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SUBJECT: "Annual Operating Rept for DC Cook Nuclear Plant for 1989."
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Indiana Michigan
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P.O Box 16631
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INDIANA
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United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

February 28, 1990

Donald C. Cook Nuclear Plant
Docket Nos. 50-315/50-316
License Nos. DPR-58/DPR-74

Document Control Manager:

Attached are two copies of the 1989 Annual Operating Report for the Donald C. Cook Nuclear Plant. The information contained in this report covers the activities delineated in the Donald C. Cook Nuclear Plant Technical Specifications, Section 6.9.1.5, and the requirements of 10 CFR 50.59. In addition, a revision to the 1988 personnel exposure submittal is included. The dose totals section of this report has been modified to include steam generator repair data, and all readings are in SRD dose. This year's report is in TLD dose to more accurately reflect the dose received during the year. We will continue to report personnel exposure in TLD dose in the future.

Copies of this report are being transmitted to the Regional Administrator, the Director of Inspection and Enforcement, the Director of the Office of Management Information and Program Control, and the NRC Resident Inspector as specified in 10 CFR 50.4 and 10 CFR 50.59.

Sincerely,

M. P. Alexich
Vice President

cc: D. H. Williams, Jr.
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DONALD C. COOK NUCLEAR PLANT

ANNUAL OPERATING REPORT

1989

COMPILED BY: J. E. Borggren
Nuclear Operations Support

REVIEWED BY: B. A. Svensson
Licensing Activities Coordinator

APPROVED BY: M. P. Alexich
Vice President

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INTRODUCTION

The Donald C. Cook Nuclear Plant, owned by Indiana Michigan Power Company is located five miles north of Bridgman, Michigan and consists of two nuclear power units. Each unit employs a pressurized water reactor nuclear steam supply system furnished by Westinghouse Electric Corporation.

The Unit 1 reactor is currently designed for a power output of 3250 MWt and the Unit 2 reactor is designed for a power output of 3411 MWt, which are their licensed ratings. The approximate gross and net electrical outputs of Unit 1 are 1056 MWe and 1020 MWe and of Unit 2 are 1100 MWe and 1060 MWe, respectively. The main condenser cooling method is open cycle using Lake Michigan water as the cooling source. The Cook Nuclear Plant was the first domestic nuclear facility to employ the ice condenser reactor containment system. The American Electric Power Service Corporation was the architect-engineer and constructor.

This report was compiled by J. E. Borggren with the following individuals contributing information as follows:

D. C. Loope	-	Personnel Exposure Summary
D. C. Loope	-	Update to 1988 Personnel Exposure Summary
C. A. Freer	-	Steam Generator ISI Summary
D. H. Malin	-	Results of Irradiated Fuel Inspections
B. A. Svensson	-	Changes to Procedures
B. A. Svensson	-	Tests and Experiments Not Described in the FSAR
B. A. Svensson	-	Challenges to Pressurizer PORVs and Safety Valves
J. B. Kingseed	-	Changes to Facility - RFCs
R. B. Bennett T. G. Harshbarger	-	Licence Amendments

PERSONNEL EXPOSURE SUMMARY

ANNUAL OPERATING REPORT - RG 1.16 FOR 1989

	# PERSONNEL >100 mR			TOTAL MAN-REM		
	STAT.	UTIL.	CONT.	STATION	UTILITY	CONTRACT
Reactor Operations & Surveillance						
Maintenance Personnel	0000	0000	0015	0.000	0.000	3.209
Operations Personnel	0055	0000	0023	13.233	0.000	4.694
Health Physics Personnel	0011	0001	0068	1.591	0.120	22.112
Supervisory Personnel	0000	0002	0001	0.000	0.165	0.252
Engineering Personnel	0002	0000	0006	0.180	0.000	0.591
Routine Maintenance						
Maintenance Personnel	0103	0001	0457	30.984	0.212	232.997
Operations Personnel	0023	0004	0064	5.774	0.845	18.627
Health Physics Personnel	0002	0000	0038	0.477	0.000	11.321
Supervisor Personnel	0001	0001	0001	0.098	0.128	0.167
Engineering Personnel	0009	0016	0017	1.729	2.295	3.222
In-Service Inspection						
Maintenance Personnel	0004	0000	0050	0.544	0.000	16.568
Operations Personnel	0002	0001	0015	0.332	0.098	3.893
Health Physics Personnel	0000	0000	0006	0.000	0.000	0.917
Supervisory Personnel	0000	0000	0001	0.000	0.000	0.401
Engineering Personnel	0003	0000	0000	0.252	0.000	0.000
Special Maintenance						
Maintenance Personnel	0002	0000	0035	0.174	0.000	5.293
Operations Personnel	0000	0000	0000	0.000	0.000	0.000
Health Physics Personnel	0000	0000	0001	0.000	0.000	0.109
Supervisory Personnel	0000	0000	0000	0.000	0.000	0.000
Engineering Personnel	0000	0000	0001	0.000	0.000	0.109
Waste Processing						
Maintenance Personnel	0007	0000	0080	0.980	0.000	25.480
Operations Personnel	0000	0000	0008	0.000	0.000	3.563
Health Physics Personnel	0002	0000	0023	0.999	0.000	7.476
Supervisory Personnel	0001	0000	0000	0.270	0.000	0.000
Engineering Personnel	0002	0000	0001	0.418	0.000	0.214
Refueling						
Maintenance Personnel	0004	0000	0030	0.833	0.000	7.805
Operations Personnel	0011	0000	0035	3.247	0.000	17.764
Health Physics Personnel	0001	0000	0007	0.120	0.000	0.714
Supervisory Personnel	0000	0000	0001	0.000	0.000	0.189
Engineering Personnel	0001	0000	0000	0.086	0.000	0.000
Totals						
Maintenance Personnel	0104	0001	0567	33.515	0.212	291.352
Operations Personnel	0077	0004	0133	22.586	0.944	48.541
Health Physics Personnel	0013	0001	0089	3.188	0.120	42.649
Supervisory Personnel	0002	0002	0003	0.368	0.293	1.008
Engineering Personnel	0016	0016	0022	2.666	2.294	4.135
Grand Totals	0212	0024	0814	62.324	3.863	387.685

NOTE: ALL VALUES ON THIS SHEET ARE MEASURED BY TLD.

PERSONNEL EXPOSURE SUMMARY

1.16 REPORT - WORK FUNCTION CATEGORIES

Reactor Operations and Surveillance

Reactor Operations and surveillance are those activities involved with operating the plant or monitoring it's operations, including chemistry, performance testing, surveillance testing, etc. The plant may be at any power level, including zero, and still have work falling into this area. Many STP's run during shutdown or refueling may still fall into this category.

Routine Maintenance

Routine maintenance includes all equipment or system maintenance, whether preventative or restorative, which does not involve significant modifications to equipment or systems. Included is I&C repair work, as well as work to adjust operable equipment to improve performance (adjusting fan blade pitch, for example).

Inservice Inspection

Inservice inspections are inspections of equipment and systems to monitor changes that would be detrimental to function or integrity. Also included is all work required to permit such inspections, such as building required scaffolding, removing or replacing supports of insulation, or disassembly of valves, pumps, etc. Not included are inspections to assess or monitor normal wear, etc. For example, disassembly of a charging pump to inspect bearing wear would not be Inservice Inspection, but disassembly to inspect for rotor cracking or casing damage would be. Inspection of a weld on a newly added line is Special Maintenance, whereas inspection of a weld repair at a leaking fitting is Routine Maintenance.

Special Maintenance

Special maintenance includes all work on equipment or systems performed to make significant modifications. Installation of new systems or equipment, replacements or addition of supports or hangers, addition of new lines or instruments, removal of existing equipment, replacement of existing equipment with significantly different equipment are all Special Maintenance. For example, replacement of a properly functioning, original equipment pressure transmitter with a different model with improved characteristics or certification would be Special Maintenance, but replacement of a malfunctioning pressure transmitter with a newer or improved model would probably be Routine Maintenance.

PERSONNEL EXPOSURE SUMMARY

Waste Processing

Waste processing includes all work associated with decontamination of equipment, areas, systems, etc. (if not an integral part of another job, such as pump repair), and collection and processing of waste, whether solid, liquid or gas. Operations in support of waste handling are also included. For example, draining a filter to permit changing it, or venting it after changing are part of Waste Processing, but valving a clean filter into the system is Reactor operations. Repair of the Baler or drumming room crane is Routine Maintenance.

Refueling

All work directly concerned with refueling the reactor, including all support operations, is classified as Refueling. Testing the polar crane or installing the cavity filter rig is part of Refueling, as is cavity decon before or after flood-up. Changing the cavity filter, however, is Waste Processing and repairing the manipulator crane is Routine Maintenance.

PERSONNEL EXPOSURE SUMMARY

ANNUAL OPERATING REPORTING - RG 1.16 for 1988

	# PERSONNEL >100 mR			TOTAL MAN-REM		
	STAT.	UTIL.	CONT.	STATION	UTILITY	CONTRACT
REACTOR OPERATIONS & SURVEILLANCE						
Maintenance Personnel	0007	0000	0010	0000.971	0000.000	0003.062
Operations Personnel	0056	0001	0033	0015.451	0000.155	0013.699
Health Physics Personnel	0018	0000	0061	0007.353	0000.000	0022.952
Supervisory Personnel	0000	0000	0001	0000.000	0000.000	0000.130
Engineering Personnel	0010	0000	0004	0001.822	0000.000	0000.674
ROUTINE MAINTENANCE						
Maintenance Personnel	0101	0002	0241	0044.339	0000.267	0155.889
Operations Personnel	0019	0001	0022	0005.910	0000.338	0006.246
Health Physics Personnel	0006	0000	0013	0001.205	0000.000	0005.106
Supervisory Personnel	0002	0000	0001	0000.285	0000.000	0000.244
Engineering Personnel	0003	0003	0000	0000.357	0000.536	0000.000
IN-SERVICE INSPECTION						
Maintenance Personnel	0005	0000	0018	0000.954	0000.000	0003.448
Operations Personnel	0003	0001	0014	0000.630	0000.650	0008.586
Health Physics Personnel	0003	0000	0003	0000.494	0000.000	0000.929
Supervisory Personnel	0001	0000	0000	0000.170	0000.000	0000.000
Engineering Personnel	0000	0000	0001	0000.000	0000.000	0000.170
SPECIAL MAINTENANCE						
Maintenance Personnel	0011	0031	0680	0001.722	0028.997	0541.633
Operations Personnel	0020	0010	0011	0004.961	0004.987	0010.332
Health Physics Personnel	0000	0000	0079	0000.000	0000.000	0103.184
Supervisory Personnel	0010	0010	0012	0001.102	0000.432	0004.257
Engineering Personnel	0010	0010	0020	0000.145	0003.130	0022.918
WASTE PROCESSING						
Maintenance Personnel	0001	0000	0054	0000.620	0000.000	0019.097
Operations Personnel	0000	0000	0004	0000.000	0000.000	0002.429
Health Physics Personnel	0000	0000	0017	0000.000	0000.000	0004.597
Supervisory Personnel	0000	0000	0000	0000.000	0000.000	0000.000
Engineering Personnel	0001	0000	0001	0000.170	0000.000	0000.175
REFUELING						
Maintenance Personnel	0004	0000	0018	0000.528	0000.000	0008.664
Operations Personnel	0010	0001	0066	0004.750	0000.106	0033.389
Health Physics Personnel	0002	0000	0014	0000.205	0000.000	0005.743
Supervisory Personnel	0000	0000	0000	0000.000	00000.00	0000.000
Engineering Personnel	0001	0001	0000	0000.121	0000.151	0000.000
						Revised
TOTALS						
Maintenance Personnel	0129	0033	1021	0049.134	0029.264	0731.793
Operations Personnel	0108	0014	0150	0031.702	0006.236	0074.681
Health Physics Personnel	0029	0000	0187	0009.257	0000.000	0142.511
Supervisory Personnel	0013	0010	0014	0001.557	0000.432	0004.631
Engineering Personnel	0025	0014	0026	0002.615	0003.817	0023.937
GRAND TOTALS	0304	0071	1398	0094.265	0039.749	0977.553

NOTE:. ALL VALUES ON THIS SHEET ARE MEASURED BY SRD.

