as a change specifically prohibited by NRC regulation, order or license condition?	☐ Yes ☒ No	If yes, explain:
Actions Required Prior to or Concurrent with Change Implementation Evaluation:		
Action		**Explanation**
6. Modification / Addition to CAS system or system coverage documentation	☐ Yes ☒ No	If yes, explain:
7. Acquire NRC pre-approval (license amendment)	☒ Yes ☐ No	If yes, explain: License renewal.
8. Conduct/modify ISA	☐ Yes ☒ No	If yes, explain:
9. ISA Database Modification	☐ Yes ☒ No	If yes, explain:
10. Modification of other safety program information / underlying analyses (PHA, RHA, FHA, NCSA, etc.)	☐ Yes ☒ No	If yes, explain:
Actions required subsequent to Change Implementation Evaluation:		
11. Update safety program information (PHA,RHA,FHA,NCSA, P&ID)	☐ Yes ☒ No	If yes, explain:

* **If this form exists as a part of a document, the document number is not required.**

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Application Date:	NRC Docket No. 70-1257
October 27, 2006	

8.0 Emergency Management

8.1 *Emergency Plan*

AREVA NP Inc. (AREVA NP) has and will maintain an emergency plan for its Richland fuel fabrication facility in accordance with the requirements of 10 CFR 70.22(i). Content of the plan is in accordance with 10 CFR 70.22(i)(3) and considers U.S. Nuclear Regulatory Commission (NRC) guidance in Regulatory Guide 3.67, "Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities."

8.2 *Emergency Plan Implementing Procedures*

AREVA NP has developed and will maintain approved written procedures to assure effective implementation of its facility emergency plan.

8.3 *Amendment of the Emergency Plan*

In accordance with 10 CFR 70.32(i), AREVA NP will not implement changes to the plan that decrease its effectiveness without prior application to, and prior approval by, the NRC. Changes that do not decrease the effectiveness of the plan may be made without NRC approval. Copies of such changes will be provided to the NRC and appropriate organizations within six months of making the changes, in accordance with 10 CFR 70.32(i).

8.4 *Agreements with Offsite Emergency Response Resources*

AREVA NP has established and maintains formal written agreements with appropriate offsite emergency response organizations (fire, police, medical, etc.) as needed to assure implementation of its emergency plan and associated response procedures. These agreements are reviewed and renewed as appropriate and maintained on file at the Richland site.

EHS&L Document

SNM-1227 - Chapter 9
Environmental Protection

Nature of Changes

Item	Paragraph	Description	Justification
1.	New Document	New Document	License Renewal
2.			
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Independent Technical Review	RK Burklin	☒		
Operability Review(s)			Mgr, Richland Operations[1]	☐
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Recovery		☐		
Ceramics		☐	Mgr, Ceramic Operations[1]	☐
Rods		☐	Mgr, Rods & Bundles[1]	☐
Bundles		☐		
Transportation		☐		
Components		☐	Mgr, Component Fabrication[1]	☐
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Radiation Protection		☐	Mgr, Safety, Security & Emergency Preparedness[2]	☐
Safety/Security		☐		
Emergency Preparedness		☐		
MC&A	LJ Maas	☒	Mgr, Licensing & Compliance[2]	☐
Transportation		☐		
Environmental		☐		
BWR Product Eng. Review		☐	Mgr, BWR Product Engineering	☐
Quality Review		☐	Mgr, Quality	☐
Ops. Projects & Planning Review		☐	Mgr, Ops. Projects & Planning	☐
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Training & Employee Dev.:[3]		☐	Training & Employee Dev.	☐

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EHS&L Change Impact Evaluation Form		
Document / ECN No*.: E10-08-009		Change Evaluator: RK Burklin
Does the change potentially impact Criticality Alarm System (CAS) coverage?	☐ Yes ☒ No	If yes, explain:
NRC Pre-Approval Evaluation:		
Is NRC Pre-approval (License Amendment) Needed? (Based on "Yes" answer to any of five questions below). (Based on "No" answer to all five questions below).	☒ Yes ☐ No	This is part of our NRC license renewal.
1. Does the change create new types of accident sequences that, unless mitigated or prevented, would exceed the performance requirements of 10 CFR 70.61 (create high or intermediate consequence events) and that have not previously been described in AREVA NP Inc's ISA Summary?	☐ Yes ☒ No	If yes, explain:
2. Does the change use new processes, technologies, or control systems for which AREVA NP Inc. has no prior experience?	☐ Yes ☒ No	If yes, explain:
3. Does the change remove, without at least an equivalent replacement of the safety function, an item relied on for safety that is listed in the ISA Summary?	☐ Yes ☒ No	If yes, explain:
4. Does the change alter any item relied on for safety, listed in the ISA Summary, that is the sole item preventing or mitigating an accident sequence of high or intermediate consequences?	☐ Yes ☒ No	If yes, explain:
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Actions Required Prior to or Concurrent with Change Implementation Evaluation:		
Action		**Explanation**
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7. Acquire NRC pre-approval (license amendment)	☒ Yes ☐ No	If yes, explain: License renewal
8. Conduct/modify ISA	☐ Yes ☒ No	If yes, explain:
9. ISA Database Modification	☐ Yes ☒ No	If yes, explain:
10. Modification of other safety program information / underlying analyses (PHA, RHA, FHA, NCSA, etc.)	☐ Yes ☒ No	If yes, explain:
Actions required subsequent to Change Implementation Evaluation:		
11. Update safety program information (PHA,RHA,FHA,NCSA, P&ID)	☐ Yes ☒ No	If yes, explain:

* **If this form exists as a part of a document, the document number is not required.**

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9.0 Environmental Protection

9.1 *Environmental ALARA*

The Richland site's as low as reasonably achievable (ALARA) program is described in Chapter 4.0, "Radiation Protection." With respect to environmental protection, AREVA NP Inc's (AREVA-NP's) goal is to maintain concentrations of radioactive materials in plant effluents and the surrounding environs at ALARA levels. Furthermore, environmental releases shall be limited and monitored such that compliance with the public dose limits of 10 CFR 20.1301 and the effluent limits of 10 CFR 20.1302 can be achieved and demonstrated.

