DEVELOPMENT OF ACCIDENT SCENARIOS FOR THE DECOMMISSIONING OF THE FORT ST. VRAIN NUCLEAR GENERATING STATION

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ABSTRACT

A set of six accident scenarios was developed for the decommissioning of the Fort St. Vrain Nuclear Generating Station. These scenarios took into consideration both internal events, such as fires and heavy load drops, as well as external events, such as tornadoes. The resultant potential accident doses were quite small and well within applicable NRC and EPA regulations.

SCENARIO DEVELOPMENT

The Fort St. Vrain decommissioning project is divided into three major work areas:

- Decontamination and dismantlement of the Prestressed Concrete Reactor Vessel (PCRV).
- Decontamination and dismantlement of the contaminated balance of plant systems.
- Site cleanup and final site radiation survey.

Key elements of the decommissioning plan include techniques used to remove internal components such as activated graphite blocks and steam generators from the PCRV. Additionally large amounts of activated concrete will be removed from the PCRV structure. The technique approach is based on filling the PCRV with water for shielding while internal components are removed. Diamond-wire cutting will be used to remove large sections of activated concrete. The accident scenarios described reflect these unique techniques.

The activities, equipment, and circumstances associated with decommissioning are markedly different from those evaluated in the Fort St. Vrain Final Safety Analysis Report for power operations and refueling (1). The risk of accidents during decommissioning is significantly less than during power operations. This is primarily due to the reduced radiological source term as the reactor will be defueled prior to decommissioning. [The activity levels of various components used in this analysis were based on a neutron activation analysis (2).] Six accident scenarios were developed. These were:

- Dropping of Contaminated Concrete Rubble
- Heavy Load Drop
- Fire
- Loss of PCRV Shielding Water
- Loss of Power
- Tornado

The following subsections describe each of these scenarios.

Dropping of Contaminated Concrete Rubble

After the majority of the PCRV top head concrete is removed in large pieces by diamond-wire cutting, the last 0.15 m. (6 in.) just above the PCRV top head liner will be removed by a mechanical breaker. This scenario assumed that radioactivity is released from the drop of a rubble transport container due to a faulty jib crane or operator error. It was conservatively assumed that 10 percent of the 0.15 m. (6 in.) thick concrete segment is involved in the accident. This is approximately 3 400 kg. (7 500 lbs.)of concrete. The primary radionuclides in the concrete were Fe-55, Co-60, tritium, and Eu-154. It was assumed that of the total releasable activity, 40 percent was due to tritium and 60 percent was due to Eu-154 which has the highest dose conversion factor of the remaining nuclides.

No credit was taken for particulate filtration by the reactor building ventilation system, nor was any credit taken for an elevated release over 30 m. (100 ft.) above grade. The major exposure pathway was air inhalation by an adult standing at the decommissioning Emergency Planning Zone (EPZ), some 100 m. (328 ft.) from the Reactor Building, for a period of 2 hours. A worst case atmospheric dispersion factor of 3.53E(-2) sec/m^3 was calculated at 100 m. (328 ft.) based on Regulatory Guide 1.145. The whole body and bone doses to an adult standing at the EPZ were calculated to be 0.049 mSv. (4.92 mRem) and 0.547 mSv. (54.7 mRem), respectively. The major contributor to both the whole body dose and bone dose was Eu-154.

Heavy Load Drop

The dismantling of the PCRV will be accomplished with the aid of three types of hoist systems, including the Reactor Building main bridge crane rated at 154 220Kg. (170 ton), the auxiliary 15 900 kg. (17.5 ton) hoist on the bridge crane, and three 1 360 kg. (1.5 ton) jib cranes on the refueling floor. There will be many heavy loads removed during decommissioning. These lifts include:

- Large graphite reflector blocks
- Large concrete sections
- Twelve steam generators
- Helium diffusers
- Concrete core support floor

A heavy load drop accident is a relatively low-probability event; a failure of the hoisting cable could cause a drop of the load, the generic probability of this happening is on the order of 1.0E-5 to 1.0E-6 per demand (hoist lift). The loss of the crane cable brakes could be due to mechanical failure, operator error, or an incorrect maintenance operation. Since the Fort St. Vrain Reactor Building bridge crane does not qualify as a single-failure-proof crane in accordance with NUREG-0544 (4) guidelines, the loss of crane brakes is postulated as a credible failure mode.