STEAM GENERATOR INSERVICE INSPECTION REPORTS

SUMMARY OF UNIT 1 STEAM GENERATOR INSPECTION

Unit One Steam Generators were Bobbin Coil Eddy Current inspected in 100% of the tubes in all four steam generators. A summary of the number of tubes inspected and the extent of the inspection of each steam generator is as follows:

Steam Generator 11

All tubes tested from the inlet (HL)
515 full length (TE-CL to TE-HL)
2,784 U-bend (7TSP-CL to TE-HL)

Steam Generator 12

All tubes tested from the inlet (HL)
499 full length (TE-CL to TE-HL)
2,781 U-bend (7TSP-CL to TE-HL)

Steam Generator 13

All tubes tested from the inlet (HL)
511 full length (TE-CL to TE-HL)
2,828 U-bend (7TSP-CL to TE-HL)

Steam Generator 14

All tubes tested from the inlet (HL)
510 Full length (TE-CL to TE-HL)
2,768 U-bend (7TSP-CL to TE-HL)

The Bobbin Coil Test frequencies used were as follows:

400 kHz as the prime test frequency.
200 kHz and 100 kHz as supplemental frequencies.
10 kHz as the locator frequency.

All tubes were tested utilizing a .720 diameter probe with the exceptions of rows 1 thru 4 U-bends that would not allow a .720 probe to pass. A .700 or .680 probe was used for the U-bend inspections only.

A total of 281 tubes were removed from service following this inspection due to the eddy current results. The attached reports for each steam generator detail the indications found in each tube and whether or not the tube was plugged.

STEAM GENERATOR INSERVICE INSPECTION REPORTS

STEAM GENERATOR NUMBER 11

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
8	2	C	1%	1	0.0	NO
8	3	C	35%	2	0.0	NO
10	3	C	38%	1	0.0	NO
13	3	C	12%	1	0.0	NO
12	4	H	DI	TE	20.5	YES
17	5	C	16%	2	0.0	NO
16	6	H	DI	TE	20.4	YES
17	7	C	12%	1	0.0	NO
15	10	H	DI	TE	20.6	YES
17	10	H	34%	TE	20.1	YES
16	12	H	DI	TE	20.4	YES
21	12	H	DI	TE	20.3	YES
25	12	H	DI	TE	20.2	YES
23	13	H	DI	TE	20.4	YES
7	14	H	DI	TE	20.8	YES
15	14	H	DI	TE	20.4	YES
28	15	H	DI	TE	20.4	YES
5	16	H	DI	TE	20.9	YES
22	16	H	DI	TE	20.6	YES
17	17	H	DI	TE	20.6	YES
27	17	H	DI	TE	20.6	YES
32	17	C	23%	2	0.0	NO
33	17	C	45%	2	0.0	YES

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
28	18	H	DI	TE	20.7	YES
35	18	C	34%	2	0.0	NO
35	18	C	DI	1	0.0	NO
5	20	H	DRI	TE	2.1	YES
6	20	H	DRI	TE	2.0	YES
7	20	H	DRI	TE	2.1	YES
23	20	H	DI	TE	20.5	YES
36	20	C	28%	2	0.0	NO
37	20	C	19%	1	0.0	NO
4	21	H	79%	TE	20.6	YES
37	21	C	28%	2	0.0	NO
38	21	C	22%	2	0.0	NO
2	22	H	DRI	TE	2.2	YES
26	22	H	DI	TE	20.3	YES
29	22	H	DI	TE	20.7	YES
38	22	C	19%	2	0.0	NO
4	23	H	DRI	TE	2.3	YES
22	23	H	18%	TE	20.8	YES
35	23	H	DI	TE	20.8	YES
36	23	H	DI	TE	20.7	YES
40	24	H	DI	TE	20.5	YES
7	25	H	DRI	TE	2.4	YES
40	25	C	31%	2	0.0	NO
39	26	H	DI	TE	20.5	YES

STEAM GENERATOR NUMBER 11

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
16	27	H	DRI	TE	2.4	YES
19	27	H	DRI	TE	2.4	YES
38	29	H	DI	TE	20.8	YES
1	30	H	DRI	TE	2.4	YES
10	30	H	DI	1	0.0	NO
12	30	H	DI	1	0.0	NO
30	30	H	DI	TE	20.6	YES
43	30	C	41%	2	0.0	YES
9	31	H	DRI	TE	2.4	YES
26	31	H	DI	TE	20.2	YES
42	31	C	22%	2	0.0	NO
43	31	C	31%	1	0.0	NO
23	32	H	DI	TE	20.1	YES
43	32	C	DI	1	0.0	NO
2	33	H	DI	TS	0.0	YES
4	33	H	DRI	TE	2.3	YES
16	33	H	DRI	TE	2.1	YES
9	35	H	DRI	TE	2.4	YES
33	35	H	DI	TE	18.8	YES
12	36	H	DRI	TE	2.3	YES
15	36	H	DRI	TE	2.2	YES
44	36	C	48%	2	0.0	YES
11	38	H	DI	TE	20.4	YES

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
18	38	H	DI	TE	20.7	YES
2	41	H	DRI	TE	2.4	YES
5	42	H	DRI	TE	2.3	YES
15	42	H	DRI	TE	2.3	YES
20	44	H	DI	TE	20.8	YES
20	45	H	DI	TE	20.2	YES
38	45	H	DI	TE	20.4	YES
6	47	H	DI	TE	20.6	YES
13	47	H	57%	TS	0.7	YES
16	48	H	DI	TS	2.1	YES
41	48	U	25%	AV4	0.0	NO
1	49	H	DRI	TE	2.3	YES
14	50	H	DI	TS	0.7	YES
19	51	H	DI	TE	20.6	YES
19	51	H	67%	TE	20.1	YES
1	52	H	DRI	TE	2.2	YES
10	52	H	DRI	TE	2.1	YES
13	52	H	DRI	TE	2.3	YES
34	52	H	DI	TE	20.7	YES
34	52	H	DI	1	0.0	YES
45	52	C	27%	2	0.0	NO
18	54	H	DI	TS	0.8	YES
19	55	H	DI	TS	1.0	YES

STEAM GENERATOR INSERVICE INSPECTION REPORTS

STEAM GENERATOR NUMBER 11

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
35	55	H	DRI	TE	2.1	YES
43	56	C	DI	2	0.0	NO
45	56	C	DI	3	0.0	NO
45	56	C	DI	2	0.0	NO
17	57	H	29%	TS	1.6	YES
45	57	C	35%	3	0.0	NO
13	58	H	DRI	TE	1.1	YES
18	58	H	35%	TS	1.4	YES
43	58	C	34%	2	0.0	NO
44	59	C	33%	2	0.0	NO
45	59	C	29%	2	0.0	NO
1	61	H	DRI	TE	2.1	YES
13	61	H	DRI	TE	2.2	YES
14	61	H	DRI	TE	2.2	YES
29	61	H	DI	1	0.0	NO
13	62	H	DRI	TE	2.2	YES
39	62	H	DI	1	0.0	NO
23	63	U	20%	AV2	0.0	NO
23	63	U	19%	AV3	0.0	NO
36	63	U	25%	AV3	0.0	NO
4	64	H	DRI	TE	2.2	YES
20	64	H	DI	TE	20.5	YES
1	65	H	DRI	TE	2.2	YES
19	65	H	DI	TE	20.9	YES
26	65	H	DI	1	0.0	NO
43	65	C	32%	1	0.0	NO

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
2	66	H	DRI	TE	2.2	YES
20	67	H	DI	TE	20.3	YES
4	68	H	DRI	TE	2.4	YES
15	68	H	DRI	TE	2.2	YES
40	70	C	14%	2	0.0	NO
1	72	H	DI	1	0.0	NO
1	74	H	DI	1	0.0	NO
14	75	H	DRI	TE	2.2	YES
29	75	H	DI	TE	20.7	YES
5	76	H	DI	TE	20.4	YES
32	76	H	DI	1	0.0	NO
36	76	C	28%	1	0.0	NO
32	77	C	DI	1	0.0	NO
35	77	C	16%	1	0.0	NO
14	78	H	DI	TE	20.9	YES
14	78	H	DRI	TE	2.2	YES
22	78	H	DI	TE	20.7	YES
31	79	C	< 20%	1	0.0	NO
30	80	C	22%	1	0.0	NO
30	81	C	26%	1	0.0	NO
19	83	H	DI	1	0.0	NO
28	83	H	DI	1	0.0	NO
27	84	C	27%	3	0.0	NO
27	84	C	28%	2	0.0	NO
10	85	H	DI	TE	21.0	YES

STEAM GENERATOR NUMBER 11

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED	ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
10	86	H	DI	1	0.0	NO							
4	87	H	DI	1	0.0	NO							
14	87	H	DI	1	0.0	NO							
9	88	H	DI	1	0.0	NO							
2	89	H	DI	1	0.0	NO							
10	90	H	DI	1	0.0	NO							
15	91	C	24%	1	0.0	NO							
10	92	H	DI	2	0.0	NO							
9	93	C	17%	1	0.0	NO							
11	93	C	17%	1	0.0	NO							

STEAM GENERATOR INSERVICE INSPECTION REPORTS

STEAM GENERATOR NUMBER 12

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
4	1	H	DI	TE	20.9	YES
9	2	C	39%	1	0.0	NO
7	3	C	35%	1	0.0	NO
13	3	C	10%	1	0.0	NO
14	3	C	69%	1	0.0	YES
13	4	C	19%	1	0.0	NO
14	4	C	26%	1	0.0	NO
15	4	C	DI	1	0.0	NO
8	5	H	DI	1	0.0	NO
16	5	C	16%	2.	0.0	NO
8	6	H	DI	3	0.0	NO
8	7	H	DI	2	0.0	NO
9	7	H	DI	1	0.0	NO
16	7	H	DI	TE	20.3	YES
12	9	H	DI	1	0.0	NO
15	10	H	DI	TE	20.3	YES
4	12	H	DI	1	0.0	NO
8	13	H	DI	1	0.0	NO
23	13	H	DI	1	0.0	NO
2	14	H	DI	1	0.0	NO
12	14	H	DI	1	0.0	NO
23	14	H	DI	1	0.0	NO
8	15	H	DI	1	0.0	NO

