NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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TITLE (4)

Two Workers Received an Internal Exposure Which Potentially Exceeded Federal Limits

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
11	02	96	96	030	02	08	13	99		
			--		--				FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	6	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	000	X	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
		X	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Richard Sexton, Health Physics Manager	(860) 267-3947

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On Saturday November 2, 1996 at 0830 hours, with the plant in Mode 6 (refueling) a plant maintenance supervisor, and a contractor refueling manager entered the fuel transfer canal area of the refueling cavity to inspect the mechanical condition of the fuel transfer system prior to fuel offload. Following completion of the inspection the individuals performed a housekeeping activity in the canal which consisted of placing debris into a plastic bag. This activity created an airborne condition in the canal and refueling cavity causing the workers to become internally contaminated. Immediate corrective action included restricting access to the canal/cavity and restricting the individuals from the radiological control area until a dose assessment was performed. Also, root cause evaluations which included the use of a licensee independent review team were initiated. The root causes of this event were a programmatic breakdown in administrative barriers and poor radiological worker practices. Interim corrective actions consisted of additional radiological work restrictions, a restatement of management's expectations and increased senior management oversight during the core offload process. Long term Corrective actions included procedure and programmatic enhancements. Final dose assessments were adjusted for the Pu-241 contribution and did not exceed any federal dose limits.

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BACKGROUND INFORMATION

In the past, the inspection of the fuel transfer system was performed using impeded visual closeouts (i.e. respirators, hard hat divers). During the last refueling outage (1995), the fuel transfer cart was found jammed by a wrench and hammer that were left on the transfer cart track. Based on this it was decided, this outage, to clean the transfer canal of radioactive materials to the appropriate level (ALARA) to allow personnel to enter without respiratory protection so that a more thorough, on-location inspection could be performed of this area.

During August 1996 the transfer canal was decontaminated to contamination levels that would allow for inspections by plant personnel without respirators. From the period August 1996 until the November 2, 1996 event date, personnel have been able to enter this area in a single set of cloth protective clothing and without respirators. Up until the November 2, 1996 event, all transfer canal entry individuals received low doses (20-50 mRem), were free of contamination on exit counters, and generated no airborne activity.

EVENT DESCRIPTION

On Saturday November 2, 1996 at 0830 hours, with the plant in Mode 6 (refueling) a plant maintenance supervisor, and a contractor refueling manager entered the refueling cavity to perform an inspection of the reactor head and cavity area and then entered the fuel transfer canal area of the refueling cavity to inspect the mechanical condition of the fuel transfer cart system prior to fuel offload. Following completion of the inspection the individuals performed a housekeeping activity in the canal which consisted of placing approximately 3 pounds of debris into a plastic bag. This activity created an airborne condition in the canal and adjacent refueling cavity causing both workers to become internally contaminated.

When the two individuals exited the refueling cavity, one individual's electronic dosimetry was in alarm. A technician performed an initial survey of the bag of debris which read 20 Rem/hr on contact and 600 mR/hr at 12" as measured by an Eberline RO-2A. The workers exited the area and alarmed the containment personnel contamination monitors (PCMs). The two individuals were then directed to the main health physics control point where they were decontaminated and sent to the whole body counter for internal contamination assessment. External dose to the individuals by electronic dosimetry devices (confirmed by thermoluminescent dosimeters (TLDs) worn by these individuals) was 239 mrem to one individual and 157 mrem to the other worker.

Immediate corrective action included restricting access to the canal/cavity and restricting the individuals from the radiological control area until a dose assessment was performed. In addition, root cause evaluations were initiated which included the use of a licensee independent review team. On November 7, 1996 an initial dose assessment by the NRC indicated the workers may have received an exposure potentially exceeding federal limits due to internal contamination. Dose calculations by the licensee indicated a dose below any federal limits. A third party was contracted to perform an independent dose assessment. On November 8, 1996, at approximately 0930 hours, notification of this event was made in accordance with 10CFR20.2202(b)(2).

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Follow-up actions consisted of continued whole body counting of the contaminated workers, periodic counting of the air samples taken during the event to verify long-lived alpha activity, and a thorough investigation of the bag of debris for alpha to beta/gamma ratio assessment and makeup of the debris. Additionally, three fecal samples were collected for each of the two workers involved, and external dose calculations were performed using time motion studies from available electronic dosimetry printouts.

The concentrations of radioactive material and levels of radiation involved are provided in Attachment A. Each individual's final dose assessment is provided in Attachment B. The information required by 10CFR20.2203(b)(2) for each individual exposed is provided in Attachment C.