The accident scenarios developed for heavy load drops at nuclear power plants consider the dropping of a heavy load, e.g., fuel shipping cask onto a very large radionuclide inventory, such as spent fuel (3). In the case of Fort St. Vrain, all fuel will have been removed from the Reactor Building prior to decommissioning, therefore, the full spectrum of heavy load drop accidents is much less severe than at an operating nuclear power plant.

The most severe heavy load drop accident was postulated to consist of dropping the component containing the largest inventory of dispersible radioactive material. While each of the 12 steam generators was estimated to have 1.75E + 13 Bq. (473 Ci.) associated with it most of this activity is activation products which are tightly bound in the steam generator and not dispersible upon impact. The largest dispersible radioactive inventory is associated with the 240 large graphite side reflector blocks. This activity was estimated to be 5.46E + 13 Bq. (1 477 Ci.) per block, with the major contributors to the activity being Fe-55, Co-60, and tritium.

The drop of a heavy load onto a highly radioactive component was evaluated and determined not to represent the worst case scenario. For instance , the dropping of a large graphite side reflector block back into the PCRV might crush portions of adjacent graphite reflector blocks. However, since all highly radioactive components are submerged until they are ready for processing, the debris and its attendant activity would remain in the water. This activity would be cleaned by in the PCRV water cleanup and clarification system.

For this accident scenario it was postulated that the Reactor Building bridge crane is hoisting 1 of the 240 large graphite side reflector blocks. It is currently planned to section these reflector blocks into smaller pieces for packaging in low specific activity (LSA) containers. However, it was conservatively assumed that a single reflector block could be transported intact in its own shipping container.

After appropriate drying and radiation surveys, the container with the unsectioned side reflector block is lowered approximately 30 m. (100 ft.) through an enlarged equipment hatch to the truck loading bay. It is assumed that the failure occurs during this lowering. The free falling container ruptures and spills its contents on the truck-loading bay floor.

Administrative controls will be in place that will prevent the tractor of the tractor trailer unit from being in the loading bay during lowering operations and will ensure that all the truck loading bay doors are closed and the Reactor Building filtered ventilation exhaust system is in operation. It was conservatively assumed that one percent of the activity of a single large reflector block is dispersed from the drop. The dust escapes the immediate area and exits through the Reactor Building ventilation stack. Credit was taken for the decontamination afforded by the Reactor Building ventilation system. No credit was taken for an elevated stack release.

The whole body and lung doses to an adult standing at the EPZ were calculated to be 0.046 mSv. (4.66 mRem) and 1.33 mSv. (133 mRem), respectively. These doses are due primarily from Fe-55 and Co-60 generated from activation of impurities in the graphite.

Fire

During decommissioning there are many possible fire initiators that could result in a release of radioactive materials. These include:

- Fires started from cutting torches
- Contamination control tent fires
- Fires associated with waste processing activities on the refueling deck level
- Electrical fires

The most likely fire initiator has been determined to be a cable tray fire started from a spark during PCRV tendon cutting operations. The fire would be quickly extinguished by the fire watch on duty for the tendon cutting operations. The radiological consequences of this accident would be negligible as the cable trays contain virtually no radioactive contamination.

The postulated worst case fire scenario involves a fire enveloping LSA waste containers. The greatest exposure for a fire is during the six month period when the highly radioactive large graphite side reflector blocks and side spacer blocks are being removed from the PCRV.

It is postulated that a tractor trailer begins to transport packaged waste from the Reactor Building truck loading bay to an off-site burial ground/processing facility. The shipment is assumed to contain 230 graphite side spacer blocks with their boron pins removed which are contained in 5 LSA containers. There are 1 152 side spacer blocks to be removed during decommissioning. It is postulated that an engine fire develops and spreads to the tractor's diesel fuel tanks. The tanks may contain a combined capacity of up to 1.13 $m.^3$ (300 gallons) of fuel. The fuel tanks are assumed to rupture from the heat and engulf the entire tractor trailer and the side spacer blocks in a diesel fuel pool fire.

