Charlie Him

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

JUL 12 1984

MEMORANDUM FOR: O. Lynch, Leader, Radiation Protection Section, RAB

FROM: L. R. Greger, Chief, Facilities Radiation Protection Section

SUBJECT: ASSISTANCE IN ESTABLISHING AN OCCUPATIONAL DOSE DATA BASE

The attached information is provided in response to F. J. Congel's July 14, 1982 memorandum, subject as above.

If you have any questions concerning the attachments, please contact Bill Grant (FTS 388-5626).

for

L. R. Greger, Chief
Facilities Radiation
Protection Section

Attachments: As stated

8502090522 840809
PDR FOIA
MAYBERR84-574 PDR

JUNE 26, 1984
03:06

MONTICELLO NUCLEAR GENERATING PLANT

DOSE ESTIMATE STATUS
FOR
RECIRCULATION PIPING REPLACEMENT PROJECT

****

DESCRIPTION		WORK STATUS	CURRENT ESTIMATE MREM	CURRENT ESTIMATE HRS	CURRENT TOTALS MREM	CURRENT TOTALS HRS
1.1 PIPING SYSTEM REMOVAL						
1.2 DRYWELL PREPARATION		100 %	235058	6285	234636	7659
1.3 LOOP 'A' SYSTEM REMOVAL		100 %	18211	1206	18201	1151
1.4 LOOP 'B' SYSTEM REMOVAL		100 %	20677	1092	20666	1043
1.5 RHR PIPING REMOVAL		100 %	72360	1206	72304	1214
1.2 PIPING SYSTEM INSTALLATION						
2.1 LOOP 'A' SYSTEM INSTALLATION		52 %	88500	8850	36467	3659
2.2 LOOP 'B' SYSTEM INSTALLATION		47 %	82500	5500	36467	3659
2.3 DRYWELL RESTORATION	*	0 %	115088	5651	4051	537
2.4 RHR PIPING INSTALLATION	*	25 %	33840	1693	11957	447
1.3 SAFE-END REPLACEMENT						
3.1 LOOP 'A' DISCHARGE S/E REPLACEMENT	*	45 %	196487	3340	113066	2025
3.2 LOOP 'B' DISCHARGE S/E REPLACEMENT	*	45 %	196692	3340	113066	2025
3.3 LOOP 'A' SUCTION S/E REPLACEMENT		100 %	13406	723	13399	757
3.4 LOOP 'B' SUCTION S/E REPLACEMENT		100 %	13406	723	13399	757
3.5 JET PUMP INSTR S/E REPLACEMENT		0 %	19056	548	0	0
3.6 SBLC S/E REPLACEMENT	*	0 %	19660	350	490	5
1.4 SITE SUPPORT						
4.1 MATERIALS & EQUIPMENT HANDLING		82 %	12600	6300	10017	350
4.5 Q/A INSPECTION & RADIOGRAPHY	*	25 %	15900	795	9572	727
4.6 GENERAL SUPERVISION		100 %	16400	820	16325	1087
4.7 SECURITY		57 %	1793	1793	965	0
4.8 GENERAL LABORER SUPPORT WORK	*	82 %	40000	4000	37727	3442
4.9 WASTE HANDLING	*	75 %	21600	864	18316	193
1.5 SEPERATE CONTRACTS						
5.2 HEALTH PHYSICS SUPPORT		82 %	69800	6980	57334	3591
1.6 AUXILIARY WORKSCOPES						
6.1 SMALL BORE PIPING REPLACEMENT	*	45 %	61604	2701	28196	1253
6.2 HANGER & RESTRAINT WORK		43 %	112935	5188	26384	1854
6.4 REFUEL FLOOR WORK	*	55 %	45925	1390	27318	1283
			1523500	71338	920326	38726

* - ACTUAL EXPOSURE ACCUMULATED EXCEEDS ESTIMATE, BASED ON THE PERCENTAGE OF WORK COMPLETED.

JUNE 26, 1984
03:09

MONTICELLO NUCLEAR GENERATING PLANT
DOSE ESTIMATE STATUS
FOR
BALANCE OF 1984 OUTAGE WORK

DESCRIPTION	WORK STATUS	CURRENT ESTIMATE MREM	CURRENT ESTIMATE HRS	CURRENT TOTALS MREM	CURRENT TOTALS HRS
2.1 REFUELING FLOOR					
1.1 GENERAL MAINT/MODS/OPERATIONS	57 %	22510	3430	17255	NA
1.2 REFUELING	50 %	2800	1100	608	NA
1.3 RECIRC PUMP MOTOR & VALVE MAINT	90 %	24930	906	21112	NA
1.4 FW SPARGER REPAIR	50 %	40000	267	14798	NA
2.2 TORUS AREA					
2.1 TORUS PIPE MODIFICATIONS	80 %	31098	7162	29473	NA
2.3 INSTALL WORK PLATFORMS	50 %	1900	400	68	NA
2.4 GENERAL MAINT/MODS/OPERATIONS	57 %	7020	3740	4367	NA
2.5 INSTALL NEW TORUS ACCESS	100 %	2392	598	2392	NA
2.6 SRV BLOWDOWN MODIFICATIONS	8 %	2180	704	322	NA
2.3 REACTOR WATER CLEANUP SYSTEM					
3.1 HEAT EXCHANGER REPLACEMENT	75 %	38106	2002	35235	NA
3.3 INSTALL WORK PLATFORMS & JIB CRANES	40 %	9000	225	895	NA
3.4 QUADREX DECON	100 %	15790	460	15769	NA
3.5 GENERAL MAINT/MODS/OPERATIONS	57 %	18150	492	4622	NA
2.4 DRYWELL					
4.1 GEN SYS MAINTENANCE & MODIFICATIONS	57 %	40570	2311	17631	NA
4.2 SRV REWORK	50 %	13320	370	9519	NA
4.3 LOCAL LEAK RATE TEST	100 %	1225	122	1226	NA
4.4 INSERVICE INSPECTION	95 %	17350	302	17002	NA
4.5 INSTALL RX. VESSEL LEVEL SYSTEM	50 %	5280	240	3258	NA
4.6 RAD PROTECTION COVER	57 %	5200	416	4156	NA
4.7 INSULATION WORK	50 %	4250	85	2075	NA
4.8 CRD MAINTENANCE	25 %	24750	351	5698	NA
4.9 GENERAL ENTRY - INSPECTION & VALVING	57 %	36000	1800	12261	NA
2.5 BALANCE OF REACTOR & RADWASTE BLDGS					
5.1 WORK PLATFORM INSTALLATION	80 %	1832	206	1725	NA
5.2 GENERAL MAINT/MODS/OPERATIONS	57 %	36913	13950	33117	NA
5.3 INSTALL CGCS SYSTEM	75 %	6000	3000	4926	NA
5.5 REWORK MOV-2030	0 %	3040	152	0	NA
5.6 MISCELLANEOUS WORK IN REACTOR BLDG	57 %	50000	5000	45389	NA
5.7 RAD MATERIAL SHIPPING & PACKAGING	57 %	15000	1500	12266	NA
2.6 ENVIRONMENTAL QUALIFICATION PROGRAM					
6.1 LIMIT SWITCH REPLACEMENT DC82M051-1	0 %	1445	173	0	NA
6.2 INSTALL ENVIRO. SEALS DC82M051-2	0 %	934	110	0	NA
6.3 INSTALL SEISMIC & ENVIRO SOV'S	0 %	318	58	21	NA
6.4 RHR MODS	10 %	16505	3301	5700	NA
2.7 TURBINE BUILDING					
7.1 TURBINE OVERHAUL	85 %	2800	2800	2813	NA
7.2 REPLACE FEEDWATER HEATERS	95 %	4562	14774	2722	NA
7.3 RE-TUBE CONDENSER	90 %	5589	8278	4994	NA
7.4 BALANCE OF TURBINE BUILDING	57 %	5000	5000	3666	NA
7.5 GEN SYS MAINT./MODS./OPERATIONS	57 %	6240	6240	4624	NA
		520000	92026	341706	NA

