D O N A L D C. C O O K N U C L E A R P L A N T

ANNUAL OPERATING REPORT

1985

Compiled By: ____________
J.F. Stietzel
QC Superintendent

Reviewed By: B.A. Svensson 2/27/86
B.A. Svensson
Assistant Plant Manager

Approved By: A. Alan Blind 2/28/86 for
W.G. Smith, Jr.
Plant Manager

8610150034 851231
PDR ADOCK 05000315
R PDR

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INTRODUCTION

The Donald C. Cook Nuclear Plant, owned by the Indiana and Mighican Electric Company and located five miles north of Bridgman, Michigan, consists of two 1100 MWe pressurized water reactors. The Nuclear Steam Supply Systems for both units are supplied by Westinghouse with a General Electric turbine-generator on Unit 1 and a Brown-Boveri turbine-generator on Unit 2. The condenser cooling method is open cycle, using Lake Michigan water as the condenser cooling source. The Donald C. Cook Nuclear Plant wwas the first nuclear facility to use the ice condenser reactor containment system, which utilizes a heat sink of borated ice in a cold storage compartment located inside the containment. The architect/engineer and constructor was the Americal Electric Power Service Corporation.

This report was compiled by Mr. J.F. Stietzel, with information contributed by the following individuals:

T.A. Kriesel	-	Personnel Exposure Summary
R.L. Otte	-	Inservice Inspection
T.P. Beilman	-	Changes to Facility

1.16 REPORT - WORK FUNCTION CATAGORIES

Reactor Operations and Surveillance

Those activities involved with operating the plant or monitoring it's operation, including chemistry, performance testing, surveillance testing, etc. The plant may be at any power le el, including zero, and still have work falling into this area. Many STP's run during shutdown or refueling may still fall into this catagory.

Routine Maintenance

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

Inservice Inspection

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 o insulation, or disassembly of valves, pumps, etc. Not included are inspections to assess or monitor normal wear, etc. For example, dissembly of a charging pump to inspect bearing wear would not be Inservice Inspection, but dissembly to inspect for rotor cracking or casing damage would be. Inspection of a weld on a newly added line is Special Maintenance, or inspection of a weld repair at a leaking fitting is Routine Maintenance.

Special Maintenance

All work on equipment or systems performed to make significant modifications. Installation of new systems or equipment, replacement 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.

Waste Processing

All work associated with decontamination of equipment, areas, systems, etc. (if not an integral part of another job, such as pump repair), 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 fixing the manipulator crane is Routine Maintenance.