9.2 *Gaseous Effluent Control*

Operating and engineered controls will be used as necessary to ensure that environmental airborne concentrations of radioactive materials attributable to gaseous effluents and resultant radiological doses to members of the public comply with the concentration limits and public dose limits specified in 10 CFR 20. The site will maintain procedures with action levels to assure that compliance with applicable limits is maintained. This includes the ALARA constraint on air emissions of 10 CFR 20.1101(d).

Dose calculations as well as environmental concentrations in 10 CFR 20, Appendix B, Table 2 for members of the public may be modified based on ICRP 66 and 68, assuming an Activity Median Aerodynamic Diameter (AMAD) of 1 micrometer.

9.2.1 Gaseous Effluent Sampling

Continuous representative sampling shall be provided on stacks exhausting air with potential concentrations of radioactive materials that are significant with respect to the site's compliance with 10 CFR 20. These samples shall be periodically analyzed for particulate radioactive material, generally on a weekly basis. Stack samples of gaseous effluents potentially containing uranium shall be analyzed for gross alpha activity. Non-uranium isotopes likely to contribute greater than 0.1 mrem total effective dose equivalent (TEDE) per year to a member of the public will be accounted for.

9.2.2 High-Efficiency Particulate Absolute (HEPA) Filtration

Air exhausted from areas, equipment, and/or activities that may contain concentrations of radioactive materials that are significant with respect to the site's compliance with 10 CFR 20 shall be passed through at least one stage of HEPA filtration prior to release via an exhaust stack. Fire-resistant HEPA filters that are certified by the manufacturer as meeting HEPA efficiency specifications shall be used.

The adequacy of final HEPA filter installations shall be verified by in-place testing prior to initiating operations with radioactive materials in the following instances:

- Startup of a new facility.
- Following replacement of final filters
- After maintenance work on the final filter bank that could have foreseeable adverse impacts on their effective operation.
- After exposure of the final filters to a condition or agent that may have adversely impacted their effective operation, if deemed necessary based on a visual/operational inspection.

9.2.3 Final HEPA Filter Surveillance

Systems exhausting air from areas or process equipment containing corrosive fumes shall be designated and the final HEPA filters periodically inspected in accordance with an established preventive maintenance (PM) procedure.

Final HEPA filter installations shall be equipped with pressure differential measuring/indicating devices. Measured differential pressures shall be used to evaluate the need for filter changeout/maintenance. See Chapter 4, Section 4.6.1 for evaluation criteria.

9.3 *Liquid Effluent Controls*

Operating and engineered controls will be used as necessary to ensure that concentrations of radioactive materials in liquid effluents and sewer discharges are compliant with 10 CFR 20.1302 and 10 CFR 20.2003, respectively. The site will maintain procedures with action levels to ensure that compliance with applicable limits is maintained.

9.3.1 Wastewater Collection/Treatment

Wastewaters contaminated with, or subject to contamination with, radioactive materials shall be managed, as appropriate, within the plant's process wastewater management system. After any necessary treatment, process wastewaters may be combined with plant sanitary sewage and non-contaminated cooling water streams for discharge to the City of Richland sewer system. The process wastewater management system shall provide treatment required to assure compliance with 10 CFR 20 radiological sewering limits.

9.3.2 Wastewater Sampling

The plant's combined liquid effluent shall be proportionately sampled for uranium and measured for flow at the AREVA NP plant effluent monitoring station prior to discharge to the city sewer. Utilization of an appropriate grab sampling program is permissible on a short-term interim basis to cover instances when the proportional sampler is inoperable. Discharge data based on grab samples shall be so noted in the plant liquid effluent sampling records.

9.4 *Environmental Monitoring*

AREVA NP shall conduct a routine environmental surveillance program relative to operation of the Richland fuel fabrication facility. Surface environmental media and groundwater samples shall be collected from strategic locations in the surrounding environs and analyzed for pertinent constituents of concern.

9.4.1 Surface Environmental Sampling

Surface-level environmental media shall be sampled for uranium and fluorides per the following schedule. See Figure 9-1 for the location of the stations.

Sample Station	Media	Frequency	Analysis
1	Soil	Quarterly	Uranium
2	Soil	Quarterly	Uranium
3	Air	Monthly	Fluoride
4	Air	Monthly	Fluoride
5	Forage	Monthly*	Fluoride
6	Forage	Monthly*	Fluoride

* During the growing season only (April-October)

9.4.2 Groundwater Sampling

Samples of groundwater shall be collected as outlined below. The program design (sampled constituents and well locations) focuses on current facilities managing the plant's major process wastewater streams as well as the legacy surface impoundment (lagoon) system. See Figure 9-2 for locations of the groundwater monitoring wells.

Well Numbers	Frequency	Analysis
GM-1, GM-2, GM-5, GM-6, GM-7, GM-8, GM-10, GM-12	Semi-Annually	Gross alpha/beta, fluoride, nitrate, pH

9.4.3 Sanitary Sewer Sludge Sampling

The release of radioactive materials to the municipal sewer system shall be controlled and sampled as described in Section 9.3. In addition to the effluent monitoring described therein, AREVA NP shall conduct a routine sampling program for the radioactive content of sewage sludge produced at the City of Richland sewage treatment plant. Samples of the sludge in its dewatered (semi-dry) state, as subsequently landfilled, shall be collected quarterly and analyzed for uranium and moisture content. The uranium-based activity in picocuries per gram of sewage sludge (as landfilled) shall be maintained below 30 pCi/gram for any confirmed single sample result and below 25 pCi/gram of sewage sludge as an annual running average. Results in excess of these values shall be cause for investigation, with a follow-up action plan as appropriate to restore compliance with limits.

