Characterisation Progress at the Windscale Pile Reactors
Challenges and Results

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ABSTRACT

The decommissioning of the Windscale Pile 1 reactor, fifty years after the 1957 fire, is one of the most technically challenging decommissioning projects in the United Kingdom, if not the world. The decommissioning is being performed by an Alliance of the United Kingdom Atomic Energy Authority (UKAEA), CH2M HILL International Nuclear Services (CHNS) Ltd. and AMEC, NNC. The 1957 Windscale Pile 1 accident is summarized. The resulting fire caused significant characterisation challenges. Challenges to intrusive characterization included hypothesized uranium hydride causing re-ignition of the core fire, unknown fuel configurations leading to a reactor criticality and graphite dust explosions. As a result, the Pile 1 facilities were sealed, isolated and managed in a monitoring and surveillance regime while plans for dismantling were developed. For years the intrusive inspection of the fire damaged region of Pile 1, estimated to contain 15 tonnes of fuel, was precluded based on safety grounds.

In June of 2006 the United Kingdom Health and Safety Directorate approved a new Pile 1 safety case that successfully demonstrated that Pile 1 presents a minimal safety risk with no credible risk of a core fire, criticality or graphite dust explosion. Adoption of the new safety case enabled the intrusive inspections of the fire damaged region. Characterisation activities planned and performed since the safety case approval, were prioritised relative to the results potential to mitigate decommissioning project risks.

D-Void examinations, irradiation foil hole intrusive inspections, bioshield and thermal shield plate characterizations were performed. Results obtained allow determination of waste stream composition and confirmation of assumed design conditions. Changes to the strategic approach to safely and efficiently decommission the two Windscale Pile Reactors include waste packaging and storage facilities and confirmation of design assumptions. Fuel channel endoscope inspections have confirmed the strategic approach to safely and efficiently decommission the Windscale Pile 1 Reactor. The first detailed images from deep within the Pile 1 fire affected zone(FAZ) since the 1957 fire are discussed. The decommissioning impacts of these images are provided. Characterisation has confirmed design assumptions and supported decommissioning option evaluation. The results confirmed that the decommissioning strategy of a top down approach, employing an array of light weight, high payload robotic arms to remove the damaged fuel, the graphite core, activated metals and concrete remains the most efficient decommissioning method.

INTRODUCTION

Windscale Pile 1 was the first of two graphite-moderated, air-cooled, reactors built in west Cumbria in the late 1940's. Fuelled with natural and/or slightly enriched uranium cartridges in horizontal fuel channels, the reactors produced plutonium and other materials for the British nuclear weapons programme. Waste heat was rejected to the atmosphere by once-through air cooling. In October 1957, during a graphite annealing operation, the core temperature unexpectedly increased resulting in a large fuel fire. During the analysis of the event, as a conservative measure, the second plant, Pile 2 was shut down. Both reactors were defueled and production isotope cartridges removed. The two facilities were later sealed and isolated and put on a monitoring and surveillance regime while plans for dismantling were developed.

PILE 1 FIRE

The fire was a complex event. The Piles at the Windscale site were specifically designed and built for the production of weapons grade plutonium necessary to maintain strong defence capabilities. One component (annealing) of routine operations was the periodic release of Wigner energy that accumulated in the graphite during operation. Thermocouples were installed during annealing to monitor the core and fuel temperatures, the temperatures indicated were used as a proxy for the amount of Wigner energy released. Shutdown fans were used during annealing to force air through the fuel channels to carry away excess energy while maintaining a temperature that allowed a self sustaining (without reactor power input) Wigner energy release. The nominal sequence of the annealing operation and subsequent fire was:

Monday 7th	Nuclear heating started.
Tuesday 8th	Self sustaining energy release started at 01.00 but appeared to have ceased by 09.00.
	Second nuclear heating applied between 11.00 and 19.25.
Wednesday 9th	High temperatures contained by cooling.
	Isotope cartridge damage detected.
Thursday 10th	Fuel damage detected.
	16.30 – 23.30 Glowing fuel cartridges seen - up to 150 channels affected.
	Firebreak created around these channels.
	Fire temperature reached indicated temperature of 1300^{o}C.
Friday 11th	CO_2 injection proved ineffective.
	08.55 Water injection started.
	10.10 Shutdown fans turned off – fire died immediately.
Saturday 12th	15.10 Pile cold; water turned off; hoses removed.