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
17	15	H	DI	TE	20.8	YES
30	15	C	DI	2	0.0	NO
30	15	C	DI	1	0.0	NO
18	16	H	DI	1	0.0	NO
31	16	H	37%	TS	1.8	YES
34	16	C	51%	1	0.0	YES
16	19	H	DI	1	0.0	NO
5	20	H	DRI	TE	2.1	YES
8	20	H	DI	2	0.0	NO
15	24	H	DRI	TE	2.1	YES
20	24	H	DI	1	0.0	NO
21	24	H	DRI	TE	2.2	YES
7	28	H	DRI	TE	2.2	YES
16	29	H	DRI	TE	2.1	YES
23	29	H	DI	TE	21.2	YES
18	31	H	DI	TS	0.8	YES
35	31	U	11%	AV1	0.0	NO
35	31	U	24%	AV2	0.0	NO
35	31	U	34%	AV3	0.0	NO
43	33	C	DI	1	0.0	NO
17	34	H	21%	TS	1.1	YES
20	36	H	DI	TS	0.5	YES
43	36	H	DI	TE	20.7	YES
12	37	H	DI	2	0.0	NO
16	37	H	DI	TS	0.7	YES

STEAM GENERATOR INSERVICE INSPECTION REPORTS

STEAM GENERATOR NUMBER 12

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
44	37	C	31%	2	0.0	NO
15	39	H	76%	TS	1.2	YES
17	39	H	90%	TE	3.4	YES
13	41	H	DRI	TE	2.1	YES
43	41	U	32%	AV3	0.0	NO
43	41	U	28%	AV4	0.0	NO
45	42	C	59%	2	0.0	YES
45	42	C	45%	1	0.0	YES
15	43	H	DRI	TE	2.1	YES
20	43	H	DRI	TE	2.0	YES
21	43	H	DRI	TE	2.3	YES
38	43	H	DI	1	0.0	NO
20	44	H	DRI	TE	2.5	YES
35	44	U	21%	AV1	0.0	NO
35	44	U	13%	AV2	0.0	NO
14	46	H	37%	TS	1.4	YES
17	46	H	DI	TS	1.0	YES
43	47	U	12%	AV1	0.0	NO
13	48	H	DI	TS	1.0	YES
15	49	H	DRI	TE	2.1	YES
16	49	H	35%	TS	0.7	YES
34	54	U	18%	AV4	0.0	NO
37	54	U	33%	AV4	0.0	NO
3	55	H	DI	1	0.0	NO
14	55	H	DRI	TE	2.3	YES
36	55	U	30%	AV1	0.0	NO
36	55	U	21%	AV2	0.0	NO
43	60	C	37%	1	0.0	NO
44	62	C	DI	2	0.0	NO
42	63	U	11%	AV3	0.0	NO
42	63	U	23%	AV4	0.0	NO
18	65	H	DI	1	0.0	NO
19	66	H	DI	TE	21.0	YES
41	66	C	DI	2	0.0	NO
41	66	C	DI	1	0.0	NO
25	74	H	DI	TE	20.9	YES
36	75	C	29%	2	0.0	NO
37	75	C	DI	2	0.0	NO
37	76	C	14%	2	0.0	NO
34	77	C	46%	2	0.0	YES
36	77	C	39%	2	0.0	NO
14	78	H	DI	TE	20.8	YES
28	78	H	DI	1	0.0	NO
35	78	C	6%	1	0.0	NO
31	79	C	24%	2	0.0	NO
31	79	C	26%	1	0.0	NO
29	82	C	DI	2	0.0	NO
30	82	C	26%	2	0.0	NO
26	85	C	41%	2	0.0	YES
24	86	C	DI	2	0.0	NO

STEAM GENERATOR INSERVICE INSPECTION REPORTS

STEAM GENERATOR NUMBER 12

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED	ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
7	89	H	DI	2	0.0	NO							
13	89	H	DI	TE	20.7	YES							
16	89	H	DI	2	0.0	NO							
19	89	C	DI	1	0.0	NO							
6	90	H	DRI	TE	2.1	YES							
14	90	C	8%	3	0.0	NO							
6	91	C	11%	1	0.0	NO							
9	91	C	3%	1	0.0	NO							
10	91	C	26%	1	0.0	NO							
11	91	C	23%	1	0.0	NO							
7	93	C	29%	1	0.0	NO							
9	93	C	40%	1	0.0	YES							

STEAM GENERATOR INSERVICE INSPECTION REPORTS

STEAM GENERATOR NUMBER 13

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
12	2	C	29%	2	0.0	NO
5	3	H	DI	TE	20.2	YES
10	3	C	2%	2	0.0	NO
14	3	C	13%	2	0.0	NO
15	5	C	13%	1	0.0	NO
10	6	H	DI	TE	20.5	YES
20	6	C	7%	1	0.2	NO
25	9	H	DI	TE	20.2	YES
26	9	U	15%	AV3	0.0	NO
26	11	H	DI	TE	20.3	YES
4	12	H	DI	TE	20.5	YES
6	12	H	DI	1	0.0	NO
28	12	C	< 20%	2	0.0	NO
30	13	H	DI	TE	20.3	YES
4	17	H	DI	TE	20.7	YES
25	17	H	DRI	TE	2.3	YES
34	17	C	44%	2	0.0	YES
6	19	H	DRI	TE	2.0	YES
10	19	H	DI	2	0.0	NO
34	19	C	19%	2	0.0	NO
28	20	H	DRI	TE	2.5	YES
35	21	C	24%	2	0.0	NO
3	22	H	DRI	TE	2.2	YES
5	22	H	DRI	TE	2.2	YES
24	23	H	DRI	TE	2.2	YES
3	24	H	DRI	TE	2.2	YES
7	24	H	DRI	TE	2.2	YES
9	24	H	DRI	TE	2.3	YES
3	25	H	DRI	TE	2.2	YES
12	25	H	DRI	TE	2.2	YES
14	25	H	DRI	TE	2.2	YES
22	25	H	DI	TE	20.5	YES
25	25	H	DRI	TE	2.2	YES
28	25	H	DRI	TE	2.1	YES
29	25	H	DRI	TE	2.2	YES
30	25	H	DI	TE	20.6	YES
2	26	H	DRI	TE	2.4	YES
12	26	H	DRI	TE	2.2	YES
4	27	H	DRI	TE	2.2	YES
4	27	H	DI	TE	20.5	YES
11	27	H	DRI	TE	2.2	YES
29	27	H	DRI	TE	2.2	YES
4	28	H	DRI	TE	2.2	YES
38	28	U	12%	AV2	0.0	NO
38	28	U	19%	AV3	0.0	NO
38	28	U	13%	AV4	0.0	NO
2	29	H	DRI	TE	2.3	YES
17	31	H	DI	TS	0.0	YES
29	32	H	DRI	TE	2.4	YES

STEAM GENERATOR NUMBER 13

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
2	34	H	DRI	TE	2.4	YES
2	34	H	DI	TE	21.1	YES
44	34	C	22%	2	0.0	NO
3	35	H	DRI	TE	2.2	YES
9	35	H	DRI	TE	2.2	YES
24	36	H	DRI	TE	2.2	YES
25	36	H	DRI	TE	2.2	YES
44	36	C	25%	2	0.0	NO
8	38	H	DRI	TE	2.3	YES
11	38	H	DRI	TE	2.4	YES
23	38	H	DRI	TE	2.3	YES
27	38	H	DRI	TE	2.2	YES
30	38	H	DRI	TE	2.4	YES
41	38	U	24%	AV4	0.0	NO
11	39	H	DRI	TE	2.3	YES
24	39	H	DRI	TE	2.3	YES
4	40	H	DRI	TE	2.3	YES
30	40	H	DRI	TE	2.4	YES
4	41	H	DRI	TE	2.2	YES
23	41	H	DRI	TE	2.1	YES
26	41	H	DRI	TE	2.2	YES
27	41	H	DRI	TE	2.2	YES
36	41	U	18%	AV1	0.0	NO
36	41	U	34%	AV2	0.0	NO
36	41	U	35%	AV3	0.0	NO

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
26	42	H	DRI	TE	2.1	YES
38	42	U	23%	AV3	0.0	NO
38	42	U	26%	AV4	0.0	NO
9	43	H	DRI	TE	2.1	YES
24	44	H	DRI	TE	2.2	YES
2	45	H	DRI	TE	2.2	YES
4	45	H	DRI	TE	2.3	YES
2	47	H	DRI	TE	2.3	YES
23	47	H	DRI	TE	2.5	YES
24	47	H	DRI	TE	2.1	YES
39	47	U	18%	AV3	0.0	NO
39	47	U	16%	AV4	0.0	NO
38	48	U	17%	AV3	0.0	NO
38	48	U	28%	AV4	0.0	NO
42	48	U	13%	AV1	0.0	NO
42	48	U	16%	AV2	0.0	NO
21	49	H	DRI	TE	2.2	YES
31	49	H	DRI	TE	2.2	YES
33	49	U	12%	AV3	0.0	NO
38	49	U	< 20%	AV1	0.0	NO
2	50	H	DRI	TE	2.3	YES
4	50	H	DRI	TE	2.3	YES
40	50	U	26%	AV3	0.0	NO
40	50	U	19%	AV4	0.0	NO
41	50	U	28%	AV3	0.0	NO
41	50	U	22%	AV4	0.0	NO
44	50	C	< 20%	TE	9.4	NO

STEAM GENERATOR NUMBER 13

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
9	51	H	DRI	TE	2.3	YES
12	51	H	DRI	TE	2.1	YES
27	51	H	DRI	TE	2.0	YES
33	52	U	16%	AV1	0.0	NO
33	52	U	25%	AV2	0.0	NO
33	52	U	33%	AV3	0.0	NO
33	52	U	16%	AV4	0.0	NO
7	53	H	DRI	TE	2.3	YES
11	53	H	DI	TS	0.1	YES
15	53	H	DI	TS	0.9	YES
16	53	H	DI	TS	2.2	YES
19	53	H	DI	TS	1.0	YES
36	53	U	25%	AV3	0.0	NO
36	53	U	13%	AV4	0.0	NO
15	54	H	DI	TS	2.6	YES
15	55	H	DI	1	0.0	NO
40	55	U	22%	AV3	0.0	NO
40	55	U	19%	AV4	0.0	NO
23	56	H	DRI	TE	2.7	YES
27	56	H	DRI	TE	2.2	YES
36	56	U	34%	AV1	0.0	NO
36	56	U	30%	AV2	0.0	NO
37	56	U	29%	AV1	0.0	NO
37	56	U	24%	AV2	0.0	NO
37	56	U	27%	AV3	0.0	NO
42	56	U	11%	AV1	0.0	YES
42	56	U	26%	AV2	0.0	YES
42	56	U	42%	AV3	0.0	YES
42	56	U	21%	AV4	0.0	YES
6	57	H	DRI	TE	2.4	YES