Final dose assessments were adjusted for the Pu-241 contribution and did not exceed any federal dose limits.

CAUSE OF EVENT

It was determined that the root causes of this event were a programmatic breakdown in administrative barriers (i.e. radiological controls, work planning and control, communication) and poor radiological worker practices.

SAFETY ASSESSMENT

This event is reportable under 10CFR20.2203(a)(2)(i) which states: "In addition to the notification required by 20.2202, each licensee shall submit a written report within 30 days after learning of any of the following occurrences: (2) Doses in excess of any of the following (i) the occupational dose limits for adults in 20.1201 (specifically 20.1201(a)(1)(ii))." 10CFR20.2202(b) (2) requires 24 hour notification upon discovery that an event may have caused, or threatens to cause, an individual to receive, in a period of 24 hours, an intake in excess of one occupational annual limit on intake.

Since the initial NRC dose assessment was made using conservative assumptions of nuclide form, path of intake, particle size, initial intake and ratios of alpha to beta-gamma activities a refined dose assessment was performed by a third party using the individual's whole body count (WBC) data and fecal sample results. Preliminary Haddam Neck internal dose calculations, as well as, refined calculations based on continued whole body count data, and the earliest fecal sample results indicated doses below any federal limits.

The potential doses and biological effects were discussed with the individuals involved. It should be noted that the doses calculated by the NRC are the maximum credible (worst case) condition and doses at this level would not place these individuals in a health threatened condition. They were also made aware of generated bounding dose assessments. Both individuals involved in this contamination event were restricted from receiving any additional exposure pending the results of the internal dose assessment. Senior management for the contractor was informed of the worker's restrictions and the possible issues regarding additional exposure at another utility.

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Final dose determinations for CEDE, DDE and CDE are complete and each individual was contacted and informed of their final dose assessment. Both individuals were relieved of RCA restrictions on January 11, 1997.

CORRECTIVE ACTION

Interim Actions

Interim corrective actions consisted of additional radiological work restrictions, the issuance of a restatement of management's expectations and increased senior management oversight during the core offload process. In addition, the immediate lessons learned from this event were communicated to the health physics technicians.

Initial Corrective Actions

Lessons learned from this event were provided to all Managers and supervisors on-site and Senior Management provided small group presentations to permit a more personal and open forum to address questions and concerns. A Stand Down Day was conducted on January 28, 1997 to focus on the areas of Radiological and Industrial Safety. Management has re-emphasized to site personnel that the primary function of the Health Physics Technicians is radiation protection.

Long Term Corrective Actions

The Independent Review Team, root cause investigations and the NRC inspection reports were evaluated and corrective actions were developed. These corrective actions were integrated into procedures and program upgrades relative to performing the following:

- Enhance RWP process to ensure proper evaluation of radiological risk and controls associated with all work in the radiological controlled areas,
- Identification of High Radiological Risk activities to ensure RWP controls are commensurate with Risk,
- Enhance alpha detection and monitoring through evaluation of new equipment, methodologies and trigger values,
- Establish minimum Radiological Controls for High Risk work to ensure oversight and job controls,
- Enhance internal dosimetry program requirements for rapid sample collection, and validation of final dose calculations by an independent reviewer,
- Increased challenges to the workers by the HP staff upon signing into the RCA in knowing the scope and conditions of their job. Any unplanned or changed job scopes will result in a stop work order,
- Documented pre-job briefings for all high risk evolutions and a limitation to only workers who have received the full brief to perform work on that RWP,
- Elimination of job site air sample checks to relax radiological controls,
- Include TEDE ALARA evaluations in the development of the RWPs.

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ADDITIONAL INFORMATION

Attachment A - Concentrations of Radioactive Material and Levels of Radiation

Attachment B - Final Dose Assessments(by job title)

Attachment C - CONFIDENTIAL - Exposed Individuals
(NRC Document Control Desk copy only)

The purpose of this supplement LER is to document that Haddam Neck has re-calculated the internal dose for the event to include an estimate of the dose contribution by Pu-241.

Commitments

None.

PREVIOUS SIMILAR EVENTS

LER 86-039-00, "Personnel Overexposure"

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ATTACHMENT A
CONCENTRATIONS OF RADIOACTIVE MATERIAL AND LEVELS OF RADIATION

1. Discussion:

The reactor cavity entry made by the Licensee maintenance supervisor and contractor refueling manager involved inspection of the reactor head flange, then inspection and housekeeping in the fuel transfer canal, where the event occurred. Concentrations of radioactive material and levels of radiation are provided for the reactor head, fuel transfer canal, and the bag of debris that was collected in the fuel transfer canal.