1984 MONTICELLO EXPOSURE (PERSON-MR) SORTED BY DAY

DAY	JAN	FEB	MAR	APR	MAY	JUN	JUL	AUG	SEP	OCT	NOV	DEC
01	94	1015	10181	12771	15476	14628						
02	41	748	17604	16540	14003	7095						
03	170	675	14314	12380	13013	575						
04	172	4402	15768	8776	9380	8044						
05	321	4336	21619	12740	7675	10948						
06	203	5524	16676	21230	3174	5850						
07	63	3448	12587	3209	8772	15469						
08	47	2295	9442	8591	10535	13694						
09	341	6072	11898	12498	8572	10937						
10	213	9101	12201	10272	7340	11272						
11	247	1339	5781	5895	10642	15870						
12	283	547	3881	8254	6903	10883						
13	425	1805	8558	6625	1988	9986						
14	74	3730	3592	8264	8385	10218						
15	9	5265	13848	3296	8936	5693						
16	259	5266	17628	7070	11503	3730						
17	201	6222	6593	6505	10334	1733						
18	319	3506	5997	9195	9570	3710						
19	263	2810	6393	7968	10355	5550						
20	344	8882	11111	7887	535	4708						
21	100	6098	18970	7798	7439	4170						
22	48	4287	16936	675	8652	5092						
23	192	5960	13767	10738	8301	1529						
24	344	8862	25881	8757	9386	2260						
25	776	1726	22606	5902	3907	3490						
26	295	5460	16561	15821	6009	0						
27	647	6922	23707	13559	2142	0						
28	207	7466	9911	10953	4806	0						
29	89	7531	14458	1545	9530	0						
30	567	0	9061	9704	11338	0						
31	583	0	15251	0	9210	0						
	7937	131300	412781	275418	257811	187134						

GRAND TOTAL 1272381 MR (DRD READINGS 152670)
ANNUAL TREND 2623835 MR (7188 MR/DAY) (JUN DRD TOTAL)

	JAN	FEB	MAR	APR	MAY	JUN	JUL	AUG	SEP	OCT	NOV	DEC
EST	10	362	530	516	418	270	230	106	10	10	10	10

RECORD TLD READINGS AND THE RATIO TO LOGGED DOSIMETER READINGS

TLD	8047	131506	429649	277347	259152
DRD	9815	141231	416884	306361	287229
%	79.4	93.1	103.1	90.5	90.2

MWD 11561 777 0 0 0 0 0 0
(GROSS ELECTRICAL) 1984 TOTAL - 12338 MWD

NORMAL OPERATING EXPOSURE - MAN-MREM / MWD ELECTRICAL
.564 2.501 .000 .000 .000 .000 .000 .000
OVERALL - .687

DOSIMETER/TLD EXPOSURE BY RWP FROM 6/25/84 TO 6/25/84

RWP#	MR	WORK DESCRIPTION
6	30	NON-POSTED CONTROLLED AREAS - GENERAL ENTRY
11	50	935 RW COMPACTOR AREA - GENERAL ENTRY
16	115	1001 RX LAUNDRY AREA - GENERAL ENTRY
17	105	1001 RX TOOL DECON AREA - GENERAL ENTRY
28	5	HOT LAB - GENERAL ENTRY & ROUTINE CHEMISTRY
30	15	RW SHIPPING BLDG - GEN ENTRY, STORAGE & SHIPPING
33	10	896 TK RM,HPCI,TORUS, RCIC, TIP DRIVE - GEN ENTR
34	5	RX A RHR - GENERAL ENTRY
35	10	RX B RHR - GENERAL ENTRY
57	175	RX & RW BLDGS - GENERAL ENTRY
62	45	CONTROLLED AREA - CLOTHING AND TRASH PICKUP
67	90	DW - RADIATION PROTECTION COVERAGE
69	210	DW - GENERAL LABORERS AND HELPERS TASK
71	175	1027 RX - GENERAL ENTRY
86	50	935 RW - BARREL COMPACTING HIGH RAD TRASH
130	5	935 RX - INSTALL/REMOVE TEMP ELECTRICAL LIGHTING
195	5	911 CR - ERECT & REMOVE SCAFFOLDING
202	5	COND RM - REMOVE & REINSTALL 13B FW HTR PIPING
204	5	COND RM - REMOVE & REINSTALL 13A FW HTR PIPING
219	15	985 RX - SET RECOMBINERS/STRAINERS/PUMPS/PIPING
251	15	951 TB - DISASSEMBLE, INSPECT, REASSEMBLE ROTORS
266	5	935 RX - SECURITY SURVEILLANCE AT DRYWELL ENT
289	40	896 RX - DO NDE ON VARIOUS WELDS,HANGERS,& SUP P
295	70	DW - INSTALL/REMOVE TEMP PLAT FOR AIR PALLET
323	10	RX BLDG - INSTALL CONDUITS, J BOXES AND SUPPORT
358	80	DW - REMOVE/STORE CONDUIT, CABLE & ASSOC WIRING
364	5	RX BLDG - INSTALL AIRLINE SUPPORTS AND AIRLINES
383	10	962 RW - CLEAN STRAINER & BACKFLUSH VALVES
452	55	896 RX - REMOVE/INSTALL CABLES FOR CORE SPRAY PM
462	20	RCIC - FAB BASE PLATES & DRILL BOLT HOLES
481	10	DW - INSTALL/REMOVE TV MONITORING SYSTEM
485	25	DW - DRILL HOLES IN DW SHELL & WELD ON PEN. NOZZ
508	40	DW - REMOVE RECIRC A LOOP HANGERS, ETC
525	190	896 RX - INSTALL FLEX LOOPS/CAP ENDS OF DRAIN LI
584	115	DW - REMOVE RHR SUCTION & 'A' RETURN PIPING
602	90	DW - SETUP, INTERFERENCES, & CUT PIPE ON RHR 2B
609	25	985 RX - INSTALL PIPING/SUPPORTS FOR RX LEVEL IN
627	15	911 COND RM - REPLACE #1 & 2 CONT VLV STAND STUD
668	60	DW - WORK ASSOC W/ A & B LOOP SYST INSTALLATION
686	5	DW - DO RADIOGRAPHIC EXAM FOR WELD JOINTS
704	65	TORUS RM - INST NEW PIPE VLVS & TUBING
706	5	962 CUPR - INST HX & SMALL BORE PIPING & HANGERS
709	25	DW - INSTALL RECIRC SYS DISCHRG NOZZLE SAFE-END
713	15	COND RM - REMOVE INSULATION FROM PIPING IN CONDE
782	250	DW - INSTALL A & B LOOP SMALL BORE PIPING
784	10	985 RX - INST TUBING FROM LINES TO METERS
788	45	928 TORUS RM - REORIENTATE 6 VALVES
791	5	931 TB - INSTALL INLET ISOL VLVES IN VALVE VAULT
810	5	931 TB - FIRE DAMPER, SCAFFOLDING IN 4 KV AREA
816	1030	1027 RX - INSPEC OF SRM/IRM-REMOVE JET PUMP PLUG
823	5	935 RX - CHECK/REPAIR TIP DRIVE MACHINE COVERS
826	5	951 TB - ASSEMBLE LONG STUB ENDS OF RHR CHECK VL
828	5	B RHR - INSTALL ANCHOR BOLTS ON RHR PIPES
833	5	985 RX - REPAIR RC49-1 LINKAGES
834	5	896 TORUS RM - DO NDE ON PENETRATIONS FROM DW
	3490	

DISCUSSIONS CONCERNING [redacted]
PACIFIC NORTHWEST LABORATORY, WISCONSIN ELECTRIC POWER COMPANY,
NUMANCO, INC., AND WESTINGHOUSE

May 18-19, 1983

1. Participants

PNL - D. W. Murphy, Senior Research Scientist
M. A. Parkhurst, Research Scientist

WEPCO - Doug Johnson, Health Physics Coordinator
Richard Bredvad, Plant Health Physicist

Numanco, Inc. - Dee Kirk, Health Physics Technician

Westinghouse - George Thompson, Training Supervisor
Eugene Ciferno, Instructor

2. General Information

The management of the Point Beach Nuclear Plant (PBNP) has made a commitment to maintaining contamination of the plant facility to a minimum. As a part of this commitment, all personnel (either plant or contractor) are responsible for contamination control at the work site and final cleanup of the work area. No plant decontamination staff is maintained. As a result of this commitment and other plant policies, the plant, outage and contractor staff requirements at PBNP are less than usually seen at most other operating plants. Due to the relatively small plant staff (about 225 people), most non-routine maintenance and outage work is contracted. Oversight of the contractor work is usually provided by PBNP or Wisconsin Electric Power Company (WEPCO) staff. The associated health physics coverage is provided under contract with Numanco, Inc.

The resleeving outage is basically a turn key operation to Westinghouse with overview provided by WEPCO. The WEPCO special projects group consists of five people: a project administrator, two health physics coordinators (one day and one night) and two site coordinators. The special projects group has overseen the activities associated with the resleeving effort. Westinghouse, the primary contractor, has 710 technical people associated with the activity at the site; Atlantic Nuclear Services provides about 80 people for channel head work; Numanco, Inc., has about 30 health physics personnel committed, and Hittman Nuclear and Development maintained 3 people for waste handling. The average staffing level associated with the resleeving is about 330 people. The relatively small staffing level and the sequent camaraderie appears to contribute to the quality of the work being performed and the maintenance of the proposed resleeving schedule.