ANNUAL OPERATING REPORT - RG 1.16 FOR 1985

	# PERSONNEL >100 mR			TOTAL MAN-REM		
	STAT.	UTIL.	CONT.	STATION	UTILITY	CONTRACT
Reactor Operations & Surveillance						
Maintenance Personnel	18	0	52	3.120	0.000	19.266
Operations Personnel	85	0	13	28.703	0.000	5.801
Health Physics Personnel	19	0	73	4.360	0.000	17.638
Supervisory Personnel	3	0	0	0.481	0.000	0.000
Engineering Personnel	1	1	2	0.229	0.328	0.231
Routine Maintenance						
Maintenance Personnel	114	0	488	74.368	0.000	265.319
Operations Personnel	23	0	39	5.487	0.000	24.879
Health Physics Personnel	17	0	101	3.994	0.000	49.511
Supervisory Personnel	10	0	1	3.762	0.000	0.398
Engineering Personnel	4	5	9	0.830	0.877	3.086
In-Service Inspection						
Maintenance Personnel	13	0	211	4.550	0.000	88.436
Operations Personnel	14	0	17	2.701	0.000	8.583
Health Physics Personnel	5	0	47	0.653	0.000	13.026
Supervisory Personnel	4	0	0	0.721	0.000	0.000
Engineering Personnel	2	4	4	0.304	0.706	0.801
Special Maintenance						
Maintenance Personnel	22	0	174	5.870	0.000	63.665
Operations Personnel	1	0	7	0.192	0.000	1.431
Health Physics Personnel	0	0	14	0.000	0.000	3.352
Supervisory Personnel	1	0	0	0.132	0.000	0.000
Engineering Personnel	3	3	8	0.886	0.391	1.531
Waste Processing						
Maintenance Personnel	37	0	120	10.343	0.000	68.615
Operations Personnel	1	0	1	0.230	0.000	0.400
Health Physics Personnel	10	0	6	2.699	0.000	1.074
Supervisory Personnel	1	0	0	2.501	0.000	0.000
Engineering Personnel	1	0	0	1.525	0.000	0.000
Refueling						
Maintenance Personnel	5	0	43	0.672	0.000	17.900
Operations Personnel	4	0	11	0.802	0.000	2.578
Health Physics Personnel	0	0	2	0.000	0.000	0.285
Supervisory Personnel	0	0	0	0.000	0.000	0.000
Engineering Personnel	0	0	1	0.000	0.000	0.140
Totals						
Maintenance Personnel	121	0	786	98.923	0.000	523.201
Operations Personnel	105	0	64	38.115	0.000	43.672
Health Physics Personnel	31	0	106	11.706	0.000	84.886
Supervisory Personnel	17	0	1	7.597	0.000	0.398
Eningeering Personnel	7	10	22	3.774	2.302	5.789
Grand Totals	281	10	979	160.115	2.302	657.946

STEAM GENERATOR INSPECTIONS

This report delineates the complete results of steam generator tube inservice inspections and any resulting plugging performed during calendar year 1985. As a result of these inspections twenty-eight (28) tubes were plugged in Unit 1 and one hundred forty-seven (147) tubes were plugged in Unit 2.

Unit 1 Steam generator activities were in progress from April 19 through June 6, 1985. This examination took place during the Unit 1 10 Year Refueling Outage and exceeded the requirements of the Unit 1 Donald C. Cook Nuclear Plant Technical Specifications.

Eddy current examination was performed on 100% of all unplugged tubes in all four steam generators. Steam Generator No. 11 had a total of three thousand three hundred seventy-six (3,376) tubes examined. Steam Generator No. 12 had a total of three thousand three hundred seventy-five (3,375) tubes examined. Steam Generator No. 13 had a total of three thousand three hundred sixty-nine (3,369) tubes examined. Steam Generator No. 14 had a total of three thousand three hundred seventy-three (3,373) tubes examined.

UNIT 1 TUBES PLUGGED

Steam Generator No. 11

Row	Column	% of Indication
19	19	45%
36	19	63%
27	25	71%
29	36	74%
31	55	73%
28	68	60%
31	73	61%
25	80	47%

Steam Generator No. 12

Row	Column	% of Indication
18	37	46%
18	38	90%
18	39	87%
19	39	85%
14	41	66%
16	42	84%
16	43	85%
17	43	85%
7	86	50%

UNIT 1 TUBES PLUGGED Con't

Steam Generator No. 13

Row	Column	% of Indication
10	2	82%
37	47	49%
20	51	56%
40	51	44%
34	56	43%
33	61	40%

Steam Generator No. 14

Row	Column	% of Indication
6	5	54%
13	39	46%
21	40	58%
20	47	54%
22	70	62%

Secondary Side Search and Retrieval and Sludge Lancing

The Combustion Engineering crew commenced sludge lancing on April 19, 1985, and completed their activities on April 28, 1985. The four generators yielded a total of two hundred seventy-four (274) pounds of wet sludge. The work consisted of inspecting the annulus between the vessel wall and the outer diameter of the tube bundle and spraying high pressure water through the tubelanes to remove the tubesheet sludge. Both activities were prformed on all four steam generators.

