SEP 25 1986

MEMORANDUM FOR: William F. Kane, Director, Division of Reactor Projects

THROUGH: Harry B. Kister, Chief, Reactor Projects Branch No. 1, DRP
Jack Strosnider, Chief, Reactor Projects Section No. 1B, DRP

FROM: Roy L. Fuhrmeister, Reactor Engineer, RPB No. 1, DRP

SUBJECT: PILGRIM RESIN RELEASE IN JUNE 1982

During recent public meetings in the vicinity of Plymouth, Massachusetts there have been numerous references to the resin release at Pilgrim in June of 1982. These references have most often been made by Mr. Abbott of the Plymouth County Nuclear Information Group in the manner of "the accident in 1982". A great deal has been made of the increased dose measured on a particular Thermo-Luminescent Dosimeter (TLD) during the summer and autumn of that year. In order to determine if there was any credence to the claims that the Pilgrim resin release contaminated the environment as far away as New hampshire, a number of TLD data points were extracted from the NUREG 0837 series and plotted on a common time line. An explanation of the data points selected from random plants, a tabulation of the data, and a plot on the common time axis are attached.

It is interesting to note that during the first half of 1984, while the plant was shut down, TLD 1 from Pilgrim showed striking increases in the dose. Also of note is the fact the TLD 49 from Pilgrim, located in Weymouth, Massachusetts, shows a consistently higher dose than TLD 13, which is only 0.7 miles from the plant.

As a check on TLD 1, the plant operation time-line and quarterly release data are also included in the figure. No correlation with plant activities is readily apparent. In fact, the high reading in early 1984, with the plant shut down and no releases being made is inconsistent with Mr. Abbott's contentions.

In conclusion, it can be seen that the off-site dose in the vicinity of Pilgrim Nuclear Power Station followed the general trend of the other sites in the Northeastern United States. This trend includes a significant drop in doses during the first quarter of 1982. This drop, if narrowly construed, could lead one to the conclusion that the second quarter 1982 dose was significantly higher. This would be an erroneous conclusion, since second quarter 1982 dose is lower than the fourth quarter 1981 dose. In general, it appears that from mid-1981 to mid-1983, Eastern Massachusetts dose data followed the decreasing trend evidenced across the Northeast United States. In fact the doses in Eastern Massachusetts, including those measured around the Pilgrim site (with the exception of TLD No. 1 which is exposed to turbine "shine"), were on the order of 70% of expected natural background throughout the period. First quarter 1983 doses show a dramatic drop in Eastern Massachusetts, despite a major release from Pilgrim during that time period (13,200 ci, higher than the 1982 release). It is also worthy of note that with the exception of the third quarter 1981, Weymouth, Massachusetts doses

8805240433 880429
PDR ADOCK 05000293
U PDR

were higher than those recorded only 0.7 miles from the site. This suggests factors other than operation of and releases from Pilgrim are affecting the results of the environmental monitoring program. This also shows that the 1982 resin release and higher dose readings are strictly coincidental.

The dose readings on TLD 1 are in the range of 1 to 3 times the expected background levels for the area. The cause of the elevated readings was originally thought to be "turbine shine". The 1984 data do not support that conclusion, and further information on plant activities in 1984 is being developed. Particular interest is being paid to temporary on-site storage of materials removed during the recirculation piping changeout.

Attachment 6 shows typical expected doses.

Roy L. Fuhrmeister

Roy L. Fuhrmeister
Reactor Engineer, RPS 1B
DRP

Attachments:
1. Explanation of Data Points
2. Tabulation of Data
3. Plot of TLD Exposure Data 1981-1984
4. PNO-I-82-42
5. PNO-I-82-42a
6. Extract from Health Physics and Radiological Health Handbook

cc:
H. Kister
J. Strosnider
L. Doerflein
M. McBride

Plot of TLD Exposure Data for 1981 - 1984
Data extracted from NUREG 0837 Series

Stations used are identified as follows

ID	Plant	Azimuth	Distance	Location
1	Pilgrim	233°	0.09 mi	Pilgrim Overlook on-site
13	Pilgrim	146°	0.7 mi	Rocky Hill Road
49	Pilgrim	301°		Weymouth, Massachusetts
VY 19	Vermont Yankee	[illegible]°	13.8 mi	Wilmington, Vermont
PI 1	Prairie Island	3[illegible]°	19.3 mi	Goodhue, Minnesota
TMI [illegible]	Three Mile Island	[illegible]°	19.7 mi	Fair Haven Road State 1

Stations 49, VY 19, PI 1, and TMI [illegible] on basis of remote location and no influence of plant on environment in order to get an indication of Regional and national trends

Station 1 was chosen as representative of turbine shine

Station 13 was chosen as representative of closest road to site

All data are normalized to a "standard quarter" of 90 days

* indicates data not available due to lost/damaged TLD

† indicates data not available due to report missing

MAR 22 '88 11:16 PILGRIM NRC RESIDENT

station 1 2

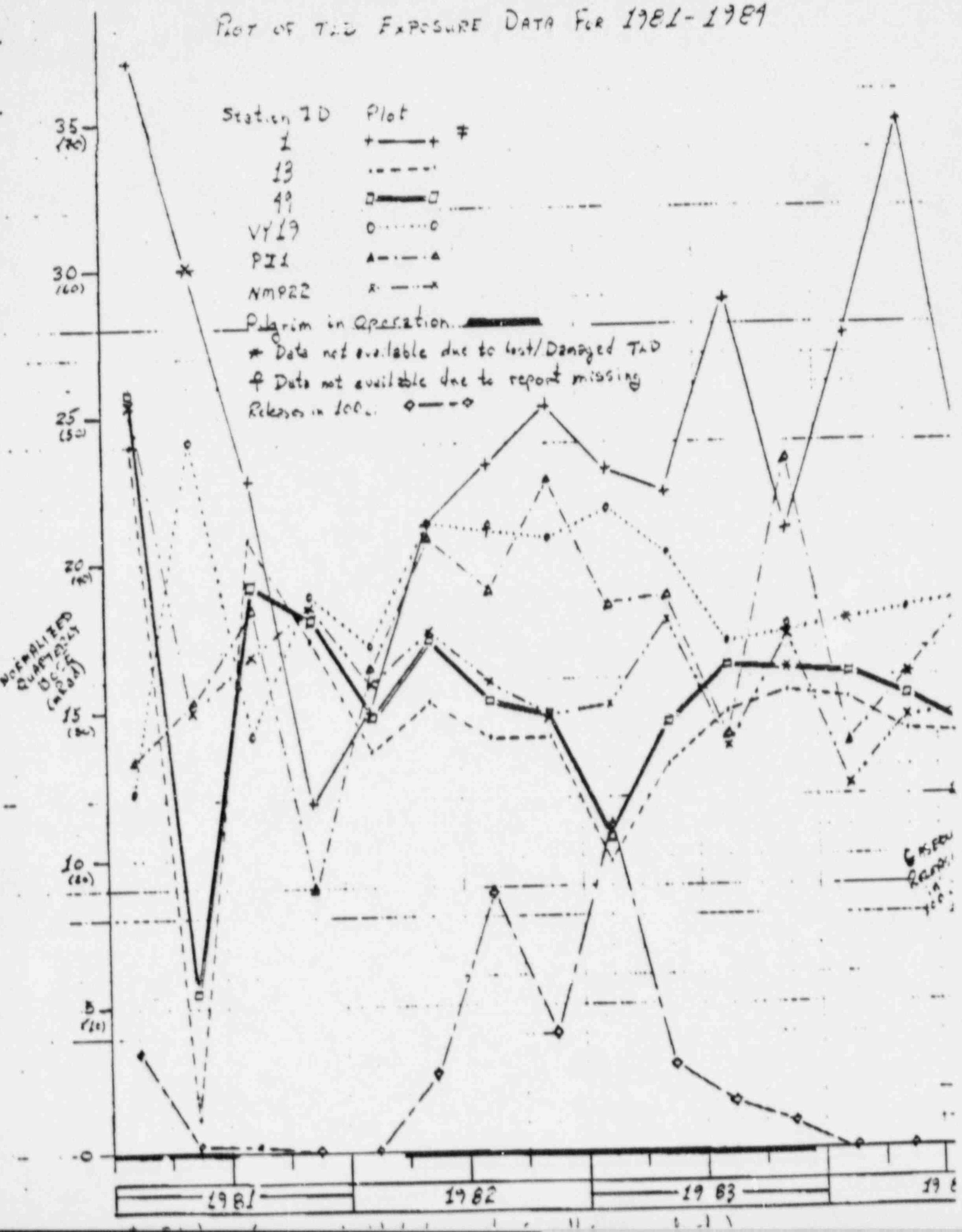
Tabulation of Results

yr	qtr	station	mR/std qtr
'91	1	1	21.1
		13	29.0
		49	25.8
		9/9 19	12.2
		PL 1	13.3
		RMT P22	25.5
	2	1	*
		13	1.2
		49	5.4
		9/9 19	21.1
		PL 1	15.2
		RMT P22	15.0
	3	1	45.5
		13	20.7
		49	19.2
		9/9 19	14.1
		PL 1	18.5
		RMT P22	16.7
	4	1	23.5
		13	17.4
		49	17.9
		9/9 19	18.9
		PL 1	6.9
		RMT P22	18.5

yr	qtr	station	mR/std qtr
'92	1	1	29.9
		13	13.6
		49	14.9
		9/9 19	17.1
		PL 1	16.4
		RMT P22	15.9
	2	1	43.5
		13	15.2
		49	17.4
		9/9 19	11.3
		PL 1	20.9
		RMT P22	17.6
	3	1	46.1
		13	14.1
		49	15.3
		9/9 19	4
		PL 1	19.0
		RMT P22	15.9
	4	1	50.4
		13	14.1
		49	14.3
		9/9 19	10.9
		PL 1	22.7
		RMT P22	14.9

Attachment 2

yr	qtr	station	mR/std qtr
'83	1	1	46.1
		13	9.8
		49	10.4
		OYG 19	21.7
		PL 1	18.5
		MMP 22	15.3
	2	1	44.9
		13	13.1
		49	14.5
		OYG 19	20.3
		PL 1	18.8
		MMP 22	13.0
	3	1	57.7
		13	14.8
		49	16.4
		OYG 19	17.1
		PL 1	14.0
		MMP 22	13.7
	4	1	42.1
		13	16.5
		49	*
		OYG 19	f
		PL 1	23.4
		MMP 22	17.6
'84	1	1	55.4
		13	15.3
		49	16.1
		OYG 19	*
		PL 1	15.8
		MMP 22	12.3
	2	1	63.9
		13	14.7
		49	15.4
		OYG 19	19.[illegible]
		PL 1	*
		MMP 22	14.6
	3	1	45.3
		13	14.1
		49	14.2
		OYG 19	*
		PL 1	18.6
		MMP 22	14.8


Plot of TLD Exposure Data For 1981-1984
Station ID
Plot
1
13
49
VY19
PI1
NMP22
Pilgrim in Operation
* Data not available due to lost/Damaged TLD
# Data not available due to report missing
Releases in 100...
NORMALIZED QUARTERLY DOSE (mrad)
35 (70)
30 (60)
25 (50)
20 (40)
15 (30)
10 (20)
5 (10)
0
1981
1982
1983

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE

This preliminary notification constitutes EARLY notice of events of possible safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by the staff.

Facility:	Pilgrim Nuclear Power Station Plymouth, Massachusetts DN 50-293	Licensee Emergency Classification:	
		___	Notification of Unusual Event
		___	Alert
		___	Site Area Emergency
		___	General Emergency
		X	Not Applicable

Subject: RELEASE OF SPENT RESIN

At approximately 1300 on June 11, 1982 spent resin was found on the ground near the Turbine Building. Subsequent surveys identified contamination of the roofs of the Turbine, Reactor, Off-Gas and Re-Tube Buildings. Contamination was also found on the ground within the site controlled areas. Contamination levels ranged from [illegible] dpm/100 cm^2 with maximum contamination of up to 100,000 dpm/100 cm^2. Gamma analysis of the resin identified primarily long lived radionuclides (Co-60, [illegible], Cs-134 and Mn-54).

No contamination was identified off-site or in storm drains. All personnel are being frisked prior to exiting the site and no personnel contamination has been [illegible].

The resin may have been released through the reactor building vent duct which exhausts to the atmosphere at an elevation of approximately 100 ft. The licensee has found approximately 10 ft^3 of resin in the Standby Gas Treatment System inlet plenum. The source of the resin is being investigated. Three radiation specialists have been dispatched to the site to evaluate the radiological aspects of the occurrence.

Media interest is expected due to public interest in the facility. The licensee is considering issuing a press release. The NRC does not plan to issue a press release but will respond to media inquiries. The Commonwealth of Massachusetts has been informed.

This PN is current as of 4:45 P.M., June 11, 1982.

CONTACT: Elsasser 488-1235 Brunner 488-1225

DISTRIBUTION:

H. St. ____	MNBB ____	Phillips 5:25 6/11/82	E/W ____	Willste ____	Mail: ADM:DMB
Chairman Palladino	EDO	NRR	IE	NMSS	
Comm. Gilinsky	PA		OIA	RES	
Comm. Ahearne	MPA		AEOD		
Comm. Roberts	ELD				
ACRS		Air Rights	INPO ____		
SECY		SP	NSAC		
CA					
PDR	Regional Offices				

~~8206170371~~ 1p.

Fk Elsasser

[illegible] 5:42 6/11

[illegible] 5:42 pm

RI Resident [illegible]
FDA [illegible]
PNO-I-82-[illegible]

DCS No '0293-820611
Date: June 14, 1982

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE--PNO-I-82-42A

This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by the Region I staff on this date.

Facility: Pilgrim Nuclear Power Station
Plymouth, Massachusetts
DN 50-293

Licensee Emergency Classification:
___ Notification of Unusual Event
___ Alert
___ Site Area Emergency
___ General Emergency
_x_ Not Applicable

Subject: RELEASE OF SPENT RESIN (UPDATE PNO-I-82-42)

Surveys of the entire site within the protected area and surveys of selected areas of the licensee controlled area were made within 3 hours of the identification of the spent resin release. The licensee's onsite surveys identified two contaminated pavement areas which were barricaded and posted. Surveys confirmed contamination of the Turbine, Administration Augmented Off-Gas and Re-Tube Building roofs. The Reactor Building Roof was found to be free of contamination. The licensee's offsite survey included surveys of cars, parking lots, shorefront, and security access areas. No contamination was identified. Routine environmental air samples covering the period June 1-15, 1982 were counted. Nothing unusual was identified. Because of the size and weight of the resins, no offsite airborne release of the beads appears to have occurred. This was confirmed by air samples collected during clean-up of the contaminated pavement areas which when counted indicated background and the identification of resins only on roof-tops under the Reactor Buiding Vent. Prelimi ary samples of storm drain residue have been counted with no contamination identified. All contaminated ventilation ducts have been vacuumed clean. A duct surveillance program has been established to identify any additional resin accumulation.

The licensee believes the resin entered the ventilation ducts from the condensate demineralizer system during resin backwashing via the Cation Regeneration Tank Vent. In addition, resin from defective condensate demineralizer vent valves may have also been released prior to their repair during the September 1981 -March 1982 refueling outage. The resin appears to have been released from the Reactor Building Ventilation Exhaust System which vents above the reactor building roof, prior to the repair of defective filters in this system in September 1981.

The licensee has suspended all transfer operations which could result in further resin releases to ventilation ducts and has initiated additional environmental sampling. The licensee's actions were monitored by three Region I Radiation Specialists throughout the weekend. Region I will issue a Confirmatory Action Letter to address planned licensee corrective actions. The licensee is continuing to review the source and cause to determine what permanent corrective action will be needed. The Resident Inspectors are closely following licensee actions concerning this event.

Media interest has occurred. The licensee has responded to media inquiries but does not plan to issue a press release. The NRC will respond to media inquiries but does not plan to issue a press release.

This PN is current as of 11:00 a.m., June 14, 1982.

8206210607

Region I Form 83
(Rev. March, 1982)

IE34

The Health Physics and Radiological Health Handbook

016

Table 1.5. Summary of average annual per capita doses to whole U.S. population

Source	Ave. per capita dose (mrem/year)	
Natural background		
Cosmic	31	
Terrestrial	68	
Tech. inhanced	4	
Sub-total		103
Man-made		
Medical		
X-ray	77	
Nuc. Med.	14	
Sub-total		91
Nuclear weapons	4-5	
Nuclear power	< 1	
Consumer products	0.5-1.5	
Sub-total		~ 8
Total		~ 200

Table 1.6. U.S. general population collective dose estimates - 1978

(From Biologic Effects of Ionizing Radiation. Report of the Science Work Group of the Interagency Task Force on Radiation, Department of Health, Education and Welfare, June, 1979)

Source	Person-rems per year (in thousands)
Natural background	20,000
Technologically enhanced	1,000
Healing arts	18,000
Nuclear weapons	
Fallout	1,000-1,600
Weapons development, testing and production	0.165
Nuclear power	56
Consumer products	6

Table 1.7. Annual per capita dose from natural radioactivity

Source	Variability	Dose[b] (mrem/year)
Cosmic	Average[a]	31
	Rock mountain states	60 > 80
	Jet flight - trans continental	2.5/Trip
Terrestrial (external)	Average[a]	40
	Colorado	75-140
(internal)	Average[a] (gonads)	28
	Lung	100-450
Tech. inhanced	Average[a]	4
Total		103

[a]Average whole-body dose to the whole population.

[b]Uncorrected for shielding of structures (reduce cosmic by 10% and terrestrial by 20%). Self-shielding by body further reduces dose.