Although exhausted by the fire fighting efforts no one at Windscale or in the surrounding area was injured in the accident. The most immediate and significant consequences of the fire was the plume of radioactivity that escaped from the core via the chimney and out into the surrounding environment. Initially heated from the fire the plume quickly cooled and fell onto the ground. The portion of the radioactive cloud that did not fall to the ground was detected in Europe and Scandinavia in the next few days. The interior of the reactor was contaminated with damaged fuel and production isotopes and potentially uranium hydride had been formed when the water injection reached damaged fuel elements. Damaged fuel was observed on the discharge of the reactor.

CHARACTERISATION CHALLENGES

Characterisation answers a series of questions. The most important being what is necessary to protect the workers while they perform decommissioning activities. This knowledge enables the philosophy of “DO IT ONCE--DO IT RIGHT”. The characterisation knowledge determines the methods of protecting the worker and the packaging of decommissioning wastes. This information allows completion of the decommissioning activities without; 1) safety case violations; 2) environmental releases; 3) lost time accidents; 4) rejected waste shipments; or 5) remaining residual Nuclear Decommissioning Authority site liability.

At the project level, the characterisation knowledge allows the resolution of uncertainties included in the project risk register. For instance in 2003, some believed the radiation levels in the graphite core were so high that if the bioshield top was removed, the sky shine would make the neighbouring vicinity uninhabitable. The same radiation levels would require radiation hardening of equipment used to perform decommissioning operations. Project planning estimates for waste streams of the same vintage assumed that a portion of the bioshield concrete and the thermal shield plates would be Intermediate Level Waste (ILW). Other areas were assumed for planning purposes to have been contaminated with fission products and fuel debris during the fire.

For years, the intrusive inspection of the fire damaged region of Pile 1, estimated to contain up to 15 tonnes of fuel, was precluded based on safety concerns. Concerns included that disturbance of the potential present uranium hydride could re-ignite the core fire. Other concerns were that unknown fuel configurations might lead to a reactor criticality and earthquakes might cause a graphite dust explosion. Consequently early plans for dismantling were developed using very pessimistic assumptions. Proposed dismantling options used complex remote systems, through small penetrations in the bioshield in a fully inerted atmosphere. The technical, financial and regulatory hurdles of these options were found to be too high to overcome.

UKAEA/CH2M HILL decided to attack these issues head on and directed an extensive programme of characterization, reactivity measurements, fuel inventory evaluation, criticality analysis and graphite performance modeling. This resulted in a proposed safety case submitted in 2005 that reduced the uncertainties associated with the previous concerns. In June of 2006 the United Kingdom Health and Safety Directorate approved a new Pile 1 safety case that successfully demonstrated that Pile 1 presents a lower safety risk than assumed previously, with no credible risk of a core fire, criticality or graphite dust explosion. Implementing the new safety case enabled the intrusive inspections of the fire damaged region. Characterization activities planned and performed since the safety case approval, were prioritized relative to the results ability to mitigate decommissioning project risks.

The five largest project risks were only resolvable by characterisation. They were, chronologically as investigated:

- The radiation levels and conditions present in the graphite;
- The radiological levels present in the discharge area of the reactor known as the D-void;
- The actual activity level in reactor components and the bioshield;
- The amount of fuel and condition of the fuel channels in the fire affected zone (FAZ);
- The radiological levels of the debris material in the fuel channels.

RESULTS

Characterisation results have impacted the strategic approach to safely and efficiently decommission the Windscale Reactors by determining that conditions confirm design assumptions and support various decommissioning option evaluations. The results have confirmed that the decommissioning top down in air approach, employing an array of light weight, high payload robotic arms to remove the damaged fuel, the graphite core, activated metals and concrete remains the most efficient decommissioning method. Characterisation outcomes and impacts are discussed in chronological sequence.

Pile 1 Foil Hole Inspections

The first intrusive characterisation was the vertical traverse of two irradiation foil holes external to the fire affected zone with cameras and radiation level instruments. The inspection revealed graphite that appeared as it did upon installation. Radiation levels were clearly so low that radiation hardened decommissioning equipment will not be required. Radiation levels in the foil hole graphite column average nominally 2 mSv/hr while approaching a 5 mSv/hr maximum. An intact fuel element was observed through a Wigner gap (estimated 75mm from the detector) and the radiation levels near that fuel element approached 20 mSv/hr.