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
25	57	H	DRI	TE	2.3	YES
36	57	U	26%	AV3	0.0	NO
36	57	U	29%	AV4	0.0	NO
6	58	H	DI	1	0.0	NO
10	58	H	DRI	TE	2.1	YES
23	58	H	DRI	TE	2.1	YES
39	58	U	19%	AV4	0.0	NO
41	58	U	33%	AV2	0.0	NO
41	58	U	35%	AV3	0.0	NO
41	58	U	23%	AV4	0.0	NO
10	59	H	DRI	TE	2.2	YES
4	60	H	DI	TE	20.9	YES
4	60	H	DRI	TE	2.3	YES
34	61	U	12%	AV3	0.0	NO
34	61	U	22%	AV4	0.0	NO
36	61	U	25%	AV3	0.0	NO
36	61	U	31%	AV4	0.0	NO
41	61	U	17%	AV3	0.0	NO
32	62	U	23%	AV3	0.0	NO
40	62	U	14%	AV1	0.0	NO
40	62	U	29%	AV2	0.0	NO
40	62	U	23%	AV3	0.0	NO
40	62	U	22%	AV4	0.0	NO
33	64	U	22%	AV1	0.0	NO
33	64	U	13%	AV2	0.0	NO
36	64	U	18%	AV3	0.0	NO
36	64	U	16%	AV4	0.0	NO
40	64	U	24%	AV1	0.0	NO
18	65	H	DI	1	0.0	NO
36	65	U	15%	AV3	0.0	NO
36	65	U	23%	AV4	0.0	NO

STEAM GENERATOR NUMBER 13

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED	ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
39	65	U	16%	AV3	0.0	NO							
39	65	U	24%	AV4	0.0	NO							
40	65	U	15%	AV2	0.0	NO							
40	65	U	23%	AV3	0.0	NO							
2	66	H	DRI	TE	2.2	YES							
39	66	H	DI	1	0.0	NO							
3	67	H	DRI	TE	2.2	YES							
3	69	H	DRI	TE	2.2	YES							
3	70	H	DRI	TE	2.1	YES							
29	70	H	DI	TE	20.7	YES							
39	70	C	4%	4	0.0	NO							
39	70	C	21%	3	0.0	NO							
31	71	H	DI	TE	20.7	YES							
39	73	C	DI	1	0.0	NO							
4	75	H	DRI	TE	1.9	YES							
9	76	H	DRI	TE	2.2	YES							
36	76	U	21%	AV3	0.0	NO							
4	77	H	DRI	TE	2.1	YES							
32	78	C	DI	1	0.0	NO							
4	80	H	DRI	TE	1.6	YES							
3	82	H	54%	1	0.0	YES							
10	83	H	DI	2	0.0	NO							
10	86	H	DI	1	0.0	NO							
10	92	C	4%	1	0.0	NO							
10	93	C	16%	1	0.0	NO							
5	94	C	13%	1	0.0	NO							

STEAM GENERATOR INSERVICE INSPECTION REPORTS

STEAM GENERATOR NUMBER 14

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
7	1	C	44%	1	0.0	YES
8	2	C	42%	1	0.0	YES
6	3	H	DI	4	0.0	NO
6	4	H	DI	4	0.0	NO
15	5	C	DI	1	0.0	NO
19	6	C	11%	1	0.0	NO
11	7	H	DI	1	0.0	NO
20	7	C	DI	1	0.0	NO
21	7	C	19%	1	0.0	NO
11	9	H	DI	1	0.0	NO
25	9	C	28%	1	0.0	NO
22	10	H	DI	TE	20.8	YES
27	10	C	22%	1	0.0	NO
13	11	H	77%	TE	20.7	YES
11	12	H	DI	TE	20.6	YES
21	12	H	15%	TS	34.8	NO
7	13	H	DRI	TE	2.2	YES
19	13	H	DI	TE	20.6	YES
28	13	H	DI	1	0.0	NO
30	13	H	DI	1	0.0	NO
20	14	H	DI	TE	20.5	YES
20	14	H	DRI	TE	2.1	YES
28	14	H	DI	TE	20.6	YES
31	14	C	17%	1	0.0	NO
31	15	C	23%	1	0.0	NO
28	16	H	DI	TE	20.8	YES
28	16	H	DI	1	0.0	YES
34	17	C	33%	1	0.0	NO
8	18	H	DI	3	0.0	NO
22	18	H	DI	TE	20.6	YES
2	19	H	DI	2	0.0	NO
14	19	H	DRI	TE	2.1	YES
9	20	H	DI	1	-.1	NO
17	20	H	DI	TE	20.9	YES
37	21	C	28%	1	0.0	NO
11	22	H	DI	1	0.0	NO
11	22	H	DI	3	0.0	NO
14	22	H	DRI	TE	2.2	YES
21	22	H	45%	1	0.0	YES
28	22	H	DI	1	0.0	NO
28	22	H	DI	2	0.0	NO
1	23	H	DI	TE	20.9	YES
19	26	H	DRI	TE	2.3	YES
13	27	H	DI	TE	21.1	YES
13	27	H	DRI	TE	2.3	YES
9	29	H	DI	1	0.0	NO
5	30	H	DI	2	0.0	NO
5	30	H	DI	3	0.0	NO
42	30	C	29%	1	0.0	NO
41	31	C	8%	1	0.0	NO
42	32	C	8%	1	0.0	NO

STEAM GENERATOR INSERVICE INSPECTION REPORTS

STEAM GENERATOR NUMBER 14

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
40	33	H	DI	TE	21.0	YES
14	34	H	DI	TS	1.2	YES
44	34	C	39%	1	0.0	NO
44	35	C	14%	1	0.0	NO
19	36	H	DRI	TE	2.4	YES
45	36	C	17%	1	0.0	NO
15	37	H	DRI	TE	2.3	YES
16	37	H	DRI	TE	2.3	YES
17	37	H	DI	1	0.0	NO
29	37	H	DI	1	0.0	NO
32	37	H	DI	1	0.0	NO
38	37	H	DI	2	0.0	NO
44	37	C	33%	2	0.0	NO
15	38	H	DI	TS	1.0	YES
14	39	H	DI	TS	0.8	YES
14	39	H	36%	TS	1.6	YES
16	39	H	DRI	TE	2.3	YES
17	39	H	22%	TS	1.2	YES
44	39	C	DI	4	0.0	NO
20	40	H	25%	TS	0.5	YES

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
42	40	H	DI	2	0.0	NO
12	41	H	DRI	TE	2.3	YES
38	41	H	DI	1	0.0	NO
41	41	H	DI	TE	20.5	YES
42	41	H	DI	1	0.0	NO
45	41	C	13%	2	0.0	NO
46	41	C	15%	2	0.0	NO
14	42	H	29%	TS	1.3	YES
17	42	H	DI	TS	1.0	YES
42	42	H	DI	TE	20.7	YES
16	43	H	DI	TS	1.5	YES
17	43	H	DI	TS	1.4	YES
44	43	C	DI	2	0.0	NO
46	43	C	18%	2	0.0	NO
16	44	H	24%	TS	1.7	YES
29	44	H	DI	1	0.0	NO
38	44	H	DI	TE	20.6	YES
39	44	H	DI	TE	20.5	YES
36	45	H	DI	TE	20.6	YES
38	45	H	DI	TE	20.8	YES
41	45	U	21%	AV3	0.0	NO
41	45	U	15%	AV4	0.0	NO

STEAM GENERATOR NUMBER 14

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
33	46	U	24%	AV2	0.0	NO
33	46	U	32%	AV3	0.0	NO
33	46	U	24%	AV4	0.0	NO
46	46	H	DI	1	0.0	NO
22	47	H	DI	TS	0.2	YES
38	48	H	9%	TS	42.9	NO
13	50	H	DI	TE	21.0	YES
18	50	H	DRI	TE	2.2	YES
28	50	H	DI	TE	21.0	YES
28	52	H	DI	TE	21.1	YES
19	53	H	DRI	TE	2.2	YES
19	54	H	DRI	TE	2.4	YES
46	54	H	DI	4	0.0	NO
44	55	H	DI	2	0.0	NO
34	60	H	DI	TE	20.7	YES
44	60	C	13%	3	0.0	NO
43	61	C	28%	1	0.0	NO
13	62	H	DI	1	0.0	NO
19	62	H	DRI	TE	2.2	YES
19	63	U	27%	AV2	0.0	NO
19	63	U	17%	AV3	0.0	NO
19	63	U	19%	AV4	0.0	NO
26	63	H	DI	1	0.0	NO
42	63	C	46%	2	0.0	YES
42	63	C	17%	1	0.0	YES
16	65	H	DRI	TE	2.3	YES

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
41	65	C	33%	2	0.0	NO
19	66	U	22%	AV1	0.0	NO
19	66	U	26%	AV2	0.0	NO
19	66	U	29%	AV3	0.0	NO
19	66	U	14%	AV4	0.0	NO
26	69	U	13%	AV1	0.0	NO
26	69	U	14%	AV2	0.0	NO
20	70	H	DI	TE	20.8	YES
39	73	C	5%	2	0.0	NO
4	74	H	DI	2	0.0	NO
21	74	H	DRI	TE	2.0	YES
13	75	H	DI	1	0.0	NO
35	75	C	18%	2	0.0	NO
36	75	C	9%	2	0.0	NO
4	76	H	DI	2	0.0	NO
36	76	C	DI	1	0.0	NO
23	78	H	DI	3	0.0	NO
31	79	H	DI	TE	20.9	YES
34	79	C	47%	2	0.0	YES
34	79	C	DI	1	0.0	YES
26	80	U	19%	AV3	0.0	NO
24	81	H	DI	2	0.0	NO
30	81	C	38%	2	0.0	NO
30	82	C	16%	2	0.0	NO
26	86	C	36%	2	0.0	NO
5	87	H	DI	2	0.0	NO

STEAM GENERATOR INSERVICE INSPECTION REPORTS

STEAM GENERATOR NUMBER 14

ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED	ROW	COL	LEG	INDIC	LOC	INCH	PLUGGED
23	87	C	25%	2	0.0	NO							
16	88	H	DI	TE	20.9	YES							
7	92	C	4%	1	0.0	NO							
12	93	C	29%	1	0.0	NO							
4	94	H	DRI	TE	2.0	YES							

IRRADIATED FUEL EXAMINATIONS

There were no irradiated fuel examinations for either Unit 1 or Unit 2 of the Donald C. Cook Nuclear Plant during 1989.

CHANGES TO PROCEDURES

This section contains a brief description of the procedure changes implemented under the provisions of 10CFR50.59 and the associated safety evaluations.

Emergency Plan Implementing Procedures

1. Procedure No. PMP 2081.EPP.304, Revision 3
   "Offsite Dose Projection"
   Revision 3 to this procedure involved changes to the offsite doses projection calculations necessitated by the installation of a new meteorological monitoring system.

