CONCRETE REMOVAL AND DISPOSAL FROM RESEARCH REACTOR DECOMMISSIONING

WASTE MANAGEMENT 1986

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ABSTRACT

Removal and disposal of neutron activated concrete from biological shields is the most significant operational task associated with research reactor decommissioning. During 1985 and 1986 Chem-Nuclear Systems, Inc. was the prime contractor for complete dismantlement and decommissioning of the Northrop Triga Reactor Facility and the Virginia Tech Argonaut Reactor.

This paper discusses operational requirements, methods employed, and results of the concrete removal, packaging, transport and disposal operations for the Northrop Triga and Virginia Tech Reactor decommissioning projects.

Methods employed for each are compared. Disposal of concrete above and below regulatory release limits for unrestricted use are discussed.

This study concludes that activated reactor biological shield concrete can be safely removed and buried under current regulations.

Operational Requirements

To meet unrestricted use and license termination criteria, removal of contaminated and activated concrete from reactor biological shielding is required. This includes removing both radioactive contamination and activation products, which are produced by reactor operations. Release criteria for these operations are defined in NRC Regulatory Guide 1.86 for fixed and loose surface contamination, and Draft Regulatory Guidance found in the NRC staff position paper on Reactor Decommissioning. This guidance specifies a dose-rate limit of 5 micro-R per hour above background measured at one meter from any surface or alternately, an annual dose to persons occupying the area of 10 mrem per year. The most limiting requirement when removing activated concrete is the 5 micro-R per hour (ur/hr) dose rate limit. In order to monitor for compliance with this requirement, sensitive instrumentation (usually a pressurized ionization chamber) is used to survey material (concrete) subject to the release limit. A number of measurements must be taken in background areas with similar construction materials and similar geometries to establish a background dose rate. The release criteria dose rate limit is then applied to the actual background measurement(s). Generally, this background level is in the neighborhood of 10 ur/hr. In Los Angeles, California, background measurements ranged from 7 to 14 ur/hr. In western Virginia, (Blacksburg), the background dose rates were approximately 6 to 12 ur/hr.

Additional requirements which govern the operations for concrete removal include proper control of dusts, airborne radioactivity, and surface contamination generated during removal; proper monitoring, training, dosimetry, and adherence to safety procedures by workers who actually perform the tasks. Localized shielding is sometimes used to reduce area dose rates for concrete removal, particularly around activated steel embedments or near surface reinforcing bars (rebar).

It is essential to have an estimate of activation product concentration as a function of depth into the concrete shield prior to actual concrete shield demolition. Estimating concrete activation depth is a relatively well-known science which is based on the integrated power output of the reactor, the materials which intervene between the reactor and the shield, and the composition of the shield itself. For most concrete shields studied, the mean free path of neutrons in the concrete is approximately the same. For a number of ordinary concrete shields studied, the mean free path is approximately 11 centimeters. Based on this data, one can use samples of the concrete shield from close to the surface to estimate the depth at which the concrete radionuclide concentration will result in meeting the release criteria. Concrete core samples can be used to verify these estimates and measure for real specific activities as a function of depth.

The principle radionuclides found in concrete samples from the Northrop and Virginia Tech reactor shields are listed on Table I and show that the nuclides, Cobalt-60, Europium-152, Europium-154, Tritium and Carbon-14 make up most of the activity in the shield. Hard to measure nuclides, such as Iron-55, Nickel-63, and other weak beta emitters, have been measured in concrete shields but at levels which are far below any concern from either a occupational safety or waste classification standpoint.

TABLE I

Concrete Activation Products

Radionuclide	Half-Life	Concentration (pCi/gm) Northrop	Virginia Tech
22 Na	2.6y	--	1050
40 K	1.29 E + 4y	20.0	7610
54 Mn	312d	38.7	--
55 Fe	2.7y	7030	1
59 Fe	44.6d	33.2	1
58 Co	70d	11.7	374
60 Co	5.27y	796	19600
134 Cs	2.06y	28.3	842
141 Ce	32.5d	2.0	--
152 Eu	13.4y	2870	65100
154 Eu	8.2y	234	1590
155 Eu	4.76y	9.8	318
226 Ra	1600y	0.3	288
228 Th	1.91y	0.7	--
232 Th	1.4 E +10y	0.6	150

Methods Employed

The two research reactor decommissioning projects discussed involved different reactor shield geometries and different concrete removal methodologies. The Northrop Triga reactor was a pool type reactor. Here, concrete removal requirements were confined most exclusively to the dry exposure room which was a cube-shaped room approximately 11 feet on a side shown in Fig.1. Concrete removal depth in this reactor was up to 40 inches. A different type of reactor shield geometry was at Virginia Tech. This was a Argonaut type reactor which employed a four foot cube reactor core cavity with two major experimental facilities, the thermal column and tank duct, which protruded from the central core area shown in Fig. 2. Concrete surrounding these areas had to be removed to a depth up to 28".