Figure 9-1 Environmental Sampling Stations

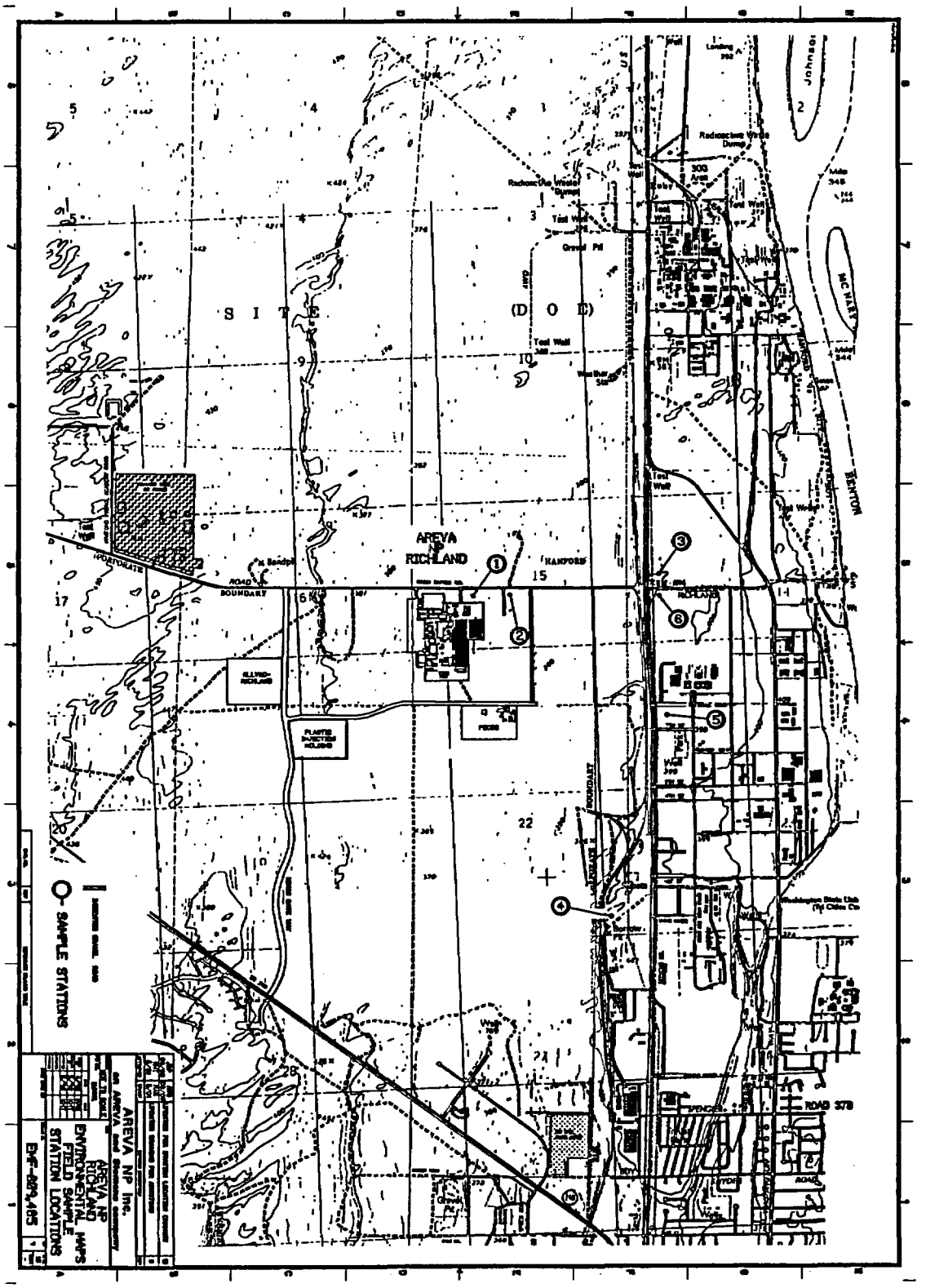
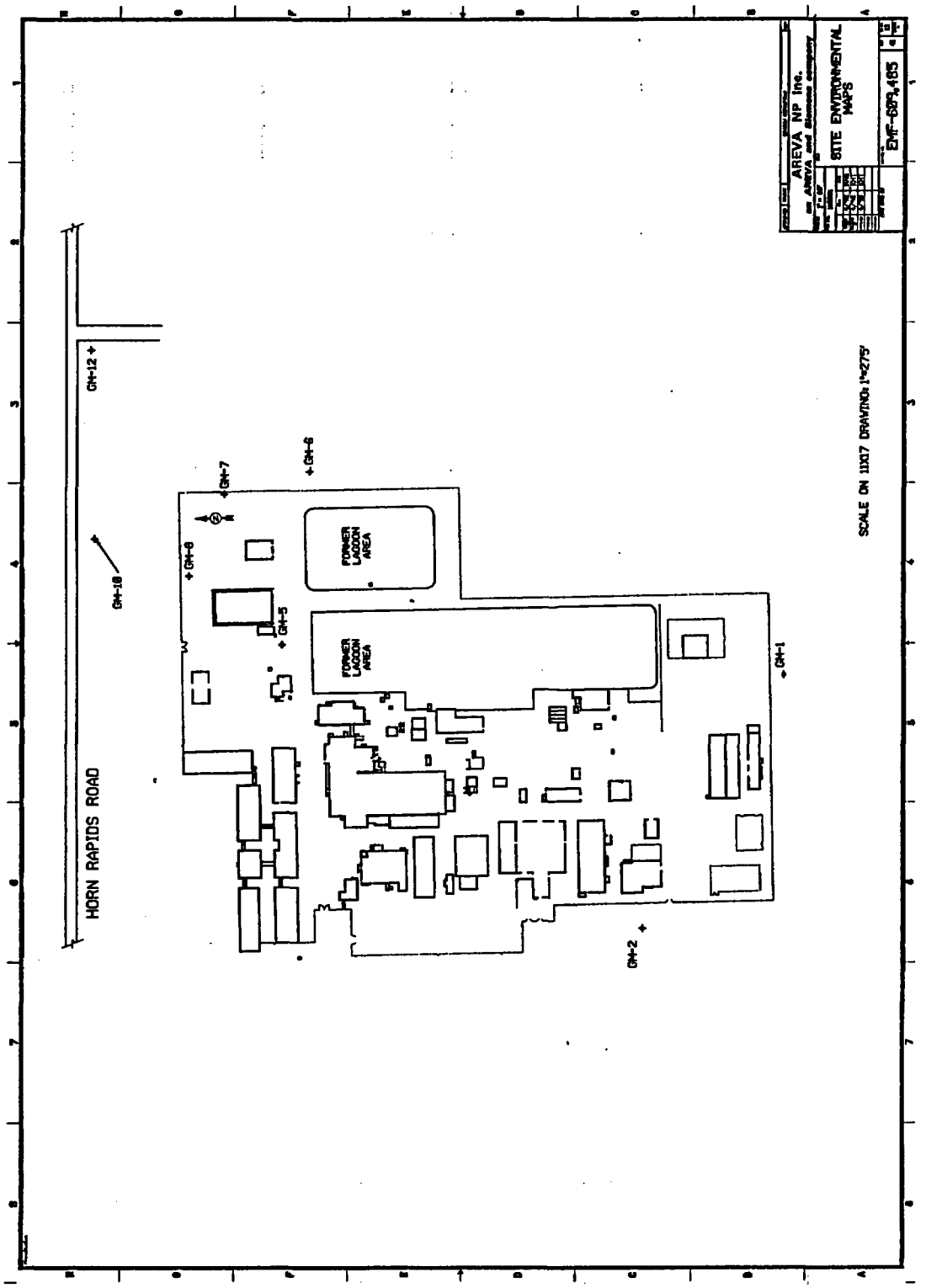