   The technical evaluation concluded that the system has been tested and calibrated in accordance with the appropriate procedures. The new meteorological monitoring system was designed to meet an NRC commitment and has been reviewed and approved by the NRC. The safety evaluation concluded that the changes implemented under Revision 3 do not constitute an unreviewed safety question as defined in 10CRF50.59.

2. Procedure No. PMP 2081.EPP.305, Revision 2,
   "Protective Action Recommendations" - Change Sheet No. 1

   The procedure was revised to accommodate a change to the dose assessment program (DAP) so that it can be run on a personal computer. All of DAP's calculation functions have been retained on the PC-based version.

   The PC-based program has been verified and validated according to the requirements of General Procedure 2.6, "Software Quality Assurance Program for the Cook Nuclear Plant".
   Based on the technical evaluation, the safety evaluation concluded that the change implemented under Change Sheet No. 1 does not involve an unreviewed safety question as defined in 10CFR50.59.

Radiation Protection Procedure

Procedure No. 12 THP 6010.RPP.009, "Emergency Equipment Inventory"
Procedure No. 12 THP 6010.RAD.222 is being replaced by 12 THP 6010 RPP.009, "Emergency Equipment Inventory." One storage location which is mentioned in the Emergency Plan will be changed from the "I-94 Gate House" to the "Training Center Dosimetry Office".

The procedure change is not altering the type of amount of available emergency equipment. The new location is not less accessible or less convenient.

Base on the technical evaluation that the change of storage

Radiation Protection Procedure (Con't.)

location will not adversely impact the Emergency Plan, the safety evaluation concluded that the procedure change does not involve an unreviewed safety question as defined in 10CFR50.59.

TEST AND EXPERIMENTS NOT DESCRIBED IN THE FSAR

A special test was conducted on Unit 1 turbine driven, auxiliary feed pump (TDAFP) motor operated discharge valves 1-FMO-211, -221, 231, -241 to steam generator 1, 2, 3, and 4 respectively to verify the valves ability to close during abnormally high flow conditions.

Background

Following a reactor trip on Unit 1 on January 16, 1989, the TDAFP discharge valve to steam generator No. 2, 1-FMO-221, failed to close automatically to the intermediate flow retention setting and could not be closed remotely from the control room. The valve moved in the closed direction but failed to close due to the torque limit switch taking the valve out. Valve 1-FMO-221 lagged the other three discharge valves in starting to close, and the resulting high flow condition and high differential pressure are believed to be the cause for the valve failing to close. Based on the investigation findings, the torque switches were changed from a setting of 1.0 to 2.0 on all four valves. After the torque switch settings were changed, valve 1-FMO-221 was successfully stroked from full open to full closed with a 400 gpm flow established.

Special Test Description

The special test was conducted to determine if the TDAFP discharge valves to the steam generators would be able to close under the abnormally high flow rate and differential pressure conditions that valve 1-FMO-221 was subjected to during the auxiliary feedwater system transient.

The test conditions were established with the TDAFP operating at rated speed and all four discharge valves in the fully closed position. One discharge valve at a time was then cycled to the fully open position, passing the full pump discharge flow, and then returned to the fully closed position. There were no problems experienced with any of the four valves during the testing.

The tests were conducted on March 18, 1989, with the reactor coolant system in Mode 3, Hot Standby, and at rated zero power temperature and pressure conditions.

Safety Evaluation Summary

In order to determine if the special test would constitute an unreviewed safety question an engineering evaluation and safety review was performed.

The TDAFP system was to be considered "inoperable" although functional during the test. The estimated test duration was 20 minutes (actual 9 minutes).

The engineering evaluation concluded that the test parameters (flow

Safety Evaluation Summary (Con't.)

rates and differential pressures) would have no adverse effect on the valves and their motor operators.

The nuclear safety review considered the effect of the TDAFP system test on the feedwater break analysis and the transient-analysis aspect of the test with respect to an overcooling transient. The test conditions were found to be bounded by the current accident analysis, and it was, therefore, concluded that the test did not involve an unreviewed safety question as defined in 10CFR50.59.

CHALLENGES TO PRESSURIZER POWER OPERATED RELIEF VALVES AND SAFETY VALVES - 1989

There were no challenges to the pressurizer power operated relief valves (PORV's) or the pressurizer safety valves as a result of the valves being called upon to mitigate an actual overpressure condition. One event occurred on Unit 1 on June 11, 1989, that caused an actuation of pressurizer PORV NRV-153. The reactor coolant system was in Mode 5, Cold Shutdown, and the PORV's were in the low temperature over-pressure protection mode.

Event Description

During cold shutdown operation, PORV, NRV-153, opened automatically one June 11, 1989, at 1141 hours. The PORV opened after starting reactor coolant pump (RCP) No. 13 during the reactor coolant system (RCS) fill and vent procedure. There were no other RCPs running at the time. Prior to the pump start, the RCS was at approximately 350 psig and 145°F with a steam bubble in the pressurizer. The RCS pressure increased to approximately 370 psig after the pump start and the PORV opened. The PORV closed after approximately two seconds and the RCS pressure stabilized at approximately 350 psig. The highest recorded pressure during this event was approximately 370 psig, which is below the 400 psig pressurizer PORV setpoint limit of Technical Specification 3.4.9.3.

The pressurizer PORV setpoints are (385 psig ± 15) and the minimum pressure for operation of the RCPs during fill and vent operations is 325 psig (to which we add 25 psig due to instrument uncertainties). This gives an effective operating window of 20 psig. Information supplied by the RCP vendor indicates that a pressure change of 30-40 psig could be expected when starting an RCP. The problem of a very small operating window is recognized within the industry and potential resolutions are being pursued.

We are currently in the process of having the PORV setpoints recalculated by the NSSS vendor using a refined method which should result in slightly higher setpoints and a larger operating window. We are also studying the use of narrow range pressure instrumentation for cold overpressure PORV control to remove some to the instrument uncertainties and allow a higher setpoint.

CHANGES TO FACILITY
REQUEST FOR CHANGE

Brief descriptions and summary safety evaluations for design changes (RFCs) made to the facility as described in the Donald C. Cook Nuclear Plant Final Safety Analysis Report (FSAR) are presented in this section. These changes were completed pursuant to the provisions of Title 10, Code of Federal Regulations subsection 50.59 (a).

RFC-02-2810

BRIEF DESCRIPTION

RFC-DC-02-2810 provided for the upgrading of the Unit 2 incore thermocouple system to comply with the requirements of NUREG-0737 and Regulatory Guide 1.97, Revision 3.

SAFETY EVALUATION

This RFC has been classified as "safety related" because (a) the incore thermocouples may be used in the safe shutdown of the reactor by notifying the operators of an inadequate core cooling incident, and (b) the thermocouple system involves a portion of the Reactor Coolant System pressure boundary.

The RFC has been reviewed in accordance with NS&L Section Procedure No. 7, "Safety Review of Design Changes." As a result of this review, special attention was focused on the requirements of NUREG-0737 and Regulatory Guide 1.97, in addition to other plant-specific design requirements.

It is concluded that RFC DC-02-2810 does not constitute an unreviwed safety question as defined in 10 CFR 50.59, nor does it create a substantial hazard to the health and safety of the general public.

CHANGES TO FACILITY
REQUEST FOR CHANGE

RFC-01-2872

RFC-02-2873

BRIEF DESCRIPTION

The purpose of these RFCs was to install ATWS Mitigation System Actuation Circuitry (AMSAC) in Unit 1 and 2. The AMSAC provides an independent means of tripping the turbine and initiating auxiliary feedwater in the unlikely event of a common mode failure within the reactor protection system (RPS). This design incorporates a 25-30 second delay on initiation of AMSAC, based on 50% or less feedwater flow to 3 out of 4 steam generators with the reactor power greater than 70%. At this point AMSAC trips the turbine, initiates the motor-driven and turbine-driven auxiliary feedpumps and conserves auxiliary feedwater. This is a generic design by Westinghouse (Ref. WOG 85-1-3 of January 23, 1985) and has an NRC Safety Evaluation Report dated September 29, 1985. The reason for the change is NRC approval of 10 CFR 50.62, regarding the reduction of risk from anticipated transients without scram (ATWS).

The AMSAC circuitry (Isolators, bistables, logic circuitry, output relays) is powered from the N-train battery. The equipment (bistables, inverter, logic relays, output relays) is housed in a new cabinet located in the control room. Existing sensors/transmitters be used to sense the feedwater flow and the 70% power level (turbine first-stage pressure).

The following alarms/indications are provided in the control room: AMSAC enabled, AMSAC in bypass/test, AMSAC disabled, AMSAC control bus unavailable, AMSAC test successful, and AMSAC initiated.

The control of AMSAC (bypass/test, disabling, enabling, manual initiation, etc.) will be performed from the control room.

Isolation from the N-train battery is accomplished through a qualified 1E breaker.
The output relays will serve the following function (when initiated): 1) trip the turbine, 2) start the turbine-driven auxiliary feedwater pump and two motor-driven auxiliary feedwater pumps and 3) conserve auxiliary feedwater.

SAFETY EVALUATION

These RFCs were classified as safety-related, because some of the components are required to be Class 1E due to their interaction with other Class I components.

It is of particular note that AMSAC utilized existing sensors from other safety-related systems, and this was carefully reviewed with Electrical Engineering to assure the appropriate safety/non-safety interfaces were maintained. We also reviewed the conceptual functionality of the AMSAC system and found it is electrically compatible with the requirements of the WOG document approved by the NRC.

Nuclear Safety and Licensing has reviewed the proposed change as per the review criteria in NS&L Procedure No. 7. As a result of the review, there were no unresolved safety questions.

It is concluded that the RFCs do not constitute an unreviewed safety question as defined in 10 CFR 50.59, nor do they create a substantial hazard to the health and safety of the public.

RFC-12-2900 Subtask B.01

BRIEF DESCRIPTION

The purpose of this Subtask was to upgrade the Nuclear Instrumentation System (NIS) to meet the guidance provided by R.G. 1.97,

Revision 3. R.G. 1.97, Revision 3 lists Neutron Flux as a Type B, Category 1 variable with a range of 10^{-6}% to 100% full power. A commitment was made to the NRC to provide neutron flux instrumentation that meets R.G. 1.97, Rev. 3 Recommendations," dated June 29, 1987.

The neutron flux instrumentation was installed so that the performance of the Reactor Protection system (RPS) is not degraded. The scope of work for this Subtask included providing two Gamma-Metrics environmentally and seismically qualified detectors inside containment, installing a pre-amplifier just outside containment, modifying the NIS cabinets, and running cable for signals and power supplies.

SAFETY EVALUATION

This Subtask has been classified as safety-related. The instrumentation installed by this Subtask will not provide any reactor trip signals to the Reactor Protection System (RPS). However, the neutron flux indicators will be mounted in channels I and III of the safety-related NIS cabinet. Furthermore, R.G. 1.97, Revision 3 classifies Neutron Flux as a Category 1 variable whose purpose is to verify accident mitigation. Category 1 classification is intended for key variables important to safety. Consequently, the installation requirements for category 1 variables include full equipment qualification, class 1E power supplies, redundancy, and quality assurance program.