Impact. Consultation with stakeholders was held. Based upon the information obtained the decision to proceed directly to the intrusive inspection of the fuel channels was made. This decision accelerated the schedule by nearly a year and reduced characterisation costs by nearly £300,000. The determination that radiation hardened tools were not necessary clearly impacts design and fabrication schedules by multiple years while no pound value for that reduction has been calculated.

Pile 1 D-Void Inspections.

The new safety case provided the knowledge that the disturbance of the D-void spaces would not have consequences that impacted the reactor or the environment. Concrete cores were removed that allowed the insertion of an infrared camera and a radiation detection instrument into the D-Void from the D-Void ceiling. The traverse from the ceiling to the floor confirmed that the physical conditions were essentially uncontaminated. No removable contamination was found on any equipment and the radiation levels were less than 1 microSv/hr.

Impact. The determination that the D-Void was uncontaminated allowed the elimination of additional characterisation efforts valued in excess of £200,000. This decision accelerated the characterisation schedule by nearly a year. The major impacts are associated with the removal of this work from the ILW removal schedule and inclusion in the exempt materials removal schedule, The D-void removal is no longer a critical path activity and is planned using conventional demolition techniques. The resulting estimate is significantly lower than that previously used by the project.

Bioshield Characterisation

Ten bioshield concrete cores were removed, five from Pile 1 and five from Pile 2. These cores included concrete, concrete reinforcement steel, rock wool insulation, insulation box steel, thermal shield plate steel coupons, and thermal shield plate support framing steel. These samples are allowing assessment of contamination and activation levels on various portions of the Piles structures.

Preliminary results have indicated that estimates of the ILW portion of the bioshields were conservative. Radiation levels at the interior surface of the bioshield concrete (from activation and contamination) for

Pile 1 approach 1 microSv/hr and for
Pile 2 are less than 1 microSv/hr

Concrete radioactivity levels at the bioshield inside face are factor of 10 below the ILW threshold thus all concrete is Low Level Waste (LLW). Concrete with tritium diffusion to levels greater than exempt levels, in the bioshield wall reached a maximum of 130 cm from the inside face (maximum thickness of High Volume Very Low Level Waste (HVVLLW)). The remainder of the bioshield is below exempt radiological levels and therefore is available for reuse or disposal.

Impact. There is no ILW concrete waste forecast from the bioshield decommissioning efforts. This lowers the estimate of volume to be stored in the temporary ILW store by nominally 1000 m^3 or 6% of the previously estimated total project ILW volumes.

The steel in the core (Thermal Shield Plates and insulation boxes) are expected to be LLW based upon instrument readings and concrete contamination levels. Analytical results are being reviewed. This reduces the total project ILW forecast of steel by 3000 m^3 and of insulation boxes by 600 m^3 or 24% of the previously estimated total project ILW volumes.

These reductions represent nominally 4600 m^3 or 30% of the storage capacity required in the temporary store. Site costs for a smaller store, and packaging and processing that volume as LLW instead of ILW reduce costs by £30-35 million, of which about half represents a Piles Project budget estimate reduction and the other half is estimated from long term stewardship and disposition charges.

Pile 1 Fuel Channel Inspection

A team of UKAEA, CH2M HILL and Hertel personnel performed the first phase of the intrusive inspection of fuel channels in the Fire Affected Zone (FAZ) of the Windscale Pile 1 Reactor. Fifty-two fuel channels (Table I) in the fire affected zone were inspected between July 27th and September 3rd, 2007.

Table I.—Listing of Channels Inspected, Images Captured and Video DVD Labels.