A fire involving 1.13 $m.^3$ of diesel fuel spilled onto a relatively flat surface will burn itself out within 30 minutes. The resultant fire temperature will be bounded by the ASTM-E119 standard fire curve (5). Most of the graphite will be exposed to temperatures well below the fire temperatures due to insulation provided by adjacent graphite blocks and some protection afforded by the LSA shipping containers. Under these conditions, it was assumed that 50 percent of the graphite inventory on the shipping trailer is oxidized during the 30 minute fire. It was conservatively assumed that all of the tritium in the oxidized fraction (50 percent of the total tritium inventory) is released. In addition to tritium release, it was assumed that 0.015 percent of the balance of the radionuclide inventory is released in the form of particulates (6).

The fire was assumed to occur at ground level immediately outside of the Reactor Building truck-loading bay. The radioactive inventory for the 230 graphite side spacer blocks is estimated to be 1.37E + 14 Bq. (3 706 Ci.). This inventory consists of 1.31E + 14 Bq. (3556 Ci.) of Fe-55, 4.51E + 12 Bq. (122 Ci.) of tritium, and 1.03E + 12 Bq. (28 Ci.) of Co-60.

The whole body and lung doses to an adult standing at the EPZ were calculated to be 1.21 mSv.(121 mRem) and 2.15 mSv. (215 mRem), respectively. This was the largest accident dose of the six accident scenarios.

Loss of Pre-Stressed Concrete Reactor Vessel (PCRV) Shielding Water

During a portion of the Fort St. Vrain decommissioning, the PCRV cavity will be flooded with water for shielding. This water will be circulated and purified by the PCRV Water Cleanup and Clarification system to gradually decrease the radioactivity in the water. This system will be in operation when the PCRV internals are removed. This accident scenario assumes that there is a leak or rupture of the PCRV Water Cleanup and Clarification system piping resulting in a liquid release. Tritiated water with dissolved iron and colbalt would be spilled into the Reactor Building sump and keyway. Assuming the worst case of emptying the whole PCRV cavity, 1 603 m.^{3} (423 500 gallons) fills the Reactor Building sump/keyway and floods the Reactor Building basement floor to a depth of 0.60m. (2 ft.). This water will be 14.9 m. (49 ft.) below grade and contained by the Reactor Building sump/keyway and basement walls. No credit was taken for the Reactor Building ventilation system.

Since the non-gaseous activity will be retained in the spilled water, tritiated water (HT0) released through evaporation is the only significant activity. This would be evaporated from the surface area of the spilled water in the Reactor Building basement. The reactor building is approximately 36.6 m. (120 ft.) long and 23.2 m. (76 ft.) wide providing a surface area for the spilled water of 848 m.^{2} (9 120 ft.^{2}). The best fit evaporation rate at 70 percent relative humidity and ventilation flow of 1 m./sec. is 0.046 gm./m.^{2}/sec. (0.00109 pounds/ft.^{2}/sec.) (7). It is predicted that tritium levels in the PCRV will be less than 3.7E + 13 Bq. (1 000 Ci.). However, for this analysis, it was assumed that the theoretical maximum amount of tritium is transferred to the PCRV shielding water from the graphite blocks. The maximum tritium concentration in the spilled water was calculated to be 6.24E + 5 Bq. /1cc. (62.4 uCi./cc.). The resultant tritium release rate is about 2.5E + 7 Bq./sec (2.5 mCi./sec.) over the 848 m.^{2} (9 120 ft.^{2}) of water surface area.

Since the dose conversion factor for tritium is the same for whole body and lung doses, the dose to an adult standing at the EPZ was calculated to be 0.348 mSv. (34.8 mRem) for a 2 hour period.