Nuclear Safety and Licensing has reviewed the proposed change in accordance with the instructions in NS&L Procedure No. QP-7, "Safety Review of Design Changes." As a result of the review process. NS&L concludes that this RFC does not involve a change in the T/S, does not constitute and unreviewed safety question as defined in 10CFR50.59, "Changes, Tests and Experiments," Section (a) (2), and does not create a substantial hazard to the health and safety of the public.

CHANGES TO FACILITY
REQUEST FOR CHANGE

RFC-12-2900 B.14 Rev. 1

BRIEF DESCRIPTION

The purpose of this revision was to expand the scope of RFC-12-2900, Subtask B.14. The original version of this RFC specified replacement of the position indication limit switches for steam generator blowdown sample valves DCR-301, -302, -303, and -304; steam generator blowdown line valves DCR-310, -320, -330, and -340; and fan cooler glycol supply valves VCR-11 and -21.

This RFC was revised to upgrade the position indication limit switches for the steam generator blowdown sample and line valves under the scope of Revision 0 while the fan cooler glycol supply valve limit switches were upgraded under the scope of Revision 1. In addition, the containment control air isolation valve (DCR-100), -101, -102, -103) position limit switches were included for upgrading under the scope of Revision 1.

Floodup tubing and feedthroughs were installed with the above valve position limit switches to meet the requirements for this plant upgrade.

SAFETY EVALUATION

This revision was classified as "safety related" for consistency with the original RFC. The installation was made to meet the requirements of Regulatory Guide 1.97, Revision 3, which classifies the valve positions of these containment isolation valves as Type B, Category 1 variables. Category 1 variables require a full equipment qualification, quality assurance program, Class IE power source, and must meet the single failure criterion.

Nuclear Safety and Licensing has reviewed the proposed revision in accordance with the instruction in NS&L Procedure No. QP-7,

"Safety Review of Design Changes." As a result of the review it is concluded the RFC DC-12-2900, Subtask B. 14, Revision 1 does not constitute and unreviewed safety question as defined in 10CFR50.95, "changes, Tests and Experiments," Section (a) (2), and that it does not create a substantial hazard to the health and safety of the public.

RFC-12-2900 B.14 Rev. 0

BRIEF DESCRIPTION

The purpose of this RFC was to install new, environmentally and seismically qualified replacement position indication limit switches on the following containment isolation valves and upgrade the corresponding isolation valve position indication lights with seismically qualified devices.

Valve Number	Function
DCR-301	Steam Generator Blowdown Sample #1
DCR-302	Steam Generator Blowdown Sample #2
DCR-303	Steam Generator Blowdown Sample #3
DCR-304	Steam Generator Blowdown Sample #4
DCR-310	Steam Generator Blowdown Lines #1
DCR-320	Steam Generator Blowdown Lines #2
DCR-330	Steam Generator Blowdown Lines #3
DCR-340	Steam Generator Blowdown Lines #4

The position indication limit switches for all steam generator blowdown isolation valves listed above are located outside containment.

SAFETY EVALUATION

Subtask B.14, of RFC DC-12-2900 was implemented to meet requirements of Revision 3 to Regulatory Guide 1.97 dated May, 1983.

The valve positions for all containment isolation valves in this RFC are classified as Type B, Category 1 variables in our submittal AEP:NRC:0773S, "Additional Information on and Requests For Deviations from Regulatory Guide 1.97, Rev. 3, page 3 states "Category 1 provides the most stringent requirements and is intended for key variables." As such, the valve position indications for these containment isolation valves had a full equipment qualification, quality assurance program and Class 1E power source, and single failure criteria was incorporated into the installation of these variables.

This RFC was classified as "safety related" since the isolation valve position indications covered are important to plant safety and the installation has many safety requirements.

The Nuclear Safety and Licensing Section (NS&L) has reviewed Subtask B.14 of RFC DC-12-2900 under the provisions of NS&L Procedure No. QP-7, "Safety Review of Design Changes." Commitment to this installation was made in AEP:NRC:07730, "June 12, 1984 Confirmatory Order - Final Status Report on R.G. 1.97 Compliance," dated October 15, 1985, and AEP:NRC:0773S. This RFC does not affect any Technical Specifications.

The review concluded that Subtask B.14, of RFC DC-12-2900 does not constitute an unreviewed safety question as defined in 10 CFR 50.59, "Changes, Tests, and Experiments," Section (a) (2), nor does it represent a substantial hazard to the health and safety of the public.

<u>RFC-12-2900</u> C.02

<u>BRIEF DESCRIPTION</u>

Subtask C.02 of RFC DC-12-2900 provided environmentally and/or seismically qualified installation for valves NRV-101, -103, NRC-105, -106, and ERV-430. These valves are part of the Post

Accident Sampling System (PASS) and provide a pathway to obtain an RCS hot leg fluid sample.

SAFETY EVALUATION

Subtask C.02 of RFC DC-12-2900 was classified "safety-related" because the change was made as an environmental and seismic qualification upgrade, and involved Class I components.

The Nuclear Safety and Licensing Section (NS&L) has reviewed Subtask C.02 of RFC DC-12-2900 under the provisions of NS&L Section Procedure No. 7, "Safety Review of Design Changes."

It was concluded that Subtask C.02 of RFC DC-12-2900 does not constitute an unreviewed safety question as defined in 10 CFR 50.59, nor does it create a substantial hazard to the health and safety of the public.

This change does not affect any Technical Specifications.

RFC-12-2900 D.02

Subtask D.02 of RFC DC-12-2900 installed new, environmentally qualified replacement thermocouples (ITI -310, 320) and cable to measure the inlet temperature to the RHR heat exchanger.

SAFETY EVALUATION

Subtask D.02 of RFC DC-12-2900 was classified "safety-interface" since the equipment interacts with the RHR system.

The Nuclear Safety and Licensing Section (NS&L) has reviewed Subtask D.02 of RFC DC-12-2900 under the provisions of NS&L Section Procedure No. 7, "Safety Review of Design Changes."
It was concluded that Subtask D.02 of RFC DC-12-2900 does not constitute an unreviewed safety question as defined in 10 CFR

50.59, nor does it create a substantial hazard to the health and safety of the public.

RFC-12-2900 D.03a & D.03b

BRIEF DESCRIPTION

Subtask D.03a of RFC DC-12-2900 installed new, environmentally qualified replacement wide range accumulator tank level transmitters ILA-110,120,130, and 140. Subtask D.03b of RFC DC-12-2900 Installed New Environmentally Qualified Replacement Accumulator Tank Pressure Transmitters IPA-110,120,130, and 140. Flood up tubes and feed throughs were installed for the above transmitters.

SAFETY EVALUATION

Subtasks D.03a and D.03b of this RFC was implemented to meet the guidance provided by R.G. 1.97, Revision 3 dated May 1983. Accumulator tank level and pressure are classified as Type D, Category 2 variables in our NRC submittal AEP:NRC:07730, "June 12, 1984 Confirmatory Order - Final Status Report on Regulatory Guide 1.97 Compliance," dated October 15, 1985. Category 2 classification requires environmental qualification and an appropriate QA program. Seismic qualification and a class 1E power supply were not required.

These RFCs were classified as "safety interface" since accumulator tank level and pressure variables have a mix of safety and non safety requirements.

Nuclear Safety and Licensing has reviewed the proposed changes in accordance with the instruction in Nuclear Safety and Licensing Section Procedure No. QP-7, "Safety Review of Design Changes." As a result of the review it is concluded that this RFC does not

constitute an unreviewed safety question as defined in 10 CFR 50.59, "Changes, Tests and Experiments," Section (a) (2), nor does it create a substantial hazard to the health and safety of the public.

RFC-12-2900 D.14

BRIEF DESCRIPTION

Subtask D.14 of RFC DC 12-2900 replaced the pressurizer relief tank temperature RTD with a device having a wider range and changed the scale on the indicator (NTA-351), and recalibrated any associated instrumentation to reflect the wider range.

SAFETY EVALUATION

Subtask D.14 of RFC DC-12-2900 was classified as "safety interface" since the system handles radioactive liquid.
The Nuclear Safety and Licensing Section (NS&L) has reviewed Subtask D.14 of RFC DC-12-2900 under the provisions of NS&L Section Procedure No. 7, "Safety Review of Design Changes."

It was concluded that Subtask D.14 of RFC DC-12-2900 does not constitute an unreviewed safety question as defined in 10 CFR 50.59, nor does it create a substantial hazard to the health and safety of the public.

This change does not affect any Technical Specifications.

RFC-12-2900 D.19

BRIEF DESCRIPTION

Subtask D.19 of RFC DC-12-2900 installed a second, seismically qualified Class IE channel to measure condensate storage tank

level. The control room indicator for the existing channel, CLI-113, was relocated to satisfy human factors criteria.

SAFETY EVALUATION

Subtask D.19 of RFC DC-12-2900 was classified as "safety related" since the equipment was installed as Class IE.

The Nuclear Safety and Licensing Section (NS&L) has reviewed Subtask D.19 of RFC DC-12-2900 under the provisions of NS&L Section Procedure No. 7, "Safety Review of Design Changes."

It is concluded that Subtask D.19 of RFC DC-12-2900 does not constitute an unreviewed safety question as defined in 10 CFR 50.59, nor does it create a substantial hazard to the health and safety of the public.

This change does not affect any Technical Specifications.

RFC-12-2900 D.22

BRIEF DESCRIPTION

The purpose of this RFC was to upgrade six of the seventeen containment atmosphere temperature RTDs to meet R.G. 1.97, Rev. 3, requirements. Commitment for this change was made in AEP:NRC:07790, "June 12, 1984 Confirmatory Order - Final Status Report on Regulatory Guide 1.97 Compliance," dated October 15, 1985. Specifically, we agreed to replace six containment atmosphere temperature RTDs (ETR-12, 14, 18, 20, 21, and 22) with environmentally qualified equipment having a temperature range of 0 to 400^{O}F. In addition, this RFC expanded the range of all seventeen containment atmosphere temperature RTDs by replacing them with wider range instruments.

SAFETY EVALUATION

This RFC was classified as safety-interface. R.G. 1.97, Rev. 3 classifies containment atmosphere temperature as a Type D, Category 2 variable. Instrumentation for this type of variable provides information indicating the operation of individual safety systems and other systems important to safety but redundancy, Class 1E power supplies, and seismic qualification are not required for these instruments. Containment atmosphere temperature serves no function associated with reactor trip or ECCS actuation, and failure of the RTDs will not result in an increase in the severity of an accident or prevent proper functioning of safety systems.

Nuclear Safety and Licensing has reviewed the proposed change in accordance with the instructions in NS&L Procedure OP-7. As a result of this review, it is concluded that Subtask D.22 of RFC DC-12-2900 does not constitute an unreviewed safety question as defined in 10 CFR 50.59, "Changes, Tests and Experiments," Section (a) (2), nor does it create a substantial hazard to the health and safety of the public.