Pile 1 FIRST LOOK IN 50 YEARS! ENDOSCOPE INSPECTION				
POSITION FROM CHARGE FACE			STILL IMAGES	
ROW	COLUMN	CHANNEL	CAPTURED	DVD #
24	49	TL	6	24 49 TL, TR, BL, BR
		TR	8	24 49 TL, TR, BL, BR
		BL	2	24 49 TL, TR, BL, BR
		BR	0	24 49 TL, TR, BL, BR
	50	TL	17	24 50 TL, TR, BL, BR
		TR	10	24 50 TL, TR, BL, BR
		BL	10	24 50 TL, TR, BL, BR
		BR	23	24 50 TL, TR, BL, BR
	51	TL	8	24 51 TL, TR, BL, BR
		TR	12	24 51 TL, TR, BL, BR
		BL	17	24 51 TL, TR, BL, BR
		BR	15	24 51 TL, TR, BL, BR
	52	TL	19	24 52 TL
		TR	4	24 52 BR, TR
		BL	42	24 52 BL
		BR	19	24 52 BR, TR
	53	TL	7	24 53 TL, TR
		TR	5	24 53 TL, TR
		BL	38	24 53 BL
		BR	10	24 53 BR
	54	TL	6	24 54 TL
		TR	8	24 54 TR
		BL	14	24 54 BL
		BR	13	24 54 BR
	55	TL	18	24 55 TL
		TR	16	24 55 TR
		BL	14	24 55 BL
		BR	9	24 55 BR
	56	TL	4	24 56 TL
		TR	23	24 56 TR
		BL	7	24 56 BL
		BR	15	24 56 BR
	57	TL	9	24 57 TR, TL
		TR	11	24 57 TR, TL
		BL	14	24 57 BL
		BR	10	24 57 BR
	58	TL	7	24 58 TL
		TR	1	24 58 TR, BL, BR
		BL	16	24 58 TR, BL, BR
		BR	1	24 58 TR, BL, BR
	59	TL	6	24 59 TR, TL
		TR	3	24 59 TR, TL
		BL	7	24 59 BL, BR
		BR	7	24 59 BL, BR
	60	TL	3	24 60 BL, BR, TL, TR
		TR	3	24 60 BL, BR, TL, TR
		BL	3	24 60 BL, BR, TL, TR
		BR	2	24 60 BL, BR, TL, TR
	61	TL	1	24 61 BL, BR, TR, TL
		TR	4	24 61 BL, BR, TR, TL
		BL	2	24 61 BL, BR, TR, TL
		BR	2	24 61 BL, BR, TR, TL
TL=Top Left TR=Top Right BL=Bottom Left BR=Bottom Right as seen from charge face.				

Systematic inspection with a 10mm diameter endoscope allowed the inspection past fuel and debris blockages (Figure I) for the first time since the 1957 fire. The 12.2 meter long insertion tube on the probe allowed the endoscope to reach the 31st graphite block of the 36 blocks that comprise the length of the reactor. The endoscope video data and the captured still images are stored on DVD's and backed up on an external hard drive.

Figure I—Fuel Channel Blocked By Three Graphite Boats.

As expected, graphite damage was present to varying degrees and the fuel was found to range from intact, apparently pristine fuel elements to materials that visually resemble volcanic ash or coal fire clinker and display no characteristics of fuel other than location in the channel. Evidence of damage to the lead spacers in the isotope channels was observed in the fuel channels.

The graphite conditions range from pristine sharp-edged blocks to blocks that show progressively increasing amounts of worn and weathered conditions. Graphite loss (Figure II) has occurred to the extent that adjacent fuel channels are connected by openings that from informal scaling appear to be multiple inches in height. Two types of "damage" are observed, physical removal of graphite and a weathering of graphite surfaces generally in the presence of re-solidified molten lead. All forms of damage vary considerably between adjacent blocks, potentially reflecting some block specific phenomena e.g. Wigner energy content or release rate.

Figure II—Graphite Damage/Displacement In A Fuel Channel.

In general block gap edges show what appears to be an erosion/ablation removal of material in that the underlying graphite appears solid and capable of load bearing. This is currently believed to be from the fire fighting water flashing to large volumes of steam with resultant high velocity flow through and between the fuel and isotope channels. Some of the erosion/ablation is possibly from the cavitation associated with the fire water flashing to steam. The third removal method is assumed from the oxidation of the surface of the graphite blocks and subsequent removal similar to that observed in the consumption of charcoal in a fire box or barbeque. The loss of materiel is distributed throughout the FAZ in such a manner that varying combinations of all three mechanisms are postulated to have occurred.

The second damage type presents a very “weathered” looking surface on the block. While some delaminating is apparent, observations appeared to indicate that the “weathering” did not penetrate the block deeply, as several inches from a weathered surface on one block the graphite appeared relatively unaffected. The extent of the damage detected will be used to ensure the design of the proposed fuel channel removal tool (FCRT) is sufficiently robust to successfully remove the fuel. No graphite that appeared susceptible to brittle fracture or failure was observed

For the first time in 50 years, images have been obtained from deep within fuel channels damaged in the 1957 fire Pile 1 Reactor fire. Displaced graphite, intact and relatively intact fuel (Figure III) and fuel debris conditions were obtained by visual inspection and the material behaviour in response to the passage and incidental contact of the inspection probe.