Loss of Power

During plant decommissioning, electric power will normally be supplied by off-site sources. No backup power is assumed available during a loss-of-power event. The primary equipment using electric power during decommissioning will be:

* Pumps:
  Deionized water
  Fire water
  Service water
  Water treatment
  PCRV Water Cleanup and Clarification
* Cranes
* PCRV work platform
* Lighting
* Demolition tools:
  Plasma arc torch
  Diamond-wire cutter
  Water jet cutter
  Drills
* Mobile laundry
* HVAC systems

This scenario postulates the loss of off-site power due to a weather-related event. Such an event could be downed power lines due to strong winds or heavy icing conditions. The likelihood of this occurrence is remote since off-site power can be supplied to the site by means of two redundant transformers through six separate 230 KV lines. Decommissioning activities would cease until local power is restored. The postulated accident scenario is the loss-of-power to the HVAC system while a large graphite side reflector block has been removed from the PCRV for cutting. These graphite blocks are grappled and hoisted by a jib crane to a refueling floor work station. At the work station a block is cut into sections in preparation for packaging into LSA containers. The loss-of-power is assumed to occur after the cutting/cleaving operation.

It was conservatively assumed that these processing operations release 1.5 percent of the total activity of a single large side reflector block. It was estimated that the combination of radiological controls in place at the work station (confinement tenting) and the confinement function provided by the Reactor Building itself will result in retention of 99 percent of the particulate (Fe-55 and Co-60) kerfing debris. It was assumed that 1 percent of the Fe-55 and Co-60 kerfing debris is released and 100 percent of the tritium in the kerfing debris is released. No credit was taken for an elevated release. The total activity of a large graphite side reflector block was estimated at 5.46E + 13 Bq. (1 477 Ci.). Using the assumption provided above, this results in a release of 8.14E + 9 Bq. (0.22 Ci.). These activities are based on a 3-year decay period.

The whole body and lung doses to an adult standing at the EPZ were calculated to be 0.015 mSv. (1.54 mRem) and 0.4 mSv. (40.0 mRem), respectively. The primary contributors to dose were Fe-55 and Co-60.

Tornadoes

A number of external accident initiators were evaluated, including earthquakes, tornadoes, floods, and range fires. Because of plant design and site conditions, only the tornado was further evaluated. The risks from a tornado at Fort St. Vrain during decommissioning are quite low: first, the probability that a tornado will strike the site is small, and second, the plant-specific vulnerability to a tornado and its consequence are also small. Unlike an operating nuclear power plant with active safety systems to contain large quantities of radioactive materials at high energy levels, all spent fuel will be removed from Fort St. Vrain prior to decommissioning. The PCRV will be a passive container of radioactive material. Possible loss-of-power caused by a tornado is covered under the loss of power scenario.

Temporary storage or staging of radioactive waste containers prior to shipment is expected. It is assumed that interim radioactive waste storage will be available for 15 LSA containers and 200 drums in the Fort St. Vrain New Fuel Storage Building. Calculations demonstrate that neither forces generated by a 325 kph (202 mph) wind loading nor the impact from the tornado-driven design basis missile will result in breach of the walls or roof of this building. In this scenario, it is assumed that a 325 kph (202 mph) tornado strikes the Fort St. Vrain site. At this wind level, the walls of the Reactor Building enclosing the PCRV will remain intact.

The tornado-driven, design-basis missile is a 3.66 m. (12 ft.) by 0.3 m. (1 ft.) by 0.1m. (4 in.) 3.66 m. (12 ft.) by 0.3m. (1 ft.) by 0.1m (4 in.) fir plank, weighing 48 kg.(105 lbs.), which impacts and penetrates the Reactor Building above the refueling deck level. It was assumed that this missile strikes and ruptures a LSA container with 46 graphite side spacer blocks. It was conservatively assumed that one percent of the activity in the LSA box is dispersed and released to the environment. No credit was taken for the Reactor Building ventilation system or an elevated release. The total radioactive inventory for the 46 graphite side spacer blocks is estimated at 2.74E + 13 Bq. (741 Ci.).

The whole body and lung doses to an adult standing at the EPZ were calculated to be 0.0058 mSv. (0.58 mRem) and 0.168 mSv. (16.8 mRem), respectively.