Figure 9-2 Groundwater Monitoring Wells



EHS&L Document

SNM-1227 - Chapter 10
Decommissioning

Nature of Changes

Item	Paragraph	Description	Justification
1.	New Document	New Document	License Renewal
2.			
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Change Author	LJ Maas	☒	Author	☒
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Components		☐	Mgr, Component Fabrication[1]	☐
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Safety/Security		☐		
Emergency Preparedness		☐		
MC&A	LJ Maas	☒	Mgr, Licensing & Compliance[2]	☐
Transportation		☐		
Environmental		☐		
BWR Product Eng. Review		☐	Mgr, BWR Product Engineering	☐
Quality Review		☐	Mgr, Quality	☐
Ops. Projects & Planning Review		☐	Mgr, Ops. Projects & Planning	☐
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EHS&L Change Impact Evaluation Form		
Document / ECN No*.: E10-08-010		Change Evaluator: RK Burklin
Does the change potentially impact Criticality Alarm System (CAS) coverage?	☐ Yes ☒ No	If yes, explain:
NRC Pre-Approval Evaluation:		
Is NRC Pre-approval (License Amendment) Needed? (Based on "Yes" answer to any of five questions below). (Based on "No" answer to all five questions below).	☒ Yes ☐ No	This is part of our NRC license renewal.
1. Does the change create new types of accident sequences that, unless mitigated or prevented, would exceed the performance requirements of 10 CFR 70.61 (create high or intermediate consequence events) and that have not previously been described in AREVA NP Inc's ISA Summary?	☐ Yes ☒ No	If yes, explain:
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Application Date:	NRC Docket No. 70-1257
October 27, 2006	

10.0 Decommissioning

AREVA NP Inc. (AREVA NP) shall fulfill the requirements of 10 CFR 70.25, "Financial Assurance and Recordkeeping for Decommissioning," by submittal of a decommissioning funding plan (DFP) to the U.S. Nuclear Regulatory Commission (NRC) and by the maintenance of records of information important to the decommissioning of the site until such time as the site is released for unrestricted use.

10.1 *Decommissioning Funding Plan (DFP)*

The DFP submitted to the NRC shall contain the elements called for in 10 CFR 70.25(e), including a decommissioning cost estimate; a financial assurance mechanism; a means for adjusting the cost estimate and associated funding levels periodically over the life of the facility; and a certification that financial assurance has been provided in an amount that covers the current estimate for decommissioning. The DFP will address the decommissioning of facilities on the Richland, Washington site with potential for contamination with licensed materials.

10.2 *Decommissioning Cost Estimate*

The decommissioning cost estimate shall be presented in the DFP. The estimate shall include a facility description, estimated decommissioning costs (labor, non-labor, and contingency), and key underlying assumptions. Consistent with 10 CFR 70.25(e), the cost estimate will be reviewed and updated at an interval not to exceed three years.

10.3 *Decommissioning Financial Assurance*

Financial assurance for decommissioning will be provided by a methodology authorized in 10 CFR 70.25(f). In conjunction with submittal of the DFP, AREVA NP will provide the NRC with signed originals of the financial instruments obtained to satisfy the requirements of 10 CFR 70.25(f).

EHS&L Document

SNM-1227 - Chapter 11
Management Measures

Nature of Changes

Item	Paragraph	Description	Justification
1.	New Document	New Document	License Renewal
2.			
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Minor Changes: If the proposed changes are limited to editorial and/or administrative changes check the box at the right. The document will be routed directly for review by EHS&L without technical review. All applicable approvals must still be obtained. ☐

Document Reviews			Document Approvals	
Purpose/Function of Review	Specify Reviewer(s) (Optional except for change author)	(Check all that apply)	Title of Approver	(Check all that Apply)
Document Control (Automatic)		☒	Document Control (Automatic)	☒
Change Author	LJ Maas	☒	Author	☒
Independent Technical Review	RE Link	☒		
Operability Review(s)			Mgr, Richland Operations[1]	☐
Conversion		☐	Mgr, Uranium Conversion & Recovery Operations[1]	☐
Recovery		☐		
Ceramics		☐	Mgr, Ceramic Operations[1]	☐
Rods		☐	Mgr, Rods & Bundles[1]	☐
Bundles		☐		
Transportation		☐		
Components		☐	Mgr, Component Fabrication[1]	☐
Maintenance Review		☐	Mgr, Maintenance[1]	☐
Lab Review		☐	Mgr, Analytical Services[1]	☐
EHS&L Review(s)			Mgr, EHS&L[2]	☒
Criticality		☐	Mgr, Criticality Safety[2]	☐
Radiation Protection		☐	Mgr, Safety, Security & Emergency Preparedness[2]	☐
Safety/Security		☐		
Emergency Preparedness		☐		
MC&A	LJ Maas	☒	Mgr, Licensing & Compliance[2]	☐
Transportation		☐		
Environmental		☐		
BWR Product Eng. Review		☐	Mgr, BWR Product Engineering	☐
Quality Review		☐	Mgr, Quality	☐
Ops. Projects & Planning Review		☐	Mgr, Ops. Projects & Planning	☐
Others:		☐	Mgr, Richland Site/Other	☐
Training & Employee Dev.:[3]		☐	Training & Employee Dev.	☐

[1]Note: If approvals include 2 or more product center managers, the Operations manager can be substituted for the applicable product center managers.