3. Radiation Dose to Workers

At the time discussions were held, the sleeving was about 60% complete and the total dose as of May 16, 1983 was 442 person-rem based on the self-reading pocket ion chambers (PIC). Results of PICs are known to be higher then TLD badge results, and the total person-rem for the resleeving project is projected to be under 1000 person-rem. The estimated exposure per sleeve was about 300 mrem.

Personnel contamination has been minimal with most contamination incidents being "spot" contaminations which were readily removed. Whole body counts indicate that the respiratory protection program has been adequate. Initially, problems with airborne particulate radioactive iodine were encountered but were controlled by increasing the ventilation in the channel head. Smears from inside the channel head indicated that 80% of the radionuclides was Co-60 and Co-58.

4. ALARA Techniques

The Point Beach plant operates with the policy of maintaining exposures and contamination to a minimum. Although many of the practices discussed in this section could be considered ALARA practices, in most instances they are standard operating procedures for the plant.

Both steam generators were decontaminated using magnetite grit to reduce worker exposure. The B generator was deconned with one pass and resulted in a dose rate reduction of about 2-1/4. The A generator was deconned with three passes and achieved a dose rate reduction of 3 to 3-1/2. The resulting dose rates were 40-50 mrem/min and 75 mrem/min for the A and B generators, respectively. Honing the inside of tubes up to 40" (which was necessary for sleeve insertion) did not significantly reduce the dose rates.

The dose rates at the work platform was 50-60 mrem/hr; however, a shield wall was erected on the platform and an area large enough for two people was taped off where the dose rate was 5-9 mrem/hr. The channel head workers stand behind this wall until they are required to make a "half" jump to load the mandril. Airborne activity at the work platform was usually in the 10^{-10} µCi/cc range or less.

The entire lower area (the 10-foot level) under the reactor and the steam generators was enclosed with plastic sheeting to maintain the outer area as a non-respiratory protection/non-contaminated area. The general area dose rates at the 10-foot level are 5-8 mrem/hr. Contamination at the 10-foot level is usually below 300 dpm. A tool decontamination area has been set up with its own air supply and filter system. Additionally, lay-down areas at the 8-foot level were roped off and contained only essential equipment.

Three important aspects of exposure control were: 1) the limited number of people in the containment dress-out area and in the controlled work areas, 2) the exceptional training program Westinghouse has developed for channel head workers, and 3) the extensive use of remotely operated closed circuit television.

At any given time in the sleeving operation, about eight people are at the access point: Two Westinghouse technicians observe the sleeving on TV monitors and are available to provide assistance as needed, two Numanco health physics technicians observe and record channel head stay times, two Numanco health physics technicians aid in the dressing and undressing of personnel entering and exiting the controlled steam generator work area, and two Atlantic Nuclear Services (ANS) personnel enter the work area. Within the controlled work area, two ANS personnel were on the work platform, and two Numanco health physics technicians (sometimes a third) are stationed at observation points. All of the people in the work area are in constant communication with the Westinghouse control trailer and the people at the access control point.

Westinghouse provided channel head worker training for the ANS personnel. The training was given for 5 to 7 days and lasted 12 hours per day. The training schedule simulated the 12-hour shifts at the plant. The training facility contained a mockup of the channel head, a display of all the manual tools used in the resleeving process and a "think-tank" template where trainees, under observation of their peers, could practice identifying tubes. The training course included full dress rehearsal for each trainee including the use of supplied air. The course was designed to allow as much one-on-one training as possible. Extra time was spent with students as was necessary. One hundred eighteen people were trained at the Point Beach facility.

Finally, Westinghouse has instituted an extensive use of remote-controlled closed circuit television (CCTV) Approximately 80 CCTV cameras were positioned around the work and access areas. CCTV monitors were located at the Westinghouse control trailer, at the access control point (one set for Westinghouse technicians and one set of health physics technicians), and one console for one of the health physics technicians in the controlled work area. The CCTV system was complimented by a direct linked headset communications systems. The combined use of the CCTV and communications network provided constant positive contact of all personnel actively involved in the resleeving process.

5. Worker Exposure Control

Monitoring of channel head worker's radiation exposure was performed by dosimeter packages on the head, chest and initially the gonads. The package for the head contained the TLD dosimeter, a high- and a low-range self-reading PICs. The chest package contained a high-, medium-, and low-range PICs and the gonads were monitored with a medium-range PIC. The gonad monitoring was stopped after 30 channel head entries when the results indicated that the gonadal dose would not be limiting. Additionally, Westinghouse personnel wear a Westinghouse-issued badge on their chest. The official dose of record is the dose to the head based on the TLD results.

For the channel head workers, doses are limited to 2500 mrem with a completed NRC Form-4, 1050 mrem without a completed NRC Form-4 and 300 mrem for females. Doses are recorded daily based on timekeeping and the results of the PICs. The highest result of the PICs is recorded daily and the updated dose records are provided for each shift. Running records for the outage are provided by daily, weekly and monthly exposure results; these are distributed to group heads, group supervisors and are posted at the health physics station. The daily records are reviewed by the plant lead health physicist, the project health physics coordinator, and the Numanco supervisor. The exposure results are transferred to the RWP and checked by the health physics technician at the control point.

General area and channel head surveys are performed twice per shift using Eberline RO-2As (or PIC-6s) and teletectors, respectively. Additional surveys were also performed whenever conditions may have changed or when deemed necessary by health physics personnel.

Although airborne activity and contamination levels in the work area are low, channel head workers on the platform work in supplied-air masks. Other controlled area workers wear full-face respirators with filters. As we stated earlier, no major personnel contamination problems have been encountered.

6. Waste and Waste Handling

The waste from the decontamination operations was handled by Hittman Nuclear and Development Corporation. The magnetite grit was dewatered using a cyclone separator and "dried" grit was piped into liners where solidification was performed. Four Hittman liners were used for the disposal of the grit (approximately 280 cubic feet). The process for handling the grit suffered few problems, and the only delay in the shipment resulted from off-site TRU analyses of the waste.

As expected, the amount of low level radioactive waste (LLW) produced by the resleeving work is significant. Although the actual amount of LLW being generated was not known, it was estimated that the plant was shipping about ten (10) times more LLW than normal. The waste consists primarily of herculite and plastics used for personnel protective clothing. A drum compactor is used to reduce the total volume.

7. Lessons Learned

Because the resleeving process was approximately 60% complete, a debriefing meeting of the outage had not yet been performed. However, there appeared to be few problems involved in the outage as evidenced by the fact that the work was on schedule. Lessons learned and problems associated with the outage will be reviewed and incorporated into the planning for the removal of the steam generators for the sister unit this fall.

DAIRYLAND POWER COOPERATIVE · P.O. BOX 817 · 2615 EAST AV SOUTH · LA CROSSE WISCONSIN 54601
(608) 788-4000

October 25, 1983

In reply, please
refer to LAC-9390

DOCKET NO. 50-409

Mr. J. A. Hind, Chairman
Region III SALP Board
Director, Division of Radiological
and Materials Safety Program
U. S. Nuclear Regulatory Commission
Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

SUBJECT: DAIRYLAND POWER COOPERATIVE
LA CROSSE BOILING WATER REACTOR (LACBWR)
PROVISIONAL OPERATING LICENSE NO. DPR-45
SYSTEMATIC ASSESSMENT OF LICENSE PERFORMANCE (SALP)

REFERENCE: (1) NRC Letter, Hind to Linder,
dated October 5, 1983

Dear Mr. Hind:

The following comments are provided regarding the SALP report (Enclosures 1 and 2) received with your letter (Reference 1). As stated, written comments may be submitted within 20 days following the meeting held October 12, 1983.

On page 6, last paragraph, the NRC states that:

The licensee's total exposures (person-rems) over the preceding five years have increased an average of about 5% per year compared to an average increase of about 20% per year for U. S. boiling water reactors over the same period.

A linear regression analysis of LACBWR's total person Rem dose for the previous five years (1978-1982) was performed by the Radiation Protection Engineer. This analysis shows that the person Rem dose per year has decreased by about 1.7%, and did not increase by 5% over the five year period as the NRC SALP report indicates. A second linear regression analysis of person Rem dose for the period 1973 through 1983 was performed by the Health & Safety Supervisor. This analysis shows that the dose per year has decreased somewhat over the ten year period.

~~8311280046~~

OCT 28 1983

In the same paragraph, last part of last sentence, the NRC states:

... *while power normalized exposures (person-rems/MWe) continue to be high.*

We feel that person-rem per megawatt is an inappropriate comparison, since it is an attempt to justify the production of more electrical power with higher radiation exposure. The NRC, in using this comparison, has become involved in the commercial aspects of nuclear power generation. The more useful SALP comparison for evaluation of individual licensee's ALARA efforts, would be to perform linear regressions of person-Rem/year for each nuclear plant and statistical comparisons between nuclear plants of similar design.