CONCLUSIONS

The results of the preceding accident scenarios postulated for Fort St. Vrain decommissioning indicate that the radiation exposures to a member of the public will be very low. A summary of the consequences of these accident scenarios is shown in Fig. 1. These evaluations have determined that in all cases, the radiological consequences at the 100 m. (328 ft.). EPZ are well within the NRC 10CRF100 regulation (8) of 0.25 Sv. (25 Rem)whole body dose, and 3 Sv. (300 Rem) to any specific organ. These doses are also a small fraction of the 0.01 Sv. (1 Rem) whole body dose and 0.05 Sv. (5 Rem) to any specific organ cited in the EPA Protective Action Guidelines (9).

These scenarios are considered to have a low probability of occurrence and their radiological consequences envelope other less severe accident scenarios. It is concluded, therefore, that the Fort St. Vrain decommissioning activities do not pose any undue risk to the health and safety of the general public.

REFERENCES

1. *Fort St. Vrain Updated Final Safety Analysis Report*, Rev 8, Public Service Company of Colorado
2. *Fort St. Vrain Activation Analysis*, EE-DEG-0010, Rev 8, Public Service Company of Colorado, October 1990.
3. NUREG-0612 *Control of Heavy Loads at Nuclear Power Plants*, July 1980.
4. NUREG-0554 *Single-Failure-Proof Cranes for Nuclear Power Plants*, May 1979.
5. ASTM E119-88 *Standard Test Methods for Fire Tests of Building Construction and Materials*, 1988.
6. J. Mishma, L.C.Schwendiman *Fractional Airborne Release of Uranium During the Burning of Contaminated Waste*, BNWL-1730, April 1973.
7. WCAP-11002 *Evaluation of Steam Generator Overfill Due to a Steam Generator Tube Rupture Accident*, Westinghouse Electric, February 1986.
8. 10CFR100, *Reactor Site Criteria*, October 1988.
9. EPA-520/1-75-001-*A Manual of Protective Action Guides and Protective Actions for Nuclear Incidents* Environmental Protection Agency, January 1990.

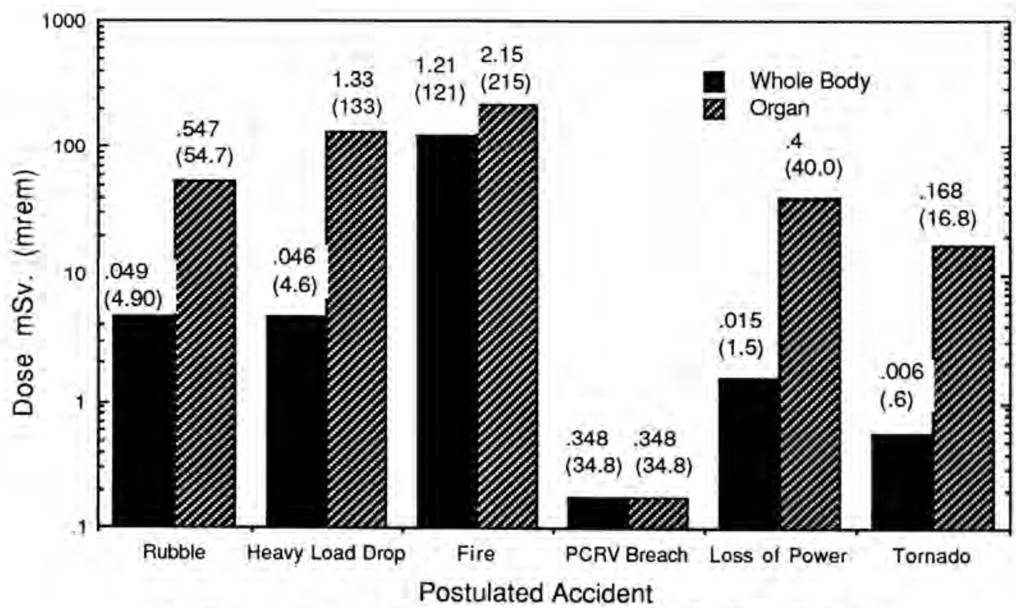

Fig. 1. Impact from Fort St. Vrain decommissioning accidents.