Table 1.8. Radiation doses from medical radiation[a]

Source	Mean active bone-marrow dose		
	mrem/exam		Ave. per capita dose[b] (mrem/year)
Diagnostic x-rays			
Chest x-ray	10		
Upper GI	500		
Lower GI	900		
Skull	80		
Full mouth (dental)	9		
Sub-total			77[c]
Radio pharmaceuticals	Dose (mrem) to organ Specified/exam		
I^{133} (function)	Thyroid	5000	
	Whole body	30	
Tc^{99M}	Whole body	180	
Xe^{133}	Whole body	5	
	Whole-body equivalent to whole population	14	
Total			91

[a]Doesn't include therapy

[b]Based on whole population (exposed and unexposed)

[c]GSD is 20 mrem/year (GSD is the Genetically Scientific Dose)

ENCLOSURE 4
TO QUESTION 7

NUREG-0837
Vol. 2, No. 3

NRC TLD Direct Radiation Monitoring Network

Progress Report
July-September 1982

U.S. Nuclear Regulatory Commission
NRC Region I

F. Costello, T. Thompson, L. Cohen

PILGRIM

TLD DIRECT RADIATION ENVIRONMENTAL MONITORING
FOR THE PERIOD 820630-821008 101 DAYS
FIELD TIME 820707-821005 91 DAYS

NRC STATION	LOCATION AZIMUTH/DIST (deg.)	(mi.)	GROSS EXPOSURE(mR) +- Std. Dev.	EXPOSURE RATE mR/Std.Qtr. +- Std. Dev.
001	288	0.10	52.1 +- 1.3	46.4 +- 1.1
002	310	0.20	19.6 +- 1.1	17.5 +- .9
005	289	0.70	19.4 +- .2	17.3 +- .2
006	261	1.70	17.9 +- .9	15.9 +- .8
007	270	0.50	19.4 +- .0	17.3 +- .0
008	247	0.30	19.2 +- .3	17.1 +- .3
009	224	0.30	17.6 +- .3	15.7 +- .2
010	205	0.30	27.1 +- 1.2	24.1 +- 1.1
011	184	0.03	22.2 +- .3	19.8 +- .2
012	159	0.40	23.2 +- .5	20.6 +- .4
013	146	0.70	15.8 +- .5	14.1 +- .4
014	155	1.00	17.4 +- .0	15.5 +- .0
016	136	1.30	20.3 +- .2	18.1 +- .2
018	212	0.60	19.7 +- .0	17.6 +- .0
019	232	1.00	14.6 +- .3	13.0 +- .3
021	256	1.60	16.8 +- .1	15.0 +- .1
022	130	2.50	15.4 +- .5	13.7 +- .5
023	146	3.40	15.3 +- .8	13.6 +- .7
025	168	1.50	15.6 +- .3	13.9 +- .2
026	180	1.30	14.9 +- .2	13.3 +- .2

PILGRIM

TLD DIRECT RADIATION ENVIRONMENTAL MONITORING
FOR THE PERIOD 820630-821008 101 DAYS
FIELD TIME 820707-821005 91 DAYS

NRC STATION	LOCATION AZIMUTH/DIST (deg.)	(mi.)	GROSS EXPOSURE(mR) +- Std. Dev.	EXPOSURE RATE mR/Std.Qtr. +- Std. Dev.
027	231	1.80	16.3 +- .7	14.5 +- .7
030	153	2.20	17.1 +- .1	15.2 +- .1
031	179	2.50	15.2 +- .0	13.5 +- .0
032	217	2.60	13.9 +- .4	12.4 +- .4
033	234	2.50	16.0 +- .2	14.2 +- .1
037	264	4.20	17.9 +- .1	15.9 +- .1
039	155	5.30	13.3 +- .2	11.9 +- .2
040	272	4.60	16.2 +- .1	14.5 +- .1
043	291	5.60	18.2 +- .6	16.2 +- .5
045	-	-	13.9 +- .0	12.3 +- .0
047	301	26.2	18.0 +- .0	16.0 +- .0
048	301	26.2	17.7 +- .2	15.8 +- .1
049	301	26.2	17.2 +- .1	15.3 +- .1

COMMENTS:

STATION 1 IS ON LICENSEE PROPERTY (PILGRIM OVERLOOK AREA).
ACCESS IS CONTROLLED

PILGRIM
FOR THE PERIOD 820630-821008 101 DAYS
TLD DIRECT RADIATION ENVIRONMENTAL MONITORING

AZIMUTH (deg.)	AVER. EXPOSURE +- Std.Dev. (mR/Std.Qtr.)	# IN GROUP
348.75-11.25 (N)	NO DATA+-NO DATA	0
11.25-33.75 (NNE)	NO DATA+-NO DATA	0
33.75-56.25 (NE)	NO DATA+-NO DATA	0
56.25-78.75 (ENE)	NO DATA+-NO DATA	0
78.75-101.25 (E)	NO DATA+-NO DATA	0
101.25-123.75(ESE)	NO DATA+-NO DATA	0
123.75-146.25(SE)	14.9 +- 2.2	4
146.25-168.75(SSE)	15.4 +- 3.3	5
168.75-191.25(S)	15.5 +- 3.7	3
191.25-213.75(SSW)	20.9 +- 4.7	2
213.75-236.25(SW)	14.0 +- 1.3	5
236.25-258.75(WSW)	16.0 +- 1.5	2
258.75-281.25(W)	15.9 +- 1.2	4
281.25-303.75(WNW)	26.7 +- 17.1	3
303.75-326.25(NW)	17.5 +- 0.0	1
326.25-348.75(NNW)	NO DATA+-NO DATA	0

DISTANCE(mi) FROM THE REACTOR	AVER.EXPOSURE +- Std.Dev. (mR/Std.Qtr.)	# IN GROUP
0-2	18.3 +- 7.4	19
2-5	14.1 +- 1.1	8
>5	14.1 +- 3.1	2
UPWIND CONTROL DATA	15.7 +- .3	3

NUREG-0837
Vol. 2, No. 1

NRC TLD Direct Radiation Monitoring Network

Progress Report
January - March 1982

U.S. Nuclear Regulatory Commission

F. Costello, T. Thompson, L. Cohen

PILGRIM
FOR THE PERIOD 811222-820415 115 DAYS

TLD DIRECT RADIATION ENVIRONMENTAL MONITORING

AZIMUTH (deg.)	AVER. EXPOSURE +- Std.Dev. (mR/Std.Qtr.)	# IN GROUP
348.75-11.25 (N)	0.0 +- 0.0	0
[illegible]1.25-33.75 (NNE)	0.0 +- 0.0	0
[illegible].75-56.25 (NE)	0.0 +- 0.0	0
[illegible].25-78.75 (ENE)	0.0 +- 0.0	0
78.75-101.25 (E)	0.0 +- 0.0	0
101.25-123.75(ESE	0.0 +- 0.0	0
123.75-146.25(SE)	15.3 +- 2.3	4
146.25-168.75(SSE)	15.3 +- 2.9	6
168.75-191.25(S)	14.8 +- 1.3	3
191.25-213.75(SSW)	18.8 +- .9	2
213.75-236.25(SW)	13.8 +- .7	5
236.25-258.75(WSW)	15.9 +- .6	2
258.75-281.25(W)	15.8 +- 1.3	5
281.25-303.75(WNW)	18.0 +- 5.9	6
303.75-326.25(NW)	16.8 +- 0.0	1
326.25-348.75(NNW)	0.0 +- 0.0	0

DISTANCE(mi) FROM THE REACTOR	AVER.EXPOSURE +- Std.Dev. (mR/Std.Qtr.)	# IN GROUP
0-2	16.9 +- 3.7	19
2-5	15.0 +- 1.6	10
>5	14.7 +- 1.4	5

PILGRIM

TLD DIRECT RADIATION ENVIRONMENTAL MONITORING
FOR THE PERIOD 811222-820415 115 DAYS
FIELD TIME 820105-820406 92 DAYS

NRC STATION	LOCATION AZIMUTH (deg.)	DIST (mi.)	INTEGRATED EXPOSURE(mR) +- Std. Dev.	EXPOSURE RA mR Std.Qtr. +- Std. Dev.
027	231	1.80	18.2 +- .3	14.3 +- .3
030	153	2.20	20.5 +- .5	16.1 +- .4
031	179	2.50	17.8 +- .1	13.9 +- .1
032	217	2.60	16.8 +- .1	13.1 +- .1
033	234	2.50	17.8 +- .3	14.0 +- .2
037	264	4.20	20.1 +- .0	15.7 +- .0
038	152	3.50	24.1 +- .3	18.9 +- .2
039	155	5.30	16.3 +- .1	12.8 +- .1
040	272	4.60	18.5 +- .6	14.5 +- .5
042	281	4.60	18.8 +- .6	14.7 +- .5
043	291	5.80	21.0 +- .1	16.4 +- .1
045	-	-	15.9 +- .4	12.4 +- .3
047	301	26.2	17.8 +- .0	13.9 +- .0
048	301	26.2	19.9 +- .4	15.6 +- .3
049	301	26.2	19.0 +- .7	14.9 +- .6
050	CTL	TLD	17.1 +- .0	13.3 +- .0

COMMENTS:

STATION 1 IS ON LICENSEE PROPERTY (PILGRIM OVERLOOK AREA).
ACCESS IS CONTROLLED

PILGRIM

TLD DIRECT RADIATION ENVIRONMENTAL MONITORING
FOR THE PERIOD 811222-820415 115 DAYS
FIELD TIME 820105-820406 92 DAYS

NRC STATION	LOCATION AZIMUTH/DIST (deg.)	(mi.)	INTEGRATED EXPOSURE(mR) +- Std. Dev.	EXPOSURE RAT mR/Std.Qtr +- Std. De
001	288	0.10	38.2 +- .8	29.9 +- .6
002	310	0.20	21.5 +- .7	16.8 +- .6
005	289	0.70	21.9 +- .3	17.2 +- .2
006	261	1.70	20.6 +- .1	16.1 +- .1
007	270	0.50	22.7 +- .4	17.7 +- .3
008	247	0.30	20.9 +- .2	16.4 +- .1
009	224	0.30	18.9 +- .1	14.8 +- .1
010	205	0.30	24.8 +- .0	19.4 +- .0
011	184	0.03	21.0 +- .1	16.4 +- .1
012	159	0.40	26.1 +- .4	20.4 +- .3
013	146	0.70	17.3 +- .2	13.6 +- .2
014	155	1.00	19.8 +- .5	15.5 +- .4
016	136	1.30	23.8 +- .0	18.6 +- .0
018	212	0.80	23.2 +- 1.0	18.1 +- .8
019	232	1.00	16.6 +- .3	13.0 +- .3
021	256	1.60	19.8 +- .1	15.5 +- .1
022	130	2.50	19.1 +- .0	14.9 +- .0
023	146	3.40	17.8 +- .6	13.9 +- .5
025	168	1.50	17.8 +- .6	13.9 +- .4
026	180	1.30	18.3 +- .3	14.3 +- .3

Updated Copy

NUREG-0837
Vol. 2, No. 2

NRC TLD Direct Radiation Monitoring Network

Progress Report
April-June 1982

U.S. Nuclear Regulatory Commission

NRC Region I

F. Costello, T. Thompson, L. Cohen

PILGRIM
FOR THE PERIOD 820325-820712 110 DAYS
TLD DIRECT RADIATION ENVIRONMENTAL MONITORING

AZIMUTH (deg.)	AVER. EXPOSURE +- Std.Dev. (mR/Std.Qtr.)	# IN GROUP
348.75-11.25 (N)	0.0 +- 0.0	0
11.25-33.75 (NNE)	0.0 +- 0.0	0
33.75-56.25 (NE)	0.0 +- 0.0	0
56.25-78.75 (ENE)	0.0 +- 0.0	0
78.75-101.25 (E)	0.0 +- 0.0	0
101.25-123.75(ESE)	0.0 +- 0.0	0
123.75-146.25(SE)	16.0 +- 1.1	4
146.25-168.75(SSE)	17.9 +- 4.0	5
168.75-191.25(S)	18.4 +- 5.8	3
191.25-213.75(SSW)	18.9 +- 2.9	2
213.75-236.25(SW)	16.6 +- 2.2	5
236.25-258.75(WSW)	17.3 +- 1.4	2
258.75-281.25(W)	17.9 +- 3.4	5
281.25-303.75(WNW)	26.5 +- 13.8	3
303.75-326.25(NW)	18.0 +- 0.0	1
326.25-348.75(NNW)	0.0 +- 0.0	0

DISTANCE(mi) FROM THE REACTOR	AVER.EXPOSURE +- Std.Dev. (mR/Std.Qtr.)	# IN GROUP
0-2	19.5 +- 6.4	19
2-5	16.4 +- 1.6	9
>5	16.3 +- 3.3	2
UPWIND CONTROL DATA	17.0 +- .4	3

PILGRIM

TLD DIRECT RADIATION ENVIRONMENTAL MONITORING
FOR THE PERIOD 820325-820712 110 DAYS
FIELD TIME 820406-820707 93 DAYS

NRC STATION	LOCATION AZIMUTH/DIST (deg.)	(mi.)	GROSS EXPOSURE(mR) +- Std. Dev.	EXPOSURE RATE mR/Std.Qtr. +- Std. Dev.
026	180	1.30	18.2 +- .5	14.9 +- .4
027	231	1.80	19.4 +- .3	15.9 +- .3
031	179	2.50	18.7 +- .2	15.3 +- .2
032	217	2.60	24.5 +- .1	20.0 +- .1
033	234	2.50	18.9 +- .6	15.5 +- .5
037	264	4.20	21.3 +- .5	17.4 +- .4
038	152	3.50	19.1 +- .5	15.6 +- .4
039	155	5.30	17.0 +- .1	13.9 +- .1
040	272	4.60	19.8 +- .3	16.2 +- .3
042	281	4.60	18.4 +- .4	15.1 +- .3
043	291	5.80	22.7 +- .2	18.6 +- .2
047	301	26.2	20.3 +- .1	16.6 +- .1
048	301	26.2	20.9 +- .5	17.1 +- .4
049	301	26.2	21.3 +- .4	17.4 +- .3

COMMENTS:

STATION 1 IS ON LICENSEE PROPERTY (PILGRIM OVERLOOK AREA).
ACCESS IS CONTROLLED

PILGRIM

TLD DIRECT RADIATION ENVIRONMENTAL MONITORING
FOR THE PERIOD 820325-820712 110 DAYS
FIELD TIME 820406-820707 93 DAYS

NRC STATION	LOCATION AZIMUTH/DIST (deg.)	(mi.)	GROSS EXPOSURE(mR) +- Std. Dev.	EXPOSURE RATE mR/Std.Qtr. +- Std. Dev.
001	288	0.10	51.9 +- .5	42.5 +- .4
002	310	0.20	22.0 +- .1	18.0 +- .1
005	289	0.70	22.5 +- .6	18.4 +- .5
006	261	1.70	21.0 +- .1	17.1 +- .1
007	270	0.50	29.0 +- .3	23.8 +- .2
008	247	0.30	22.4 +- .0	18.3 +- .0
009	224	0.30	21.0 +- .6	17.2 +- .5
010	205	0.30	25.7 +- .4	21.0 +- .3
011	184	0.03	30.7 +- .9	25.1 +- .7
012	159	0.40	28.1 +- .2	23.0 +- .2
013	146	0.70	18.6 +- .3	15.2 +- .2
014	155	1.00	26.1 +- .0	21.4 +- .0
016	136	1.30	20.0 +- .4	16.4 +- .3
018	212	0.80	20.6 +- .2	16.9 +- .1
019	232	1.00	17.5 +- 1.1	14.4 +- .9
021	256	1.60	19.9 +- .3	16.3 +- .2
022	130	2.50	18.4 +- .2	15.1 +- .2
023	146	3.40	21.2 +- .1	17.3 +- .1
025	168	1.50	19.1 +- .1	15.6 +- .0

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ENCLOSURE 5
TO QUESTION 7

Docket No. 50-293

JUL 8 1982

MEMORANDUM FOR: H. R. Denton, Director, ONRR

FROM: R. J. Mattson, Director, DSI/ONRR

SUBJECT: GENERIC IMPLICATIONS OF THE RELEASE OF SPENT DEMINERALIZER RESINS FROM PILGRIM, UNIT NO. 1

Reference: PNO-I-82-42/42A

The release of radioactive spent resins from the Pilgrim Power Station, reported in PNO-I-82-42, June 11, 1982, has been reviewed for generic implications in accordance with your request. Based on information in the PN and its update of June 14, 1982, on information in the docket file, and on information obtained in telephone discussions with Region I representatives, a licensee representative, and the Operating Project Manager (DL), it is our conclusion that there are several related factors in this incident which have both generic and licensee - specific implications. These are discussed in items (1) through (5) below.

(1) It is probable that the resins observed and reported in the PN originally escaped from operations involved in a resin cleaning operation for condensate demineralizer resins. Resins were apparently forced up a vent pipe into a ventilation exhaust duct, from which the resins were transported by ventilation air flow. Vent pipes are designed to maintain tank pressure close to atmospheric as tank levels fluctuate and gases evolve from tank contents. Such a design provides a controlled exhaust system rather than a discharge into the building atmosphere; many such vents are present in plant designs. While it is considered good design practice to install screens or filters in such vent lines, there were apparently no such devices in the Pilgrim vents. The Standard Review Plans 11.2 (Liquid Waste Management Systems) and 11.3 (Gaseous Waste Management Systems) and Regulatory Guide 1.143 (Radwaste System Design Guidance) do not specifically address such a design criterion.

(2) It is probable that water entered the ventilation exhaust ducts along with the resins noted in (1), above. While it is not known if this water was significantly radioactive, the presence of the water may have been a factor in the deterioration of filters and filter frames (see (3), below). Vent lines serving liquid systems should be designed to incorporate a device or mechanism, such as a water trap, to prevent the flow of liquids into vent pipes discharging to ventilation exhaust ducts. Neither the applicable Standard Review Plans nor the applicable Regulatory Guide address such a design feature.

~~8208240032~~ 6pp

(3) The licensee considers the most probable source of the discharge of radioactively contaminated resins to the roof and ground areas of the plant to be the reactor building ventilation exhaust duct. Based on the dispersal pattern of the resins, we arrived at the same conclusion. As noted in (1) and (2), above, resins are presumed to have entered tank vent pipes leading to ventilation ducts, probably in the form of a slurry. The continuous flow of warm dry air would cause the resin to dry out, leaving a residue of small beads or particles of low density, which can be carried along the duct by the ventilation exhaust air current. In the filtration plenum, air from the ventilation exhaust ducts is passed first through a fiberglass prefilter media and then through a HEPA (High Efficiency Particulate Air) filter. Air flow through the filters is horizontal and there is about a four-foot space (measured horizontally) between the prefilter banks and HEPA filter banks. Linear face flow velocity (design) of the prefilters is about 250 linear feet per minute, or about 3 mph. Each HEPA filter module has a dimensional cross-section of about 4 ft^2 and has a rated capacity, when new, of 1,000 cfm at a 1" (water) pressure drop; the face velocity for a HEPA filter is also about 250 linear feet per minute or about 3 mph.

An IE Health Physics appraisal team visited Pilgrim in January and February, 1980. The team's report, dated July, 1980, noted that the prefilters were "disintegrating in place" (Section 4.2.3.2, page 55) but that no damage to the HEPA filters could be observed by visual inspection. This situation was apparently not corrected until the refueling outage which began in September, 1981. In fairness to the licensee, though, it should be noted that the prefilter disintegration was not included as a "significant finding" by the NRC in the appraisal. While there may be extenuating circumstances which are not apparent from the IE appraisal, there appear to be no reasons why these non ESF systems could not have been taken out of service for replacement or repair in a more expeditious manner.

While we have not been able to determine the exact condition of the HEPA filters at the time of their replacement in September, 1981, licensee representatives did state many of the HEPA filters were found to be damaged. It should be pointed out that no release of resins had been identified at that time and no tests were performed to determine the nature or extent of leakage or damage. The staff considers that the Pilgrim occurrence has no direct implications as to the integrity of adequately tested and maintained HEPA filters in ESF filter systems but, rather, emphasizes the need for regular testing and surveillance where a specified level of performance is to be achieved and maintained. The occurence is, however, a clear demonstration that plant operators cannot neglect HEPA filter systems indefinitely and then expect them to perform as designed.

We note, however, that in the present regulatory climate, licensees, in general, have no compelling motivation to perform surveillance which is not formally required of them, especially when inoperability of a system will not lead to noncompliance. The fact that deteriorating prefilters were observed during the Pilgrim Health Physics appraisal and that radioactive resins were found to be present in the ventilation exhaust ducts was not evidence that Technical Specification release limits or Appendix I criteria were being exceeded and, therefore, there was no violation of regulatory requirements to initiate corrective action. The periodic testing, or replacement of non-ESF filtration system components represents an expenditure of money and manpower with little tangible benefit when only routine normal operation is considered; in an era of tight money and budgetary restraints, plant managers may be hard-pressed to justify to upper levels of utility management the expenditure of even a few thousands of dollars at a very high cost-benefit ratio.

(4) Technical Specifications require periodic testing of ESF filter systems at nearly all plants, as well as surveillance of parameters such as pressure drop, which are indicative of system condition and performance. Normal ventilation exhaust air filter systems are not ESF systems and, therefore, are not subject to Technical Specification requirements for testing and surveillance. Non-ESF ventilation exhaust filter systems are installed in nuclear power plant buildings to reduce releases of airborne material to levels that satisfy the criteria of Appendix I to 10 CFR Part 50; Pilgrim, Unit 1, is only one of many plants which do not regularly inspect, check, or test their non-ESF filter systems.

While the failure or procrastination on the part of operating plants to regularly test and assure the proper functioning of these systems may be interpreted by some parties as failing to provide maximum protection to the environment, making such testing a firm commitment would necessitate a substantial revision in the basic NRC philosophy of plant safety and environmental protection. Commitments made by applicants in their FSAR to Regulatory Guide 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," is the method currently used by NRR to implement design guidance and testing programs for non-ESF filter systems. Such criteria had not been established by the NRC when Pilgrim 1 was licensed in 1972, so it is likely that no commitment was ever made by Boston Edison to provide surveillance testing of the non-ESF filters at Pilgrim 1.