The concentrations of airborne radioactive material in the fuel transfer canal resulting from the event are determined based on analysis of the air samples taken during and after the event that were used for the initial NRC dose assessment. The concentrations of removable surface radioactive contamination, in Section 4, on the reactor head and in the fuel transfer canal are based on radiological surveys. The concentrations of radioactive material constituting the bag of debris are based on analysis of five samples of the debris material. The analyses of the air samples and debris samples were performed by a licensed vendor laboratory using gamma and alpha spectroscopy. The Tables in Sections 3 and 5, below, include only those isotopes with measured activity by spectroscopy. All other isotopes were less than the minimum detectable activity. The strontium values as reported are provided.

The levels of radiation, in Section 6, are based on radiological surveys of the reactor head and fuel transfer canal, the electronic dosimetry system exposure history (integrated accumulated dose and differentiated dose rate in one minute intervals) retrieved from the system for the electronic dosimeter worn by the Licensee maintenance supervisor, and a TLD study of the bag of debris.

2. Summary:

Since the air sample beta-gamma to alpha ratios used for initial internal dose assessments were applied to the known concentration of Co-60 from whole body counts, they should have been based on the known concentration of Co-60 on the air samples (from licensee gamma spectroscopy), instead of the sample gross beta-gamma counts. This would have resulted in ratios of 100.7 to 1 and 98 to 1 versus the gross ratios of 88.4 to 1 and 81.3 to 1.

The results of vendor gamma and alpha spectroscopy analysis of the air samples presented in Section 3, indicates that the gross alpha analysis is conservative, with actual Co-60 to alpha ratios of 146 to 1 and 122 to 1 respectively. The average Co-60 to alpha ratio based on gamma and alpha spectroscopy analysis of the five debris samples is 190 to 1 (refer to Section 5).
The dry active waste (DAW) conservative scaling factor for Co-60 to transuranics for the last cycle indicates a ratio of 147 to 1 which matches well with the results above.

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3. Concentrations of Airborne Radioactive Material:

a. Fuel Transfer Canal Job Coverage, 11/2/96: Air Sample Number 110201; Sample Time = 35 minutes (0830-0905); Sample Volume = 9.91 E5 ml

Licensee Analysis:

Type of Analysis	Gross Beta	Gamma Spectroscopy (Co-60)	Gross Alpha
Activity (µCi/ml)	7.4E-9	8.21E-9	8.15E-11
DAC Fraction	0.74	0.82	20.38*
		Co-60 to Gross Alpha Ratio	101-1

* Stochastic Am-241 DAC 4.0 E-12 µCi/ml

Vendor Laboratory Analysis:

DAC Used	Stochastic		Non-Stochastic					
Isotope (Class)	C0-60 (Y)	Cs-137 (D)	Gross Alpha	Pu-238 (W)	Pu-239 (W)	Am-241 (W)	Cm-242 (W)	Cm-244 (W)
Activity (µCi/ml)	7.23E-9	7.96E-11	5.1E-11	1.2E-11	5.0E-12	2.0E-11	1.6E-12	1.1E-11
DAC Fraction	0.72	0.0013	12.75	4.0	1.67	6.67	0.016	2.2
Co-60 to Transuranic Isotope Activity Ratios (µCi/ml)			142-1	603-1	1,446-1	362-1	4,519-1	657-1

Total Stochastic DAC Fraction; Vendor Analysis = 0.721
Total Stochastic DAC Fraction; Licensee Analysis = 0.82
Total Non-Stochastic DAC Fraction; Vendor Analysis = 14.556 (actual activities)
Total Non-Stochastic DAC Fraction; Licensee Analysis = 20.73 (gross alpha counts)
Co-60 to Gross Alpha Ratio: Licensee = 100.7-1; Vendor = 142-1

Actual Alpha Activity by Spectroscopy = 4.96E-11 µCi/ml
Co-60 to Actual Alpha Activity Ratio = 146-1

Vendor Laboratory Analysis:

Type of Analysis	Sr-89	Sr-90
Activity(µCi)	< 3. E-05	<2. E-05
Concentration (uCi/ml)	< 3.03 E-11	<2.02 E-11

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b. Reactor Cavity, Vacuum Stud Holes, Job Coverage, 11/2/96 (after the event):
Air Sample Number 110203; Sample Time = 15 minutes (0910-0925);
Sample Volume = 4.25 E3 ml