Figure 3—Partially melted fuel cartridge with fuel pin exposed.

The reactor contains fuel ranging from pristine appearing intact fuel cartridges to clinker-like (Figure IV) and pseudo-cryptocrystalline appearing clumps of material.

Figure IV— Fuel debris containing fragments of partially melted fuel cartridge fins.

The radioactivity levels and particle sizes affect waste processing requirements for the removed fuel materials. If the ratio of radioactive dusts to the remainder of fuel channel materials is too high container loadings may be affected. Additional waste processing systems may be required. No estimate is available but costs and schedule easily can be affected by tens of millions of pounds and 1-2 years of schedule increase (assuming additional design and construction are critical path activities).

Only miniscule amounts of dust/fly ash like materials were observed, typically as a light surface coating. The inspection methods also provided small samples of the materials in the channels which were retrieved on decontamination swabs of the endoscope. The potential behaviour of the observed “dust” particles in the proposed new ventilation systems was demonstrated by monitoring results at the inlet and outlet of the filters on the existing Pile 1 ventilation system. The cumulative project dose from the Electronic Personnel Dosimetry was 1030 micro Sieverts.

Team dose was reduced by determining in the field that only one charge plug removal per charge pan inspection. This reduced the dose from charge plug handling by nominally 50%. After the first two channels were inspected it was determined that the inspection of all four fuel channels in a pan was both efficient and dose reducing. Performing all four inspections, eliminated the duplicate charge plug removal and associated doses while eliminating approximately 30 minutes of operations in respirators. These practices were adopted. A third dose reduction was achieved by swabbing only the portion of the endoscope necessary to be retrieved into the confinement box to allow the alignment for the next channel. The swab of the full length of the endoscope 12.2 meter long insertion tube was performed only when relocating to the next charge plug. This final swab for each charge pan was retained and later sent to the laboratory for analysis.

No quantities of dust or fine ash-like materials sufficient to bury or block the endoscope camera views were observed. Occasionally dust particles would be observed in the camera view being drawn in the direction of the ventilation flow typically lasting for only 2-3 seconds.

The data from the field data sheets was manually transferred to an Excel spreadsheet to visually display two layers of channel conditions. The layers depict the conditions in color observed in the top layer (Figure 5) and the bottom layer (Figure 6) of fuel channels in row 24 of the charge face.

light green	bright green	gold	yellow	light orange	red	grey	clear	grey
Empty	Blocked by Boats	Blocked Boat&Fuel	Fuel Intact	Fuel damaged	Fuel clinker or Sand	CRV-ControlRodVisible	Not Traversed	HOLES

Row	Column	Position																																				
24	49	TL	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
		TR	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
	50	TL	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	X BF	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
		TR	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	X BF	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
	51	TL	1	2	3	4	X B 5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
		TR	1	2	3	4	5	6	7	8	X B 9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
																																HOLE						
	52	TL	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	hole2	21	22	23	CRV	25	26	27	28	29	30	31	32	33	34	35	36
															to LR	To LR																						
		TR	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	Hole	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
																							to BR															
	53	TL	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
		TR	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
	54	TL	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
		TR	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
	55	TL	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
		TR	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
	56	TL	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
		TR	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
	57	TL	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
		TR	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
	58	TL	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
		TR	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
	59	TL	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
		TR	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
	60	TL	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
		TR	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
	61	TL	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
		TR	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36

Figure 5.—Map Of The Top Layer Of Row 24 Fuel Channels.