(5) The Licensee and IE (reference IE Health Physics Appraisal Report for Pilgrim, dated June 22, 1980, page 54) have been aware for over two years that radioactive resin beads and fines were present in Pilgrim ventilation exhaust ducts. The same appraisal report, page 55 notes serious deficiencies in the condition of ventilation exhaust prefilters and the presence of approximately six inches of spilled radioactive (2R/hr) resins on the floor of a room in the Radwaste Building (p. 48), as well as loose contamination up to 90 mrads/hr on the floor immediately outside that room. In view of the unique and highly visible nature of resin beads, the rather high radioactive contamination levels associated with the resin, and the knowledge that resins had been a problem in several areas of the plant for over two years, the Licensee's statement (PN Update June 14, 1982) that the resins had probably been released prior to September 1981 seems to indicate, at best, an absence of recognition of potential problems on the part of plant management. To admit that external plant contamination of this order of magnitude had gone unnoticed and undetected for over eight months would seem to admit to the existence of inadequacies in the Health Physics program.

IE COORDINATION

Our review has been coordinated with IE personnel at Bethesda, Region I, and the Resident Inspectors' office. The Radiological Safety Branch (IE) is currently reviewing completed Health Physics appraisal reports for other plants to identify any similar circumstances to confirm the generic nature of the Pilgrim incident and support the need for issuance o guidance to licensees; this review has not been completed but will be made available at a later date.

SUMMARY

As the result of our review of the Pilgrim, Unit 1, PNO of June 11, 1982 (PNO-I-82-42), the staff suggests the following:

(1) As a short-term action, recommend to IE that an information notice be issued to all operating reactors which (a) describes the Pilgrim 1 resin dispersal event, (b) requests plants to voluntarily institute a surveillance program for existing non-ESF filtration systems if one does not exist and (c) requests that tank vent designs be reviewed and that, if appropriate and feasible, modifications be made to prevent inadvertent release of resins or liquids to the ventilation system. NRR staff is available to provide assistance to IE in the preparation of such a circular.

(2) As a longer term action, revise Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components installed in Light-Water-Cooled Nuclear Power Plants," and Standard Review Plan 11.2, "Liquid Waste Management Systems," to include design guidance and acceptance criteria which address (a) the incorporation of filters or screens in the design of vents from tanks which may contain resins, and (b) the incorporation of provisions into the vent design such as filters traps or check valves to prevent or minimize the flow of liquids through vent lines while permitting pressure equalization within the tank.

Roger J Mattson

R. J. Mattson, Director
Division of Systems Integration
Office of Nuclear Reactor Regulation

JUL 8 1982

cc: E. Case
D. Eisenhut
S. Hanauer
G. Laines
T. Novak
W. Houston
W. Gammill
D. Vassallo
F. Congel
L. Hulman
R. Bangart
C. Willis
R. Capra
L. Cunningham
K. Eccleston
P. Stoddart

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ENCLOSURE 6
TO QUESTION 7

APR 19 1983

MEMORANDUM FOR: Karl V. Seyfrit, Chief
Reactor Operations Analysis Branch
Office for Analysis and Evaluation
of Operational Data

AEOD/T307

THRU: Stuart D. Rubin, Lead Engineer
Reactor Systems 4
Reactor Operations Analysis Branch

FROM: John L. Pellet
Reactor Systems 4
Reactor Operations Analysis Branch

SUBJECT: TECHNICAL REVIEW REPORT ON PILGRIM 1 RESIN MIGRATION

Enclosed find the technical review report titled "Condensate Demineralizer Resin Migration Through the Plant Vent and Standby Gas Treatment System." This report concludes that no additional AEOD/ROAB involvement is necessary for this event.

John L. Pellet

John L. Pellet
Reactor Systems 4
Reactor Operations Analysis Branch

~~8305040149~~ 1p.

AEOD TECHNICAL REVIEW REPORT*

UNIT: Pilgrim 1
DOCKET: 50-293
LICENSEE: Boston Edison Company
NSSS/AE: General Electric/Bechtel

TR REPORT NO.: AEOD/T307
DATE: April 19, 1983
EVALUATOR/CONTACT: J. Pellet

SUBJECT: CONDENSATE DEMINERALIZER RESIN MIGRATION THROUGH THE PLANT VENT AND THE STANDBY GAS TREATMENT SYSTEM

EVENT DATE: June 11, 1982

SUMMARY

This report reviews the safety significance of the June 1982 discovery at Pilgrim that demineralizer resins had migrated throughout the plant contaminated exhaust vent to external plant areas inside the protected area fencing. Also, sufficient resin had migrated through the reactor building ventilation system to block proper operation of the Standby Gas Treatment System (SBGTS). References are cited which show that resin migration into the ventilation system and SBGTS had occurred at least three years previously. This report finds that the event was of minimal safety significance and concludes that current NRC efforts are adequate without additional AEOD involvement.

DISCUSSION

Plant & Status

Pilgrim 1 was in steady state power operation on June 11, 1982 while performing a surveillance instruction (SI) on the SBGTS.

Occurrence-Cause & Effect[1]

The SBGTS failed its routine SI due to low flow. The low flow was caused by carryover of resin beads from the condensate demineralizer vent piping to the reactor building ventilation system and contaminated exhaust vent and from there to the SBGTS. This carryover occurred during backwashing of the demineralizer. Backwashing with air and water resulted in resin fines, particulates, and some resin beads being entrained in the air/water backwash. An air scrubber was installed during initial startup to prevent resin migration into the ventilation system. However, it did not perform as expected since installation. As a result, substantial resin migrated to the radwaste and ventilation systems over a considerable time period.

After this event, contaminated resin beads were discovered outside of the plant buildings (but not offsite) as well as inside the vent system. Less than 70 cubic feet of resin was removed from the ventilation system and less than 1/2 of a cubic foot was found inside the protected area. Root cause of the substantial resin migration appears to be inadequate design of the scrubber intended to preclude such migration.

~~8305040155~~ 3pp.

*This document supports ongoing AEOD and NRC activities and does not represent the position or requirements of the responsible NRC program office.

History

At least two cases of resin intrusion into the SBGTS have been previously reported[2,3] since June, 1979. This indicates that resin intrusion into the ventilation system and SBGTS has been a recognized problem at Pilgrim for several years without adequate resolution. However, prior to the June 11, 1982 event there was no evidence of contamination outside of the plant buildings.

Consequences

The consequences of this event may be broken down into three categories: 1) offsite release, 2) personnel exposure, and (3) system performance or availability. The resin migration problem produced no evidence of offsite release during this review. However, the resin migration clearly has resulted in added equipment contamination and substantial cleanup efforts by plant personnel over a period of several years, but this review found no indication of unacceptable personnel exposure. From a system viewpoint, this event demonstrates the potential for failure in a nonsafety system to act as a common cause initiator affecting multiple trains of a safety system (in this case SBGTS). This potential is mitigated because failure is as a result of flow restriction due to resin buildup and is therefore very slow with respect to the test interval (i.e., only two failures over the last three years). Also, even though one train of SBGTS was inoperable due to low air flow, the train was capable of performing at a reduced level. In summary, the resin migration produced minimal actual consequences in the three areas of concern.

Corrective Actions

The licensee actions to preclude further resin migration into the vent system may be divided into short-term and long-term efforts. The immediate actions by the licensee to remove existing resin and preclude additional migration were set out in Confirmative Action Letter No. 82-19[4]. Additionally, the licensee disconnected the ventilation system from the poorly functioning gas scrubber and rerouted the scrubber discharge (liquid, air, and resin) to the Reactor Building Equipment Sump. However, the equipment sump was not intended for either the quantity of air/water mixture or the entrained resins produced by demineralizer backwashing. This resulted in sump discharge to the HPCI room during demineralizer backwash. Due to a loose cap on a floor drain, approximately 12 inches of water accumulated in the B RHR pump room as well as in the HPCI room. Resin contamination was also evident in the HPCI room[5]. The licensee corrected this problem by securing the leaking floor drain and administratively requiring low sump level prior to demineralizer backwash. The above details introduce considerable uncertainty as to the long-term efficacy of the corrective actions implemented by the licensee thus far. The licensee is currently studying potential long-term corrective actions and can be expected to implement such actions when they are determined. The NRC Resident Inspector is following this subject and can be expected to require an adequate resolution based on his past efforts.

FINDINGS

Findings for this investigation were:

1) Resin migration through the ventilation system can produce a common mode failure of both trains of SBGTS.

2) The safety significance of this event is minimal due to the slow propogation rate and limited actual consequences of the resin migration.

3) Corrective actions by the licensee are adequate at present.

CONCLUSIONS

The safety significance of this event is relatively minor given the radiological release and system performance effects previously discussed. The personnel exposure effects may be more significant, especially since this has evidently been a problem for over three years. However, this review produced no evidence of excess personnel exposure. Given the limited significance discussed above, followup and resolution of this event by the resident inspector appears to be adequate. At present there is no need for additional AEOD involvement on this event. However, this type of common mode failure is potentially generic, depending on plant specific arrangement of demineralizer vents, SBGTS, and reactor building ventilation.

REFERENCES

1. LER 82-019/03L-0 on Pilgrim unit 1.
2. LER 79-020/03L-0 on Pilgrim Unit 1.
3. IE Inspection No. 50-293/82-20
4. Confirmative Action Letter 82-19
5. IE Inspection No. 50-293/82-30

ENCLOSURE 7
TO QUESTION 7

Event Evaluation Sheet

Initial Receiver	Date	Information Source	Subject
WIGGINTON	6/14/82	D.O. LOG (PN ISSUED) (82-42)	Spent resin release

Licensee/Facility Type/Location
PILGRIM / BWR / RI

Regional Contact
Greenman / NIMITZ

Event Summary: 6/11/82, spent resin discovered on routine rad survey on plant — spill occurred sometime during past few weeks, resin on ground near buildings, on TB, RB, etc roofs — believed to have exited building @ RB vent @ 100' elevation — some found on inlet plenum of STBGT SYSTEM. Region dispatched 2-3 HP INSPECTORS TO SITE.

(SEE ~FEB 4, 1982 EVENT)

(Section Chief)

Further Action Required: **Yes** No

Assigned to WIGGINTON

(Assign Code #)*

Followup Actions/Results

Looks like defect VENT ISOLATION VALVES ON demin, ... VENTILATION SYSTEM — allowed ...

→ Followup PN coming — RI got a confirmation

ACTION LETTER from LICENSEE

Evaluate further w/ PN issued.

Item Closed by/Date

Concurrence/Date
(Sect. Chief)

DCS No: 50293-820611
Date: June 11, 1982

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE--PNO-I-82-42

This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by the Region I staff on this date.

Facility: Pilgrim Nuclear Power Station
Plymouth, Massachusetts
DN 50-293

Licensee Emergency Classification:
___ Notification of Unusual Event
___ Alert
___ Site Area Emergency
___ General Emergency
_X_ Not Applicable

Subject: RELEASE OF SPENT RESIN

At approximately 1300 on June 11, 1982 spent resin was found on the ground near the Turbine Building. Subsequent surveys identified contamination of the roofs of the Turbine, Reactor, Off-Gas and Re-Tube Buildings. Contamination was also found on the ground within the site controlled areas. Contamination levels ranged from 20-30,000 dpm/100 cm^2 with maximum contamination of up to 100,000 dpm/100 cm^2. Gamma isotopic analysis of the resin identified primarily long lived radionuclides (Co-60, Cs-137, Cs-134 and Mn-54).

No contamination was identified off-site or in storm drains. All personnel are being frisked prior to exiting the site and no personnel contamination has been identified.

The resin may have been released through the reactor building vent duct which exhausts to the atmosphere at an elevation of approximately 100 ft. The licensee has found approximately 10 ft^3 of resin in the Standby Gas Treatment System inlet plenum. The source of the resin is being investigated. Three radiation specialists have been dispatched to the site to evaluate the radiological aspects of the occurrence.

Media interest is expected due to public interest in the facility. The licensee is considering issuing a press release. The NRC does not plan to issue a press release but will respond to media inquiries. The Commonwealth of Massachusetts has been informed.

This PN is current as of 4:45 P.M., June 11, 1982.

CONTACT: Elsasser 488-1235 | Brunner 488-1225

DISTRIBUTION:

H St. ____	MNBB ____	Phillips ____	E/W ____	Willste ____	Mail: ADM:DMB
Chairman Palladino	EDO	NRR	IE	NMSS	DOT:Trans. Onl
Comm. Gilinsky	PA		OIA	RES	
Comm. Ahearne	MPA		AEOD		
Comm. Roberts	ELD				
ACRS		Air Rights	INPO ____		
SECY		SP	NSAC ____		
CA					
PDR	Regional Offices ____				

TMI Resident Section ____
RI Resident Office ____

Licensee: ____
(Reactor Licensees)

Region I Form 83
(Rev May 1982)

~~8206170377~~ 1p.

DCS No: 50293-820611
Date: June 14, 1982

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE--PNO-I-82-42A

This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by the Region I staff on this date.

Facility: Pilgrim Nuclear Power Station
Plymouth, Massachusetts
DN 50-293

Licensee Emergency Classification:
___ Notification of Unusual Event
___ Alert
___ Site Area Emergency
___ General Emergency
_x_ Not Applicable

Subject: RELEASE OF SPENT RESIN (UPDATE PNO-I-82-42)

Surveys of the entire site within the protected area and surveys of selected areas of the licensee controlled area were made within 3 hours of the identification of the spent resin release. The licensee's onsite surveys identified two contaminated pavement areas which were barricaded and posted. Surveys confirmed contamination of the Turbine, Administration, Augmented Off-Gas and Re-Tube Building roofs. The Reactor Building Roof was found to be free of contamination. The licensee's offsite survey included surveys of cars, parking lots, shorefront, and security access areas. No contamination was identified. Routine environmental air samples covering the period June 1-15, 1982 were counted. Nothing unusual was identified. Because of the size and weight of the resins, no offsite airborne release of the beads appears to have occurred. This was confirmed by air samples collected during clean-up of the contaminated pavement areas which when counted indicated background and the identification of resins only on roof-tops under the Reactor Buiding Vent. Preliminary samples of storm drain residue have been counted with no contamination identified. All contaminated ventilation ducts have been vacuumed clean. A duct surveillance program has been established to identify any additional resin accumulation.

The licensee believes the resin entered the ventilation ducts from the condensate demineralizer system during resin backwashing via the Cation Regeneration Tank Vent. In addition, resin from defective condensate demineralizer vent valves may have also been released prior to their repair during the September 1981 -March 1982 refueling outage. The resin appears to have been released from the Reactor Building Ventilation Exhaust System which vents above the reactor building roof, prior to the repair of defective filters in this system in September 1981.

The licensee has suspended all transfer operations which could result in further resin releases to ventilation ducts and has initiated additional environmental sampling. The licensee's actions were monitored by three Region I Radiation Specialists throughout the weekend. Region I will issue a Confirmatory Action Letter to address planned licensee corrective actions. The licensee is continuing to review the source and cause to determine what permanent corrective action will be needed. The Resident Inspectors are closely following licensee actions concerning this event.

Media interest has occurred. The licensee has responded to media inquiries but does not plan to issue a press release. The NRC will respond to media inquiries but does not plan to issue a press release.

This PN is current as of 11:00 a.m., June 14, 1982.

~~8206210607~~ 1p

Region I Form 83
(Rev. March 1982)

ENCLOSURE 3
TO QUESTION 7

SSINS No.: 6835
IN 82-43

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D. C. 20555

November 16, 1982

IE INFORMATION NOTICE NO. 82-43: DEFICIENCIES IN LWR AIR FILTRATION/VENTILATION SYSTEMS

Addressees:

All nuclear power reactor facilities holding an operating license (OL) or construction permit (CP).

Purpose:

This information notice is provided as notification of events that had actual or potential radiological impact on the plant environs. It is expected that recipients will review the information for applicability to their facilities. No specific action or response is required.

Description of Circumstances:

Within the past 2-1/2 years, air filtration/ventilation systems at five facilities were found to have serious deficiencies, ranging from overloaded prefilters to evidence of a wetted high-efficiency particulate air (HEPA) filter bank, to penetration of HEPA filter banks by substantive quantities of radioactive resin beads. Deficiencies occurred in both safety-related and non-safety-related systems.

In June 1982, radioactive spent resin was found on the grounds and roof areas at Pilgrim 1. Principal radionuclides were Co-60, Cs-137, Cs-134, and Mn-54; contamination ranged from 20,000 dpm/100 cm^2 to 100,000 dpm/100 cm^2. The contamination penetrated damaged filters in a non-safety-grade HEPA filter plenum. The degraded condition of these filters was not detected in a timely manner because of a lack of surveillance or testing of the filtration system. The HEPA filter failure occurred possibly as an end result of a combination of high dust loadings and mechanical damage resulting from the impact of disintegrating prefilters, as well as the probable warping or distortion of HEPA filter frames under prolonged exposure to water and high humidity.

In December 1980, the SGTS trains at Brunswick 1 were found to be operating at close to 100% humidity, and condensation was observed on the interior walls. Regulatory Guide 1.52 recommends operation at humidity of 70% or less; operation at high humidity is known to cause substantial degradation of the iodine-retention capacity of charcoal adsorbers. Also, in December 1980, both filter trains in the turbine building filter system at Brunswick were found to be operating with the upstream HEPA differential pressure gauges offscale high. Also, in the turbine building filter system, 43% of the upstream HEPA filters were improperly installed.

~~8208190248~~ 2pp

In August 1980, filters and charcoal adsorbers in the Surry 1 process vent exhaust air treatment system were determined to have been half submerged in water, and the HEPA filters were caked with dust. No pressure drop instrumentation was provided across the filter banks to ascertain their state of loading. Also, in August 1980, pressure drop gauges across the HEPA filter banks in the ventilation exhaust treatment system of the auxiliary building at Surry 1 exceeded 5 inches, which is offscale high; this condition had existed since May 1980.

In May 1980, the normal containment building exhaust filters at Turkey Point were found to be overloaded with dust to such an extent that the filter medium was separated from its frame in more than 50% of the filters. This apparently allowed radioactive contamination resulting from explosive plugging of steam generator tubes to be transported to the southeast sector of the plant site.

In March 1980, it was determined that HEPA filters in the Big Rock Point offgas and chemistry laboratory exhaust treatment systems were not being tested for leakage in place. No records were maintained of pressure differential across the laboratory HEPA filters which had not been replaced for at least five years.

In each case described above, licensees initiated programs and procedures to correct the deficiencies and to prevent or minimize their potential for reoccurrence.

Air treatment systems which incorporate filtration or adsorption media are provided to reduce the potential release of radioactive materials to the environs. In order to function as designed, such systems should be installed, tested, and maintained to a degree consistent with their intended function.

Guidance on installation, maintenance, and testing programs, of a degree and nature which have been demonstrated to ensure proper system functioning, is provided in Regulatory Guides 1.52 and 1.140.

No written response to this information notice is required. If you need additional information about this matter, please contact the Regional Administrator of the appropriate NRC Regional Office or this office.

/S/ William Mills for
Edward L. Jordan, Director
Division of Engineering and
Quality Assurance
Office of Inspection and Enforcement

Technical Contacts: L. J. Cunningham, IE
301-492-8073

P. G. Stoddart, NRR
301-492-7633

ENCLOSURE 9
TO QUESTION 7

Pilgrim Nuclear Power Station

Radioactive Effluent and Waste Disposal Report including Radiological Impact on Humans

January 1 through June 30, 1982

By: Nuclear Operations Support Department
Environmental and Radiological
Health and Safety Group

Date: September 1, 1982

Boston Edison Company

8209160303 820831
PDR ADOCK 05000293
R PDR

IE25

PILGRIM NUCLEAR POWER STATION

RADIOACTIVE EFFLUENT AND WASTE DISPOSAL REPORT

INCLUDING RADIOLOGICAL IMPACT ON HUMANS

JANUARY 1 THROUGH JUNE 30, 1982

Prepared by: Christine E. Bowman

Christine E. Bowman
Sr. Radiological Engineer

Approved by: Tom Sowdon

Thomas L. Sowdon
Environmental and Radiological Health and Safety Group Leader

Date of Submittal: September 1, 1982

TABLE OF CONTENTS

LIST OF TABLES

LIST OF TABLES (Cont.)

Table	LIST OF TABLES (Cont.)	Page

1. INTRODUCTION AND SUMMARY

This report is issued for the period January-June 1982 in accordance with NRC Regulatory Guide 1.21 "Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water Cooled Nuclear Power Plants" (Rev. 1). The information supplied includes actual effluent releases, radioactive waste and meteorological data; doses from liquid releases, doses from gaseous releases and direct gamma radiation doses.