We also wish to address No. 4, Surveillance and Inservice Testing, Item a, last paragraph, last sentence, which states:

The licensee's program, although lacking formal detail, continues to be effective with few surveillances being performed late.

We would like to change that to read..."with no surveillances being performed late during this report period." As a matter of history, in 17 years of Technical Specification testing, only two tests have ever been performed late, the last one being in 1980.

If you have any questions, please contact us.

Sincerely,

DAIRYLAND POWER COOPERATIVE

Frank Linder

Frank Linder
General Manager

FL:JDP:eme

cc: J. G. Keppler, Regional Administrator

CH

DUKE POWER COMPANY
P.O. BOX 33189
CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

September 13, 1983

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Ms. E. G. Adensam, Chief
Licensing Branch No. 4

Re: McGuire Nuclear Station, Unit 2
Docket No. 50-370
Steam Generator Modification ALARA Report

Dear Mr. Denton:

My letter of June 15, 1983 transmitted the subject report for Unit 1. Attached is a supplemental report which provides the information on Unit 2.

Please advise if there are questions concerning this report.

Very truly yours,

H.B. Tucker

Hal B. Tucker

GAC/php
Attachment

cc: Mr. W. T. Orders
NRC Resident Inspector
McGuire Nuclear Station

Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30303

B001
1/1

McGuire Nuclear Station - Unit 2
Steam Generator Modification ALARA Report

McGuire Unit 2 shut down on June 17, 1983 after 4.86 effective full power days to perform the necessary modifications to the preheater section of the steam generator. By comparison, Unit 1 had operated 191 effective full power days at the time of shutdown for the modifications.

Due to the limited operating time on the unit, Health Physics surveys indicated 90 percent of the Reactor Building as having a general area dose rate of less than 5 mR/hr., and Reactor Coolant System contact dose rates were less than 25 mR/hr. In addition, smear survey results for Unit 2 showed no detectable levels of contamination, allowing the modification to proceed without protective clothing dress requirements. The negligible dose rates on contact with the SG shell also allowed the job to be accomplished without shielding.

Unit 2, being a mirror image of the Unit 1 containment, required the same interferences to be removed to access the feedwater nozzle area and the same temporary work platforms to be built to stage the equipment necessary to support the modification. A separate RWP was written for dose accountability for the staging and interference removal and replacement. A total of 1.085 person-Rem was accumulated on these activities.

Lessons learned during the Unit 1 modification and subsequent modifications at other utilities required several changes to be incorporated in the procedures and to determine component fitups prior to final welds. Extremely low radiation levels of 70 mR/hr. on contact with the tubes through the feedwater nozzle allowed experienced Westinghouse personnel to install the protective shroud, catch basin and bolts. (Higher than estimated radiation levels of 3.2 R/hr. prevented the use of these personnel on Unit 1.) This contributed to the lower than estimated exposure for the modification.

The exposure accumulated on Unit 2 is shown on Table 1. The exposure was considerably lower than the estimate due to:

1) Lower dose rates than projected on steam generator tube bundles and tube sheets;

2) Experience of the modification crews; and

3) Efficiency of the modification crews since they were not encumbered with protective clothing.

Table 2 shows the doses, estimated and actual, for the eddy current testing performed following the modification work. Two factors contributed to the lower than estimated doses. These were:

1) Lower dose rates on the primary side of the tube sheets than estimated; and

2) Excellent performance of the hardware which resulted in less time than estimated to accomplish testing.

TABLE 1

Dose for Modification Work

TASK	ESTIMATE	ACTUAL
Steam Generator 2A	1.780	0.730 person-Rem
Steam Generator 2B	1.470	0.775 person-Rem
Steam Generator 2C	1.470	0.550 person-Rem
Steam Generator 2D	1.470	0.760 person-Rem
Area Staging and Interference Removal	1.030	1.085 person-Rem
TOTAL	7.220	3.900 person-Rem

The estimate for Steam Generator 2A was higher due to its being the first to be modified.

TABLE 2

Dose for Eddy Current Testing

TASK	ESTIMATE	ACTUAL
Steam Generator 2A	0.590	0.225 person-Rem
Steam Generator 2B	0.590	0.395 person-Rem
Steam Generator 2C	0.590	0.210 person-Rem
Steam Generator 2D	0.590	0.125 person-Rem
Work Platform Staging	0.100	0.075 person-Rem
TOTAL	2.460	1.030 person-Rem

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July 20, 1983
4410-83-L-0151

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TMI Program Office
Attn: Dr. B. J. Snyder
Program Director
US Nuclear Regulatory Commission
Washington, DC 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
ALARA Exception to Technical Specification Surveillance

NRC letter dated June 15, 1983, requested that GPUNC provide adequate justification for each instance in which a Technical Specification Surveillance Exemption has been invoked due to occupational exposure considerations. Adequate justification should include (but is not limited to): calculations showing yearly man-rem expenditure to perform the surveillance at the required frequency and the criteria used for determining that the man-rem expended is too high (include comparison to yearly exposure for all TMI-2 activities). Additionally, your letter requested GPUNC's Interim Plan (until the revision to the Technical Specifications and the Recovery Operations Plan is implemented) for demonstrating that the affected systems or components will operate as required by the Technical Specifications.

In response to the information requested above, the Technical Specification Surveillances for which GPUNC is requesting ALARA exemption are provided separately in attachments to this letter (Attachments 1 through 10). The justification for each surveillance includes: calculations showing yearly man-rem expenditure to perform the surveillance at the required frequency, risk analysis associated with performing each surveillance, and GPUNC's Interim Plan for the affected systems or components of the surveillance. The dose rates shown in the man-rem calculations have not changed significantly from the dates given in the calculations.

The decision to exclude a surveillance because of ALARA considerations was not based solely upon the radiation exposure incurred in completing the surveillance. Instead, a subjective evaluation was made comparing the reliability of achieving the primary objective which the surveillance sought to assure and the potential impact of the radiation exposure to the occupationally exposed individuals working at TMI-2.

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For comparison purposes, the total radiation exposure incurred for all TMI-2 activities in 1982 was 383 man-rem. The total exposure for the first six months of 1983 was 221 man-rem.

If you have any questions, please feel free to contact Mr. J. J. Byrne of my staff.

Sincerely,

B. K. Kanga

B. K. Kanga
Director, TMI-2

BKK/JJB:RDW/jep

Attachments

CC: Mr. L. H. Barrett, Deputy Program Director - TMI Program Office

Sworn before me this 20th day of July, 1983.

Darla Jean Berry
Notary Public

DARLA JEAN BERRY, NOTARY PUBLIC
MIDDLETOWN BORO, DAUPHIN COUNTY
MY COMMISSION EXPIRES JUNE 17, 1985
Member, Pennsylvania Association of Notaries

BORON INJECTION FLOW PATH SURVEILLANCE: RECOVERY OPERATIONS PLAN SECTION 4.1.1.1.b

An exception is being taken to the monthly GPUNC Surveillance Procedure No. 4303-M4 of four (4) valves associated with the flow path from the Borated Water Storage Tank (BWST) to the reactor vessel. Specifically, the valves are DH-V101A, DH-V101B, DH-V178A, and DH-V178B.

The man-rem calculations for this surveillance are as follows:

Surveillance Location	Area Dose Rates (Date)	Total Estimated Yearly Man-Rem Expenditure to Perform Surveillance
Decay Heat Vault "A"	200 - 400 mR/hr (April, 1982)	1.1 - 2.4
Decay Heat Vault "B"	0.7 - 1.2 R/hr (March, 1982)	4.2 - 7.2
	TOTAL:	5.3 - 9.6 Man-Rem

The valves DH-V178A and DH-V178B are manual, locked open valves on the discharge side of redundant Decay Heat Removal (DHR) coolers A and B, respectively. The personnel risk associated with conducting the surveillance of DH-V178 A and B is due to occupational dose and possible industrial hazards (i.e., the valves are located in the Decay Heat Vaults which are now without lighting). The risk associated with not surveilling these valves is that they could be inadvertently closed and thus prevent injection or recirculation of DHR flow, if required. The increase in plant risk due to omitting this surveillance is judged to be small for the following reasons:

- The position of these valves was verified for the transfer of 50,000 gallons of water from the BWST to Reactor Coolant Bleed Tank (RCBT) A in April, 1982, per GPUNC Special Operating Procedure (SOP) No. R-2-82-018.
- Neither Decay Heat Vaults A or B, which contain DH-V178 A and B have been entered since the performance of SOP No. R-2-82-018. (Last survey of Decay Heat Vault B was performed March 1, 1982; last survey of Vault A was April 24, 1982.) The Decay Heat Vaults which contain DH-V178 A and B are locked high radiation areas. Consequently, access is tightly controlled and can be verified.
- As stated above, according to operating records, the valves are locked open.