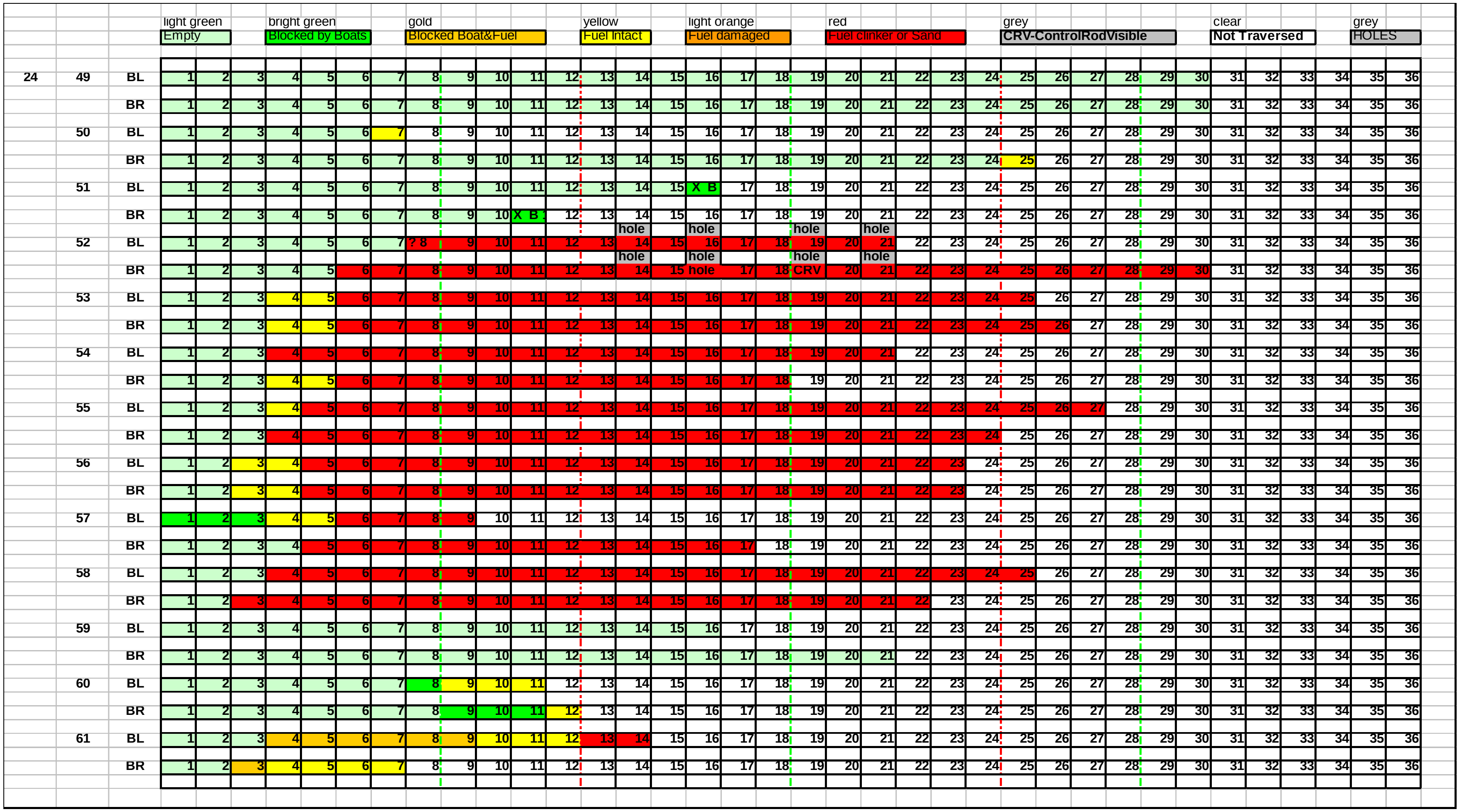

Figure 6.—Map Of The Bottom Layer Of Row 24 Fuel Channels.

Projection of observations to the entire FAZ and Reactor Core

Whilst nearly every channel was a one-off sample of conditions, an estimate of volume of the damage was made from the preliminary review of the field notes. There is observed damage in 8-12 channels in the form of enlarged openings at block joints, enlarged fuel channel interiors, graphite displacement and erosion (loss) of graphite sufficient to require additional operational considerations. Assuming that Row 24 is the midline of the FAZ, the damaged channels were rotated 360 degrees. That circular projection of the assumed damage area overlaps nominally 8 charge pans in total. Eight charge pans contain roughly 5% of the damaged fuel and 1% of the graphite in the core. This is the basis for the statement that “1) For at least 95% of the fuel channels containing intact and damaged fuel, the Fuel Channel Removal Tool (FCRT) is expected to be successful in removing accessible fuel; and 2) At least 99% of the reactor graphite can be removed by the bulk graphite removal methods.” This estimate assumes no fuel or graphite is unloaded from the discharge or charge faces prior to encountering the damage zone.

This zone will require operation adjustments and possibly some tooling adjustments and/or equipment design revisions to be removed. This estimate will be revised as new information is available from the inspection of the remaining fuel channels.