2. EFFLUENT, WASTE DISPOSAL AND WIND DATA

Radioactive liquid and gaseous releases, wind speed data together with measurement errors and solid waste disposal information are given in Tables 1A, 1B, 1C, 2A, 2B, 3, 4A-1, 4A-2, and supplemental information section in the standard Regulatory Guide 1.21 format.

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT

Supplemental Information
January - June 1982

Facility Pilgrim Nuclear Power Station Licensee DPR-35

1. Regulatory Limits

   a. Fission and activation gases $\frac{Qs}{0.25/\bar{E}} + \frac{Qv}{0.10/\bar{E}} \leq 1$

   b. Iodines 2Ci/Quarter

   c. Particulates, half-lives >8 days 13(1.8E4Qs+1.8E5Qv)≤1

   d. Liquid effluents: 10Ci/Quarter

2. Maximum Permissible Concentration

   Provide the MPCs used in determining allowable release rates or concentrations.

   a. Fission and activation gases: } 10 CFR 20
   b. Iodines: } Appendix B
   c. Particulates, half-lives >8 days: } Table II
   d. Liquid effluents: H - 3 = 1 X 10^{-3} μCi/ml; all rest, 10 CFR 20, Appendix B, Table II

3. Average Energy

   Provide the average energy ($\bar{E}$) of the radionuclide mixture in releases of fission and activation gases, if applicable.
   MS =0.324:RBV=0.503

4. Measurements and Approximations of Total Radioactivity

   Provide the methods used to measure or approximate the total radioactivity in effluents and the methods used to determine radionuclide composition.

   a. Fission and activation gases: } GeLi
   b. Iodines: } Isotopic
   c. Particulates: } Analysis
   d. Liquid effluents:

5. Batch Releases

   Provide the following information relating to batch releases of radioactive materials in liquid and gaseous effluents.

   a. Liquid

      1. Number of batch releases: 121
      2. Total time period for batch releases: 192.92hrs
      3. Maximum time period for a batch release: - 7.75hrs
      4. Average time period for batch releases: 1.59hrs
      5. Minimum time period for a batch release - 0.25hrs
      6. Average stream flow during periods of release of effluent into a flowing stream: 1.90E+5GPM

   b. Gaseous (Not Applicable)

6. Abnormal Releases

   a.
   b. None

TABLE 1A
EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT
GASEOUS EFFLUENTS - SUMMATION OF ALL RELEASES
January - June 1982

	Unit	Quarter 1	Quarter 2	Est. Total Error, %
A. Fission and activation gases				
1. Total release	Ci	-	3.55E+3	2.50E+1
2. Average release rate for period	μCi/sec	-	4.52E+2	
3. Percent of Technical Specification limit	%	-	6.92E-2	
B. Iodines				
1. Total iodine-131	Ci	-	3.97E-3	2.54E+1
2. Average release rate for period	μCi/sec	-	5.05E-4	
3. Percent of Technical Specification limit	%	-	1.99E-1	
C. Particulates				
1. Particulates with half-lives > 8 days	Ci	<3.68E-4	4.26E-3	3.05E+1
2. Average release rate for period	μCi/sec	<4.73E-5	5.42E-4	
3. Percent of Technical Specification limit	%	<8.39E-3	6.98E-2	
4. Gross alpha radioactivity	Ci	<4.52E-7	<5.61E-7	
D. Tritium				
1. Total release	Ci	2.34E0	5.92E0	3.20E+1
2. Average release rate for period	μCi/sec	3.01E-1	7.52E-1	
3. Percent of Technical Specification limit	%	-	-	

TABLE 1B
EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1982)
GASEOUS EFFLUENTS – ELEVATED RELEASE
January - June 1982

Nuclides Released	Unit	CONTINUOUS MODE Quarter	CONTINUOUS MODE Quarter	BATCH MODE Quarter	BATCH MODE Quarter
1. Fission gases					
krypton-85	Ci	-	1.37E-2		
krypton-85m	Ci	-	2.93E+2		
krypton-87	Ci	-	6.55E+1		
krypton-88	Ci	-	3.62E+2		
xenon-133	Ci	-	2.28E+3		
xenon-135	Ci	-	2.61E+2		
xenon-135m	Ci	-	<6.06E+0		
xenon-138	Ci	-	<2.38E+1		
xenon-131m	Ci	-	-		
xenon-137	Ci	-	-		
xenon-133m	Ci	-	4.28E+1		
Total for period	Ci	-	3.33E+3		
2. Iodines					
iodine-131	Ci	-	2.53E-3		
iodine-133	Ci	-	7.90E-3		
iodine-135	Ci	-	<6.55E-3		
Total for period	Ci	-	<1.70E-2		
3. Particulates					
strontium-89	Ci	<6.32E-7	5.16E-4		
strontium-90	Ci	<6.26E-8	5.50E-6		
cesium-134	Ci				
cesium-137	Ci	<1.04E-5	1.14E-5		
barium-lanthanum-140	Ci		1.57E-3		
chromium-51	Ci				
manganese-54	Ci	8.90E-6	2.90E-6		
cobalt-58	Ci				
iron-59	Ci				
cobalt-60	Ci	<7.86E-5	3.00E-5		
zinc-65	Ci				
zirconium-niobium-95	Ci				
cerium-141	Ci				
cerium-144	Ci				
ruthenium-103	Ci				
ruthenium-106	Ci				

TABLE 1C

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1982)

GASEOUS EFFLUENTS - GROUND LEVEL RELEASE

January - June 1982

		CONTINUOUS MODE		BATCH MODE	
Nuclides Released	Unit	Quarter	Quarter	Quarter	Quarter
1. Fission gases					
krypton-85	Ci	-	1.01E-5		
krypton-85m	Ci	-	2.47E+1		
krypton-87	Ci	-	2.51E+0		
krypton-88	Ci	-	4.55E+1		
xenon-133	Ci	-	4.19E+1		
xenon-135	Ci	-	1.07E+2		
xenon-135m	Ci	-	-		
xenon-138	Ci	-	-		
Total for period	Ci	-	2.22E+2		
2. Iodines					
iodine-131	Ci	-	1.44E-3		
iodine-133	Ci	-	6.50E-3		
iodine-135	Ci	-	<1.02E-2		
Total for period	Ci	-	<1.81E-2		
3. Particulates					
strontium-89	Ci	1.64E-5	1.46E-3		
strontium-90	Ci	4.76E-7	1.44E-6		
cesium-134	Ci	1.17E-6			
cesium-137	Ci	2.42E-5	3.67E-5		
barium-lanthanum-140	Ci		3.95E-4		
manganese-54	Ci	1.08E-5	5.88E-6		
cobalt-58	Ci				
iron-59	Ci				
cobalt-60	Ci	2.16E-4	2.27E-4		
zinc-65	Ci				
zirconium-niobium-95	Ci				
cerium-141	Ci				
ruthenium-103	Ci				
ruthenium-106	Ci				

TABLE 2A

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1982)

LIQUID EFFLUENTS - SUMMATION OF ALL RELEASES

January - June 1982

	Unit	Quarter 1	Quarter 2	Est. Total Error, %
A. Fission and activation products				
1. Total release (not including tritium, noble gases, or alpha)	Ci	5.72E-1	1.44E-1	3.00E+1
2. Average diluted concentration during period	μCi/ml	8.91E-8	7.58E-8	
3. Percent of applicable limit	%	5.72E0	1.44E0	
B. Tritium				
1. Total release	Ci	5.26E0	1.99E-1	3.00E+1
2. Average diluted concentration during period	μCi/ml	8.19E-7	1.05E-7	
3. Percent of applicable limit	%	8.19E0	1.05E0	
C. Dissolved and entrained gases				
1. Total release	Ci	-	-	-
2. Average diluted concentration during period	μCi/ml	-	-	
3. Percent of applicable limit	%	-	-	
D. Gross alpha radioactivity				
1. Total release	Ci	< 1.44E-4	< 1.73E-5	4.00E+1
E. Volume of waste released (prior to dilution)	liters	1.61E6	1.10E5	2.00E+1
F. Volume of dilution water used during period	liters	6.42E9	1.90E9	2.00E+1

TABLE 2B
EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1982)

LIQUID EFFLUENTS
January - June 1982

Nuclides Released	Unit	CONTINUOUS MODE Quarter	CONTINUOUS MODE Quarter	BATCH MODE Quarter	BATCH MODE Quarter
strontium-89	Ci			6.70E-4	1.89E-3
strontium-90	Ci			4.17E-4	1.65E-4
cesium-134	Ci			1.46E-2	7.42E-4
cesium-137	Ci			1.08E-1	6.60E-3
iodine-131	Ci			-	2.25E-6
cobalt-58	Ci			2.54E-3	8.23E-4
cobalt-60	Ci			2.44E-1	7.00E-2
iron-59	Ci			4.27E-5	3.06E-6
zinc-65	Ci			4.28E-3	1.20E-3
manganese-54	Ci			2.61E-2	1.01E-2
chromium-51	Ci			-	1.20E-5
zirconimum-niobium-95	Ci			5.16E-4	6.74E-4
molybdenum 99-technetium 99m	Ci			-	-
barium-lanthanum-140	Ci			-	4.96E-5
cerium-141	Ci			1.65E-5	-
iodine-133	Ci			-	2.70E-6
cerium-144	Ci			-	1.75E-5
silver-110m	Ci			-	-
iron-55	Ci			1.47E-1	2.43E-2
unidentified	Ci			2.40E-2	2.72E-2
Total for period (above)	Ci			5.72E-1	1.44E-1
xenon-133	Ci			-	-
xenon-135	Ci			- -	-

TABLE 3

EFFLUENT AND WASTE DISPOSAL SEMI-ANNUAL REPORT (1982)
SOLID WASTE AND IRRADIATED FUEL SHIPMENTS
JANUARY - JUNE 1982

A. SOLID WASTE SHIPPED OFF SITE FOR BURIAL OR DISPOSAL. (Not irradiated fuel.)

1. TYPE OF WASTE	UNIT	6 MONTH PERIOD	EST. TOTAL ERROR %
a. Spent resins, filter sludges, evaporator bottoms, etc.	m^3 Ci	97.299 123.60353	N/A N/A
b. Dry compressible waste, contaminated equipment, etc.	m^3 Ci	1539.11 10.67373	N/A N/A
c. Irradiated components, control rods, etc.	m^3 Ci	NONE	N/A
d. Other (Describe) Miscellaneous low-level waste	m^3 Ci	NONE	N/A

2. ESTIMATE OF MAJOR NUCLIDE COMPOSITION. (By Type of Waste)

		%	E(Curies)
a. Spent Resins, Filter Sludges, Evap. Bottoms, Diatomateous Earth, Etc.	Sr90	.522	.64564
	Sr89	19.972	24.68618
	Fe55	12.697	15.69454
	Cs134	4.156	5.13671
	Cs137	26.327	32.54062
	Co58	1.220	1.50773
	Mn54	2.712	3.35228
	Zn65	.450	.55669
	Co60	31.633	39.09916
	La-140	.019	.02323
	Ba-140	.005	.00623
	I-131	.004	.0049[illegible]
	Cr-51	.283	[illegible]
	TOTALS	100.000	123.60353

		%	E(Curies)
b. Dry Compressible Waste Contaminated Equipment	Co60	50.24	5.36260
	Co58	7.63	.81467
	Cs137	22.48	2.39956
	Cs134	6.75	.72011
	Fe55	1.75	.18635
	Fe59	1.14	.12171
	Sr89	.12	.01328
	Sr90	.01	.00027
	Zn65	.23	.02488
	Mn54	9.65	1.03030
	TOTALS	100.00	10.67373

c. N/A

d. N/A

3. SOLID WASTE DISPOSITION

Number of Shipments	Mode of Transportation	Destination
20	Tractor Trailer	Richland, Wash.
32	Tractor Trailer	Barnwell, S.C.

B. IRRADIATED FUEL SHIPMENTS (Disposition)

Number of Shipments	Mode of Transportation	Destination
NONE	N/A	N/A

ENCLOSURE 10
TO QUESTION 7

PILGRIM NUCLEAR POWER STATION

Radioactive Effluent and Waste Disposal Report
including
Radiological Impact on Humans

July 1 through December 31, 1982

BY: NUCLEAR OPERATIONS SUPPORT DEPARTMENT
ENVIRONMENTAL AND RADIOLOGICAL
HEALTH AND SAFETY GROUP

Date: March 1, 1983

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

RADIOACTIVE EFFLUENT AND WASTE DISPOSAL REPORT

INCLUDING RADIOLOGICAL IMPACT ON HUMANS

JULY 1 THROUGH DECEMBER 31, 1982

Prepared By: Christine E. Bowman

Christine E. Bowman
Senior Radiological Engineer

Approved By: Thomas L. Sowdon

Thomas L. Sowdon
Environmental Radiological
Health and Safety Group Leader

Date of Submittal: March 1, 1983

TABLE OF CONTENTS

LIST OF TABLES

LIST OF TABLES (cont.)

LIST OF TABLES (cont.)

1. INTRODUCTION AND SUMMARY

This report is issued for the period July-December 1982 in accordance with NRC Regulatory Guide 1.21, "Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water Cooled Nuclear Power Plants" (Rev. 1). The information supplied includes actual effluent releases, radioactive waste and meteorological data; doses from liquid releases, doses from gaseous releases and direct gamma radiation doses.

2. EFFLUENT, WASTE DISPOSAL AND WIND DATA

Radioactive liquid and gaseous releases, wind speed data together with measurement errors and solid waste disposal information are given in Tables 1A, 1B, 1C, 2A, 2B, 3, 4A-1, 4A-2, and supplemental information section in the standard Regulatory Guide 1.21 format.

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT

Supplemental Information

July-December 1982

Facility Pilgrim Nuclear Power Station Licensee DPR-35

1. **Regulatory Limits**

   a. Fission and activation gases: $\frac{Qs}{0.25/\bar{E}} + \frac{Qv}{0.10/\bar{E}} = \leq 1$

   b. Iodines: 2Ci per quarter

   c. Particulates, half-lives >8 days: $13(1.8E4Qs + 1.8E5Qv) \leq 1$

   d. Liquid effluents: 10Ci per quarter

2. **Maximum Permissible Concentration**

   Provide the MPCs used in determining allowable release rates or concentrations.

   a. Fission and activation gases: } 10 CFR 20
   b. Iodines: } Appendix B
   c. Particulates, half-lives >8 days: } Table II
   d. Liquid effluents: H-3 = 1 X 10^{-3} µCi/ml; all rest, 10 CFR 20, Appendix B, Table II

3. **Average Energy**

   Provide the average energy ($\bar{E}$) of the radionuclide mixture in releases of fission and activation gases, if applicable. $\bar{E}$ = 1 Mev

   MS = 0.304&0.287; RBV = 0.391&0.494 (3rd & 4th quarter)

4. **Measurements and Approximations of Total Radioactivity**

   Provide the methods used to measure or approximate the total radioactivity in effluents and the methods used to determine radionuclide composition.

   a. Fission and activation gases: } GeLi
   b. Iodines: } Istopic
   c. Particulates: } Analysis
   d. Liquid effluents:

5. **Batch Releases**

   Provide the following information relating to batch releases of radioactive materials in liquid and gaseous effluents.

   a. Liquid

      1. Number of batch releases: 77
      2. Total time period for batch releases: 87.48hrs
      3. Maximum time period for a batch release - 4.08hrs
      4. Average time period for batch releases: 1.14hrs
      5. Minimum time period for a batch release - 0.33hrs
      6. Average stream flow during periods of release of effluent into a flowing stream: 3.05E+5 GPM

   b. Gaseous (Not Applicable)

6. **Abnormal Releases**

   a.
   b. None

TABLE 1A

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT

GASEOUS EFFLUENTS - SUMMATION OF ALL RELEASES

July-December 1982

	Unit	Quarter (3)	Quarter (4)	Est. Total Error, %
A. Fission and activation gases				
1. Total release	Ci	< 1.07E+4	< 5.19E+3	2.49E+1
2. Average release rate for period	μCi/sec	< 1.35E+3	< 6.53E+2	
3. Percent of Technical Specification limit	%	< 1.77E-1	< 8.25E-2	
B. Iodines				
1. Total iodine-131	Ci	1.03E-2	9.32E-3	2.51E+1
2. Average release rate for period	μCi/sec	1.30E-3	1.17E-3	
3. Percent of Technical Specification limit	%	5.15E-1	4.66E-1	
C. Particulates				
1. Particulates with half-lives > 8 days	Ci	8.20E-3	8.01E-3	3.03E+1
2. Average release rate for period	μCi/sec	1.03E-3	1.01E-3	
3. Percent of Technical Specification limit	%	9.67E-2	8.72E-2	
4. Gross alpha radioactivity	Ci	< 5.14E-7	< 4.50E-7	
D. Tritium				
1. Total release	Ci	4.90E0	5.93E0	3.30E+1
2. Average release rate for period	μCi/sec	6.16E-1	7.46E-1	
3. Percent of Technical Specification limit	%	-	-	

TABLE 1B

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1982)

GASEOUS EFFLUENTS – ELEVATED RELEASE

July-December 1982

		CONTINUOUS MODE		BATCH MODE	
Nuclides Released	Unit	Quarter (3)	Quarter (4)	Quarter	Quarter
1. Fission gases					
krypton-85	Ci	1.62E-2	1.60E-2		
krypton-85m	Ci	7.69E+2	5.47E+2		
krypton-87	Ci	< 1.87E+2	< 4.58E+1		
krypton-88	Ci	8.99E+2	4.99E+2		
xenon-133	Ci	4.51E+3	3.07E+3		
xenon-135	Ci	3.73E+3	7.36E+2		
xenon-135m	Ci	< 1.54E+1	< 9.26E0		
xenon-138	Ci	< 3.75E+1	< 3.90E+1		
xenon-131m	Ci	-	-		
xenon-137	Ci	-	-		
xenon-133m	Ci	1.30E+2	8.49E+1		
Total for period	Ci	< 1.03E+4	5.03E+3		
2. Iodines					
iodine-131	Ci	4.66E-3	6.53E-3		
iodine-133	Ci	1.68E-2	2.24E-2		
iodine-135	Ci	< 1.22E-2	< 1.48E-2		
Total for period	Ci	< 3.37E-2	< 4.37E-2		
3. Particulates					
strontium-89	Ci	1.62E-3	2.78E-3		
strontium-90	Ci	1.73E-5	1.83E-5		
cesium-134	Ci	8.15E-6	2.61E-6		
cesium-137	Ci	7.38E-5	5.76E-5		
barium-lanthanum-140	Ci	3.55E-3	2.68E-3		
chromium-51	Ci	-	-		
manganese-54	Ci	1.28E-5	3.65E-6		
cobalt-58	Ci	-	2.09E-6		
iron-59	Ci	-	-		
cobalt-60	Ci	1.55E-4	3.97E-5		
zinc-65	Ci	-	-		
zirconium-niobium-95	Ci	-	-		
cerium-141	Ci	-	-		
cerium-144	Ci	-	1.53E-5		
ruthenium-103	Ci	-	-		
ruthenium-106	Ci	2.70E-5	-		

TABLE 1C

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1982)

GASEOUS EFFLUENTS - GROUND LEVEL RELEASE

July-December 1982

Nuclides Released	Unit	CONTINUOUS MODE Quarter (3)	CONTINUOUS MODE Quarter (4)	BATCH MODE Quarter	BATCH MODE Quarter
1. Fission gases					
krypton-85	Ci	< 1.49E-5	5.03E-6		
krypton-85m	Ci	< 3.46E+1	1.21E+1		
krypton-87	Ci	< 9.16E0	< 4.07E0		
krypton-88	Ci	< 1.55E+1	2.43E+1		
xenon-133	Ci	1.41E+2	5.99E+1		
xenon-135	Ci	1.86E+2	5.86E+1		
xenon-135m	Ci	-	-		
xenon-138	Ci	-	-		
Total for period	Ci	< 3.86E+2	< 1.59E+2		
2. Iodines					
iodine-131	Ci	5.66E-3	2.79E-3		
iodine-133	Ci	2.63E-2	1.18E-2		
iodine-135	Ci	4.26E-2	2.10E-2		
Total for period	Ci	7.46E-2	3.56E-2		
3. Particulates					
strontium-89	Ci	1.29E-3	1.53E-3		
strontium-90	Ci	2.55E-6	2.53E-6		
cesium-134	Ci	1.89E-6	4.46E-6		
cesium-137	Ci	6.64E-5	2.14E-5		
barium-lanthanum-140	Ci	1.24E-3	7.85E-4		
manganese-54	Ci	1.25E-5	1.31E-6		
cobalt-58	Ci	-	3.74E-6		
iron-59	Ci	-	-		
cobalt-60	Ci	1.29E-4	5.90E-5		
zinc-65	Ci	-	-		
zirconium-niobium-95	Ci	-	-		
cerium-141	Ci	-	-		
ruthenium-103	Ci	-	-		
ruthenium-106	Ci	-	2.60E-5		