Thus, because of the previous verification and limited access to the area containing the valves, it is judged that the probability of DH-V178 A and B being mispositioned upon a potential accident demand is very small. It is further judged that, under the current circumstances, monthly surveillance of the valves will not significantly improve their availability. Therefore, the risk reduction gained in monthly valve surveillance does not merit the dose and occupational hazard incurred.

MAKEUP PUMP OPERATIONAL TEST: RECOVERY OPERATIONS PLAN SECTION 4.1.1.1.c

An ALARA exception is being taken for the monthly operational tests on the Makeup Pumps. These tests are covered by GPUNC Surveillance Procedure No. 4303-M1.

The man-rem calculations for this surveillance are as follows:

Surveillance Location	Area Dose Rates (Date)	Total Estimated Yearly Man-Rem Expenditure to Perform Surveillance
Makeup Pump 1A Cubicle	35 - 60 mR/hr (June, 1982)	.036 - .072
281' Fuel Handling Building East Valve Alley	0.5 - 1.5 R/hr	3.0 - 7.0
281' Fuel Handling Building Valve Alley	0.2 - 0.6 R/hr	1.2 - 3.6
Makeup Pump 1B Cubicle	80 - 160 mR/hr	.008 - .016
Makeup Pump 1C Cubicle	200 - 400 mR/hr	.020 - 0.40
	TOTAL:	4.34 - 12.73 Man-Rem

GPUNC believes that the plant safety risk associated with the non-performance of this procedure is minimal. The reactor coolant pressure and temperature levels are low in the present condition of the plant. The chance of a recriticality and, in turn, the potential need for high pressure injection is very small. In addition, the Mini-Decay Heat System and Standby Pressure Control System are on standby to supply low pressure injection in the unlikely event of deboration or loss of coolant. The surveillance of the Mini-Decay Heat Removal valves and pump operability test is being performed using GPUNC Surveillance Procedure No. 4303-Q3, 4303-Q4, and 4303-M32 in compliance with Technical Specification 4.7.3.3. The Standby Pressure Control (SPC) System, which is presently operating in place of the Makeup System, is on continuous standby and is presently operational. Tests and surveillances with respect to the SPC System via GPUNC Surveillance Procedure Nos. 4303-M3 and 4303-W2 are being performed in compliance with Technical Specifications 4.1.1.1.j.1 and 4.1.1.k. Consequently, the risks associated with not performing this surveillance do not warrant incurring the risks associated with the radiation exposure.

Interim Plan

These pumps are made inoperable in accordance with Technical Specifications, therefore, no Interim Plan is required. Technical Specification Change Request No. 39 (GPUNC Letter No. 4410-83-L-0013) dated January 12, 1983, requested deletion of the Makeup Pumps.

DECAY HEAT REMOVAL PUMP OPERABILITY TEST: RECOVERY OPERATIONS PLAN SECTION 4.1.1.1.d

An exception is being taken to the monthly test of the Decay Heat Removal Pump and surveillance of associated valves (Surveillance Procedure No. 4303-M2).

The man-rem calculations for this surveillance are as follows:

Surveillance Location	Area Dose Rates (Date)	Total Estimated Yearly Man-Rem Expenditure to Perform Surveillance
Decay Heat Vault "A"	200 - 400 mR/hr (September, 1982)	3.0 - 6.0
Decay Heat Vault "B"	0.7 - 1.4 R/hr (March, 1982)	4.2 - 8.4
	TOTAL:	7.2 - 14.4 Man-Rem*

The risk associated with not inspecting these valves prior to pump test is that one or more valves may be mispositioned, this could cause pump damage or liquid contamination of the cubicle, and could inhibit injection of borated water from the Borated Water Storage Tank to the reactor vessel. Cleanup of the cubicle or pump repair could result in large occupational doses. This would particularly be the case if the subject pump test is performed before the cubicle is decontaminated (i.e., under present radiation conditions) and pump maintenance is performed.

The risks associated with conducting the pump test and associated surveillance are due to the occupational dose and possible industrial hazards in the Decay Heat Vault. An industrial hazard exists because the Decay Heat Vaults are without lighting. The valves listed below are necessary for the test and are located in the Decay Heat Vault.

DH-V5 A and B
DH-V100 A and B
DH-V101 A and B
DH-V102 A and B
DH-V106 A and B
DH-V111 A and B
DH-V112 A and B
DH-V122 A and B
DH-V124 A and B
DH-V125 A and B
DH-V129 A and B
DH-V130 A and B
DH-V131 A and B
DH-V167 A and B
DH-V168 A and B
DH-V178 A and B
DH-V179 A and B
DH-V180 A and B
DH-V193 A and B

As is the case with DH-V101 A and B and DH-V178 A and B, (See discussion on exception to Recovery Operations Plan 4.1.1.1.b.) the manual valves in the list were verified in the proper position during the performance of Special Operating Procedure No. R-2-82-018. The positions of the motor operated valves is indicated in the Control Room. Thus, due to the previous verification, inaccessibility of the valve location (which inhibits valve mispositioning) and the Control Room indication on the motor operated valves, the probability of a valve mispositioning is judged to be small.

* This total includes dose rate calculations for Surveillance No. 4630-R3 (Pressure Transmitters Loop Calibration), which ensures that pressure instrument loops used in performance of Tech Spec surveillances are within calibration.

SEISMIC INSTRUMENTATION CHECK, CHANNEL CALIBRATION, AND FUNCTIONAL TEST:
Recovery Operations Plan Sections 4.3.3.3.1 and 4.3.3.3.2

Checks, calibrations, and functional testing of the TMI-2 seismic instrumentation are performed in accordance with GPUNC Surveillance Procedures 4301-M3, 4302-R5, and 4303-SA1, respectively. Exceptions have been taken to these surveillances due to the high rad levels in the area of much of the seismic monitoring instrumentation.

The man-rem calculations for this surveillance are as follows:

Surveillance Location	Area Dose Rates (Date)	Total Estimated Yearly Man-Rem Expenditure to Perform Surveillance
281' Annulus	500 mR/hr (May, 1983)	1.0
328' Auxiliary Bldg.	< 1 mR/hr (July, 1983)	<0.001
315' Reactor Bldg. (Core Flood Tank 1B)	2.3 - 4.0 R/hr (April, 1983)	2.3 - 4.0
347' Reactor Bldg.	50 - 60 mR/hr	.05 - .06
	TOTAL:	3.35 - 5.06 Man-Rem

The following instrumentation is available for Unit 2 to sense and record seismic activity:

2 Triaxial Acceleration Sensors	- one located in the Annulus at elevation 281'-6"; one located atop the Reactor Building, outside containment at elevation 454'-8".
1 Seismic Switch (trigger)	- located in the Annulus at elevation 281'-6".
	- Control Room alarm is activated by this switch.
1 Remote Starter (vertical and horizontal)	- located in the Annulus at elevation 281'-6".
1 Strong Motion Recorder and Playback Unit	- located in the Cable Room.

CONTAINMENT INTEGRITY VERIFICATION: RECOVERY OPERATIONS PLAN SECTION 4.6.1.1

An exception is being taken to the monthly GPUNC Surveillance Procedure No. 4301-M8 associated with the following nineteen (19) valves and flanges:

Valve/Flange	Penetration	Location	Description
SF-V104	R-524	281' R.B.	Fuel Transfer Canal Fill
SV-V54	R-530	281' R.B.	OTSG Secondary Vent
DC-V114	R-531	281' R.B.	Leakage Cooling
DW-V139	R-535	281' R.B.	Demin Service Water
SV-V18	R-569	Seal Injection Room	Secondary Flush and Drain
SV-V17	R-569	281' R.B.	Secondary Flush and Drain
MU-V315	R-570	SIR	Isolation Valve Test
MU-V323	R-570	SIR	Isolation Valve Test
Blank Flange	R-571A	SIR	RB Leak Rate Test
Blank Flange	R-571A	SIR	RB Leak Rate Test
Blank Flange	R-571D	SIR	RB Leak Rate Test
Blank Flange	R-571D	SIR	RB Leak Rate Test
MU-V316	R-572	SIR	Isolation Valve Test
MU-V274	R-573 thru R-576	SIR	Isolation Valve Test
MU-V275	R-573 thru R-576	SIR	Isolation Valve Test
MU-V330	R-573 thru R-576	SIR	Seal Injection Isolation
MU-V364	R-573 thru R-576	SIR	Isolation Valve Test
MU-V365	R-573 thru R-576	SIR	Isolation Valve Test
MU-V439	R-573 thru R-576	SIR	RCP Seal Supply

The man-rem calculations for this surveillance are as indicated on the next page.