Licensee Analysis:

Type of Analysis	Gross Beta	Gamma Spectroscopy (Co-60)	Gross Alpha
Activity (µCi/ml)	3.0E-8	3.47E-8	3.54E-10
DAC Fraction	3.0	3.47	88.5*
		Co-60 to Gross Alpha Ratio	98-1

* Stochastic Am-241 DAC 4.0 E-12 µCi/ml

Vendor Laboratory Analysis:

DAC Used	Stochastic		Non-Stochastic					
Isotope (Class)	C0-60 (Y)	Cs-137 (D)	Gross Alpha	Pu-238 (W)	Pu-239 (W)	Am-241 (W)	Cm-242 (W)	Cm-244 (W)
Activity (µCi/ml)	2.95E-8	2.5E-10	2.5E-10	6.2E-11	2.5E-11	9.6E-11	7.3E-12	5.2E-11
DAC Fraction	2.95	0.025	62.5	20.67	8.33	32.0	0.073	10.4
Co-60 to Transuranic Isotope Activity Ratios (µCi/ml)			118-1	471-1	1,168-1	304-1	4,000-1	562-1

Total Stochastic DAC Fraction; Vendor Analysis = 2.975
Total Stochastic DAC Fraction; Licensee Analysis = 3.47
Total Non-Stochastic DAC Fraction; Vendor Analysis = 71.47 (actual activities)
Total Non-Stochastic DAC Fraction; Licensee Analysis = 85.5 (gross alpha counts)
Co-60 to Gross Alpha Ratio: Licensee = 98-1; Vendor = 118-1

Actual Alpha Activity by Spectroscopy = 2.42E-10 µCi/ml
Co-60 to Actual Alpha Activity Ratio = 122-1

Vendor Laboratory Analysis:

Type of Analysis	Sr-89	Sr-90
Activity(µCi)	< 4. E-05	1.8 E-05
Concentration (uCi/ml)	< 9.4 E-09	4.2 E-09

c. Fuel Transfer Canal follow-up Decon, Job Coverage, 11/2/96 (after the event):
Air Sample Number 110212; Sample Time = 20 minutes (1625-1645);
Sample Volume = 5.66 E5 ml

Licensee Analysis:

Type of Analysis	Gross Beta	Gamma Spectroscopy (Co-60)	Gross Alpha
Activity in µCi/ml	2.9E-8	2.99E-8	9.04E-11
DAC Fraction	2.9	2.99	22.6*
		Co-60 to Gross Alpha Ratio	331-1

* Stochastic Am-241 DAC 4.0 E-12 µCi/ml

Vendor Laboratory Analysis:

DAC Used	Stochastic		Non-Stochastic					
Isotope (Class)	C0-60 (Y)	Cs-137 (D)	Gross Alpha	Pu-238 (W)	Pu-239 (W)	Am-241 (W)	Cm-242 (W)	Cm-244 (W)
Activity (µCi/ml)	2.67E-8	2.6E-10	3.7E-11	6.8E-12	1.6E-12	1.3E-11	2.1E-12	5.4E-12
DAC Fraction	2.67	0.0044	9.25	2.27	0.53	4.33	0.021	1.08
Co-60 to Transuranic Isotope Activity Ratios (µCi/ml)			722-1	3,927-1	16,688-1	2,054-1	12,714-1	4,944-1

Total Stochastic DAC Fraction; Vendor Analysis = 2.674
Total Stochastic DAC Fraction; Licensee Analysis = 2.99
Total Non-Stochastic DAC Fraction; Vendor Analysis = 8.23 (actual activities)
Total Non-Stochastic DAC Fraction; Licensee Analysis = 22.6 (gross alpha counts)
Co-60 to Gross Alpha Ratio: Licensee = 331-1; Vendor = 722-1

Actual Alpha Activity by Spectroscopy = 2.89 E-11 µCi/ml
Co-60 to Actual Alpha Activity Ratio = 924-1

Vendor Laboratory Analysis:

Type of Analysis	Sr-89	Sr-90
Activity(µCi)	< 2. E-05	< 8. E-06
Concentration (uCi/ml)	< 3.53 E-11	< 1.41 E-11