TABLE 2A

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1982)

LIQUID EFFLUENTS - SUMMATION OF ALL RELEASES

JULY-December 1982

	Unit	3rd Quarter	4th Quarter	Est. Total Error, %
A. Fission and activation products				
1. Total release (not including tritium, noble gases, or alpha)	Ci	3.09E-2	1.25E-1	2.98E+1
2. Average diluted concentration during period	μCi/ml	7.39E-9	6.65E-8	
3. Percent of applicable limit	%	3.09E-1	1.25E0	
B. Tritium				
1. Total release	Ci	8.29E-4	4.55E-1	3.00E+1
2. Average diluted concentration during period	μCi/ml	1.98E-10	2.42E-7	
3. Percent of applicable limit	%	1.98E-3	2.42E0	
C. Dissolved and entrained gases				
1. Total release	Ci	-	5.39E-3	3.98E+1
2. Average diluted concentration during period	μCi/ml	-	2.87E-9	
3. Percent of applicable limit	%	-	-	
D. Gross alpha radioactivity				
1. Total release	Ci	< 6.60E-6	< 1.65E-5	4.01E+1
E. Volume of waste released (prior to dilution)	liters	8.47E+4	2.01E+5	2.00E+1
F. Volume of dilution water used during period	liters	4.18E+9	1.88E+9	2.00E+1

TABLE 2B

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1982

LIQUID EFFLUENTS

July-December 1982

Nuclides Released	Unit	CONTINUOUS MODE 3rd Quarter	CONTINUOUS MODE 4th Quarter	BATCH MODE Quarter	BATCH MODE Quarter
strontium-89	Ci	1.64E-5	2.10E-5		
strontium-90	Ci	4.70E-5	7.78E-5		
cesium-134	Ci	3.30E-4	7.05E-4		
cesium-137	Ci	3.73E-3	9.65E-3		
iodine-131	Ci	5.87E-6	4.12E-5		
cobalt-58	Ci	4.42E-5	1.96E-3		
cobalt-60	Ci	8.67E-3	3.66E-2		
iron-59	Ci	3.49E-6	5.30E-4		
zinc-65	Ci	5.09E-5	5.37E-5		
manganese-54	Ci	6.49E-4	3.74E-3		
chromium-51	Ci	4.02E-5	6.57E-3		
zirconimum-niobium-95	Ci	-	1.21E-6		
molybdenum 99-technetium 99m	Ci	-	5.71E-5		
barium-lanthanum-140	Ci	1.03E-6	4.38E-5		
cerium-141	Ci	2.14E-6	1.10E-4		
iodine-133	Ci	-	3.04E-6		
cerium-144	Ci	-	-		
silver-110m	Ci	-	8.01E-4		
iron-55	Ci	1.28E-2	2.41E-2		
unidentified	Ci	4.49E-3	3.95E-2		
Total for period (above)	Ci	3.09E-2	1.25E-1		
xenon-133	Ci	-	2.18E-[illegible]		
xenon-135	Ci	-	3.21E-[illegible]		

TABLE 3

EFFLUENT AND WASTE DISPOSAL SEMI-ANNUAL REPORT (1982)
SOLID WASTE AND IRRADIATED FUEL SHIPMENTS
JULY - DECEMBER 1982

A. SOLID WASTE SHIPPED OFF SITE FOR BURIAL OR DISPOSAL. (not irradiated fuel)

1. TYPE OF WASTE	UNIT	6 MONTH PERIOD	EST. TOTAL ERROR %
a. Spent resins, filter sludges, evaporator bottoms, etc.	m^3 Ci	99.007 819.10	N/A N/A
b. Dry compressible waste, contaminated equipment, etc.	m^3 Ci	547.666 5.14564	N/A N/A
c. Irradiated components, control rods, etc.	m^3 Ci	none none	N/A N/A
d. Other (describe) Miscellaneous low-level waste	m^3 Ci	none none	N/A N/A

2. ESTIMATE OF MAJOR NUCLIDE COMPOSITION. (by type of waste)

		%	E(Curies)
a. Spent Resins, Filter	Co-60	41.324	338.48620
Sludges, Evaporator	Co-58	3.864	31.65107
Bottoms, etc.	Cs-137	13.426	109.97068
	Cs-134	1.489	12.19371
	Fe-55	11.164	99.44832
	Fe-59	.597	4.89055
	I-131	.464	3.79925
	I-133	.070	.57668
	La-140	.220	1.80569
	Ba-140	.019	.15592
	Sr-89	15.478	126.78505
	Sr-90	.345	2.82477
	Sr-91	.003	.02146
	Tc-99m	.040	.32557
	Zn-65	.723	5.92615
	Mn-54	4.614	37.79740

2. ESTIMATE OF MAJOR NUCLIDE COMPOSITION. (by type of waste)

CONTINUED

		%	E(Curies)
a. Spent Resins, Filter Sludges, Evap. Bottoms, Diatomateous Earth, etc. continued	Nb-95	.002	.01495
	Cr-51	6.090	49.88606
	Ag-110m	< .001	.00641
	Ce-141	.030	.24916
	Ru-103	.014	.11290
	Sr-92	.001	.00691
	Sb-124	.010	.08267
	Xe-133	< .001	.00034
	Xe-135	.004	.03266
	Mo-99	.007	.05629
	TOTAL:	100.000	819.10682

		%	E(Curies)
b. Dry Compressible Waste, Contaminated Equipment	Co-60	17.46	.89843
	Co-58	6.32	.32546
	Cs-137	6.04	.31058
	Cs-134	1.65	.08565
	Fe-59	1.17	.06038
	I-131	2.74	.14116
	Ba-140	3.76	.19341
	Zn-65	.86	.04430
	Mn-54	3.39	.17448
	Cr-51	56.60	2.91179
	TOTAL:	100.000	5.14564

c. N/A

d. N/A

3. SOLID WASTE DISPOSITION

Number of Shipments	Mode of Transportation	Destination
37	Tractor Trailer	Barnwell, S.C.
2	Tractor Trailer	Richland, Wash.

4. IRRADIATED FUEL SHIPMENTS (Disposition)

Number of Shipments	Mode of Transportation	Destination
none	N/A	N/A

QUESTION 8. In recent years, Boston Edison has had unsatisfactory ratings in the area of fire protection. I would like to know if Pilgrim is now in full compliance with fire protection requirements? Are all barriers, fire doors and penetration seals repaired and capable of passing required testing? Are fire watches still required in certain areas of the plant? How many fire watches are still needed? Will the NRC require Edison to complete the upgrading of the entire fire protection system prior to allowing restart? How many maintenance requests are still outstanding in the area of fire protection? Please also comment on the condition of the Halon system in the computer room at the plant and the smoke detectors over the spent fuel pool.

ANSWER.

Pilgrim is either in compliance or will be in compliance with its fire protection requirements prior to restart.

During the last one and one-half to two years, Boston Edison Company has made significant improvements in their entire fire protection program. Additional personnel with extensive experience in nuclear power plant fire protection have been hired. Realignment of responsibilities and authority among these licensee personnel have strengthened the entire fire protection program and provided a higher level of accountability and continuity of effort that has resulted in substantial improvement in the program. This is evidenced by the methodology and thoroughness exhibited in identifying and correcting deficiencies.

One activity of the additional licensee fire protection personnel described above was the licensee has performed a reevaluation of plant fire protection features, comparing those features against NRC requirements and guidance, in an effort to determine (a) the level of actual compliance, and (b) the adequacy of the features provided to prevent unacceptable fire damage.

During the course of this reevaluation the licensee found several cases where they did not literally comply with the NRC requirements or specific commitments they had made earlier. The licensee, however, provided justification to demonstrate adequate protection against unacceptable fire damage and on that basis, asked for exemptions from those requirements. In most cases the staff granted the exemptions. In those cases where the staff did not agree with the justification provided, the licensee made modifications so as to be in compliance.

Because of the more or less constant activity at operating plants, temporary changes, repairs and, modifications, may result in a particular condition that is not in compliance. These situations are contemplated by the licensee and provisions are in place to assist in identifying the situation beforehand, providing interim protection measures (such as fire watches) and maintaining administrative control of the situation to assure that the out-of-compliance condition is corrected.

The licensee has indicated that all modifications and work associated with upgrading required fire barriers, fire doors and penetration seals has been completed. The licensee has committed to having all of the necessary documentation concerning the above work completed prior to plant startup.

Fire watches continue to be used in some areas at Pilgrim as well as most operating plants. At the beginning of the present outage approximately 18 months ago, eight persons per shift were assigned full time responsibility for continuous or roving fire watches covering approximately 180 individual deficiencies. As of March 17, 1988, no continuous fire watches are required. Two persons per shift are assigned roving as fire watches covering 41 separate deficiencies throughout the entire plant. Of those 41 deficiencies, 25 are related to fire barriers, 15 are related to maintenance activities, and one is related specifically to activities pertaining to the outage.

Some minor upgrading to the fire protection systems may remain at the time Pilgrim restarts. However, those modifications yet to be completed will have been identified and the schedules for completion will have been reviewed for acceptability by the staff.

One hundred and sixty-one maintenance requests were still outstanding in the area of fire protection on March 17, 1988. However, this number by itself does not give an accurate picture of the Pilgrim fire protection maintenance program. On January 5, 1987 there were 260 open maintenance requests related to fire protection. Since January 1, 1987, approximately 1,480 new fire protection-related maintenance requests have been generated and approximately 1,580 have been closed.

You also asked for our comments on the condition of the Halon System in the computer room, and smoke detectors over the spent fuel pool. A computer located in a small room adjacent to the Cable Spreading Room is being phased out. The room is protected by an operable automatic Halon fire suppression system. A new plant computer has been installed next to the Technical Support Center and the primary fire protection is provided by a sprinkler system with secondary protection provided by an automatic Halon fire suppression system. Both of these systems are operable.

Six smoke detectors are located over the Spent Fuel Pool in the ventilation system exhaust ducts. Four of the six detectors have already been tested during this current plant outage. The other two are scheduled for testing prior to plant startup.

QUESTION 9. How many automatic and manual scrams have occurred at Pilgrim since the plant became operational? What is the annual industry-wide average?

ANSWER.

Table 1 provides data on unplanned automatic and manual scrams during operational modes (criticality to 100% power) for Pilgrim from 1984 through 1987 compiled from licensee event reports submitted pursuant to 10 CFR Part 50.72 and 10 CFR Part 50.73. The comparable industry average rates are also provided in Table 1. Prior to 1984, reactor scrams were not directly reportable to the NRC (Pilgrim entered commercial service December 1, 1972).

Enclosure:

Table of Unplanned Scrams
When Critical for Pilgrim
and Industry

Enclosure to Question 9

Table 1

Unplanned Scrams When Critical for Pilgrim and Industry

1984 - 1987

	1984*	1985	1986	1987**
Pilgrim				
Automatic	0	4	4	0
Manual	0	0	0	0
Industry Average				
Automatic	5.4	5.0	4.0	3.2
Manual	0.6	0.5	0.5	0.6

*Pilgrim critical hours for 1984 = 170.

**Pilgrim critical hours for 1987 = 0.

QUESTION 10. How many "Unusual Events" and how many "Alerts" have been declared at Pilgrim since 1972? Please describe and give the date of each report. How does this compare to the industry-wide average?

ANSWER.

The NRC did not use the terms "unusual events" and "alerts" until 1980 and did not established them as reportable categories in our regulations until 1983. Our computer records of notifications to the NRC Operations Center show that Pilgrim has declared 12 Unusual Events and no Alerts since 1983. Of the 12 Unusual Events, 2 were caused by fires in nonsafety related equipment, and 1 was due to a potentially contaminated individual being transferred offsite for medical treatment. The remainder were attributed to safety system inoperability, which necessitated shutdown of the plant in accordance with the plant's Technical Specifications. Two tables are enclosed - the first compares the number of unusual events at Pilgrim since 1983 with the industry average per year; and the second provides descriptive data and the date for each unusual event at plants.

Enclosure:

Tables of Unusual Events at
Pilgrim Nuclear Station

A comparison of Pilgrim Unusual Events versus the industry average follows:

Year	Industry Unusual Events	Licensed Units	Industry Average	Pilgrim Unusual Events
*1982	-	-	-	-
1983	205	85	2.4	0
1984	224	91	2.0	1
1985	312	98	3.2	5
1986	209	104	2.0	5
1987	231	109	2.1	0
*1988	-	-	-	-
5 Year Total			11.7	11

*This table was prepared from data contained in computerized data base from August 1982 to the present. For comparison purposes, incomplete data for 1982 and 1988 are not shown. However, Pilgrim did report Unusual Events (a fire in a face mask fitting machine) on August 18, 1982 and on February 11, 1988 (a fire in the machine shop). Pilgrim also had one Alert on June 3, 1982 relating to a withdrawn incore detector resulting in abnormal radiation levels. This event lasted approximately 2 hours. Pilgrim had no other Alerts from 1983 to 1987; however, Alerts have been reported from other licensed facilities.

Enclosure to Question 10

Unusual Events at Pilgrim Nuclear Station

August 1983 to Present

Event	Description
4/26/84	Potentially contaminated man taken to hospital.
5/16/85	2 safety system trains inoperable.
05/23/85	2 safety system trains inoperable.
09/20/85	2 safety system trains inoperable.
10/15/85	2 safety system trains inoperable.
11/04/85	Residual Heat Removal safety train A inoperable.
01/04/86	2 of 8 Main Steam Isolation Valves fail closure time test.
01/09/86	Fire in line to hydrogen storage tanks.
02/11/86	Low pressure coolant injection inoperable.
02/14/86	2 safety system trains inoperable.
04/11/86	Loss of containment integrity.
02/11/88	Fire in machine shop.

QUESTION 11. How many violations of NRC regulations have occurred at Pilgrim since it began operation? What is the industry-wide average?

ANSWER.

The NRC does not maintain industry wide statistics on the total numbers of violations per plant.

In order to provide this requested data for the Pilgrim facilties, a review of inspection report data was performed. Our review indicated that Pilgrim was cited approximately 425 times for violations or deviations since the plant began operation in June, 1972 through the end of 1987. This number however, does not reflect whether the citations involved individual or multiple violations, whether the citations were subsequently withdrawn, or the severity level of the violations. Moreover, enforcement history is only one of a variety of factors NRC considers in assessing licensee performance.

QUESTION 12. There have been a number of allegations concerning the illegal dumping of radioactive waste on Boston Edison property. Concerns have also been raised over Edison's use of the town dump for disposal of radioactive material. Would you please describe what monitoring the NRC conducts or requires on materials and waste leaving the Pilgrim site. Has the NRC or the licensee performed tests on Edison property and at the town dump to ensure that there are no elevated levels of radiation at areas suspected of containing radioactive waste? Where and when were tests conducted? What were the results?

ANSWER.

The NRC staff does not itself monitor materials and waste leaving the Pilgrim site. The licensee is required to monitor all items containing or contaminated with radioactivity that leave the site and there are several facility procedures that provide specific guidance and instructions to plant health physics workers regarding this activity. All radioactive wastes that are sent to sites specifically intended for burial must meet federal regulations for radiation dose rate and contamination levels as well as special requirements of the burial sites. NRC performs routine inspections of the radioactive transportation area to ensure that licensees are conforming to these regulatory requirements. Further, onsite materials that have the potential of being contaminated and are being shipped offsite are surveyed prior to being shipped. The licensee is not allowed to dispose of contaminated objects in non-radwaste facilities without obtaining a special variance required by in 10 CFR Part 20.302(a). BECo has not applied for these variances. To our knowledge, no contaminated objects have been disposed of in the town dump or in other public facilities not specifically intended for contaminated objects.

The NRC received allegations that contaminated shrubs had been removed from the site and improperly disposed of on BECo property in 1987. NRC inspectors determined that appropriate surveys were performed, measurements were within established limits and properly recorded prior to offsite disposal. An NRC inspector accompanied by the licensee collected clippings from the shrubs which were disposed of offsite. The clippings were independently analyzed by the NRC. Only one sample had detectable levels when we used sensitive laboratory instruments but was not detectable using standard survey meters.

The contamination levels were lower than typical soil background levels and they posed no health hazard (see pages 12 - 13 of the enclosed Inspection Report 50-293/87-57, dated March 11, 1988, p.12). NRC has not performed surveys for contamination of the town dump or at other BECo properties and does not routinely perform contamination surveys of this type. As stated in the Inspection Report, the inspectors reviewed the licensee's program for release of material from the site and concluded that it was adequate.

Enclosure:
Inspection Report dated 3/11/88

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406

Enclosure to Question 1

MAR 1 1 1988

Docket No. 50-293

Boston Edison Company
ATTN: Mr. Ralph G. Bird
Senior Vice President - Nuclear
800 Boylston Street
Boston, Massachusetts 02199

Gentlemen:

Subject: Region I Inspection Report No. 50-293/87-57

This refers to the routine safety inspection (50-293/87-57) conducted by Messrs. C. Warren, J. Lyash and T. Kim of this office on December 7, 1987 to January 19, 1988 at the Pilgrim Nuclear Power Station, Plymouth, Massachusetts. Areas examined during this inspection are described in the NRC Region I Inspection Report which is enclosed with this letter.

Based on the results of this inspection, it appears that one of your activities related to high radiation area access control was not conducted in full compliance with NRC requirements, as set forth in the Notice of Violation enclosed herewith as Appendix A. The problem was identified by your staff. However, a Notice of Violation is being issued because effective corrective actions apparently have not been taken for previous problems with high radiation area access control. In addition to following the instructions of Appendix A in preparing the required response, please include those actions you intend to take to preclude recurrence of this problem by insuring that your corrective actions are effective and lasting.

Two significant integrated plant tests were successfully executed during the inspection period. Preplanning and control of these activities was generally strong. We also observed that increased management involvement in assuring effective problem followup has resulted in substantial improvement. Equipment failures identified as a result of an unanticipated safety system actuation however, indicate the need for stronger post-work test practices and a thorough power ascension test program.

The response directed by this letter and the accompanying Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511.

~~8803210184~~ 2pp.

Your cooperation with us in this matter is appreciated.

Sincerely,

Samuel J. Collins

Samuel J. Collins, Deputy Director
Division of Reactor Projects

Enclosures:
1. Appendix A, Notice of Violation
2. NRC Region I Inspection Report No. 50-293/87-57

cc w/encls:
R. Barrett, Nuclear Operations Manager
B. McIntyre, Chairman, Department of Public Utilities
Chairman, Plymouth Board of Selectmen
Chairman, Duxbury Board of Selectmen
Plymouth Civil Defense Director
J. Keyes, Boston Edison Regulatory Affairs and Programs
E. Robinson, Nuclear Information Manager
R. Swanson, Nuclear Engineering Department Manager
The Honorable Edward J. Markey
The Honorable Edward P. Kirby
The Honorable Peter V. Forman
S. Pollard, Secretary of Energy Resources
P. Agnes, Assistant Secretary of Public Safety, Commonwealth of Massachusetts
R. Shimshak, MASSPIRG
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
NRC Resident Inspector
Commonwealth of Massachusetts (2)

bcc w/encls:
Region I Docket Room (with concurrences)
W. Russell, RA
M. Perkins, DRMA (w/o encls)
R. Blough, DRP
L. Doerflein, DRP
R. Bores, DRSS
S. Collins, DRP
C. Anderson, DRS
D. McDonald, LPM, NRR
T. Chandrasekaran, SPLB, NRR
M. Callahan, OCA
J. Nickerson

APPENDIX A

NOTICE OF VIOLATION

Boston Edison Company
Pilgrim Nuclear Power Station

Docket No. 50-293
License No. DPR-35

As a result of the inspection conducted on December 7, 1987 to January 19, 1988, and in accordance with the NRC Enforcement Policy (10 CFR 2, Appendix C), the following violation was identified. Three previous Notices of Violation dated March 13, March 23, 1987, and April 28, 1987 were issued for problems related to the control of Locked High Radiation Areas. It is evident that corrective actions taken in response to these Notices of Violation have not been effective in precluding recurrence.