Surveillance Location	Area Dose Rates (Date)	Total Estimated Yearly Man-Rem Expenditure to Perform Surveillance
281' Containment Building	(1)	(1)
281' Annulus Area	500 mR/hr	1.0
Seal Injection Valve Room	15 R/hr (2) >200 Rad/hr	18 man-rem gamma
	TOTAL:	>18 man-rem gamma (1)

The justification for the aforementioned valves are discussed in detail in the following sections:

SF-V104: Fuel Transfer Canal Fill

This valve is located in the Reactor Building. While its position is not being checked, the valve downstream of it (SF-V105) is being verified closed, therefore, a containment boundary is being maintained.

SV-V54: OTSG Secondary Vent

This valve is located in the Reactor Building. While its position is not being checked, the valve downstream of it (SV-V55) is being verified closed, therefore, a containment boundary is being maintained.

DC-V114: Leakage Cooling

This valve is located inside the Reactor Building. The valve immediately downstream of it (DC-V115) is located just outside the Reactor Building. The position of DC-V115 is being verified closed, therefore, a containment boundary is being maintained.

DW-V139: Demin Service Water

This valve is located in the Reactor Building. While its position is not being checked, the valve downstream of it (DW-V28) is being verified closed, therefore, a containment boundary is being maintained.

SV-V18 and SV-V17: Secondary Flush and Drain Valves of Penetration R-569

This set of double isolation valves (one inside the Reactor Building; the other outside) has not been surveilled. However, they were locked closed before the March 28, 1979, accident and are still locked closed according to operating records. Therefore, a containment boundary is being maintained.

(1) Areas inaccessible due to high radiological conditions and ALARA considerations.

(2) Dose rates taken at door entrance to the Seal Injection Valve Room.

FIRE PENETRATION SEAL INSPECTION: RECOVERY OPS PLAN SECTION 4.7.11

This specification involves Surveillance Procedures #4331-A1 and #4331-R3 which require inspection to verify that specified fire barriers are functional. ALARA exceptions are being taken on checks of the fire penetration seals in the following areas:

a) Fuel Handling Building Makeup Valve Alley - elevation 305'

b) Fuel Handling Building Makeup Valve Alley - elevation 281'

c) Makeup Demineralizer Room

d) Makeup Filter Room

e) Makeup Pump Cubicles (1A, 1B, and 1C)

f) Seal Return Cooler Room

g) Reactor Coolant Bleed Tank Rooms

h) Neutralizer Tank and Pump Rooms

i) Annulus between the Reactor Building and Fuel Handling Building

j) Spent Resin Tank 'A' Room

The man-rem calculations for this surveillance are as follows:

Surveillance Location	Area Dose Rates (Dates)	Total Estimated Yearly Man Rem To Perform Surveillance
305' F.H. Bldg. Makeup Valve Alley	5 R/hr. at door Hot Spots up to 120 R/hr. (8/82)	Estimates exceed legal limits
281' F.H. Bldg. East Valve Alley	0.5→1.5R/hr. (7/82)	2.25→6.75
M/U Demin Cubicles	300→1150 R/hr (8/6/82)	1350→5175
M/U Filter Cubicles (2B+5B)	1→2R/hr. (6/23/83)	1.5→3

Surveillance Location	Area Dose Rates (Date)	Total Estimated Yearly Man Rem To Perform Surveillance
M/U Pump 1A Cubicle	35→60mR/hr. (6/11/82)	0.1→0.18
M/U Pump 1B Cubicle	80→160mR/hr. (6/16/83)	0.24→0.48
M/U Pump 1C Cubicle	200→400mR/hr. (7/21/82)	0.6→1.2
Seal Return Cooler Room	500→800mR/hr. (5/19/82)	1.5→2.4
R.C. Bleed Tank 1A Cubicle	10→20 mR/hr (7/83)	0.03→0.06
RC Bleed Tank 1B&1C Cubicle	100→400mR/hr (6/83)	0.3→1.2
Neut. Tk & Pump Rooms	200→400mR/hr (5/83)	1.2→2.4
Annulus between RB and FHB	500 mR/hr. (5/83)	1.0
Spent Resin Tank "A" Room	25-30 mR/hr.	0.2

TOTAL: 1358.9→5206 Man-Rem

Any risk from the non-surveillance of fire barrier penetration seals would arise from a potential loss of barrier integrity in the event of fire. Installed combustible loadings in these cubicles are known; transient combustible inventories of these areas have not been evaluated since the accident. However, it is not believed that significant transient combustible inventories exist in these cubicles.

a) In the Fuel Handling Building Makeup Valve Alley at elevation 305', ALARA exceptions have been taken for thirty five (35) fire seals. All 35 are silicon foam seals which are very reliable. Based on the results of previous inspections, no generic problems exist for seals of this type. Risk associated with non-surveillance of these seals seems minimal.

b) In the Fuel Handling Building Makeup Valve Alley at elevation 281', ALARA exceptions have been taken for twelve (12) fire seals. Six (6) of these seals are silicon and four (4) are a boot-type flexible seal. The other two (2) are Firewall-50 seals which are less reliable. All of these seals are located on the South wall of the Makeup Valve Alley so their non-surveillance presents no potential hazard to the Oil Drum Storage Cubicle located north of the Makeup Valve Alley.

c) In the Makeup Demineralizer Room, ALARA exception has been taken for ten (10) fire seals located on the West wall leading into the Makeup Valve Alley. All penetration seals in this room are of highly reliable silicon foam for which no generic problems have been noted to date.

d) In the Makeup Filter Room (elevation 305'), ALARA exception has been taken for surveillance of four (4) seals: three (3) silicon foam and one (1) flexible boot type.

e) In the three Makeup Pump Cubicles which house MU-P-1A, MU-P-1B and MU-P-1C, ALARA exceptions have been taken for a total of thirty-eight (38) fire seals on the West wall of the Auxiliary Building. All of these seals are of highly reliable silicon foam except for six (6) flexible boot-type seals and six (6) silicon/Firewall-50 composite seals.

f) In the Seal Return Coolers Room (elevation 305'), ALARA exceptions have been taken for six (6) fire seals. All six of these seals are of highly reliable silicon foam and are located on the West wall of the Auxiliary Building.

g) In the Reactor Coolant Bleed Tank Rooms (elevation 280'), ALARA exceptions have been taken for nine (9) fire seals on the West wall of the Auxiliary Building. Eight of these seals are silicon foam seals. The other seal is a silicon/Firewall-50 composite. The penetration locations lead into the Fuel Handling Building at the Reclaimed Boric Acid Tank Room.

h) In the Neutralizer Tank and Pump Rooms (elevation 280'), ALARA exceptions have been taken for fourteen (14) fire seals: eight (8) on the North wall of the Fuel Handling Building and six (6) on the West wall of the Fuel Handling Building in the Neutralizer Tank Pump Room. All fourteen (14) seals are made of silicon foam.

i) In the annulus between the Reactor Building and the Fuel Handling Building (at elevation 280'), ALARA exception has been taken for eleven (11) fire seals. In each of these cases surveillance was performed on the side of the seal which faces into the Fuel Handling Building (South wall). Only the side inside the annulus was left unchecked. No problems have been noted on the inspected side of the seals, which have been subjected to similar atmospheric environments as the uninspected side.

j) In the Spent Resin Tank 'A' Room (elevation 280'), ALARA exception is taken for one seal made of a silicon/Firewall-50 composite on the West wall (4 ft. thick) of the Auxiliary Building.

As detailed in the preceding sections, most of the unsurveilled fire penetration seals are made of reliable silicon foam materials. In addition, there is very little equipment operating in most of the cubicles, which thereby decreases the probability of fire ignition. Also, personnel traffic through these areas is (justifiably) low, which reduces the probability of fire.

Interim Plan - Based on the above justifications, these areas will be evaluated to determine the feasibility of moving the surveillance out from their locations into less restrictive, more accessible locations.

This is justified for all areas except:

a) RCBT Rooms
b) Neutralizer Tank and Pump Room
c) Annulus between RB and FHB

due to the fact the remaining areas do not have, nor will have, operating equipment or personal access until decon of both cubicles and systems are complete.

The three areas mentioned above (RCBT Rooms, Neutralizer Tank and Pump Rooms, and Annulus Area) do not necessarily meet the above criteria, therefore, evaluations will be made in order to dètermine if decon of these areas can be accelerated in order to perform the required surveillances.