Northrop Triga Reactor

This pool type reactor shield had an internal cavity which served as a dry exposure room. This exposure room measured approximately 10 foot cube and was located immediately adjacent to the reactor pool space (separated by a 1/4 inch thick sheet of aluminum). Concrete removal was required from the exposure room in roughly a spherical pattern at depths up to 40 inches deep into the shield. This reactor had operated a nominal power level of 1 megawatt from 1963 until 1985. The shut-down and decay period was only six months, thus activity from a number of short-lived radionuclides such as Iron-55 predominated in the shield during dismantlement and concrete removal.

The large cavity size inside the shield permitted the use of a small, tractor-mounted impact hammer known as a Mini-Max for concrete breaking. It could be operated remotely via an umbilical and a hand-held control panel. The Mini-Max worked in a top-down fashion, breaking chunks of concrete up to one cubic foot from all interior surfaces of the exposure room. It could remove concrete at a rate of approximately 120 cubic feet per eight-hour shift. During concrete breaking, a fine water mist was sprayed immediately around the impact hammer to minimize dust. The work area was completely contained inside a prefabricated containment structure. Negative pressure within this area was maintained using two 2,000 cfm HEPA-filtered exhaust units. Pre-filters upstream of the HEPA-filters were replaced every 4 to 6 hours due to high dust loading. Airborne radioactivity monitors were used to ensure proper control of activated concrete dusts.

General area dose rate inside the exposure room during removal was up to 5 mR/hr, hot spots from embedded steel and nails measured up to 40 mR/hr. Rubble ranging in size from very small debris up to one cubic foot chunks were loaded into steel boxes, approximately 90 cubic feet each. Due to the presence of incidental quantities of water and some hydraulic fluid which leaked from the concrete breaking machine, absorbent was placed in the bottom and dispersed throughout the box to eliminate free water. The measured swell factor or the ratio of inplace concrete volume to final waste packaged volume varied from approximately 1.2 to 1.35. Overall, concrete removal operations took six weeks and involved a crew of six men working 10 hour shifts, six days a week. A total of 2800 cu. ft. of concrete weighing 68 tons was packaged and shipped for disposal at the Hanford site operated by U.S. Ecology.

Argonaut Reactor

The Virginia Tech Argonaut Reactor was operated from 1962 until 1981 with a total energy output of 2200 MW hours at a nominal power level of 100KW. Because of the five years of decay time, only relatively long-lived activation products remained in the shield. The Argonaut design involved a four foot cube reactor core area, surrounded by a 188 ton shield of ordinary concrete. After removal of the aluminum control rod drives, fuel water boxes and graphite core moderator, dose rates inside the core cavity were up to 2 R/hr on contact with embedded steel members. General cavity area dose rates measured up to 300 mR/hr.

During initial planning for the decommissioning activity, a two-inch concrete core was drilled from outside to inside the reactor shield. This 6 foot sample was segmented into two-inch long increments and analyzed for radioactivity. Calculations of reactor core neutron flux were made to predict the amount of concrete activation in the concrete shield. Due to the long decay period, nuclides found were mostly Cobalt-60 and Europium-152. A concrete removal plan was developed based the activation profile expected. This activation profile relied on the single core sample drilled through the core and a relative neutron flux profile derived from previous measurements of neutron flux and calculations. Because of the restrictive size inside the core cavity as well as the high dose rates, concrete removal was planned to progress from outside to inside, removing the non-radioactive concrete first.

Due to the restricted reactor room access and weight limitations associated with handling materials, the maximum practical size for any single piece of concrete to be removed from this shield was approximately 10,000 lbs. Immediately outside the access door from the reactor room, a major facility construction project was underway during the term of the decommissioning work. This resulted in having a narrow, difficult path comprised of dirt and gravel for passage of equipment and materials.

The Reactor was first separated into two major pieces using a diamond-impregnated wire saw. The first cut was made through the horizontial beltline of the reactor shield. A second cut was made at the base of the shield. These two horizontal cuts then separated the biological shield into two large slabs; approximately 90 tons each. The wire saw performed each cut in approximately four days each. Much of this time was dedicated to set-up and control of cutting water needed to lubricate the wire. Water was channeled from the cut area to a sump where solids were allowed to settle out. Careful monitoring of the sludge and water resulted in determinations that most of the cutting fluid met free release criterion. Higher activity salvage which was collected while cutting through the central region of the shield (near the reactor core) was solidified and disposed of as radioactive waste.