> The Station Technical Specification 6.11, "Radiation Protection Program," requires that "procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure."
>
> The Station Procedure 6.1-012, "Access to High Radiation Areas," requires in part that the areas controlled under this procedure remain locked or guarded at all times.
>
> Contrary to the above, on December 15, 1987, December 27, 1987, and on January 8, 1988, doors to the areas being controlled as Locked High Radiation Areas were found to be unlocked and unattended, in violation of the Station Procedure 6.1-012.
>
> This is a Severity Level IV Violation (Supplement I).

Pursuant to the provisions of 10 CFR 2.201, Boston Edison Company is hereby required to submit to this office within thirty days from the receipt of the letter which transmitted this Notice, a written statement or explanation in reply, including: (1) the corrective steps which have been taken and the results achieved; (2) corrective steps which will be taken to avoid further violations; and (3) the date when full compliance will be achieved. Where good cause is shown, consideration will be given to extending this response time.

~~8803210187~~ 1p.

U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket/Report No. 50-293/87-57

Licensee: Boston Edison Company
800 Boylston Street
Boston, Massachusetts 02199

Facility: Pilgrim Nuclear Power Station

Location: Plymouth, Massachusetts

Dates: December 7, 1987 - January 19, 1988

Inspectors: C. Warren, Senior Resident Inspector
J. Lyash, Resident Inspector
T. Kim, Resident Inspector

Approved By: [signature] 3-11-88
A. Randy Blough, Chief Date
Reactor Projects Section No. 3B

Areas Inspected: Routine resident inspection of plant operations, radiation protection, physical security, plant events, maintenance, surveillance, outage activities, and reports to the NRC. The inspection consisted of 350 hours of direct inspection. Principal licensee management representatives contacted are listed in Attachment I. Observations made by the NRC Region I, Regional Administrator during a tour on December 8, 1987 are documented in Attachment II of this report. A copy of Attachment II was provided to licensee management for followup.

Results:

Violation: Repeated occurrences of locked high radiation area doors being left open and unattended were identified by the licensee. Problems with high radiation area access control have been previously identified and were the subject of violations during inspections 50-293/87-03 and 50-293/87-11. Corrective actions taken in response to these findings have not prevented their recurrence. (Section 3.b, VIO 87-57-01)

Unresolved Item: The licensee identified that two reactor vessel level gauges were incorrectly installed. A licensee investigation is currently ongoing to determine the cause and to assess the adequacy of post installation test. (Section 4.d, UNR 87-57-02)

~~8803210189~~ 30 pp.

Concerns:

1. The licensee experienced safety related equipment malfunctions upon receiving a spurious reactor scram signal on January 17, 1988. (Section 4.d)

2. Inadequate procedures and planning of surveillance tests resulted in unnecessary engineered safety feature actuations. (Section 3.a)

3. Poor preplanning and control of maintenance was noted during an electrical relay replacement. A similar problem was the subject of a violation during inspection 50-293/87-50. (Section 4.c)

4. Weak identification and tracking of lifted leads and jumpers led to a water spill in the high pressure coolant injection system room during the integrated leak rate test. (Section 6.0)

5. The prelube pump for the "B" emergency diesel generator failed to restart during a surveillance test. An identical failure occurred during a loss of offsite power event on November 12, 1987. Licensee followup appeared adequate but the failure root cause has not been identified. (Section 3.b)

6. The inspectors evaluated the erosion of construction dirt into wetlands area. The inspector's independent survey of the area, and the licensee's analyses indicate that the level of activity does not represent a health or safety concern. However, the material should not be allowed to erode. (Section 3.c)

Strengths:

1. The licensee's preparation and execution of the reactor vessel hydrostatic test was well organized and controlled. (Section 5.0)

2. The licensee's response to a January 17, 1988 reactor scram signal and subsequent equipment malfunctions was prompt, thorough and effective. (Section 4.d)

3. Using non-nuclear steam for testing of high pressure coolant injection system and reactor core isolation cooling system enabled the licensee to discover problems which may not have been easily identifiable using nuclear steam due to radiological conditions. (Section 3.b)

TABLE OF CONTENTS

DETAILS

1.0 Summary of Facility Activities

The plant was shutdown on April 12, 1986 for unscheduled maintenance. On July 25, 1986, Boston Edison announced that the outage would be extended to include refueling and completion of certain modifications. The reactor core was defueled on February 13, 1987. The licensee completed fuel reload on October 14, 1987. Reinstallation of the reactor vessel internal components and the vessel head was also subsequently completed.

During this report period, the licensee performed the reactor vessel hydrostatic test and the primary containment integrated leak rate test (ILRT) as described in Sections 5.0 and 6.0. On December 9, 1987, Pilgrim Station conducted a partial participation emergency preparedness exercise. On December 14, 1987 the licensee announced as part of a planned management realignment, the appointment of eight managers to key management positions in the licensee nuclear organization at Pilgrim Station. The details of the management realignment are described in Section 7.0.

NRC inspection activities during the report period included: 1) observation of the licensee's annual emergency preparedness exercise on December 9, 1987, 2) NRC Reactor Operator Licensing examinations were administered to eight candidates on the week of December 7, 1987, 3) observation of the primary containment ILRT and review of the test results during the week of December 21, 1987. The results of these inspections are documented in inspection reports 50-293/87-54, 50-293/87-56, and 50-293/87-58. In addition, representatives of the NRC's Office of Investigation were onsite December 3, December 7, and December 8, 1987 to interview onsite security personnel. On December 8, 1987, the NRC Regional Administrator for Region I, Mr. William T. Russell, toured the plant with the resident inspectors. On January 7, 1988, Dr. Thomas E. Murley, Director of the Office of Nuclear Reactor Regulation (NRR) and other NRC representatives toured the plant with the resident inspectors.

2.0 Followup on Previous Inspection Findings

(Closed) Unresolved Item 82-24-02 - Discrepancies in the Licensee's Response to IE Bulletin 79-08

Previous reviews of this item are documented in the inspection reports 50-293/82-30, 50-293/83-01, 50-293/83-14, and 50-293/84-26. IE Bulletin (IEB) 79-08 and the TMI Action Plan Item II.E.4.2 required licensees to review the containment isolation initiation design and procedures to ensure proper initiation of containment isolation, upon receipt of an automatic containment isolation signal. The licensee provided the results of their review in letters dated April 25, and August 21, 1979.

The licensee stated that the RBCCW supply and return lines, instrument air line, RHR to spent fuel pool cooling tie line, and torus make up line would be manually isolated and that station procedures would specify the requirements for manual isolation if a containment isolation signal was received. This was documented as acceptable by NRC:NRR in letters to the licensee dated December 18, 1979 and April 3, 1980. However, an inspector identified that manual isolation of these lines with qualified valves is not possible. Any valve which is used for primary containment isolation must meet Seismic Class I (FSAR section 12.2) and applicable 10 CFR 50, Appendix J, containment leakage testing criteria. Further, if manual operation of a valve is required to effect containment isolation, the isolation point for the valve must also be accessible under those conditions which make its use necessary.

In response to the inspector's questions, the licensee re-evaluated their response to the IEB 79-08 and TMI Action Plan Item II.E.4.2, and concluded that isolation of these lines is assured by the use of Seismic Class I check valves. The licensee also agreed that isolation for the RBCCW supply line, instrument air line, RHR to spent fuel pool cooling tie line, and torus makeup line cannot be performed by manual valve closure. The RBCCW return line from the drywell can meet the isolation valve criteria with MOV-4002 which is seismic class I, local leak rate tested and can be closed by a control switch located in the main control room. The licensee subsequently submitted a supplemental response to IE Bulletin 79-08 and TMI Action Plan Item II.E.4.2 on October 24, 1984 correcting the previous response. The inspector reviewed the supplemental response and verified that the contents were consistent with the conclusions drawn from the licensee's re-evaluation and the FSAR. Both RBCCW supply line and instrument air line are considered Class C lines in Section 7.3 of the FSAR since they penetrate containment but have no interaction with the primary containment free space or the reactor vessel. According to the original design criteria, a single check valve is provided to attain isolation for a Class C line. These check valves are seismic class I and local leak rate tested. The inspector reviewed the results of local leak rate test data for these check valves which were performed on June 12 and July 26, 1987 and found no discrepancies. The torus makeup line is identified as Class B in Section 7.3 of the FSAR. The torus makeup line is non-essential and ties the condensate transfer system into the RHR test line, which penetrate primary containment and ends below the torus water level. For water-sealed Class B lines such as the torus makeup system, the original plant design bases allow one isolation valve in addition to the water seal to meet isolation requirements. Also, the Safety Evaluation by the NRR on Appendix J Review indicate that Type C testing is not required for valves in lines which terminate below the level of the suppression pool. As for the RHR to spent fuel pool line, the licensee revised the operating procedures 2.2.85, Fuel Pool Cooling and Filtering System, prohibiting the use of the RHR to spent fuel pool lines except in cold shutdown. The inspector had no further questions. This item is closed.

(Closed) Inspector Follow Item (IFI 87-27-02) - Cracking of Surge Ring Brackets in Large GE Motors

On July 2, 1987, IE Information Notice 87-30, Cracking of Surge Ring Brackets in large GE motors, was issued. The purpose of the notice was to alert recipients of a potential for failure of surge ring brackets and cracking of felt blocks in large, vertical electric motors manufactured by General Electric Co. Felt blocks are used in large electric motors to keep the windings separated where they loop back at the end of the stator. The blocks are attached to a surge ring that is held in place by L-shaped surge ring brackets welded to the surge ring and bolted to the motor casing. Failure of these surge ring brackets and cracking of the felt blocks allows movement and wear of the end-turns, leading to a reduction in insulation resistance and possible motor failure. In addition, broken pieces of the surge ring bracket may enter the space between the stator and the rotor, resulting in electrical or mechanical motor degradation.

Following an investigation to determine the applicability of the subject notice to the Pilgrim Station, the licensee found that RHR, core spray, and recirculation pump motors were potentially affected. RHR and core spray pump motors were overhauled on site by GE under contract with the licensee in 1986. The surge ring brackets were not inspected during the overhaul. However, small cracks were found on the "A" and "C" RHR pump motor winding felt blocks. The amount of cracking found was dispositioned by GE to be acceptable and a normal phenomenon found in form-wound motors. On July 27 through August 5, 1987, GE performed a surge ring bracket inspection of the RHR and recirculation pump motors using a boroscope with the motors in place. The inspection of the RHR motors (A thru D) revealed absence of cracks on the surge ring brackets. During the inspection of the "B" recirculation pump motor, it was noted that the recirc motor surge ring bracket construction is of the bolt and stud design, whereas the RHR and core spray motor brackets are of the L-shaped design. The L-shaped design configuration is known to have the potential of cracking, according to the IE Notice 87-30 and the GE letter to the licensee dated July 14, 1987.

During the week of October 26, 1987, "B" core spray pump motor was disassembled and the surge ring brackets inspected by G.E. Due to the geometry of the core spray pump motor internals, there is limited access for the bore scope, therefore, this inspection could not be accomplished without partial disassembly of the motor. It was verified that the design had 12 brackets per surge ring and two surge rings for the top end turn assembly and two surge rings for the bottom end turn assembly. None of the brackets had indications of cracking. The licensee scheduled the inspection of the "A" core spray pump motor during the next outage because of scheduling conflicts. The licensee indicated that based on the inspection results of the RHR and "B" core spray pump motors, postponement of the "A" core spray pump motor inspection is justified. The licensee also added that the number of operating hours and starts are similar between the A and B core spray pump motors since both core spray systems' testing and surveillance requirements are similar. The inspector had no further questions. This item is closed.

(Closed) Unresolved Item 87-45-05 - Failure to Issue Licensee Event Reports

In inspection report 50-293/87-45 the NRC identified three engineered safety feature actuations which appeared to be reportable under 10 CFR 50.73 but had not been reported by the licensee. The licensee reviewed the three actuations, agreed that they should have been reported and agreed to issue License Event Reports (LER) to document the occurrences. In addition the licensee agreed to perform a review of previous actuations to determine if any additional reports were needed.

During this inspection period the licensee's compliance section conducted a review of all Failure and Malfunction Reports (F&MR) issued from April 1986 through the present. This review identified four F&MRs that fit the description of an ESF actuation under the current BECo interpretation of NUREG 1022. The licensee will submit LERs to document the following ESF actuations at a later date.

-- 4/28/87 Initiation signal to both Emergency Diesel Generators (EDG)

-- 6/7/87 Actuation of Reactor Building Isolation and Standby Gas Treatment System start signal

-- 9/17/87 Auto start of "A" EDG

-- 10/6/87 Reactor Water Cleanup and Shutdown Cooling System Isolation

These LERs will be reviewed upon issue as part of the normal resident inspection program. The inspector has reviewed the licensee's actions in addressing open item 87-45-05 and is satisfied that those actions were thorough and timely. This item is closed.

3.0 Routine Periodic Inspections

The inspectors routinely toured the facility during normal and backshift hours to assess general plant and equipment conditions, housekeeping, and adherence to fire protection, security and radiological control measures. Inspections were conducted between 10:00 p.m. and 6:00 a.m. on January 17, 18, and 19, 1988 for a total of four hours and during the weekends of December 12, 19, 27, 1987 and January 3, 9, 17, 1988 for a total of 17 hours. Ongoing work activities were monitored to verify that they were being conducted in accordance with approved administrative and technical procedures, and that proper communications with the control room staff had been established. The inspector observed valve, instrument and electrical equipment lineups in the field to ensure that they were consistent with system operability requirements and operating procedures.

During tours of the control room the inspectors verified proper staffing, access control and operator attentiveness. Adherence to procedures and limiting conditions for operations was evaluated. The inspectors examined equipment lineup and operability, instrument traces and status of control room annunciators. Various control room logs and other available licensee documentation were reviewed.

The inspector observed and reviewed outage, maintenance and problem investigation activities to verify compliance with regulations, procedures, codes and standards. Involvement of QA/QC, safety tag use, personnel qualifications, fire protection precautions, retest requirements, and reportability were assessed.

The inspector observed tests to verify performance in accordance with approved procedures and LCO's, collection of valid test results, removal and restoration of equipment, and deficiency review and resolution.

Radiological controls were observed on a routine basis during the reporting period. Standard industry radiological work practices, conformance to radiological control procedures and 10 CFR Part 20 requirements were observed. Independent surveys of radiological boundaries and random surveys of nonradiological points throughout the facility were taken by the inspector.

Checks were made to determine whether security conditions met regulatory requirements, the physical security plan, and approved procedures. Those checks included security staffing, protected and vital area barriers, personnel identification, access control, badging, and compensatory measures when required.

a. Surveillance Testing

-- Diesel Generator Prelube Pump Failure

On December 13, 1987, the prelube pump for the "B" emergency diesel generator (EDG) failed to restart on demand during a routine surveillance test. Upon disassembly it was identified that a small piece of metal had become lodged between the pump rotor and idler gear. The interference from the metal caused the pump motor breaker to trip on pump start. An identical failure occurred during a loss of offsite power event on November 12, 1987. In that case the failure caused a lengthy delay in returning an idle diesel to service. While not required for diesel operation, the prelube system reduces EDG bearing wear during equipment start.

In response to the failures, the licensee drained and inspected the lube oil sump, and disassembled and inspected the lube oil filters, strainers and heater. The lube oil heater was found to have failed in the energized mode resulting in significant carbon deposits in the heater and filter. No appreciable deposits were found in the lube oil sump. In addition, a piece of filter element packaging material was found in the lube oil filter housing. No foreign material which could have contributed to the prelube pump failure, however, was found. The pump was replaced and the diesel was returned to service. No additional failures occurred during the inspection period. The two pumps which failed had in-sequence serial numbers. Licensee Quality Control personnel performed magnetic particle and dye-penetrant testing of the internals of a third in-sequence pump in the warehouse. No flaws were noted. The licensee is pursuing the root cause of the failures in cooperation with the pump vendor, Viking Pump. The licensee stated at the exit interview that the "A" EDG prelube pump and lube oil heater would be inspected during the next "A" diesel outage. The inspector will continue to monitor licensee followup to this problem.

- Steam Testing of the High Pressure Coolant Injection and Reactor Core Isolation Cooling Systems

The licensee completed full pressure steam testing of the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) system turbines by utilizing temporary oil fired auxiliary boilers as a source of non-nuclear steam. The full pressure steam testing is part of a post-maintenance and system operability check. Both HPCI and RCIC systems were overhauled during the current outage. Utilizing temporary test procedures TP 87-198 and TP 87-199, the HPCI/RCIC testing included turbine overspeed trip, pump full flow capacity and operation from the alternate shutdown panels. Also during the test, the suction path was changed from the condensate storage tank to the torus and back.

During the testing, several problems were identified by the licensee in both HPCI and RCIC systems. In HPCI, problems with the governor control system were noted including a minor oil leak in the servo-motor. Steam leaks at gauges and turbine drain line were also discovered. In RCIC, the licensee discovered a previously installed blank flange in the turbine steam leak off line which caused steam leaks. A few problems were also noted on the RCIC governor control system. The licensee is in the process of dispositioning these items. The inspector noted that using non-nuclear steam for the testing enabled the licensee to discover problems which may not have been easily identifiable using nuclear steam due to the radiological conditions. The inspector will review the results of the tests and dispositioning of the problems identified during the tests.

- Incorrect Installation of Fire Dampers

On December 17, 1987, during performance of a routine surveillance test the licensee inadvertently actuated two fire dampers. One of the dampers failed to fully close due to interference with a hook used to secure it in the open position. When the fusible link was energized, the metal damper retaining strap should have fallen away allowing full closure. The hook attaching the strap to the fusible link was oriented with the open side toward the damper. The damper caught on the hook and remained partially open. Upon discovery the licensee immediately stationed fire watches at all areas containing suspect dampers. Inspections were promptly conducted and it was identified that all of the installed hooks were oriented in this manner. The hooks were repositioned so that the open side faces away from the damper. Three dampers were inaccessible and compensatory measures remain in place pending inspectic i.

The dampers were originally supplied to the licensee without the hooks. A revision to the plant design change (PDC) package added the hooks to facilitate surveillance testing. Installation instructions contained in the PDC specified hook orientation with the open side toward the damper. The vendor data sheet supplied by Air Balance Inc. also showed the hook installed in this manner.

Licensee event report (LER) 87-020-00 was issued describing the problem and corrective actions taken. The LER states that preliminary licensee assessment of the issue determined that it did not meet the reporting threshold of 10 CFR Part 21. The inspector discussed the Part 21 reportability with the licensee's Nuclear Engineering Department (NED). NED personnel stated that the failure mechanism was created by the licensee when the hook was added. In addition the presence of mitigating factors such as fire detection and suppression, and control of combustible materials support the conclusion that a substantial safety hazard did not exist. The licensee also feels that LER 87-020-00 contains sufficient information to clearly define the problem. The inspector had no further questions in this area.

The inspector examined two dampers in the cable spreading room to verify that the hooks had been reoriented. Both hooks had been modified, however, neither of the dampers had locking rings installed at the hook to retaining strap connection as required by the installation instructions in the PDC. The licensee reviewed the function of the locking rings and concluded that they were not required. A change to the PDC was initiated to delete the ring. The inspector had no further questions.

b. Radiation Protection and Chemistry

- Locked High Radiation Area Access Control

During the period covered by inspection report 87-57, four instances occurred in which the licensee failed to properly control access to areas that had been designated as locked high radiation areas. In three of these cases, doors to locked high radiation areas were found closed but not locked and in the fourth case a door into a locked high radiation area was found to not be on the list of doors that were being controlled under the locked high radiation area door procedure.