[2]Note: If approvals include 2 or more EHS&L functional managers, the EHS&L manager can be substituted for the applicable EHS&L functional managers.

[3]Note: Training department review is required for all procedures that require or affect a Learning Plan and if additional training materials or curriculum must be revised before issuing procedure.

EHS&L Change Impact Evaluation Form		
Document / ECN No*.: E10-08-011	Change Evaluator: RE Link	
Does the change potentially impact Criticality Alarm System (CAS) coverage?	☐ Yes ☒ No	If yes, explain:
NRC Pre-Approval Evaluation:		
Is NRC Pre-approval (License Amendment) Needed? (Based on "Yes" answer to any of five questions below). (Based on "No" answer to all five questions below).	☒ Yes ☐ No	This is part of our NRC license renewal.
1. Does the change create new types of accident sequences that, unless mitigated or prevented, would exceed the performance requirements of 10 CFR 70.61 (create high or intermediate consequence events) and that have not previously been described in AREVA NP Inc's ISA Summary?	☐ Yes ☒ No	If yes, explain:
2. Does the change use new processes, technologies, or control systems for which AREVA NP Inc. has no prior experience?	☐ Yes ☒ No	If yes, explain:
3. Does the change remove, without at least an equivalent replacement of the safety function, an item relied on for safety that is listed in the ISA Summary?	☐ Yes ☒ No	If yes, explain:
4. Does the change alter any item relied on for safety, listed in the ISA Summary, that is the sole item preventing or mitigating an accident sequence of high or intermediate consequences?	☐ Yes ☒ No	If yes, explain:
5. Does the change qualify as a change specifically prohibited by NRC regulation, order or license condition?	☐ Yes ☒ No	If yes, explain:
Actions Required Prior to or Concurrent with Change Implementation Evaluation:		
Action		**Explanation**
6. Modification / Addition to CAS system or system coverage documentation	☐ Yes ☒ No	If yes, explain:
7. Acquire NRC pre-approval (license amendment)	☒ Yes ☐ No	If yes, explain: License renewal.
8. Conduct/modify ISA	☐ Yes ☒ No	If yes, explain:
9. ISA Database Modification	☐ Yes ☒ No	If yes, explain:
10. Modification of other safety program information / underlying analyses (PHA, RHA, FHA, NCSA, etc.)	☐ Yes ☒ No	If yes, explain:
Actions required subsequent to Change Implementation Evaluation:		
11. Update safety program information (PHA,RHA,FHA,NCSA, P&ID)	☐ Yes ☒ No	If yes, explain:

* **If this form exists as a part of a document, the document number is not required.**

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11.0 Management Measures

Management measures are applied to items relied on for safety (IROFS) to provide reasonable assurance that the IROFS are available and able to perform their functions when needed. AREVA NP's ISA Summary [see Chapter 3, Integrated Safety Analysis (ISA) and ISA Summary] identifies IROFS applied to plant operating systems to assure that those systems function within the performance criteria of 10 CFR 70.61. Those IROFS may be engineered features (passive or active) or administrative features (actions of people). Management measures are applied to IROFS as described in Section 11.8 of this chapter.

11.1 *Configuration Management (CM)*

To ensure that configuration changes do not adversely impact currently implemented IROFS or add a new process or system which would pose an unacceptable risk, a formal review process will be used to assess new systems and components, or to modify existing systems or components. The CM function captures formal documentation governing the design and continued modification of the site structures, processes, systems, equipment, components, computer programs, personnel activities, and supporting management measures. The process for evaluating safety of configuration changes is discussed in Chapter 3.

11.1.1 CM Policy

It is management's policy to control facilities and processes so that the safety basis is maintained and that changes to facilities and processes are evaluated according to approved written procedures and consistent with 10 CFR 70.72. The CM process provides assurance that consistency is established and maintained between facility design, operational requirements, physical configuration, and facility documentation.

CM is applied uniformly to IROFS regardless of safety significance, i.e., a graded approach is not used.

Overall responsibility for configuration management resides with the Richland Site Manager. Key responsibilities within the configuration management program are delegated as follows:

- Plant Projects Component - overall ownership of the plant's configuration management system for facilities, equipment, and software. Includes control of plant projects (additions and modifications) from design to operational turnover.
- Environmental, Health, Safety and Licensing (EHS&L) Component - evaluation of safety and licensing impacts of plant additions/modifications.
- Fuel Operations (and Maintenance) Component - assurance that operational procedures and operator training are consistent with the equipment, process, and safety bases of the plant and that these bases are maintained over the course of maintenance activities.

The organizational relationships of these components are depicted in Chapter 2, Organization and Administration.

11.1.2 Configuration Control

AREVA shall maintain a configuration control program to ensure that proper reviews are undertaken prior to and after changes to facilities, equipment, or software systems, particularly with regards to facilities, equipment, and software involved in the handling or processing of licensed material. The reviews will ensure that facility changes are properly evaluated with respect to:

- Impact on safety and health or the control of licensed material, and
- Impacts to the pre-existing integrated safety analysis, integrated safety analysis summary, or other safety program information developed in accordance with 10 CFR 70.62.

The evaluation determines, before the change is implemented, if an amendment to the site's NRC license is required to be submitted in accordance with 10 CFR 70.72(c).

The configuration control program shall be implemented via approved procedures to which appropriate personnel shall be trained. Those procedures shall define the overall process for change control, including as appropriate:

- establishment of technical design bases/criteria;
- design development, review, and control;
- project approval, initiation, and control;
- evaluation of safety/licensing implications, including the need for new or revised safety analyses;
- needs for post-modification testing;
- project readiness review/startup approval, operational turnover, and closeout (control points applicable to the projects, safety, and operations organizations are identified and documented); and
- maintaining/updating of safety basis documentation.