This RFC does not require a revision of the Technical Specifications.

RFC-12-2900 D.23

BRIEF DESCRIPTION

RFC DC-12-2900, Subtask D.23 installed new, environmentally-qualified, replacement resistance temperature detectors (RTDs) (ITR-311 and 321) and resistance-to-current (R/I) converters, that measure the inlet temperature to the residual heat removal (RHR) heat exchangers. This change is in response to the recommendations contained in Regulatory Guide 1.97, Revision 3.

Because environmentally-qualified RTDs of the same sensor metal as the original RTDs were not available, new R/I converters were required to match the thermal characteristics of the new RTDs.

SAFETY EVALUATION

Subtask D.23 of RFC DC-12-2900 was classified as safety-interface because the RTDs and the R/I converters provide information to indicate the operation of the RHR system, which is a safety system. The RTDs and the R/I converters are used to determine the containment sump water temperature during the recirculation phase of a loss-of-coolant accident (LOCA) as stated in Attachment 1 of our submittal AEP:NRC:0773S dated June 29, 1987. In R.G. 1.97, Table 3, page 1.97-27, the containment sump water temperature is classified as a Type D, Category 2 variable, and as such, environmental qualification and a QA program are required.

Nuclear Safety and Licensing reviewed the proposed change in accordance with the instructions in Nuclear Safety and Licensing Section Procedure No. 7, "Safety Review of Design Changes." The following items were considered in this safety review: environmental qualification of the RTDs, hardware installation of the RTDs and the R/I converters, and cabling. The cabling is already environmentally qualified. In addition, the indication range required for the RTDs and R/I converters were 50 to 400°F, which satisfies the R.G. 1.97 recommended range of 50 to 250°F for containment sump temperature.

The safety review concluded that RFC DC-12-2900, Subtask D.23 does not constitute an unreviewed safety question as defined in 10 DFR 50.59, "Changes, Tests and Experiments," paragraph (a) (2), and that it does not create a substantial hazard to the health and safety of the public.

RFC-12-2900 D.31

BRIEF DESCRIPTION

Subtask D.31 of RFC DC-12-2900 installed new, environmentally qualified replacement limit switches on valves VCR-201 and VCR-202. VCR-201 and VCR-202 are containment purge supply and exhaust isolation valves located in the auxiliary building that serve the containment instrumentation room ventilation units. VCR-201 is installed in series with VCR-101 in the containment instrumentation room purge air inlet line, while VCR-202 is installed in series with VCR-102 in the containment instrumentation room purge air outlet line.

SAFETY EVALUATION

Subtask D.31 of RFC DC-12-2900 was installed to meet requirements of Revision 3 to R.G. 1.97 dated May, 1983. It is classified in our submittal AEP:NRC:0773S, dated June 29, 1987, as a Type D, Category 2 variable. As such, environmental qualification and an appropriate QA program are required. Seismic qualification and a Class 1E power supply are not required.

The only function of limit switches VCR-201 and VCR-202 is to indicate whether the valve is open or closed. These limit switches in no way affect valve operation.

This RFC is classified as "safety interface" since the limit switches do not affect the safety function of the valves and have a mix of safety and nonsafety requirements.

The Nuclear Safety and Licensing Section (NS&L) has reviewed Subtask D.31 of RFC DC-12-2900 under the provisions of NS&L Section Procedure No. 7, "Safety Review of Design Changes."

Commitment for this installation was made in AEP:NRC:07730, "June 12, 1984 Confirmatory Order - Final Status Report of R.G. 1.97 Compliance," dated October 15, 1985, and was not a part of the R.G. 1.97 submittal, dated June 29, 1987. This change does not affect and Technical Specifications.

The safety review concluded that Subtask D.31 of RFC DC-12-2900 does not constitute an unreviewed safety question as defined in 10 CFR 50.59, "Changes, Tests and Experiments," Section (a) (2), nor does it represent a substantial hazard to the health and safety of the public.

RFC-12-2900 E.02

BRIEF EVALUATION

RFC DC-1202900 Subtask E.02 was for the installation of an Eberline data acquisition module (DAM) and 25 Eberline area radiation monitors in fulfillment of commitments made in response to the Regulatory Guide 1.97, Rev. 3 recommendation to provide area radiation monitors in areas to which plant personnel might require access to service equipment important to safety following an accident. Of the 25 monitors installed, channels from 5 of the monitors feed the DAM installed under this RFC. The DAMs associated with the remaining 20 channels were installed under RFC DC-12-4036. The 25 radiation monitors installed under RFC DC-12-2900 Subtask E.02 are located as follows:

- o East and west centrifugal charging pump rooms (both units)
- o East and west RHR pump rooms (both units)
- o North and south safety injection system pump rooms (both units)
- o Unit 1 and Unit 2 control rooms
- o Access control facility
- o Radiochemistry laboratory
- o 633', 609', 587' and 573' passageways

- o Emergency sampling station
- o Unit 1 and Unit 2 vent sampling area
- o Unit 1 and Unit 2 vent sampling flow adjustment area

SAFETY EVALUATION

RFC DC-12-2900 Subtask E.02 was classified as safety interface. While these area monitors perform no safety-related function, they are important in assisting in detection of significant radioactivity releases, radioactivity release assessment and long-term surveillance following an accident. In addition, the monitors are installed in areas containing safety-related equipment.

RFC DC-12-2900 Subtask E.02 has been reviewed in accordance with NS&L Section Procedure No. 7, "Safety Review of Design Changes." During this review particular emphasis was put on commitments made to the NRC by AEP for installation of this type of radiation monitoring instrumentation in accordance with the recommendations of Regulatory Guide 1.97, Rev. 3. These commitments were made in submittals dated October 15, 1985 (AEP:NRC:0773 O) and June 29, 1987 (AEP:NRC:0773 5)

As a result of this review, it is concluded that installation of this instrumentation creates no substantial hazard to the public health and safety. In addition, it is concluded that no unreviewed safety question exists as defined by 10 CFR 50.59, "Changes, Tests and Experiments", Section (a) (2). This change does not affect the Cook Nuclear Plant Technical Specifications.

RFC-12-2912

BRIEF DESCRIPTION

RFC DC-12-2912 provided for the removal of the AFW automatic pump trip on low suction pressure. The trip function was replaced by

operator/alarm action. In addition, the existing low suction pressure trip system, which consists of (per pump) three pressure switches, one gauge protector, 2 out of 3 logic and a 5.5 second time delay relay, was be replaced with one pressure switch, which does not require a gauge protector, and a 10 to 20 second time delay relay.

The time delay has been increased due to an inherent pressure oscillation in the AFW suction line which occurs during pump start-up and has a duration of about 10 seconds.

The pressure switch circuit was tied directly to an alarm in the control room which alerts the operators of a potential problem with the pump.

Since we had previously committed to the NRC to have low suction pressure trip (AEP:NRC:00300), it was necessary to formally, by way of letter (AEP:NRC:0976), revise our position. NRC's acceptance of our new position was formalized in a letter dated June 20, 1986 from B. J. Youngblood to John E. Dolan.

SAFETY EVALUATION

RFC DC-12-2912 is classified "safety-interface" because the pressure switch is used to activate an alarm and alert the operators of a potential problem with a safety related system (AFW).

The Nuclear Safety and Licensing Section (NS&L) has reviewed RFC DC-12-2912 under the provisions of NS&L Section Procedure No. 7, "Safety Review of Design Changes."

The safety review concluded that RFC DC-12-2912 does not constitute an unreviewed safety question as defined in 10 CFR 50.59, nor does it create a substantial hazard to the health and safety of the public.

This change does not affect any Technical Specifications.

RFC-12-2980 Rev. 1

BRIEF DESCRIPTION

The purpose of Revision 1 to RFC DC-12-2980 was to expand the scope of the RFC to include the replacement of additional 250 VDC breaker panels with fuse panels. The original RFC was initiated to correct a "potentially deficient design" that was identified in Licensee Event Report (LER) No. 87-020, "Lack of Isolation Between Balance of Plant and Essential Safety System Loads Due to Potential Design Deficiency."

The replacement fuse panels purchased under this RFC were specified to be seismically qualified in the original RFC packet. Seismic qualification is required because the panels included in the original scope of the RFC are safety-related panels which supply Class 1E power to engineered safety systems.

The specific panels added to the scope of this RFC under Revision 1 are 1-AFC-1, 1-AFC-2, 2-AFC-1, and 2-AFC-2. These panels are 250 VDC annunciator feeder panels located in the Unit 1 and Unit 2 control rooms (two panels per control room). These balance of plant panels supply power to the non-safety related annunciators in the control room.

SAFETY EVALUATION

The annunciator feeder panels replaced under Revision 1 of this RFC are non-safety related. Therefore, there were no requirements for these panels to be seismically qualified. However, the panels are mounted in the control room and must therefore be mounted such that adverse interaction with safety-related components during a seismic event was precluded. Due to this concern,

Nuclear Safety and Licensing (NS&L) classifies RFC DC-12-2980 Revision 1 as "safety interface."

NS&L has reviewed Revision 1 to RFC DC-12-2980 in accordance with the instructions in NS&L Procedure QP-7, "Safety Reviews," Revision 4.

The safety review concluded that the proposed revision does not represent a change in Cook Nuclear Plant as described in the Updated Final Safety Analysis Report (FSAR). The proposed revision does not involve a change in the Technical Specifications. NS&L concludes that this revision does not constitute a significant hazard to the health and safety of the public.

RFC-12-2974

BRIEF DESCRIPTION

The purpose of this RFC was to upgrade containment sump level instrumentation to the standards given in Regulatory Guide (R.G.) 1.97, Rev. 3. The previous instrumentation was unsatisfactory due to inherent instrument drift.

Containment sump level instrumentation was installed as a Category 2 variable with appropriate environmental qualification and quality assurance requirements. Containment level instrumentation was installed per category 1 requirements for Type A variable.

This RFC was being implemented to meet a commitment made to the NRC in AEP:NRC:0856T, "Technical Specification Change Request For Containment Water Level Monitoring Instrumentation (NUREG--737, Item II.F.1.5)", dated July 16, 1987.

SAFETY EVALUATION

This RFC is classified as "safety related" since the containment water level monitoring instrumentation is necessary for plant safety and the installation has many safety requirements.

The safety review of this RFC considered the qualifications needed for containment sump level and containment level instrumentation in order to comply with our R.G. 1.97 commitments for accident monitoring instrumentation.

The modification to containment sump level and containment level instrumentation is an upgrade to comply with our commitments regarding accident monitoring instrumentation. It is concluded that this modification does not constitute an unreviewed safety question as defined in 10 CFR 50.59, "Changes, Tests and Experiments," Section (a) (2) and that the change does not create a substantial hazard to the health and safety of the public.

RFC-12-3011

NRC Bulletin No. 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," and its two supplements describe circumstances which can lead to fatigue failure of unisolated sections of piping connected to the reactor coolant system (RCS). According to the bulletin, thermal fatigue can occur when the connected piping is isolated by a leaking block valve, the pressure upstream from the block valve is higher than RCS pressure, and the temperature upstream is significantly cooler than RCS temperature.