On December 15, 1987, a contract painter failed to check that the door to the locked high radiation area he was exiting was properly latched. The unlatched door was identified during the next routine check of high radiation area doors. Licensee personnel immediately latched the door and initiated a radiological occurrence report (ROR) to document the occurrence and track all actions taken during the investigation. Surveys of the area showed no dose rates greater than 1000 millirems per hour (MR/hr). Interviews with the individual involved determined that the procedures and requirements were well understood and that the HP technician had informed them of their responsibilities prior to entry into the area.

On December 27, 1987, and again on January 8, 1988, instances similar to the one described above took place. In both cases the licensee initiated RORs and took steps to determine: 1) who had been in the area, 2) were they aware of the procedure, and 3) had they been properly briefed prior to entry into the areas involved. In both of these cases the root cause has been determined as personnel error.

In one instance the licensee identified that one of the multiple doors into an area classified as a locked high radiation area was not on the list of doors to be checked on a routine basis. The door was immediately checked and found to be locked. Records have been audited to determine if any unauthorized entry into the area had occurred and no instances were identified. The door has been placed on the list and is now routinely checked.

The inspector reviewed licensee actions as a result of these instances and is satisfied that in all cases, the immediate and followup actions were timely and complete. Surveys taken were comprehensive and conducted almost immediately after discovery of unlocked areas. Dose calculations were performed and dosimetry read in all cases. Involvement by senior HP and plant management was evident in all instances.

Inadequate control of locked high radiation areas has been an area of longstanding NRC concern. Notices of Violation have been issued in the past, during inspections 50-293/87-03, 50-293/87-11, and 50-293/87-19 which addressed these concerns. In regard to these violations the licensee instituted corrective actions which have been successful in addressing segments of the problem but have not been successful in preventing recurrence of events involving high radiation area door control.

The inspector has independently reviewed the licensee's program for control of high radiation areas and high radiation area key control and has found them adequate. Although the programs themselves are adequate and personnel have been trained on those programs, instances still occur where locked high radiation areas are not adequately controlled.

Based on review of these four instances coupled with the review of Unresolved Item 87-50-08, the inspector determined that the licensee actions in response to these previous findings have not prevented recurrence. Failure to comply with the requirements of Technical Specification 6.11 and Implementing Procedure 6.1-012 is an apparent violation of NRC requirements as documented in Appendix A of the cover letter to this report (87-57-01). Licensee response to Appendix A should include those measures taken to insure that corrective actions are effective and lasting.

- Contaminated Clothing Offsite

On December 17, 1987, at 7:26 p.m. hours a Bechtel pipefitter who was exiting the reactor building, set off a whole body portal monitor alarm. The portal monitor indicated contamination of his chest area and left hand. The health physics technician on duty at the access point removed the individual from the portal monitor and began performing a survey using a RM-14 with DT 260 probe. The HP technician identified; 1) contamination on the individual's left hand, 1-2 thousand dpm per 100 square centimeters (K DPM), which was removed by washing, 2) contamination on the shirt in both the chest (80K DPM) and lower stomach area (1K DPM). The shirt contamination was removed by tape (80K DPM) and washing with soap and water (1K DPM). The employee, now wearing an undershirt and trousers, was then sent to clear the portal monitor which again alarmed and indicated contamination in the chest area. The HP technician again surveyed the individual and identified contamination on the undershirt in the chest area (70K DPM). The individual was then sent into the portal monitor bare chested and was cleared. The individual was given his outer shirt, which was still wet from decontamination and cleared through portal monitor. At this point, the individual removed the wet shirt, put on his jacket, cleared the portal monitor again, and left for his home.

Upon returning to work December 18, 1987, the individual was given a whole body count to determine if any internal contamination had occurred. The whole body count showed no internal contamination. After completion of the whole body count the individual was interviewed to determine how he had been contaminated, where the occurrence took place and how long he was contaminated prior to detection, to calculate skin dose received.

The interview revealed that the individual had been contaminated when he disconnected a partially pressurized service air hose and depressurized it. The interview also revealed that the individual used the portal monitor at the 91 ft. elevation of the reactor building, received an alarm, did not call for HP assistance but instead tried to decontaminate himself prior to proceeding to the reactor building access. Station procedures require that an individual who finds himself contaminated is to call health physics for assistance. The individual stated that he was aware of this requirement. During the interview the individual expressed concern about whether his heavy winter jacket could have shielded the contamination on his shirt and undershirt from detection by the portal monitors. To demonstrate that this could not happen, a HP supervisor placed plastic bags, which contained the contamination removed from his shirt, inside the coat and attempted to exit through two portals. The portal monitors alarmed on each attempt. The individual appeared satisfied with the demonstration put his jacket back on, with the plastic bags removed and attempted to leave the reactor building. An alarm was actuated on the portal monitor and contamination was indicated on the left arm. The on duty HP technician removed the individual from the portal monitor and identified 3K DPM contamination on the upper right sleeve (outside) of the jacket even though the jacket had not been worn into the reactor building. At this juncture the individual expressed concern over whether the shirt that he had worn the previous day could still be contaminated. The licensee had a HP technician accompany the individual to his home. The individual's shirt was found to be contaminated, was bagged and returned to the site. Surveys of the individual's home and vehicle identified no further contamination.

Efforts to determine how the contaminated shirt was worn through the portal monitors without setting of an alarm yielded positive results. The individual stated that he had purposely kept himself away from the portal monitor in an attempt to keep his wet shirt away from his skin. The licensee taped the plastic bags, with the contamination in them, back onto the shirt and an HP supervisor attempted to pass through the portal monitors by mimicking the body posture used by the individual when he cleared the monitor. The HP supervisor was able to pass through six different monitors without setting off an alarm. The HP supervisor then used the portal monitors in the correct manner and all six monitors alarmed proving that the equipment was functional.

The licensee has evaluated the occurrence to identify the root causes and immediately implemented corrective action. This occurrence was caused by one sequence of events that involved two distinct personnel errors. The primary cause involved the failure of the HP technician to perform an adequate survey of the contaminated individual's clothing when the portal monitor alarm was received. The second problem involved the failure to properly use the installed portal monitors at the reactor building access.

In addition to personnel interviews to identify the sequence of events the licensee also reviewed procedural adequacy, personnel training and portal monitor calibration and performance. These reviews verified that training was adequate and portal monitor performance was as designed. Procedures for control of contaminated individuals at the reactor building access did not specifically require that all articles of clothing require a 100% frisk prior to this occurrence. Instructions have been posted at the reactor building access which now clarify the procedure to be followed when an individual is found to be contaminated.

The portal monitors in use at Pilgrim do not presently have a switch at chest level which must be actuated to start the monitoring process. Lack of this feature allowed the individual wearing a contaminated shirt to lean away from the machine sufficiently to clear the monitor without any alarm. The licensee has determined that the manufacture of the portal monitor now produces a chest high switch for the installed model and will install them in the future.

Calculations have been performed by the licensee to determine the radiation dose received by the individual and the amount of radioactive material that was released from the site on the contaminated shirt. The results of these calculations show that the individual received a localized radiation dose to the skin of 260 MRem, which is below the federal limits for skin exposure, and that the amount of radioactive material on the individuals clothing was 0.2 microcuries which meets the federal criteria as an exempt quantity of Co-60. The inspector is satisfied with the licensee's analysis and corrective actions and has no further questions.

- Allegation of Improper Disposal of Radioactively Contaminated Shrubs (RI-87-A-0107)

On August 31 and September 11, 1987, the NRC resident office at Pilgrim received allegations that radioactively contaminated shrubs had been removed from the site and improperly disposed. The alleged improper disposal occurred on July 23, August 26 and August 28, 1987. During this time period the licensee removed a large number of shrubs from various areas of the site, including those planted near the old administration building and the switchyard. The shrubs were removed to facilitate site construction activities and to alleviate certain security concerns. Upon receipt of the first allegation on August 31, 1987 the NRC requested that the licensee perform an evaluation and provide the results for review. In addition an independent NRC review was initiated.

Resident and specialist inspectors reviewed the licensee's conclusions. The licensee evaluated material release records and interviewed personnel regarding removal of shrubs during the week of July 20, 1987. Several truckloads of shrubs that were transported offsite during the midnight shift on July 24 were examined in detail. Because trace amounts of Cobalt-60 had previously been found in soil onsite, some of the shrubs had the soil removed from the roots prior to release. Each shrub was hand surveyed and found to meet established offsite release criteria. They were transported first to the licensee's shorefront area and later to a dump site on licensee property. The licensee concluded that the shrubs had been adequately surveyed and that no radioactive material had been improperly released.

The resident inspectors reviewed the licensee's program for control of release of material from the site. This area was also evaluated by NRC specialist inspectors during inspection 50-293/87-19. Both inspections concluded that appropriate surveys and release limits have been established and implemented. Resident and specialist inspectors examined licensee release records for the dates in question to verify that vehicles leaving the protected area had been properly surveyed. No discrepancies were identified. An NRC resident inspector accompanied by a licensee representative collected four samples of the shrubs which had been deposited in the dump site discussed above. Each of the four samples consisted of root, branch and foliage clippings from a number of different shrubs. The samples were independently analyzed by the NRC. Three of the samples indicated no contamination. One sample indicated only trace levels of Cobalt -60. Measurements showed that the amount of CO-60 present in this sample was about 2% of the average radioactivity typically found in soil due to naturally occurring isotopes.

The licensee's program for release of material from the site appears adequate. Appropriate survey techniques and release limits have been established. Review of records confirmed that the program is being implemented. Samples of the shrubs collected by the NRC showed zero or negligible contamination and pose no health and safety concern. Based on the above this allegation is considered closed. NRC Region I staff provided status briefings concerning this allegation to Senator Kennedy's staff and to the Massachusetts Department of Public Health.

- Allegation of Airborne Radioactivity in the Trash Compaction Facility (RI-87-A-0120)

On October 5, 1987, the resident office received an anonymous allegation that personnel working at the sort table in the trash compaction facility (TCF) were being routinely exposed to airborne radioactive contamination. The alleger stated that the two filter systems designed to treat exhaust air from the sort table prior to discharge into the room were not functioning, and that the filter differential pressure alarm circuits had been disabled.

On October 7 and 8, 1987, NRC specialist inspectors toured the TCF and examined the design and condition of the equipment. The sort table is used to separate contaminated materials for compaction and disposal. Potentially contaminated air is exhausted from the table, passed through two filters operating in parallel, and released into the room. Airborne radiation levels in the room are measured by means of a separate air monitor which is operated whenever the sorting table is used. The alarm is typically set at 3 X 10 -10 (3E-10) microcuries per cubic centimeter (cc). In addition the filters are surveyed daily and changed if contact dose rates exceed 2mR per hour. The inspectors also examined the trash compaction unit in the area and found that similar controls had been applied. Based on the above, no immediate health and safety concerns were indicated.

On January 15, 1988, the resident inspectors toured the TCF, examined equipment operation and interviewed licensee and contractor personnel involved in ongoing work activities. A radiation work permit specifying protective clothing, health physics coverage, and use of a continuous air monitor is in place to control work at the sort table. Personnel involved stated that trash bags were surveyed prior to sorting and rejected if radiation levels exceeded 5mr/hr, if they contained liquid, or if any powdery material was present. The health physics technician on duty stated that filter radiation levels are monitored daily.

Workers and health physics personnel also stated that filter differential pressure (dp) instruments are monitored to detect filter plugging, however no one had been clearly assigned this responsibility and no dp limit was established. The inspector observed the operation of the table and noted that the "filter restricted" alarm actuated for one of the two filters. The alarm actuated for the filter displaying the lower differential pressure. When questioned workers stated that much of the monitoring and alarm circuitry for the table was not functional, and that the filter alarm was not reliable. The table was originally part of a larger processing system and much of the disconnected circuitry was intended to perform functions which are no longer needed. The inspector verified that current filter dp readings are consistent with the manufactures name plate data.

It appears that the general process applied, including inspection and survey of trash bags prior to sorting, daily filter surveys and continuous air monitoring would preclude airborne radioactivity problems. Based on the above this allegation is closed. However, the inspector noted that no work instructions existed describing the controls applied and equipment monitoring requirements. When discussed with licensee radiation protection management they promptly committed to review the situation and issue appropriate guidance. This was confirmed during the inspector's exit interview.

- <u>Erosion of Construction Dirt into Wetland</u>

On January 15, 1988, at 5:45 p.m. the licensee made an ENS notification in accordance with 10 CFR 50.72 (b)(2)(vi) which requires the licensee to inform the NRC of an event or situation related to health and safety of public for which a news release was made or notification of another government agency has been made. During routine environmental monitoring, the licensee observed erosion from a pile of construction dirt into an adjacent licensee controlled wetland. The Plymouth Conservation Commission and the Massachusetts Department of Public Health were notified and the press release was made by the licensee. Also on January 16, 1988 two representatives from the Plymouth Conservation Commission toured the area.

In the last several years during onsite excavation for plant modifications, dirt, asphalt and concrete containing low levels of contamination were stored in a fenced in storage area outside the protected area on the licensee's property. The licensee estimated that the storage area contains 110,000 cubic feet of material. Before removal from the protected area, samples of material were obtained and isotopic analyses was performed by the licensee. The activity found was reasonably uniform at levels of 10(1E-6) and 10(1E-7) microcuries of Cobalt-60 and Cesium-137 per gram. Sampling and storage of this material was previously reviewed during inspection 50-293/87-18. On January 21, 1988 the inspector toured the area, accompanied by a licensee health physics technician, and performed a survey of the storage area and found no detectable radiation above background levels. During the tour the inspector noted that bales of hay had been put around the perimeter of the fence which borders wetlands area to prevent further erosion of material. The fenced in storage area was secured with a locked gate. The inspector's survey of the area and review of licensee's analyses indicate that the level of activity does not represent a health or safety concern. However, the inspector raised a concern to the licensee management that the material should not be allowed to erode. The inspectors will continue to monitor the licensee actions in formulating long term solution to properly dispose of the material.

c. Fire Protection

On January 17, 1988, at 4:55 a.m. the control room received a report from a security guard of smoke coming from a contractor lavatory trailer, which is located adjacent to the Bechtel warehouse inside the protected area fence. The onshift fire brigade chief was dispatched to the scene and confirmed smoke and smoldering in the area. The fire brigade was immediately dispatched and fire was extinguished using a portable dry chemical extinguisher and a hose from a nearby hydrant house. Electrical maintenance was called to shut off the power to the trailer. By 5:30 a.m., the fire brigade members had cleared the scene and a continuous fire watch was posted in the area. The cause of the fire was believed to be overheating of an overhead heating unit for the trailer. No personnel injury occurred. The inspector toured the scene with a licensee fire protection engineer on January 18, 1988. Minor damage to a small area of the ceiling in the trailer was observed. The Plymouth Fire Department was notified by the licensee in the morning of January 18, 1988.

4.0 Review of Plant Events

The inspectors followed up on events occurring during the period to determine if licensee response was thorough and effective. Independent reviews of the events were conducted to verify the accuracy and completeness of licensee information.

a. Spurious Isolations of RHR Shutdown Cooling System

On December 7, 1987, at 2:28 p.m., an inadvertent isolation of both inboard and outboard containment isolation valves on the RHR shutdown cooling suction line occurred. Preparation for the reactor vessel hydrostatic test was in progress. As part of the hydrostatic test procedure, a technician was installing an electrical jumper in the primary containment isolation system logic panel C-941 to bypass the reactor coolant system (RCS) high pressure interlock on the inboard isolation valve. When the termination screws were loosened to install the jumper, the leads lost contact and caused a false high pressure isolation signal. RHR was in its shutdown cooling mode when the isolation signal was generated, and the shutdown cooling suction valves (MOV 1001-47, 1000-50) automatically closed as designed. Coincident with the closure of the valves, the "A" and "C" RHR pumps tripped automatically to protect the pumps from loss of adequate suction. The licensee determined the actuation was due to a personnel error. The licensee revised Procedure 2.1.8.1, Class I System Hydrostatic Test, to caution the I&C technician of potential isolation of RHR shutdown cooling system while installing the jumper.

On December 8, 1987, at 9:45 p.m., the inboard isolation valve (MOV 1001-50) on the RHR shutdown cooling suction line automatically closed. The automatic isolation occurred when the plant reached 100 psig during pressurization for performance of the class I hydrostatic test. The outboard isolation valve (MOV 1001-47) was already closed to form a pressure boundary for the test. The licensee's investigation determined that the cause of the isolation was that Procedure 2.1.8.1 did not identify all the jumpers necessary to bypass the RCS high pressure interlock on the inboard isolation valve.

As immediate corrective action, the licensee halted the pressurization of RCS and reviewed the logic prints. The licensee revised Procedure 2.1.8.1 to reflect the need to install an additional jumper in panel C-942. In reviewing this event along with other similar events documented in previous inspection reports, the inspector noted that inadequate planning and inadequate procedures appear to be a common root cause for several ESF actuations which occurred on September 17, September 22, October 15 and October 24, 1987. The inspector expressed this concern at the exit meeting with licensee management. The licensee informed the inspector that the Technical Group is in the process of developing generic guidance for isolating or jumpering an electrical component which may cause inadvertent safety system actuations. The inspector will continue to monitor the effectiveness of licensee's corrective action to prevent further ESF actuations due to inadequate planning and inadequate procedures.

b. Reactor Water Cleanup System Spurious Isolation

On December 17, 1987, at 11:05 a.m., the inboard primary containment isolation valve on the reactor water cleanup (RWCU) system suction line automatically isolated. I&C technicians conducting a routine surveillance of the RWCU high area temperature isolation logic inadvertently grounded a lead which had been lifted during the test. Grounding the lead resulted in a blown logic power fuse and isolation of the valve (MOV 1201-2). Following investigation by the control room supervisor, the fuse was replaced and the isolation was reset. The licensee's investigation concluded that the root cause is a personnel error. The licensee informed the inspector that the procedure, 8.M.2-1.2.2, Reactor Water Cleanup Area High Temperature, will be revised to provide cautions to the control room operators and the I&C technicians. Also, an effort is ongoing to review recent ESF actuations caused by personnel error to formulate appropriate corrective actions.

c. Engineered Safety Feature Actuations Due to a Failed Logic Relay

On January 6, 1988, at 2:50 p.m., the coil of primary containment isolation system (PCIS) electrical relay 16A-K57 failed, creating a fault and resulting in blown logic power fuses. The deenergization of this portion of the PCIS logic caused a partial primary containment isolation along with a reactor building isolation and start of the "B" Standby Gas Treatment system (SBGT). The licensee notified the NRC at 5:12 p.m. via ENS. The failed relay was a GE type CR120A relay. The licensee has experienced several failures of this type of relay in the last few years. The licensee's evaluation of this high failure rate and corrective actions to address it are described in the inspection report 50-293/87-50.

On January 7, 1988, the inspector reviewed maintenance request (MR) 88-9 which had been initiated to investigate the cause of the above mentioned ESF actuations and to replace the blown fuse and the faulty relay. The inspector noted that the relay replacement was performed using only procedure 3.M.1-11, Routine Maintenance. This procedure contains general guidance and its stated use is for performing maintenance activities which are not complicated or critical enough to require detailed written procedures. In this case, no step-by-step instruction was initiated to control the sequence of work, to control and tag lifted leads and jumpers, and to ensure verification and independent verification of system restoration. A similar problem involving lack of a sufficiently detailed controlling procedure and the appropriate reviews during an electric relay replacement on November 24, 1987 was the subject of a violation as documented in the inspection report 50-293/87-50. The licensee informed the inspector that the corrective actions to address the violation are being formulated and will be submitted to the NRC.

d. Spurious Reactor Protection System Actuation

On January 17, 1988, at 1:13 a.m., a spurious reactor scram signal was generated during the performance of a reactor level instrument calibration. The full scram signal on low water level was received due to a disturbance in the reactor water instrument line when an I&C technician was valving a level instrument (LI-263-59A) back in service. The Rosemount level transmitters (LT-263-57 A&B) which initiated the scram signal are on the same instrument rack. The licensee's preliminary investigation indicated that the root cause of the event is attributed to a combination of personnel error and inadequate procedure. The investigation also identified that the level instruments (LI-263-59 A&B) were incorrectly installed in that the sensing lines were reversed. The new Barton level instruments (LI-263-59 A&B) were recently installed during this outage and would only be used for local indication during a shutdown from outside the control room. The licensee is currently reviewing the plant design change (PDC 85-07) records and post-installation test data to determine the cause. Surveillance test records are also being reviewed by the licensee. This item is unresolved pending the completion of the licensee investigation (87-57-02).