4. Concentrations of Removable Surface Radioactive Contamination:

a. Reactor Cavity Floor - Average 113K dpm/100 cm^2 beta-gamma; Maximum 250K dpm/100 cm^2
- Average 1.6K dpm/100 cm^2 alpha; Maximum 3K dpm/100 cm^2

b. Reactor Head Flange - Range from 20K to 40K dpm/100 cm^2 beta-gamma
- Less than 20 dpm/100 cm^2 alpha

c. Fuel Transfer Canal - Average 28.4 mrad/100 cm^2 beta; Maximum 80 mrad/100 cm^2
- Average 18.6K dpm/100 cm^2 alpha; Maximum 35K dpm/100 cm^2

5. Radioactivity Concentrations of Debris Material:

a. Sample 2-C

Vendor Laboratory Analysis:

Isotope	Co-60	Cs-137	Pu-238	Pu-239	Am-241	Cm-242	Cm-244
Activity (µCi/gm)	6.23E+0	3.79E-2	8.3E-3	2.2E-3	1.0E-2	2.8E-4	6.0E-3
Co-60 to Transuranic Isotope Activity Ratios			751-1	2,832-1	623-1	22,250-1	1,038-1

Actual Co-60 Activity to Total Alpha Activity Ratio = 233-1

Vendor Laboratory Analysis:

Type of Analysis	Sr-89	Sr-90
Activity(µCi/gm)	< 5. E-03	< 1.7 E-02

b. Sample 5-A

Vendor Laboratory Analysis:

Isotope	Co-60	Cs-137	Mn-54	Pu-238	Pu-239	Am-241	Cm-242	Cm-244
Activity (µCi/gm)	5.48E+1	1.21E+0	4.1E-1	1.2E-1	5.1E-2	1.3E-1	4.8E-4	7.3E-2
Co-60 to Transuranic Isotope Activity Ratios				457-1	1,075-1	422-1	114K-1	751-1

Actual Co-60 Activity to Total Alpha Activity Ratio = 146-1

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Vendor Laboratory Analysis:

Type of Analysis	Sr-89	Sr-90
Activity(μCi/gm)	< 1.2 E-02	< 6.5 E-01

c. Sample 5-D

Vendor Laboratory Analysis:

Isotope	Co-60	Cs-137	Pu-238	Pu-239	Am-241	Cm-242	Cm-244	U-234
Activity (μCi/gm)	2.89E+1	5.6E-1	6.8E-2	2.5E-2	9.1E-2	4.0E-4	5.6E-2	3.1E-5
Co-60 to Transuranic Isotope Activity Ratios			425-1	1,156-1	318-1	72K-1	516-1	932K-1

Actual Co-60 Activity to Total Alpha Activity Ratio = 120-1

Vendor Laboratory Analysis:

Type of Analysis	Sr-89	Sr-90
Activity(μCi/gm))	< 5. E-03	< 4.9 E-01

d. Sample 8-B

Vendor Laboratory Analysis:

Isotope	Co-60	Cs-137	Pu-238	Pu-239	Am-241	Cm-242	Cm-244
Activity (μCi/gm)	1.16E+2	3.4E+0	1.9E-1	1.0E-1	3.3E-1	9.4E-4	1.5E-1
Co-60 to Transuranic Isotope Activity Ratios			611-1	1,160-1	352-1	123K-1	773-1

Actual Co-60 Activity to Total Alpha Activity Ratio = 150-1

Vendor Laboratory Analysis:

Type of Analysis	Sr-89	Sr-90
Activity(μCi/gm)	< 1. E-02	4.8 E-01

e. Sample 8-D

Vendor Laboratory Analysis:

Isotope	Co-60	Cs-137	Mn-54	Pu-238	Pu-239	Am-241	Cm-242	Cm-244	U-234
Activity (μCi/gm)	1.32E+2	1.3E-0	4.6E+0	1.4E-1	6.7E-2	1.6E-1	5.7E-4	7.0E-2	4.1E-5
Co-60 to Transuranic Isotope Activity Ratios				943-1	1,970-1	825-1	232K-1	1,886-1	322K-1

Actual Co-60 Activity to Total Alpha Activity Ratio = 302-1

Vendor Laboratory Analysis:

Type of Analysis	Sr-89	Sr-90
Activity(μCi/gm)	< 6. E-03	5.0 E-01

6. Levels of Radiation:

a. Reactor Head - General area radiation levels at waist level around the base of the reactor head averaged 113 mRem/hr, with an average general area level of 580 mRem/hr at waist level on the reactor head flange. The average dose rate recorded by the electronic dosimeter worn by the Licensee maintenance supervisor during the reactor head flange inspection was 429 mRem/hr.