Our initial response to NRC Bulletin No. 88-08, AEP:NRC:1069 dated September 29, 1988, identified the piping sections at Cook Nuclear Plant that have the potential for being subjected to high-cycle thermal stress caused by leaking valves. The identi-

fied piping sections include four-1 1/2" high-pressure emergency core cooling system (ECCS) injection lines (one injection line to each cold leg). Our bulletin response also committed to plan and implement a program to ensure that applicable piping sections will not be subjected to sufficient stresses that could cause fatigue failure.

The purpose for installing this RFC was to provide a means for monitoring leakage through SI-125. This block valve is installed in the 1" flushing line that bypasses the boron injection tank (BIT). Leakage past this valve to the RCS has the potential for creating high thermal stress at the cold leg connections for the four high-pressure ECCS injection lines identified above. Leakage monitoring capability is achieved by the installation of a 1" stainless steel locked closed globe valve downstream of SI-125 and a 1/2" tell tale line between SI-125 and the new globe valve. This tell tale line has a 1/2" stainless steel globe valve and a threaded pipe cap. The tell tale line will be opened periodically in order to check the leak tightness of SI-125.

SAFETY EVALUATION

The 1" flushing line being modified by this RFC is not used to accomplish the safety function of the ECCS. However, the line being modified is Seismic Class I and this classification must be maintained for this line when it is modified. Therefore, Nuclear Safety and Licensing (NS&L) concluded that this RFC should be classified as "safety-related".

NS&L has reviewed the proposed change in accordance with the instructions in NS&L Procedure QP-7, "Safety Review of Design Changes."

NS&L concluded that RFC-DC-12-3011 does not involve a change in Technical Specifications, does not constitute an unreviewed safety question as defined in 10 DFR 50.59, "Changes, Tests and

Experiments," Section (a) (2), and does not create a substantial hazard to the health and safety of the public.

RFC-12-3011 Rev. 1

The original RFC-DC-12-3011 was issued to address the concerns in NRC Bulletin No. 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems." The purpose of the RFC was to provide a means for monitoring leakage through block valve SI-125 installed in the one-inch flushing line that bypasses the boron injection tank (BIT). Leakage monitoring capability was to be achieved by installing a one-inch stainless steel locked-closed globe valve downstream of SI-125 and a half-inch tell tale line between SI-125 and the new globe valve. The tell tale line was to have a half-inch stainless steel globe valve and a threaded pipe cap, and would be opened periodically in order to check the leak tightness of SI-125.

RFC-DC-12-3011 was written to comply with a commitment made to the NRC in AEP:NRC:1069, dated September 29, 1988 (our initial response to NRC Bulletin No. 88-08).

This revision involved the abandonment of block valve SI-125, the addition of another valve in the one-inch flushing line, and a change in the location of the new valves (See Attachment 1). The method of abandonment of SI-125 was to lock it open. The half-inch tell tale line remains part of the RFC revision.

SAFETY EVALUATION

The safety classification of this revision is "safety interface". The line being modified is Seismic Class I, although it is not required to accomplish any of the safety functions of the ECCS.

NS&L reviewed the original packet for RFC-DC-12-3011 and the reasons given for the revision to this RFC.

NS&L concludes that this RFC does not involve a change in the Technical Specifications, does not constitute an unreviewed safety question as defined in 10 CFR 50.59, "Changes, Tests and Experiments," Section (a) (2), and does not create a substantial hazard to the health and safety of the public.

RFC-1-3014

BRIEF DESCRIPTION

RFC DC-1-3014 modified instrumentation circuitry for the Reduced Temperature and Pressure Program on Unit 1. The changes were necessary in order to support operation of Unit 1 at reduced temperature and pressure (RTP) conditions. Analyses and Technical Specification (T/S) changes associated with the RTP program were submitted to the NRC via letter no. AEP:NRC:1067 on October 14, 1988.

The specific changes are as follows:

1. Dynamic Compensators
   The F delta I input signal to the dynamic compensator modules for the overpower delta T reactor trip was jumpered out. The jumpering was accomplished by relocating the terminal ends of existing wires within the reactor protection cabinets. (The dynamic compensator modules penalize the overpower delta T setpoint if the axial offset in the core exceeds specified values.) The Westinghouse methodology used for the RTP program sets the value of the F delta I function equal to 9 for the overpower delta T trip.

2. Steam Flow Pressure Compensators

Resistors were changed in compensators which are part of the signal conditioning circuitry for steam line flow channels. The circuitry is part of the reactor trip and ESF actuation functions. The change in resistors was necessary to allow the circuitry to be calibrated for the revised steam line pressures of the RTP program. (The RTP program full load steam pressure is 665 psia, versus the old value of approximately 750 psia.)

3. T_{ref} Loop

Resistors were changed in the T_{ref} circuitry. The circuitry involved provides a control function for automatic rod control and steam dumps. The changes were necessary to allow the circuitry to be calibrated to the revised conditions of the RTP program. (The full load RTP temperature is approximately 550°F versus the old 567.8°F .)

SAFETY EVALUATION

Items 1 and 2 above impact reactor trip and ESF actuation circuitry, and are therefore classified as safety related. Item 3 impacts only control circuitry. since this circuitry involves the control rods; however, it is classified as safety interface.

The Nuclear Safety and Licensing Section (NS&L) has reviewed RFC DC-1-3014 in accordance with Section Procedure No. 7, "Safety Review of Design Changes."

It was concluded that RFC DC-1-3014 does not constitute an unreviewed safety question as defined in 10 CFR 50.59, and that

it does not pose a significant threat to public health and safety.

Because the RTP conditions required T/S changes, the analyses had to be submitted to the NRC for review prior to implementation. The changes implemented in RFC DC-1-3014 were consistent with the proposed RTP operating conditions. However, the equipment with the changes installed could not be declared operational prior to completion of NRC review of the analyses.

CHANGES TO FACILITY
TEMPORARY MODIFICATIONS

Brief descriptions and summary safety evaluations for design changes (Temporary Modifications) made to the facility as described in the Donald C. Cook Nuclear Plant Final Safety Analysis Report (FSAR) are presented in this section. These changes were completed pursuant to the provisions of Title 10, Code of Federal Regulations subsection 50.59 (a).

TEMPORARY MODIFICATION #221 - Unit 1
TEMPORARY MODIFICATION #124 - Unit 2

BRIEF DESCRIPTION

These Temporary Modifications removed and capped shut signal lines to twelve (12 air-operated valves in the steam dump system for each Unit. This modification made these valves inoperable by leaving the 85 psig air supply in service to keep the valve closed. Removing the air signal line will prevent these valves from opening. This reduced the capacity of the steam dump system from 85% of full load to approximately 40%.

SAFETY EVALUATION

A detailed Safety Evaluation was previously performed for RFC DC-12-2845 to reduce the capacity of the Steam Dump System. This Safety Review adequately covers the temporary modifications described above. The safety review for RFC DC-12-2845 concluded that the RFC did not involve a change in the T/S and did not constitute an unreviewed safety question or create a substantial hazard to the health and safety of the public.

TEMPORARY MODIFICATION #240 - Unit 1

BRIEF DESCRIPTION

RFC-2999 provided for the replacement of the incore instrumenta-

tion thimble tubes. Fifty-seven (57) of the fifty-eight (58) Unit 1 thimble tubes were successfully replaced during the outage; however, after thimble tube #51 (core location "F-1") was removed, attempts to install the new thimble tube were unsuccessful due to an obstruction in the guide tube. A temporary cap was installed on the guide tube for thimble tube #51 by this Temporary Modification.

SAFETY EVALUATION

Since the proposed Temporary Modification forms part of the reactor coolant pressure boundary, it is classified as safety related.

NS&L has reviewed this Temporary Modification according to the requirements of NS&L QP-7.

It was determined that the proposed change does not represent an unreviewed safety question. An engineering evaluation has been completed that shows that this temporary modification meets or exceeds the design pressure of the RCS. In addition, Technical Specification 3.3.3.2, "Instrumentation, Movable Incore Detectors," allows for gathering flux mapping data with up to 25% of the thimble locations unavailable. If rod position must be verified with the flux mapping system, another thimble is located in the immediate vicinity of F-1.

Based on the above review actions taken, NS&L concludes that the proposed Temporary Modification is not an Unresolved Safety Question as defined in 10 CFR 50.59, "Changes, Tests and Experiments," Section (a) (2), and therefore, does not represent a significant hazard to the public health and safety.

TEMPORARY MODIFICATION #238 - Unit 1
TEMPORARY MODIFICATION #134 - Unit 2

BRIEF DESCRIPTION

The purpose of these Temporary Modifications was to provide for the removal of Units 1 and 2 back-up nitrogen to weld channel pressurization system low nitrogen pressure alarms.

RFC DC-12-2895 provides for abandoning the containment penetration and weld channel pressurization system (CPWCPS) except for the control air supply to the zone 3 electrical penetrations. Disconnecting alarms associated with the CPWCPS, such as the back-up nitrogen to weld channel pressurization system low nitrogen pressure alarm, is included in the scope of the RFC.

SAFETY EVALUATION

RFC DC-12-2895 was classified as "safety-related" because the Seismic Class I Portions of the CPSCPS supply leaders at the containment penetrations will be cut and capped. Removal of back-up nitrogen to weld channel pressurization system low pressure alarms is not considered to be a safety-related design change. These alarm circuits are balance of plant (non-safety related) circuits. It is concluded that the safety classification of alarm removal is non-safety related and therefore, this design change may be processed as a plant modification.

Nuclear Safety & Licensing (NS&L), provided a safety review of RFC DC-12-2895. Since alarm removal is included in the scope of the RFC, no further analyses was needed for removing the alarms.

Therefore, there are no safety concerns with alarm removal. NS&L concludes that alarm removal does not involve a change in

Technical Specifications and does not constitute an unreviewed safety question as defined in 10 CFR 50.59.

SUPPLEMENTAL LICENSE AMENDMENTS

Brief descriptions of two significant licensing amendments are included in this section.

RFC-02-2908 - SGRP

RFC-DC-02-2908 provided for the repair of the Unit 2 steam generators. The repair was undertaken to rectify the deterioration of the steam generator tubes and restore the unit to 100% capacity. The repair involved the replacement of the steam generator lower assemblies, refurbishment if the steam generator moisture separation equipment and restoration of the structures and systems required to allow access to the steam generators. A detailed description of the repairs can be found in the "Donald C. Cook Nuclear Plant Steam Generator Repair Report."

This RFC was conducted under Amendment No. 100 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit No. 2.

REDUCED TEMPERATURE AND PRESSURE

Amendment No. 126 to the Cook Nuclear Plant Unit 1 Technical Specifications allowing plant operation on a reduced temperature and pressure program was implemented beginning Cycle 11. The intent of this lowered operating program was to reduce the rate of steam generator U-tube stress corrosion cracking of the kind observed in Unit 2. The scope of the work included safety analyses, instrumentation changes, and numerous procedural revisions to support the new operating program.