Sampling of material in the fuel channels

The endoscope cabling was cleaned by swabbing as it was removed from the each channel to minimize the dose to operators who were inserting and removing the endoscope through gloves in the containment box wall. Initially swab dose readings were very low but by the 5th charge pan (24-53) dose were 120 times the initial swab. A work stoppage was called while the dose to the operator’s hands from finger dosimetry was obtained. With the confirmation that hand doses were below expected levels, leaded gloves were installed. This minimised the workers projected dose and provided margin against potentially higher levels of exposure. Also starting with charge pan 24-54, swabbing of the12.2 meter long insertion tube was restricted to only that portion retrieved to allow the transfer to the next fuel channel in that charge pan. On the retrieval from the last fuel channel inspecting in the charge pan, the cable and camera were swabbed and the full retrieval reading is recorded in Table II.
Table II.—Endoscope Swab Radiation Count Rates By Date And Charge Pan.

Date	Highest Swab from Camera after Inspections (microSv/hr Beta / Gamma)											
	27-Jul	30-Jul	31-Jul	1-Aug	2-Aug	17-Aug	20-Aug	23-Aug	24-Aug	29-Aug	30-Aug	3-Sep
24-49	250											
24-50		800										
24-51			4200									
24-52				6000								
24-53					30000							
24-54						30000						
24-55							20000					
24-56								26000				
24-57									15000			
24-58										15000		
24-59											8000	
24-60												6000
24-61												1600
	250	800	4200	6000	30000	30000	20000	26000	15000	15000	8000	7600

No attempt has been made to correlate the radiation levels with the endoscope travel distance through damaged fuel debris to determine if the contamination levels are uniform. As a result of the high radiation levels obtained it was realized that the swabs could be processed as sample for analysis at the NIRAS laboratory. Analysis indicates the Cs^{137} to beta emitter ratio is significantly lower than that expected for fission products. Detailed analysis of the beta emitters is in progress. NIRAS is attempting to obtain article size counts from the swab also.

Waste Management and Ventilation System Implications

As mentioned previously, small bursts of “dust” particles were observed. These particles were light enough that once disturbed, the air flow from the existing ventilation system would suspend the particles and transport them in the direction of the air flow. The endoscope lighting system would reflect off the particles and they would dance like dust motes in the sunlight as they exited the light. No particles were observed to fall to the channel floor. It was realised that this behaviour had the potential to provide significant information about the particles size. Figures 7 and 8 display the beta readings and alpha readings respectively as a function of time. Readings are from the inlets to the east and west discharge ducts filter and from the stack monitor which is the official stack release monitor.

Date	VPR (W)	VPR (E)	Stack
24-Jul	12	12	1
25-Jul	8	9	2
26-Jul	4	22	5
27-Jul	11	7	1
31-Jul	1432	2367	4
1-Aug	564	639	1
2-Aug	48	34	1
3-Aug	2985	5158	7
6-Aug	95	106	5
7-Aug	49	48	4
8-Aug	10	15	0
9-Aug	9	4	4
10-Aug	296	440	2
13-Aug	15	34	1
14-Aug	5	13	0
15-Aug	12	13	1
16-Aug	26	35	3
17-Aug	462	570	3
20-Aug	2135	3565	3
21-Aug	91	484	2
28-Aug	42	49	3
29-Aug	2168	2574	5
30-Aug	491	318	3

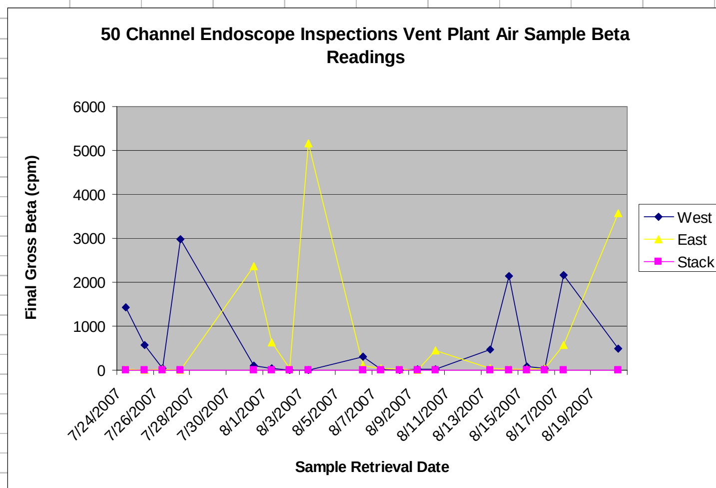

Fig. 7.—Beta Readings At The Vent Plant Air Samplers And The Stack Monitor.