Relevant procedures and associated documentation shall be readily available for review onsite. Procedures are subject to the controls set forth in Section 11.4.

11.1.3 CM Program Review

The effectiveness of the CM Program shall be assessed on a periodic basis in accordance with 11.5.2, Assessments.

11.2 *Maintenance*

Management will establish and maintain maintenance programs designed to ensure the availability and proper performance of features essential to the safe operation of the Richland facility.

11.2.1 Corrective Maintenance

Corrective maintenance will be conducted as required to ensure that IROFS and other systems or features necessary to the safe operation of the facility are properly repaired and restored to service in a manner that maintains facility safety and the function of the safety system/feature.

Corrective maintenance shall be authorized, initiated, and documented via a formally established procedure to which appropriate personnel have been trained. The procedure shall include steps to invoke appropriate coordination between the maintenance and operating organizations.

The corrective maintenance procedure shall include an evaluation step to determine if IROFS have been, or may be, affected by the equipment failure/malfunction or the ensuing maintenance. This evaluation step, performed using skill of the maintenance craft and operator, shall give appropriate consideration to the need for post-maintenance functional testing of IROFS.

11.2.2 Preventive Maintenance (PM)

AREVA conducts a preventive maintenance program covering equipment, facilities, systems and support activities with emphasis on safety items, including designated IROFS. The Richland program includes two components - preventive maintenance on equipment other than instruments (PM program) and instrument repetitive maintenance (IRM program). The PM program includes routine functional testing of IROFS.

PMs/IRMs for safety-related features, including IROFS, shall be managed within formal programs for scheduling, initiation, tracking, and documentation. These programs shall be governed by approved procedures.

IRMs and PMs shall be conducted by qualified personnel in accordance with written and approved instructions.

Safety-related IRMs/PMs shall be designated as such within the preventive maintenance management system. These IRMs/PMs are identified by a unique numbering system as well as appropriate precautionary statements on their respective data sheets.

Frequencies for safety-related IRMs/PMs are established with, and cannot be modified without, concurrence of the Environmental, Health, Safety and Licensing (EHS&L) function. Similarly, due dates for safety-related IRMs/PMs cannot be extended without concurrence of the EHS&L function.

Safety-related items, including IROFS, found to be non-functional or out-of-tolerance shall result in timely notification of the safety organization.

11.2.3 Functional Testing

AREVA will provide appropriate functional testing to assure that IROFS are available and able to perform their functions when needed. Functional testing may be provided in conjunction with corrective maintenance, preventive maintenance, or equipment installation/modification.

Corrective maintenance (see Section 11.2.1) will be conducted via a formally established and controlled procedure which shall include an evaluation step for post-maintenance functional testing.

Functional testing activities conducted under the preventive maintenance program (see Section 11.2.2) shall be specified within the applicable PM or IRM instructions.

Functional testing requirements associated with newly installed or modified safety-related equipment shall be specified and controlled via AREVA's established configuration control program (see Section 11.1.2).

11.2.4 Surveillance/Monitoring

AREVA will utilize established surveillance activities to monitor the current and long-term performance of IROFS. These activities include preventive maintenance and calibration (11.2.2), functional testing (11.2.3), and follow-up to corrective maintenance (11.2.1). IROFS found to be out-of-tolerance or unable to perform their intended function are reported in a timely manner to the safety function.

Failed IROFS are entered into AREVA's corrective action program (CAP). The CAP will provide management of the process for evaluation of the IROFS failure, cause identification, and identification/assignment of corrective actions.

Records of IROFS surveillance, performance, failures, and corrective actions will be maintained within the maintenance and corrective action programs, as applicable.

11.3 *Training and Qualification*

Workers shall be provided training to allow them to conduct their assigned activities with licensed materials in a manner that is protective of their personal health and safety and the health and safety of their co-workers. Furthermore, these activities must be conducted in a manner that does not endanger the public or the surrounding environment. This training typically falls into one of two categories, namely:

- general health and safety training not specific to particular workstations and work activities; and
- training to assure proper performance for particular positions and work activities that are relied on for safety, in particular those affecting measures designated as IROFS in the ISA summary.

11.3.1 General Health and Safety Training

General health and safety training shall be provided in radiation protection, criticality safety, emergency procedures, fire safety, and chemical safety as it relates to the safety of licensed materials. The degree of training shall be commensurate with the workers' duties and their potential interactions with licensed materials. The minimum safety-related training requirements shall be established by the EHS&L component.

Training for radiation workers shall comply with the regulatory requirements of 10 CFR Parts 19 and 20. Specifics of the radiation protection training program are provided in Chapter 4, Radiation Protection.

Initial Training

Employees shall be provided with initial general health and safety training to allow them to safely access areas containing licensed materials and, if applicable, start on-the-job training. Prior to assignment to independent operation and as appropriate to their positions, employees are required to have been instructed in radiation protection, criticality safety, fire protection, emergency requirements, and chemical safety as it relates to the safety of licensed materials. This training shall be documented and records appropriately maintained.

Recurrent Training

Each employee routinely working with licensed materials shall receive annual refresher training (including an examination) as part of the facility's continuing program in radiation protection and

criticality safety awareness. When changes are made relative to radiation protection, criticality safety, or emergency response requirements, provisions shall be made to assure that affected employees are appropriately informed and instructed on the change. Recurrent and update training shall be documented and records appropriately maintained.

11.3.2 Training and Qualification for Positions/Activities Impacting IROFS

Employees assigned to positions/activities involving licensed materials shall be appropriately qualified and trained so as to conduct their job duties in a way that does not adversely impact safety, and in particular the availability and reliability of measures designated as IROFS in the ISA Summary. Qualification for selected positions is achieved and maintained through a combination of education/experience requirements, general employee training, initial qualification, continuing training, and requalification. The training and qualification process may include classroom and/or on-the-job activities and shall be documented.

Organization and Management of Training

The assurance of appropriately trained and qualified workers is the responsibility of the Richland Site Manager and pertinent line management. Support to line management for the development, implementation, and administration of plant training and qualification programs is provided by the Training function (see Chapter 2, Organization and Administration).

Overall implementation of the training and qualification program for employees conducting activities or impacting measures relied on for safety shall be governed by a formal procedure(s).