The next step after wire sawing was to cut near-surface rebar using a diamond-tipped circular concrete saw. Vertical cuts on roughly 2-1/2 foot centers were made to define the edges of each five-ton block which would be produced from rock drilling and splitting. This saw-cutting was done to a depth of roughly four inches. Following these operations, a pneumatically powered rock drill was used to drill 1-7/8 inch holes on approximately 10 inch centers. These holes, drilled to depths up to 6 feet, permitted introduction of hydraulically powered concrete splitters which were used to segment the slab into blocks. Embedded conduit and steel supports presented unique problems with segmenting the blocks.

Release Surveys

At Virginia Tech a 10-ton overhead crane was used to rig each concrete block as it was separated from the shield. The block was weighed and moved to a low background area where it was surveyed. This involved an assessment of the degree of surface contamination and measurements of the dose rate at one meter from each side of the block. Blocks and "chunks" which were below the dose rate and contamination criteria were removed to the University owned landfill. The radioactive portions of the biological shield amounted to a total volume of approximately 1300 cubic feet.

In the case of the Northrop reactor, after removal of approximately 40 inches of concrete, dose rates inside the shield's 4 Pi geometry indicated the presence of activation products above background levels. Removal of the shield to 2 Pi geometry and re-survey of the material showed that the dose rates were less than the 5 uR/hr criteria. Based on calculations and verified by actual measurements, this dose rate elevation factor attributable to the 4 Pi compared to 2 Pi geometry was approximately 1.8 to 2.0.

In the case of the Virginia Tech reactor shield, all surfaces measured were approximately 2 Pi with the exception of the excavated area below the reactor itself. Removal of concrete and below-grade soil along the neutron flux profile created a dish-shaped area in the building floor. With this greater than 2 Pi geometry, surveys indicated readings of 5-10 uR/hr above background. Once this material was removed and deployed onto a 2 Pi surface, it measured to be less than 5 uR/hr above background at one meter.

Of the 1,098 tons of concrete material removed from the Northrop reactor, approximately 118 tons was shipped as low-level radioactive waste. It contained about 3.6 millicuries of natural activity, and 1.5 curies of induced (man-made) radioactivity. The remaining concrete below release criteria amounted to approximately 980 tons and contained 30 millicuries of natural activity and 0.5 millicuries of induced activation products.

From the Virginia Tech reactor, the amount of material removed as LLW was 30% of the total shield mass. Of the 188 ton shield, 68 tons containing 0.65 curies of induced activity and 36 millicuries of primordial activity was shipped as low-level radioactive waste. Non-radioactive waste (120 tons) contained about about 64 millicuries of primordial activity and 4.8 microcuries of induced activity and was buried at the local University landfill. Table II summarizes the concrete waste removed from both Northrop and Virginia Tech.

TABLE II

Concrete Waste Summary

	MASS (TONS)	VOLUME	ACTIVITY (Ci) INDUCED	PRIMORDIAL
Northrop				
LLW	118T	2800 ft^3	1.5	3.6E-3
Released	980T	23,000 ft^3	5.5E-4	3.ØE-2
Virginia Tech				
LLW	68T	1,300 ft^3	Ø.65	3.6E-2
Released	120T	2,100 ft^3	4.8E-6	6.4E-2

Conclusions

Methods for demolition of research reactor biological shielding are relatively straightforward and readily available within the industry today. The methods employed are dependent on the geometry of the shield and activation pattern within the shield.

Reactor biological shielding can be safely segregated into radioactive and non-radioactive components in accordance with NRC Regulations and branch technical position on research reactor decommissioning.

Monitoring methods for compliance with release criteria must consider the geometry of the source.

Disposal of concrete waste from reactor decommissioning, in accordance with 10CFR 61 and NRC regulations poses no unusual challenges or risks.