Upon receiving the spurious scram the control room staff noted that scram discharge instrument volume (SDIV) vent valve CV302-23B primary containment vent and purge valves AO5044B and AO5035B and one of two redundant secondary containment isolation dampers in each line did not close. In addition the "B" standby gas treatment system (SGTS) did not start. Based on the initiating event, these components should have actuated. The licensee notified the NRC of the failures via ENS at 5:00 a.m. on January 17, 1988.

The control room staff conducted an immediate critique with available I&C personnel, and documented observations for management followup. Later on January 17, the licensee inspected the physical condition of the SDIV vent and drain valves and noted paint on the stem of CV302-23B. The paint was removed and the valve successfully stroke timed. The licensee held a second critique with management representatives on the morning of January 18, 1988 to assess the situation. Subsequently, a walkdown of involved isolation logic components was performed to verify relay contact configuration and to identify any jumpers or lifted leads. This walkdown was performed to the extent possible without disturbing components. No discrepancies were noted. Early on January 19, the licensee performed a test in which a reactor scram was intentionally initiated. The same equipment failed to actuate as during the January 17 scram. Based on this licensee management stopped all work on the affected components. A task force composed of members from the technical staff, systems group, I&C and operations was designated to investigate the incident. This team reviewed available information and developed an action plan.

Walkdowns of the air system piping and components supplying motive air to SDIV vent valve CV302-23B were performed to verify that the as built configuration is in accordance with design documents and that components are in good physical condition. No discrepancies were identified. Valves CV302-23B and CV302-22B are supplied air by the same solenoid operated valves. The licensee deenergized these solenoid valves and observed that CV302-22B closed while CV302-23B did not. This indicates a mechanical problem with the valve or operator. The licensee was identifying replacement parts and preparing to disassemble the valve by the close of the inspection period. The inspectors will continue to monitor licensee followup to this failure.

Licensee review of logic drawings confirmed that the remaining equipment which had not properly actuated shared common isolation logic components. A series of surveillance tests was performed to allow monitoring of key relay actuations. A single contact on a General Electric (GE) HFA relay was determined to be misfunctioning. The contact is required to close when an isolation signal is received, actuating the affected equipment. However, contact resistance remained high with the contact closed. The relay was replaced and the system successfully tested. The licensee contacted GE to coordinate disassembly and inspection of the relay. Dissassembly had not begun by the close of the inspection period. The inspector will continue to monitor licensee investigation of this failure.

The inspector expressed concern that three separate equipment malfunctions had occurred during the inadvertent actuation. This may reflect weakness in the surveillance and post-work test program. However, the licensee's response to the actuation and subsequent malfunctions was prompt, thorough and effective. Control room operators quickly recognized each of the failures. They held a critique on the same shift with involved personnel. Critique observations were clearly documented and provided to management. An additional critique with management present established priorities. Action was taken to freeze equipment until an investigation plan could be developed and implemented. Followup was well coordinated and involved representatives of several portions of the organization. In this case licensee commitment to determining and correcting the problem root cause was evident.

5.0 Review of Reactor Vessel Hydrostatic Test Procedure and Test Results

During the inspection period the licensee completed the reactor vessel hydrostatic test. Several reactor vessel instrument nozzles were repaired during this outage, prompting performance of a hydrostatic test rather than a system leakage test. The reactor vessel reached minimum test pressure and all inspections were completed on December 9, 1987. Only minor leakage associated with mechanical connections, such as flanges and valve packing was identified. The reactor vessel was depressurized on December 12, 1987 after completion of excess flow check valve testing.

The inspector reviewed the licensee's hydrostatic test procedure to verify that appropriate prerequisites, precautions and instructions had been included. A sample of valve lineups was reviewed to determine the adequacy of established test boundaries. Completed valve lineups were also examined. Control of temporary electrical and mechanical jumpers was evaluated to ensure proper documentation and restoration. The inspector observed installed pressure instrumentation and verified appropriate range and calibration status. The adequacy of staffing to support test performance was periodically verified. The inspector reviewed test results and discussed them with engineering, operations, and quality control personnel to ensure that test changes were properly processed, adequate inspections were conducted, and that inspection results were promptly dispositioned.

The licensee's preparation for and execution of the test was generally well organized and controlled. Procedures for test performance and conduct of visual inspections were clear and comprehensive. A detailed Quality Control (QC) work instruction was developed specifying components and piping requiring inspection. Inspection assignments were broken down by location, elevation and component. This QC instruction also included a series of piping diagrams depicting the test boundaries which were utilized to assist in inspection performance and documentation. The licensee's Technical Engineering Section, Quality Control staff and Nuclear Engineering Department each reviewed test boundary adequacy. Inspection results were well documented, and maintenance requests were promptly initiated to correct identified leakage.

The licensee experienced two shutdown cooling isolations during implementation of the test procedure. These isolations are discussed in detail in section 4.a of this report. During the test the licensee identified leakage past the seal ring at the stuffing box to pump casing joint on both recirculation pumps. Leakage flow was estimated to be one to two gallons per minute for each pump. The leakage wet the pump casings and portions of the suction piping, and acceptable inspections could not be completed in these areas. The licensee stated that similar leakage on at least one of the pumps was noted during the last outage. That leak sealed as system temperature increased during startup. The licensee believes that the leakage observed during the recent test will also stop as temperature is increased, and no pump repairs are planned. The licensee stated at the inspector's exit interview that the pump casings and suction piping will be reinspected during startup.

The inspector noted that the test procedure did not contain valve lineups for manual instrument isolation valves within the test boundary. Many instruments and a significant portion of instrument piping has been replaced this outage. Visual inspections were performed of class I piping downstream of these valves. The inspector questioned the basis for licensee confidence in instrument line isolation valve positions during the test. The licensee pointed out that hydrostatic testing of these lines was not required during this outage. In addition excess flow check valve testing was conducted immediately after completion of the hydrostatic test with the system still pressurized. Successful completion of the check valve testing requires proper alignment of the manual isolation valves, and provides assurance that the piping was pressurized during the visual inspections. The licensee however, agreed that the intent of the test had been to pressurize and inspect this piping and that the current procedure does not adequately assure the correct valve alignment. Licensee management stated that the procedure would be revised to address this weakness.

6.0 Integrated Leak Rate Testing

On December 21, 1987, the licensee began performance of the primary containment integrated leak rate test (ILRT). The containment was pressurized with air to the full test pressure of 45 pounds per square inch and maintained at this pressure for 24 hours. The 24 hour test period started at 10:15 p.m. on December 21, 1987. A regional specialist inspector was onsite during the ILRT to review the adequacy of the test procedure and to observe the conduct of the test. The preliminary licensee test results indicated a successful test, with measured leakage slightly greater than 20 percent of the allowable leakage. A primary contributor to the observed leakage was identified as a drywell pressure transmitter piping cap which had not been fully tightened. Upon completion of the specialist inspector's review of the ILRT results, inspection report 50-293/87-58 will be issued documenting the inspectors findings.

While preparing for the primary containment integrated leak rate test (ILRT) the licensee observed that several torus temperature and moisture elements were not functioning properly. Troubleshooting identified circuit faults at a torus electrical penetration assembly. The licensee removed the penetration assembly protective cover inside the torus and found that it was filled with water. The penetration is installed vertically through the top of the torus. On both the inboard and outboard sides of the penetration a metal frame is attached on which 28 terminal boards are mounted. Cables passing through the penetration, and supplying instrumentation in the torus also landed on these terminal boards. A protective cover is bolted over the frame and terminal boards on both sides of the penetration. Design drawings specify that cover joints are to be sealed with silicone tape. The licensee stated that the protective cover had not been properly sealed, allowing water intrusion and buildup. The water caused significant corrosion of the cable connectors, terminal boards and metal framework. This corrosion and water buildup resulted in the observed electrical circuit faults. Licensee inspection of the other torus electrical penetration identified similar conditions. Temporary repairs of the temperature and moisture elements were made to allow ILRT performance. Cables for communications, lighting, and torus to drywell vacuum breaker indication also run through the penetration. The penetration is not considered by the licensee to require environmental qualification but is designated as a "Q" component. The licensee is evaluating the root cause of the water intrusion and is developing a temporary procedure to control repair and testing of the penetration. The inspectors will continue to monitor licensee followup and corrective actions.

The licensee informed the inspector that penetration repairs would not be completed until after ILRT performance. The inspector questioned the effect of the planned repairs on the penetration leak tightness, and the ability to perform adequate leakage test after the planned rework. The licensee stated that the work would not affect penetration leakage but that adequate testing could be performed after work completion. Based on available drawings however, the licensee could not demonstrate adequate testability. In response to NRC concern the licensee obtained the needed drawings from the vendor and verified that the penetration was completely testable. The inspector had no further questions.

During the ILRT, the licensee identified a water leak in the high pressure coolant injection (HPCI) turbine room. It was determined that the increasing pressure in the torus air space caused the suppression pool water to back up through the HPCI turbine exhaust line and through the drain piping, overflowing the HPCI gland seal condenser onto the HPCI room floor. The turbine exhaust line discharges to the torus through a check valve and a locked open stop-check valve. To prevent any condensation from collecting in the turbine exhaust line downstream of the check valve, a drain piping drains any condensation to the HPCI gland seal condenser through a drain pot. Two solenoid operated drain valves on the drain pot close automatically on a HPCI (Group IV) isolation signal. This is to provide the isolation from the torus to the gland seal condenser. The licensee's investigation determined that leads had been lifted in the HPCI isolation interlock logic circuit since October 30, 1987 in support of the HPCI steam testing utilizing temporary oil-fired auxiliary boilers. With the HPCI isolation signal bypassed, the drain valves remained open as the drain pot was filled with the suppression pool water. The licensee subsequently relanded the leads in the HPCI isolation interlock logic circuit and the drain valves closed.

After reviewing the ILRT procedure, HPCI test procedure and interviewing licensee personnel, the inspector concluded that licensee review of the active maintenance requests prior to the ILRT was not thorough in that the lifted leads controlled by the MR 87-663 were not identified. The MR tags were attached on the HPCI isolation logic circuit inside a logic panel and thus the tags were not identified during a system walkdown prior to the ILRT. The drain valve positions were verified by the light indications on the control room panel 903 as prescribed in the ILRT procedure.

The inspector also determined that the maintenance request above may not be an adequate method of identifying and tracking jumpers and lifted leads, especially for a long term application and for components which could affect other ongoing maintenance or surveillance. Station procedures do not require temporary modification controls for jumpers and lifted leads which are controlled by active maintenance requests. The inspector discussed these findings at the exit interview with licensee management. The licensee informed the inspector that a lifted leads and jumper log will be kept in the control room to aid the operators in controlling lifted leads and jumpers.

7.0 Licensee Nuclear Organization Management Realignment

On December 14, and on December 31, 1987, the Boston Edison Co. announced, as part of a planned realignment occurring over the next several weeks, the appointment of the following managers to key management positions in the licensee nuclear organization at Pilgrim Station.

-- Mr. Kenneth L. Highfill was named to assume the new position of Station Director. In this capacity, Mr. Highfill will oversee day to day operation of the Pilgrim Station including plant operations, planning and outage, nuclear training, plant support functions, and administrative services. Mr. Highfill will report directly to Mr. Ralph G. Bird, Senior Vice President-Nuclear.

-- Mr. Robert J. Barrett was named the new Plant Manager. Mr. Barrett will report to Mr. Highfill, the Station Director.

-- Mr. Roy Anderson, currently Deputy Outage Manager, was named to assume the new position of Planning and Outage Manager. Mr. Anderson will report to Mr. Highfill, the Station Director.

-- Mr. Ed Kraft was named to assume the new position of Plant Support Manager. In this capacity, Mr. Kraft will oversee radiological, security, industrial safety and fire protection, and other station support functions. Mr. Kraft will report to Mr. Highfill, the Station Director.

-- Mr. Donald Gillespie, currently Director of Planning and Restart, was appointed to the position of Quality Assurance Department Manager. Mr. Gillespie will assume the position after completing his Senior Reactor Operator training. The Quality Assurance Department Manager reports to Mr. J. E. Howard, Vice President-Engineering.

-- Mr. Frank Famulari, currently Operations Quality Control Group Leader, was named to assume the newly created position of Deputy Quality Assurance Department Manager. Mr. Famulari will report to Mr. Gillespie, and be acting Department Manager until Mr. Gillespie assumes the position after completing the Senior Reactor Operator training.

-- Mr. John F. Alexander was named to assume the position of Operations Section Manager. Mr. Alexander will report to Mr. Barrett, the Plant Manager.

-- Mr. Donald J. Long was named Security Section Manager. Mr. Long will report to Mr. Kraft, the Plant Support Manager.

8.0 Management Meetings

At periodic intervals during the course of the inspection period, meetings were held with senior facility management to discuss the inspection scope and preliminary findings of the resident inspectors. On January 26, 1988, the inspectors conducted a final inspection exit interview to formally present inspection findings.

Attachment I to Inspection Report 50-293/87-57

Persons Contacted

* R. Bird, Senior Vice President - Nuclear
* K. Highfill, Station Director
K. Roberts, Plant Manager
R. Barrett, Deputy Plant Manager
R. Anderson, Planning and Outage Manager
E. Kraft, Plant Support Manager
F. Famulari, Deputy Quality Assurance Manager
D. Swanson, Nuclear Engineering Department Manager
J. Alexander, Operations Manager
N. Brosee, Maintenance Manager
J. Jens, Radiological Protection Manager
J. Seery, Technical Manager
R. Grazio, Field Engineering Manager
P. Mastrangelo, Chief Operating Engineer
R. Sherry, Chief Maintenance Engineer
N. Gannon, Chief Radiological Engineer
D. Long, Security Manager
F. Wozniak, Fire Protection Manager

*Senior licensee representatives present at the exit meeting.

ATTACHMENT II

January 6, 1988

MEMORANDUM FOR: Ken Roberts
Plant Manager

FROM: Clay Warren
Senior Resident Inspector - Pilgrim

SUBJECT: FACILITY TOUR FINDINGS, DECEMBER 8, 1987

The items on the attachment were noted during the facility tour on December 8, 1987. Please contact the Resident Inspector Office when your staff is ready to discuss the evaluation of the items and the status of any actions taken. Please note the items and the facility response will be addressed in a routine inspection report.

Thank you for your time and attention to these matters.

Sincerely,

Clay C. Warren
Senior Resident Inspector

Attachment:
As stated

cc w/attachment:
R. Blough
W. Kane
W. Russell
J. Wiggins

ATTACHMENT

-- Numerous motors appear to have failed grease seals caused by overgreasing without first removing grease drains. This condition causes a buildup of grease and dirt in the cooling airflow path and in extreme cases grease in the motor windings. (SBGT fans and SLC pumps)

-- Nuts and bolts were noted laying inside an electrical cabinet in the RCIC room.

-- Multiple cases of open junction boxes, terminal boxes and conduit pulled away from terminal boxes were noted.

-- Motor heaters for the "B" RHR pump appear to have overheated causing the insulation on the heaters to melt.

-- HPCI room cooler drip pan is full of paint scrappings which could lead to drain clogging.

-- Standby Liquid Control system relief valves have boric acid crystal buildup which could alter setpoints.

-- Painting effort should be more closely controlled to prevent painting inappropriate surfaces, i.e., linkages, valve packing glands, trip throttle valves, limit switches, etc.

-- Numerous instances of scaffolding materials, i.e., nails and wood chips, laying on floors. This material could migrate to drain systems and cause pump or valve damage. Scaffolding was also noted attached to permanent equipment such as piping and conduit.

-- Valve 1001-36A motor operator conduit had melted plastic cover.

QUESTION 13. Has Pilgrim ever violated established radiation emission levels; i.e., have there been any releases from the plant which exceeded standards set by the NRC?

ANSWER.

The permissible levels of radiation in unrestricted areas and of radioactivity in effluents to unrestricted areas are established in NRC regulations embodied in 10 CFR Part 20, Standards for Protection Against Radiation. These regulations specify limits on levels of radiation and limits on concentrations of radionuclides in the facility's effluent releases to the air and water (above natural background) under which the reactor must operate. Further, the regulations require that there be no unmonitored release paths from the plant. The regulations are structured to provide reasonable assurance that no member of the general public in unrestricted areas will receive a radiation dose, as a result of facility operation, of more than 0.5 rem in 1 calendar year. These radiation-dose limits are established to protect the health and safety of the public.

In addition to the Radiation Protection Standards of 10 CFR 20, 10 CFR 50.36a establishes license requirements in the form of license Technical Specifications on effluents from nuclear power reactors. The purpose of the Technical Specifications on effluents is to keep releases of radioactive materials to unrestricted areas during normal operations, including expected operational occurrences, as low as is reasonably achievable (ALARA). Appendix I of 10 CFR Part 50 provides numerical guidance on dose-design objectives for light water reactors to meet this ALARA requirement. The dose-design objectives are low, about 1% of the Radiation Protection Standards of 10 CFR Part 20. Thus, it is possible for a licensee to exceed the dose-design objectives, but still be within the Radiation Protection Standards.

The NRC staff has reviewed the agency records on radioactivity releases from the Pilgrim nuclear power plant. Although there were situations when the radioactivity releases exceeded Pilgrim's Technical Specifications, these releases did not exceed the Radiation Protection Standards of 10 CFR Part 20.

We have also reviewed the agency records on the amounts of radioactivity measured in the environment around the Pilgrim nuclear power plant. The licensee has reported elevated levels above normal background of some radionuclides in some environmental samples over the time period 1978 through 1981. However, it should be noted that Pilgrim's previous guidelines for reporting elevated levels of radioactivity in environmental samples were conservative. Under Pilgrim's current Technical Specifications, many (if not all) of the previously reported elevated levels would no longer be considered reportable. The previously reported elevated levels of radioactivity in environmental samples would lead to doses less than specified in the Radiation Protection Standards and thus would be below NRC regulatory limits.

NRC FORM 8
(4–79)
NRCM 0240

INCOMING AND SIGNATURE TAB

Use this side of the sheet to precede the incoming material when assembling correspondence.

(USE REVERSE SIDE FOR SIGNATURE TAB)

NRC FORM 8
(4-79)
NRCM 0240

INCOMING AND SIGNATURE TAB

Use this side of the sheet to precede the signature page when assembling correspondence.

(USE REVERSE SIDE FOR INCOMING TAB)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

EDO PRINCIPAL CORRESPONDENCE CONTROL

FROM: DUE: 03/25/88 EDO CONTROL: 003569
DOC DT: 03/08/88
SEN. EDWARD M. KENNEDY FINAL REPLY:

TO:

MURLEY

FOR SIGNATURE OF: ** PRIORITY ** SECY NO:

DESC: ROUTING:

Q'S FROM 1/7/88 HEARING ON PROPOSED RESTART OF PILGRIM

STELLO
TAYLOR
REHM
RUSSELL
MURRAY
THOMPSON
BECKJORD
OCA
SECY

DATE: 03/14/88
ASSIGNED TO: NRR CONTACT: MURLEY

SPECIAL INSTRUCTIONS OR REMARKS:

XDSEND VIA 5520 TO EDO IN Q&A FORMAT.
FORMAT ATTACHED.

ACB

OFFICE OF THE SECRETARY
CORRESPONDENCE CONTROL TICKET

PAPER NUMBER: CRC-88-0220 LOGGING DATE: Mar 16 88

ACTION OFFICE: EDO

AUTHOR: E.M. Kennedy
AFFILIATION: U.S. SENATE

LETTER DATE: Mar 8 88 FILE CODE: ID&R-5 Pilgrim

SUBJECT: Questions re the proposed restart of Pilgrim

ACTION: Appropriate

DISTRIBUTION: RF, OCA, Docket

SPECIAL HANDLING: None

NOTES:

DATE DUE:

SIGNATURE: DATE SIGNED:
AFFILIATION:

Rec'd Off. EDO
Date 3-18-88
Time 8A