Training records shall be created and maintained so as to allow the verification of the training and qualification status of individuals potentially impacting the safety of licensed material operations.

Activities Requiring Training

Positions impacting the availability/reliability of IROFS shall be assessed to determine training and qualification requirements. The assessment is based on a graded approach that considers hazards and rules associated with the positions and utilizes input of pertinent subject matter experts (SMEs) as appropriate.

Training and qualification requirements for job functions shall be maintained to ensure consistency with current systems, procedures, and policies.

Position Training Requirements

Position training requirements are established for positions whose incumbents conduct activities relied on for safety or who perform actions that prevent or mitigate accident sequences described in the ISA Summary. These requirements include, as applicable, procedure reviews, classroom instruction, and on-the-job training (OJT).

Bases for Training

The objective of training shall be to ensure safe and efficient operation of the facility and compliance with applicable established regulations and requirements. Learning objectives shall be established for those positions/activities impacting the safety of licensed material operations, and in particular the availability/reliability of designated IROFS. Objectives include, as applicable, the knowledge, skills, and abilities the trainee should demonstrate; the conditions under which required actions will take place; and the standards of performance the trainee should achieve upon completion of the training activity.

Training Materials

Training materials shall be appropriately formalized to assure the consistent conduct of training and should be based on learning objectives derived from training needs assessments of the safety/IROFS-related activities. Materials may take the form of instructor guides, lesson plans, or similar training tools and shall be subject to established review and approval criteria. Computer-based training may also be utilized and shall be subject to a review and approval cycle consistent with that provided to other training materials.

Evaluation of Trainee Accomplishment

Trainee understanding and command of learning objectives shall be evaluated. The evaluation may be accomplished through a combination of observation/skills demonstration, written tests, or oral interviews. The results of trainee evaluations shall be documented.

On-The-Job Training (OJT)

OJT requirements for activities relied on for safety and listed in the ISA Summary, if applicable, shall be specified as part of pertinent position training requirements. Completion of OJT may be demonstrated by actual task performance (preferred) or task simulation. Completion of OJT requirements shall be documented.

Continued Assurance of Training/Qualification

Positions/activities impacting IROFS shall be evaluated for needed recurrent training/ qualification activities. Any such periodic retraining/requalification shall be provided as part of the formal training/qualification program and shall be documented.

Training Program Review

The effectiveness of the training and qualification program for positions impacting IROFS shall be assessed on a periodic basis (see 11.5.2, Assessments).

11.4 *Procedures Development and Implementation*

AREVA conducts its licensed activities in accordance with a system of written operating procedures. Activities involving licensed SNM and/or IROFS will be conducted in accordance with approved procedures.

Procedures may take various forms [standard operating procedures (SOPs), standard work instructions (SWIs), management control procedures (MCPs), maintenance instructions (PMs, IRMs)] and are subject to formal review and approval protocols. Safety-related procedures, and specifically those involving IROFS, are subject to formal review and approval by the EHS&L function, including when those procedures are revised.

The site process for the development and implementation of safety-related procedures shall be controlled via formal administrative/management control procedures. Those procedures shall address how procedures are developed, reviewed, approved, distributed, revised, and deleted. The system shall assure that the most current revisions of procedures are readily available to workers within their work areas.

The process for creating new and revising existing safety-related procedures, specifically those procedures involving IROFS-related activities, shall include a mechanism for triggering necessary training and/or qualification updates.

Revisions to procedures covering licensed material operations shall be evaluated in accordance with 10 CFR 70.72 as to their potential impacts to IROFS and the site integrated safety analysis, including the need for NRC pre-approval of the change.

Temporary changes to procedures, if allowed, shall be governed by a formal process, specifying the conditions under which temporary changes may be implemented and the associated review and approval requirements.

Procedures governing activities relied on for safety shall be subject to a formal program of periodic review with defined review frequencies. The review frequency may be graded based on importance to safety.

The issue investigation/corrective action program (see Section 11.6) shall include provisions to assess the role of procedures in adverse conditions/events evaluated within the program. Correction of procedural deficiencies shall be tracked to completion within the system

11.5 *Audits and Assessments*

AREVA will implement and maintain a program of audits and assessments of activities significant to facility safety and environmental protection.

Overall responsibility for the audit and assessment program rests with the manager of the Environmental, Health, Safety and Licensing function, who reports to the Richland Site Manager independently of the operations function (see Chapter 2, Organization and Administration). That overall responsibility includes:

- determining the appropriate utilization of internal and/or external personnel for particular audit/assessment activities;
- assuring that audit/assessment personnel have expertise and backgrounds sufficient to successfully conduct their assigned audit/assessment activities, and in the case of assessments, are sufficiently independent of the area being assessed.
- assuring the utilization of an effective corrective action system to address the findings of audits and assessments.

11.5.1 Audits

Audits are compliance-based evaluation activities with an objective of verifying the compliance of operations with regulatory requirements and license commitments. The audit program will apply as a minimum to radiation protection, nuclear criticality safety, fire protection, environmental protection, and hazardous chemical safety as it relates to maintaining the safety of licensed material operations.

Audits of activities significant to facility safety and environmental protection are typically conducted by personnel within the EHS&L function and as such, organizationally independent of the activities being audited. Full independence may not be practicable for the auditing of activities actually performed by EHS&L personnel, e.g., performance of contamination surveys. In these cases, the auditor will be designated such that he was not directly involved in the performance of the activities being audited.

Audits shall be conducted in accordance with written guidance (e.g., audit plans or checklists) and results shall be documented.

Audit reports shall be directed to appropriate EHS&L management and to affected operational management. Records of audits shall be maintained in accordance with the Records Management section of this chapter.

Audit findings indicating non-compliance with regulatory and/or license requirements shall be entered into the corrective action program for appropriate documentation, evaluation, and corrective action. The AREVA corrective action program is described under Incident Investigation and Corrective Action (Section 11.6).

Audits shall be conducted in accordance with Table 11-1, Schedule of Audits.

11.5.2 Assessments

Assessments are performance-based evaluation activities conducted to access the effectiveness of health, safety and environmental compliance functions in achieving their designed purposes, particularly in providing reasonable assurance of the continued availability and reliability of IROFS.