b. Fuel Transfer Canal - The average radiation levels if an individual is standing on the fuel transfer cart track are 333 mRem/hr on contact with the track (maximum 800 mRem/hr) and 233 mRem/hr at waist level (maximum 600 mRem/hr). The average radiation levels if an individual is standing on the floor on either side of the transfer cart track are 361 mRem/hr on contact with the floor and 280 mRem/hr at waist level. The maximum dose rates on either side of the track are 25 Rem/hr contact and 8 Rem/hr at waist level from a small localized spot on the floor. These dose rates were not included in the averages due to the highly localized nature of the radiation field.

c. Bag of Debris - As noted in the EVENT DESCRIPTION, the initial survey of the bag of debris indicated 20 Rem/hr on contact and 600 mRem/hr at 12" as measured by an Eberline RO-2A. After the event, a TLD study was conducted of the bag of debris collected in the fuel transfer canal to calculate average gamma dose rates per minute at contact, 1 foot and 2 feet. The results of the study are as follows:

- Contact (extremity TLD ring): - 76.65 mRem/minute
  Contact (whole body TLD): - 61.28 mRem/minute
- 1 Foot (side of bag; whole body TLD): - 10.48 mRem/minute
- 1 Foot (bottom of bag; whole body TLD: - 31 mRem/minute
- 2 Feet (side of bag; whole body TLD): - 5.14 mRem/minute

Three of eight whole body TLDs on contact with the bag indicated an average beta dose rate of 1,083 mrad/hr. The two whole body TLDs positioned one foot from the bottom of the bag indicated an average beta dose rate of 429 mrad/hr.

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ATTACHMENT B
FINAL DOSE ASSESSMENTS

Dose for Event - Licensee Maintenance Supervisor

DOSES (in rem)		
DEEP DOSE EQUIVALENT	(DDE)	0.385
EYE DOSE EQUIVALENT TO LENS OF EYE	(LDE)	0.397
SHALLOW DOSE EQUIVALENT, WHOLE BODY	(SDE,WB)	0.399
SHALLOW DOSE EQUIVALENT, MAX EXTREMITY	(SDE,ME)	1.165
COMMITTED EFFECTIVE DOSE EQUIVALENT	(CEDE)	0.290
COMMITTED DOSE EQUIVALENT, MAXIMALLY EXPOSED ORGAN	(CDE)	3.537
TOTAL EFFECTIVE DOSE EQUIVALENT (DDE + CEDE)	(TEDE)	0.675
TOTAL ORGAN DOSE EQUIVALENT, MAX ORGAN (DDE + CDE)	(TODE)	3.922
COMMENTS: Maximally Exposed Organ is the Bone Surface. Individual has no prior exposure during 1996.		

Dose for Event - Contractor Refueling Manager

DOSES (in rem)		
DEEP DOSE EQUIVALENT	(DDE)	0.213
EYE DOSE EQUIVALENT TO LENS OF EYE	(LDE)	0.213
SHALLOW DOSE EQUIVALENT, WHOLE BODY	(SDE,WB)	0.213
SHALLOW DOSE EQUIVALENT, MAX EXTREMITY	(SDE,ME)	0.441
COMMITTED EFFECTIVE DOSE EQUIVALENT	(CEDE)	0.497
COMMITTED DOSE EQUIVALENT, MAXIMALLY EXPOSED ORGAN	(CDE)	6.132
TOTAL EFFECTIVE DOSE EQUIVALENT (DDE + CEDE)	(TEDE)	0.710
TOTAL ORGAN DOSE EQUIVALENT, MAX ORGAN (DDE + CDE)	(TODE)	6.345
COMMENTS: Maximally Exposed Organ is the Bone Surface. Individual has 260 mrem DDE, 260 mrem SDE,WB and 183 mrem SDE, ME from other licensees during 1996 which are not included in the totals above.		

Haddam Neck has re-calculated the internal dose for the event to include an estimate of the dose contributed by Pu-241.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)					PAGE (3)
		YEAR	SEQUENTIAL NUMBER			REVISION NUMBER	
Haddam Neck	05000213	96	--	030	--	02	15 of 15

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

ATTACHMENT C
INFORMATION REQUIRED BY 10CFR20.2203(b)(2)
(EXPOSED INDIVIDUALS)

The names, social security account numbers and dates of birth of the exposed individuals is confidential information and was submitted as a separate and detachable part of the original LER in accordance with 10CFR20.2203(b)(2). The original LER that was forwarded to the NRC Document Control Desk included this information.

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