Date	VPR (W)	VPR (E)	Stack
24-Jul	0	1	0
25-Jul	0	0	0
26-Jul	0	0	0
27-Jul	0	1	0
31-Jul	1	4	0
1-Aug	3	2	0
2-Aug	0	0	0
3-Aug	10	35	1
6-Aug	0	3	0
7-Aug	0	0	0
8-Aug	0	0	0
9-Aug	0	0	0
10-Aug	3	0	0
13-Aug	0	0	0
14-Aug	0	0	0
15-Aug	0	0	0
16-Aug	1	1	0
17-Aug	3	7	0
20-Aug	11	19	0
21-Aug	0	1	0
28-Aug	0	0	1
29-Aug	9	12	0
30-Aug	1	1	0

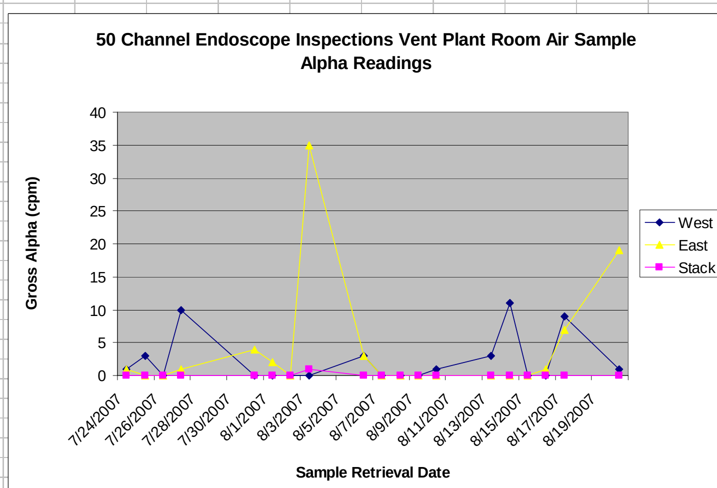

Fig. 8.—Alpha Readings At The Vent Plant Air Samplers And The Stack Monitor.

As can be seen from the data, despite significant changes in the daily reading, essentially no release occurred. This indicates that the particle size is large enough to be captured by HEPA filters. This information can be used to support the assessment of waste encapsulation methods. The quantities of dust seem to be a small fraction of the total material to be removed from the channels.

Recommendations:

Two recommendations have been reached from this preliminary review of the field notes of essentially fifty-two one off channel inspections and a review of the value of and requirements for additional endoscope surveys.

- Inspection of the remaining fuel channels should be started with emphasis on those channels containing fuel in the FAZ.

- A detailed decommissioning evaluation of the fifty-two channels inspection video should be performed;

These recommendations are based upon the evaluation of the following project activities.

Characterisation:

Cost estimates based upon the 52 channels inspected so far, show that the inspection of channels by endoscope is the most cost effective method to locate and confirm fuel channel contents.

Early Characterization is more cost effective than postponement

F&I Removal design:

The identification of each fuel channel that the FCRT must visit is required to establish the operating locations the FCRT must be able to reach. The location and the distribution of fuel in each channel is required to perform the OR studies of operations and is input to the RAM assessments. The existing tooling is constrained by the discharge void depth to a "scoop" or "shoe" length of roughly 20 inches. The "scoop" or "shoe" diameter is constrained to nominally 3 inches. The detection of holes of greater than nominally three blocks in length or 3 inches wide in the floor of fuel channels is required to identify locations presenting additional operational difficulties. Displaced graphite blocks and channels that appear closed in with displaced blocks should be identified.

Characterization of more channels will significantly shorten the duration of FCRT duty, identifying significant cost savings during F&I retrieval from those presently estimated. The additional characterization data will also reduce the risk of having to redesign the FCRT because unexpected conditions are encountered during retrieval activities.

Waste processing:

The locations and the distributions of fuel cartridges and fuel debris in each channel in the FAZ are required to support the Letter Of Compliance production and waste route design. The polymer trialing planned in support of the letter of compliance submittals will in the relative near future, require addition inventory descriptions and quantity estimates. This information is an integral portion of the design of the waste out facilities.

More Characterization will provide more accurate Waste Inventories and increase the likelihood of earlier Letter of Compliance receipt.

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