The RCBT 1A cubicle man-rem estimate indicates the seals in that area can be inspected with minimum man-rem exposure, therefore, inspections in this area will be completed.

Additionally, it is expected that current deconning efforts in the Spent Resin Tank "A" Room will allow inspections in this area to be completed.

DECAY HEAT CLOSED COOLING WATER VALVE LINEUP VERIFICATION: RECOVERY OPERATIONS PLAN SECTION 4.7.3.2.6

An exception is being taken to the quarterly cycling of valves in the flowpath through the decay heat coolers in accordance with GPUNC Surveillance Procedure #4303-M25. Currently, the only valves affected by the exception are DC-V8A and DC-V8B.

The man-rem calculations for this surveillance are as follows:

Surveillance Location	Area Dose Rates (Date)	Total Estimated Yearly Man-Rem Expenditure to Perform Surveillance
Decay Heat Vault "A"	200-400 mr/hr (April, 1982)	0.24 - 0.96
Decay Heat Vault "B"	0.7 - 1.2 R/hr (March, 1982)	1.92 - 3.84
	TOTAL:	2.16 - 4.80 Man-Rem

DC-V8A and DC-V8B are located in the decay heat vaults and are wired and sealed open. By this method, the valve is wired to prevent closure by entwining wire in the manual operator, then a lead seal is added to the wire so that any tampering can be identified.

DC-V8A and V8B are not required to close under normal operation of the DHCCW System. These valves are located on the inlet side of shell-side cooling water flow to the decay heat removal coolers. The throttle valves for this flow are located on the outlet side of the coolers (DC-V73A and DC-V73B) thus closure of DC-V8A and -V8B is required only for maintenance on the heat exchanger/coolers.

Thus, an exception to this surveillance is justified from a risk perspective because:

- Current records indicate that DC-V8A and B are physically opened (wired and sealed).
- The act of cycling introduces the potential for valve mispositioning.
- The dose and occupational hazard incurred in performing the procedure.
- The DH Coolers serve as backup to the MDH Coolers in performing emergency heat removal from the reactor. However, as noted in the justification for the exception to Recovery Operations Plan Section 4.1.1.d, the reactor will be cooled via ambient heat losses even with the vessel drained to the bottom of its nozzles.

FIRE SYSTEM VALVE CHECKS: RECOVERY OPERATIONS PLAN SECTION 4.7.10.1.1.c

An ALARA exception is being taken for the monthly inspection to verify the position of Fire Service Valves FS-V633 and FS-V634. These two valves are the shut-off valves for the hose reels at elevation 282'-6" in the Reactor Building. All other valve inspections in Surveillance Procedure No. 3301-M1 are being performed.

The estimated yearly man-rem expenditure required to perform this surveillance has not been calculated since surveys have not been made in this area. However, the estimated radiological conditions for this area of the building (greater than 50 R/hr) indicate that insufficient stay-time would be available, consistent with Federal Exposure Limitations, to perform meaningful activity.

The major risk associated with the non-surveillance of these two valves is that if they (or either one) were left in the open position, they would leak additional water into the RB basement. Sludge samples taken from the RB basement seem to indicate that criticality is not a problem in this area, so deboration presents no potential hazard from a reactivity standpoint.

When there are no personnel inside the RB, the Fire Service System is isolated from the containment. When a containment entry is being performed that would result in an increased fire risk, the isolation valve is opened. At this point, if either FS-V633 or FS-V634 are open, the fire pumps would kick on automatically since these pumps are pressure sensitive. Control Room Operators would recognize the actuation of the fire pumps coincided with the opening of the RB isolation valve, and they could reisolate the fire service water from the containment. Therefore, the amount of water added to the Reactor Building basement would be minimized. The fact that the above scenario has not occurred indicates that these valves are closed. (There would be virtually no chance of depleting fire protection water sources.) Also, since no energized electrical equipment is presently on the 282' Reactor Building elevation, there would be no potentially hazardous electrical consequences associated with fire hose leakage. There are portable fire extinguishers and fire hose stations on the 305' and 347' elevations of the Reactor Building.

GPUNC therefore believes that the risk associated with deferral of this surveillance is very low.

Interim Plan

Verification of the position of FS-V633 and FS-V634 is not required due to:

- As previously mentioned, if either FS-V633 or FS-V634 are open, Control Room Operators would recognize the actuation of the fire pumps coincide with the opening of the RB isolation valve and could reisolate the fire service water from the containment

- Accessibility to the Reactor Building basement is not possible due to ALARA considerations. Therefore, there is no need to have an operable fire station on this elevation

FIRE SYSTEM VALVE CYCLING: RECOVERY OPERATIONS PLAN SECTION 4.7.10.1.1.e.2

This Technical Specification is covered by GPUNC Surveillance Procedure No. 4333-R5, which requires the cycling of valves FS-V633, 634, 635, 637, 638, 643, 644, and other surveillance requirements. All cycling is being performed with the exception of FS-V-633 and FS-V-634 for which ALARA exception has been taken.

The major risk associated with this non-surveillance is the potential for the valve to rust shut and would therefore be unavailable in the event of an emergency demand. These two valves are in the Reactor Building basement.

The estimated yearly man-rem expenditure to perform this surveillance has not been calculated since radiation doses have not been surveyed. However, the estimated radiological conditions for this area of the building (greater than 50 R/hr) indicate that insufficient stay-time would be available, consistent with Federal Exposure Limitations, to perform meaningful activity.

Since personnel access to the area is extremely limited and electrical equipment in the area is deenergized, the risk due to non-performance of these two valve cyclings is very low. Additionally, there are portable fire extinguishers and fire hose stations available on the 305' and 347' elevations of the Reactor Building which can be used.

Interim Plan

See response to Recovery Operations Plan Section 4.7.10.1.1.c (Fire System Valve Checks).

NOTE: GPUNC Letter 4410 83-L-0102, B. K. Kanga to L. H. Barrett, mistakenly stated that ALARA exceptions were being taken for two valves in the RB required by Recovery Operations Plan Section 4.7.10.1.1.d.2. Cycling of these valves is covered in Technical Specification Section 4.7.10.1.1.e.2 which is fulfilled by Surveillance Procedure No. 4333-R5, not No. 3303-A1. Therefore, Recovery Operations Plan Section 4.7.10.1.1.d.2 does not need to be addressed in this submittal.

CONTAINMENT AIRLOCKS: RECOVERY OPERATIONS PLAN SECTION 4.6.1.3

All of the surveillance requirements for the airlocks are complied with excepting deferral of the semi-annual overall airlock leakage test at 56.2 psig (Surveillance Procedure #4303-SA2). Technical Specification Change Request No. 39 (4410-83-L-0013 dated January 12, 1983) will decrease this requirement to 6.5 psig.

The dose rate calculations associated with this surveillance are as follows:

Surveillance Location	Area Dose Rates	Total Estimated Yearly Man-Rem Expenditure to Perform Surveillance
Personnel Hatch	110 mr/hr	2.0
Equipment Hatch	200 mr/hr	3.6
	TOTAL:	5.6 Man-Rem

The airlocks are designed as Seismic Class I structures fabricated in accordance with Section III, Subsection B of ASME B & P.V. Code and ANSI N 6.2. The structural integrity of the airlocks as part of the containment was originally verified by a proof test at 115 percent of the design pressure; and an initial integrated leakage rate test was performed satisfactorily prior to initial startup. These tests established the structural integrity of the airlocks for the design peak containment internal pressure. The airlocks have since been observed to satisfactorily hold 28 psig pressure spike and a maximum temperature of 182°F that occurred approximately 10 hours into the incident of March, 1979. A sustained elevated temperature and pressure of 150°F (first 24 hours) and 1.5 psig (first 14 hours) occurred until a steady state temperature and pressure of 130°F and 0 psig was attained. Also, the airlock was exposed to Reactor Building spray solution (boric acid, sodium hydroxide, and sodium thiosulfate) runoff via the containment walls. The airlocks are designed to withstand a maximum pressure of 60 psig and temperature at 286°F. Corrosion of the airlocks has been evaluated and not found to be significant.

GPUNC believes that the airlock structural design instills sufficient confidence in the structure. Therefore, the risk of potential leakage from the structure can be considered insignificant. The major contribution to airlock mechanical failures is attributed to their active components. These active components are the bulkhead doors, interlock mechanisms, door seals, latch mechanisms, handwheel shafts and bushings, solenoid locking assemblies, and various valves and gears. Industry-wise, however, airlock failures have been dominated by leakage of the door seals (which are made of silicone rubber). The airlock door seals are regularly tested for leak rate (i.e., after each entry) and interlock verification is done in conformity with Surveillance Procedure #4311-5. The other active components have been checked regularly via Technical Specification Surveillance 4.6.1.3(b), which requires that inspection of these components be performed on a quarterly basis (GPUN Surveillance Procedure #4303-Q5).

NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

July 29, 1983

Attention: Mr. Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulations
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: Nine Mile Point Unit 1
Docket No. 50-220
DPR-63

Dear Mr. Eisenhut:

The attached report regarding occupational doses received during the recirculation system replacement is submitted in accordance with the requirements of Amendment 50 to Operating License DPR-63. This report covers the period from April through June 1983 and satisfies the requirements of Operating License paragraph 2.D(6) c and d for a quarterly progress report and a final report due within 60 days after completion of the repair.

Sincerely,

C.V. Mangan

C. V. Mangan
Vice President
Nuclear Engineering and Licensing

CVM/SMK:bd

Attachment

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QUARTERLY TASK PERSON-REM REPORT

In accordance with the requirements of Operating License DPR-63, paragraphs 2.D(6)c and d, information pertaining to the occupational dose received during the recirculation system safe-end and piping replacement is presented below and in Appendices A and B.

Table 1, Task Person-Rem Summary Report by Master Task (Revision 7), gives estimates of person-rem and person hours. The total estimated person-rem has remained at 1561 since the last reporting period.

Weekly Task Person-Rem Summary Reports were used to compare actual accumulated exposure and person hours with the estimate. Table 2, Summary of the Weekly Task Person-Rem Summary Reports, gives the status of this comparison each week during the reporting period.

Appendix A, the Task Person-Rem Summary Report includes:

1. A summary of the occupational dose received for the outage by master task category.
2. A comparison of actual dose and person-hour totals with the latest estimate.

The totals presented in Table 2 and Appendix A are the final totals for the recirculation system safe-end and piping replacement with one exception. Task 9009, Decontamination of Tools and Equipment, was still in progress at the time of startup. Therefore, an estimate of expected hours and exposure for this task is given.

Appendix B, Task Person-Rem Summary Analysis, identifies and explains major differences between the Revision 7 estimate and the final totals for the outage.

A radiation exposure monitoring system was utilized to compare film badge and dosimeter records for personnel involved in the recirculation system replacement. The ratio of film badge to recorded dosimeter exposures was 0.82 for the outage. Application of this ratio to the total accumulated exposure (actual person-rem) listed in Table 2 yields a total exposure of 1200 person-rem for this outage. This is 77% of the Revision 7 estimated person-rem.

TOTAL JOB : 1,465 PERS. REM MEASURED W/DOSIMETERS
~1,201 PERS. REM MEASURED W/FILM BADGES

TABLE 1

TASK PERSON-REM SUMMARY

REPORT BY MASTER TASK (REVISION 7)

MASTER TASK NUMBER	TASK DESCRIPTION	ESTIMATED EXPOSURE (Person-Rem)	ESTIMATED PERSON-HOURS
1xxx	Preliminary drywell work	571.188	57,703
2xxx	Removal of recirculation loop - suction-side	174.775	4,010
3xxx	Replacement of safe-ends and elbows - suction-side	116 659	3,429
4xxx	Loop replacement - suction	111.811	6,327
5xxx	Loop replacement - discharge	93.086	4,855
6xxx	Removal of discharge loop	266.585	5,348
7xxx	Safe-end/elbow install discharge side	72.018	3,188
8xxx	Post welding drywell work	63.730	3,983
9xxx	Indirect recirculation system work	90.929	6,246
TOTAL		1,560.781	95,089

TABLE 2

SUMMARY OF WEEKLY TASK PERSON-REM SUMMARY REPORTS

WEEK ENDING	TOTAL ACCUMULATED PERSON-REM	PERCENTAGE OF CURRENT ESTIMATED PERSON-REM	TOTAL ACCUMULATED PERSON-HOURS	PERCENTAGE OF CURRENT ESTIMATED PERSON-HOURS	APPROPRIATE REVISION NUMBER
4/08/83	1410	95	87,843	99	7
4/15/83	1418	94	88,647	98	7
4/21/83	1428	94	91,024	99	7
4/30/83	1437	94	93,110	100	7
5/05/83	1446	93	94,580	100	7
5/12/83	1455	94	95,380	101	7
5/19/83	1459	94	95,670	101	7
5/26/83	1462	94	96,020	102	7
6/02/83	1465 *FILM B.*.82: = ~ 1201 DOSIMETERS	94	100,249 *	105	7

* Includes 2.1 Rem and 5163 hours estimated for completion of Task 9009, Decontamination of Tools and Equipment.

APPENDIX B

TASK PERSON-REM SUMMARY ANALYSIS

The current person-rem and person-hour estimates (Revision 7) were prepared in March 1983. At that time, about 90% of the recirculation piping replacement was completed. Revision 7 was based on the actual person-rem and person-hours incurred up to that time. Since then, the estimates and actual totals have remained in good agreement. The final person-rem total is 94% of the estimate while the final person-hour total is 105% of the estimate. Because the difference between the Revision 7 estimates and the final outage totals is so small, further revision to the estimates was deemed unnecessary. Although the difference between the estimates and actual totals is insignificant, there are some disparities between individual task estimates and actual task person-rem and/or person-hours.

Individual task totals which were either significantly above or below the Revision 7 person-rem and person-hour estimates are discussed in this section. Tasks whose totals were significantly below the estimate (person-rem savings) are found in Section A. Section B discusses individual tasks whose totals are significantly above the estimate.

A. ACTUAL TOTALS BELOW THE ESTIMATE (PERSON-REM SAVINGS)

Tasks 8660,8690 - Removal of Shielding and Scaffolding from Drywell

A significant reduction in person-hours for the removal of shielding and scaffolding from the drywell saved about 12 person-rem. By establishing easily accessible storage areas near the equipment hatch in the drywell and restricting removal of scaffolding only at selected intervals, person-hours were kept to a minimum. Only 115 of an estimated 775 person-hours were used for these tasks.

Task 1010 - General Inspection

The estimates for general inspection in Revision 7 were based on the average daily person-rem and person-hours attributable to this task throughout the outage. However, as the outage drew to a close, less inspection was required. Reduced drywell inspection requirements resulted in 21 fewer person-rem and 326 fewer person-hours than estimated.

Task 8700 - Replace Decking

Prior to pipe removal, a minimal amount of decking was removed to facilitate rigging out the piping. The decking was stored in LSA boxes outside the reactor building. Prior to re-installation, most decking pieces were thoroughly decontaminated to reduce loose surface contamination levels. This enabled workers to replace grating without respirators in most cases, greatly increasing worker efficiency and reducing time spent in the drywell.

B. ACTUAL TOTALS HIGHER THAN ESTIMATE

8670 - Re-Installation of Insulation

Recirculation piping insulation reinstallation was performed by a separate contractor specializing in insulation work. Nevertheless, this task exceeded its estimate by over 1222 person-hours and 8 person-rem. Major reasons for this overrun were:

a. At first, reinstallation of insulation was not progressing as well as expected. Although job progress is usually slow in the beginning, no improvement in job quality was seen in the reinstallation work. A brief investigation discovered that the foreman assigned to this project was unfamiliar with mirror insulation work and often misinterpreted job assignments. A person more experienced in mirror insulation work was assigned the foreman's position. Soon after, errors significantly dropped and job progress was restored to a normal pace.

b. Prior to reinstallation, the recirculation pipe insulation was laid out on the turbine deck to identify each piece's exact location before moving it to the drywell. This allowed workers to locate worn out pieces, replace them and ensure that no parts were missing. Although most pieces were clearly marked and identified, some small pieces of insulation could not be identified or their location determined. Finding their location took extra time because many of these pieces were similar in appearance.

c. Recirculation gate valves were not all reinstalled in the same orientation. This necessitated recutting and fitting insulation pieces in the drywell.

8730 - Final Drywell Cleanup

The estimate for this task was based upon the assumption that final drywell cleanup would be a once-through intensive cleaning just prior to startup. However, final drywell cleanup was a continuous process that lasted for more than a month.

9009 - Decontamination of Tools and Machines

The Revision 7 estimate assumed decontamination of small tools and specialized machinery that belonged to the contractor and needed to be shipped offsite. However, due to the success of the ultrasonic/freon rig brought to the site for this purpose, it was decided that as many items as possible (chain falls, cables, tools, etc.) would be decontaminated. The rig was set up in a low radiation area (1 mR/hr) and a massive decontamination effort was undertaken. This resulted in a significant increase in person-hours over the estimate (1555 hours for the outage period), but no increase in person-rem over the estimate.

1014 - Security

Security exceeded estimated person-hours by 380. Person-rem was not a problem. Plant procedures required a guard to be posted on duty at the drywell entrance during the outage. About 18,400 person-hours were spent posting guard at the drywell entrance, using approximately 32 person-rem.