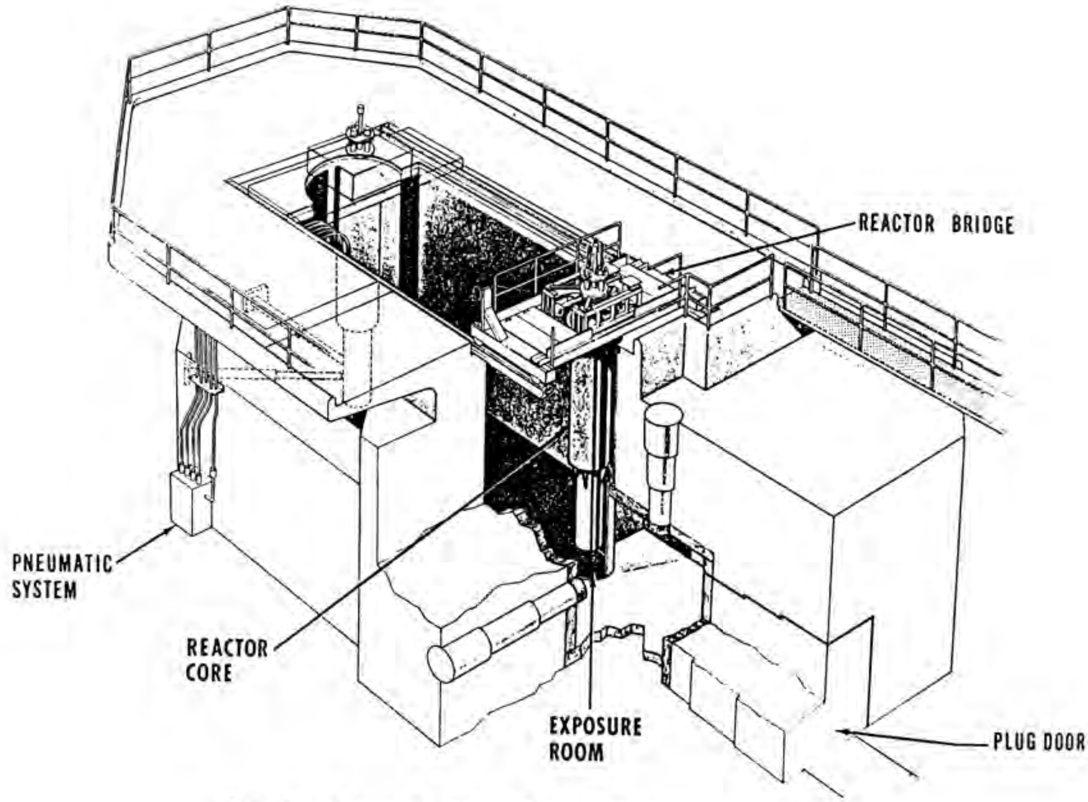

Fig. 1. Isometric View of Northrop Reactor.

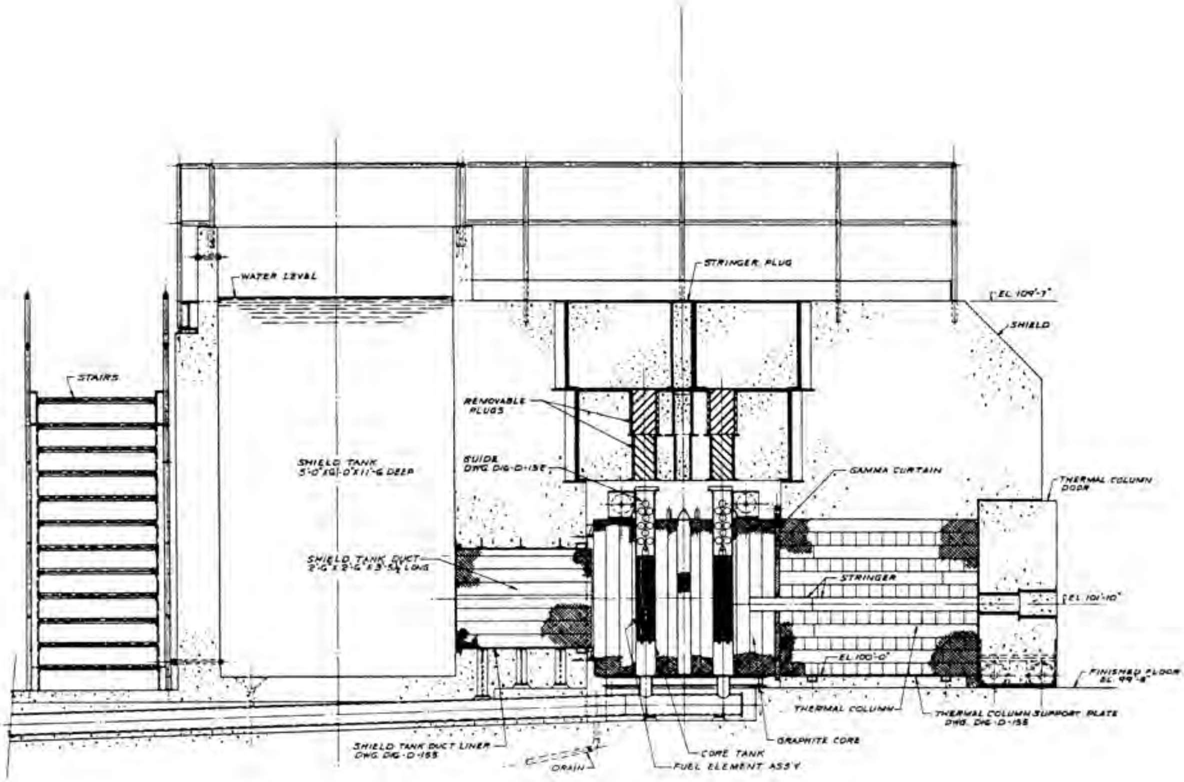

Fig. 2. Longitudinal Section of Virginia Tech Argonaut Reactor.