Assessments will utilize personnel who are independent of the area being reviewed, however some utilization of individuals from the function being assessed may be required to ensure adequate knowledge of the area under review.

Assessments shall be conducted in accordance with written guidance (e.g., assessment objectives or plans) and results shall be documented.

Assessment reports shall be directed to appropriate EHS&L management and to affected operational management. Records of assessments shall be maintained in accordance with the Records Management section of this chapter.

Assessment findings indicative of program deficiencies and/or vulnerabilities shall be entered into the corrective action program for appropriate documentation, evaluation, and corrective action. (Certain sensitive issues, e.g., security or safeguards-related issues, may be dispositioned outside the general corrective action program.) Recommendations for program improvements/enhancements may be entered into the corrective action program at the discretion of the manager of the EHS&L function.

The assessment program will be applied at a minimum to the areas of radiation protection, nuclear criticality safety, chemical safety, fire safety, environmental protection, emergency preparedness, configuration management, and training and qualification. Assessments in these areas shall be scheduled such that each area is assessed on at least a triennial basis. Actions to require interim assessments, to require more frequent assessments of any area, or to add other functional safety areas to the assessment program shall be at the discretion of the manager of the EHS&L function.

Table 11-1 Schedule of Audits

Audit	Frequency	Comments
Radiation Protection	Monthly	Plantwide inspection of radiation protection practices and exposure controls
Nuclear Criticality Safety	Monthly	Audit of various plant systems/facilities where fissile materials are processed or stored. Monthly inspections conducted such that each area of plant is audited at least biennially.
Fire Protection	Monthly	Inspections of selected areas of plant for housekeeping and industrial safety conditions pertinent to fire protection. In addition, monthly inspection of plant fire extinguishers, conducted in conjunction with or independently of the area inspections.
Environmental Protection (Radiological)	Quarterly	Inspection of environmental protection practices and exposure controls relative to licensed materials.
Hazardous Chemical Safety	Biannually	Inspections of areas of plant where hazardous chemicals which could affect licensed materials are stored, processed, or otherwise handled.

11.6 *Incident Investigation and Corrective Action*

AREVA will implement and maintain an integrated incident investigation/corrective action program to assure that safety-adverse incidents or conditions are appropriately identified, evaluated, and reported, and that suitable corrective actions are identified and applied. This integrated program will include incidents and adverse conditions involving the control and processing of licensed materials, including those with actual or potential adverse impacts to items relied on for safety (IROFS). The incident investigation and corrective action programs will be implemented via formally approved procedures.

11.6.1 Corrective Action Program

The Richland corrective action program (CAP) as applied to the safety of operations using or affecting licensed materials, including the effectiveness and reliability of IROFS, will contain the following features:

- a formal condition reporting mechanism readily available to all individuals involved in licensed material operations;
- provision for EHS&L function involvement in the safety significance screening of reported events/conditions;
- formal assignment of issue owner/issue investigator responsibilities;
- assignment of required level of incident investigation/cause analysis based on safety significance;

- formal assignment and tracking to completion of corrective actions;
- evaluation of regulatory, including NRC, reportability; and
- creation of permanent documentation of the issue identification, evaluation, and corrective action activities.

11.6.2 Issue Investigation and Causal Analysis

Level of issue investigation/causal analysis is driven by safety significance of the incident or condition, as determined under the corrective action program. Management discretion may be used to elevate the level of a particular issue. Response to low safety significance events may be limited to application of corrective actions as deemed necessary, without formal cause analysis and application of preventive actions. More significant safety-related incidents or conditions require formal investigation and cause analysis as dictated by an approved issue investigation/causal analysis procedure. This procedure defines:

- responsibilities to issue owners and investigators;
- general methods for conduct of investigations;
- requirements relative to identification of cause and generic implications;
- requirements for report preparation and approval, and for records maintenance.

Issue investigations/cause analyses of safety-related incidents and conditions require review/approval by the EHS&L function. Identified corrective actions are assigned and tracked via the formal CAP.

11.7 *Records Management*

AREVA will establish pertinent controls to assure that records documenting its health, safety and environmental activities and the safety bases/controls of its facilities and processes are appropriately created, distributed, stored, protected, and, if necessary, restored.

Records will be identified and stored such that current records are readily accessible for use and historic records can be retrieved within a reasonable timeframe. Records that have been reconstructed due to inadvertent loss or destruction of the original record shall be identified as such.

Records documenting plant alterations or additions, abnormal occurrences involving licensed materials, events associated with radioactive releases, criticality safety analyses, audits, assessments, safety-related instrument calibrations and preventive maintenance, ALARA program results, worker training and retraining, personnel exposures, routine radiation surveys, environmental surveys, decommissioning plans and activities, emergency preparedness events/drills, and IROFS and/or management measures degradations or failures resulting in non-compliance with the performance requirements of 10 CFR 70.61 shall be maintained on file for a minimum of two years or as otherwise required by federal regulation or other license condition.

11.8 *Quality Assurance for IROFS*

Quality assurance (QA) elements are applied to IROFS as management measures to assure that there is reasonable assurance that IROFS are available and able to perform their functions when needed. The QA elements are applied to IROFS based on IROFS type as identified in

Table 11-2. Commitments relative to the implementation of these QA elements are described elsewhere in this chapter.

Table 11-2 Management Measures for IROFS

Management Measure	Passive Engineered Control (PEC)	Active Engineered Control (AEC)	Administrative Control
Configuration Management (design & change control for hardware & software)	X	X	
Document Control (includes records management and procedures)	X	X	X
Corrective Maintenance	X	X	
Preventive Maintenance (PM/IRM and Surveillance Monitoring)	X	X	
Functional Testing (initial, periodic, following maintenance)	X	X	
Personnel Training (initial, periodic, following changes) & Evaluation (testing, observation)			X
Problem Identification & Corrective Action Follow-up	X	X	X

The management organization responsible for the application of QA elements to IROFS is as described in Chapter 2, Organization and Administration.

Additional QA elements/management measures may be applied to IROFS based on the specific operational/maintenance characteristics of the individual IROFS. If so, the measures shall be applied in accordance with written instructions and the results documented and appropriately retained.