Prelance inspections were performed on all four steam generators of the annulus area around the tube bundle. No objects were found during this inspection. One small piece of weld wire approximately three inches long was removed from Steam Generator No. 11 with a suction wand. A postlance inspection of random tubelanes in the sludge pile region of Steam Generator No. 14 with a fiberscope showed the generator to be clean.

The sludge was removed from each steam generator, separated from the slurry by the precoat filter and then deposited in drums. The amount of sludge removed from each steam generator was weighed and recorded as follows:

Steam Generator No. 11 - 98 lbs.
Steam Generator No. 12 - 30 lbs.
Steam Generator No. 13 - 62 lbs.
Steam Generator No. 14 - 84 lbs.

Secondary Side Search and Retrieval and Sludge Lancing Con't

The amounts of sludge removed from Steam Generator Nos. 11, 13 and 14 were of the same order of magnitude but the amount in Steam Generator No. 12 was considerably smaller. During previous sludge removal operations the amount removed from both Steam Generator Nos. 12 and 14 had been less than from the other generators. This was considered to be due to the different feed trains of the generators. Some material is carried over in the precoat filter from one generator to the next but this is normally only 10 to 15 pounds of material. This explains the slightly larger amount from Steam Generator No. 11 which received several dumps during cleaning operations at the end of the job and some of the discrepancy on Steam Generator No. 12 which was the first generator sprayed.

The Unit 2 eddy current examination of selected steam generators was performed during three different time intervals in 1985. These examinations were mandated due to primary to secondary side leakage. The following will explain the program and the course of action taken by Indiana and Michigan Electric Company.

Unit 2 was removed from service on July 15, 1985, with a primary to secondary leak rate of 0.22 gpm. Visual examination under a static head showed one leaking tube (R16-C56) in Steam Generator 23. Helium leak detection revealed no other leakage. Westinghouse Electric Corporation was brought in to perform eddy current (EC) testing of the leaking tube. Their test revealed a defect about 1 inch below the top of the tubesheet. Additional EC testing of a block of 24 tubes around R16-C56 revealed tube R16-C55 to have a pluggable indication. Reanalysis of the 1984 data for R16-C56 confirmed that no reportable indications were present in 1984, but R15-C55 had an indication of 20% that was not identified. In view of the fact that previous inspections had not disclosed widespread tube degradation problems in the tubesheet region, and with only 90 days of fuel left in Cycle 5, a decision was made to plug the two tubes and return to service.

The unit was restarted on August 2. During the start-up, increases in activity at radiation monitors on air ejectors and blowdown samples indicated slight additional leakage in Steam Generator No. 23. The unit was again removed from service. Visual inspection under a static head revealed 2 leaking tubes (R7-C28 and R14-C70). Since we believed that the cause of the problem was associated with crevice leakage, helium testing was not performed. Eddy current testing was begun on an initial sample of about 460 tubes and subsequently expanded to approximately 1500 tubes in the sludge pile region of the tubesheet, which is an area where industry experience and our own previous examinations had indicated potential problems.

During this inspection a primary and secondary eddy current data review was conducted to ensure accuracy. Westinghouse conducted the primary data review and Conam Inspection, an independent contractor, conducted the secondary review. As a result of that review, 35 tubes were plugged, 20 due to defects as defined by the Technical Specifications, and 15 due to our administrative plugging criteria. Many of the tubes plugged had no previous indication of degradation.

The unit was restarted, but on August 23, during the hold at 30 percent power for the boric acid soak, steam generator leakage was observed. The unit was removed from service when the leak rate reached 0.20 gpm. All four steam generators were opened and visually inspected under a static head. Two leaking tubes were identified, one each in Steam Generators 22 and 24. Since it was judged that we could not operate the unit with continual steam generator leakage problems, a decision was made to perform an extensive eddy current inspection of steam generator tubes. This was later expanded to include 100% of the tubes in Steam Generators 21, 22 and 24 and to inspect the balance of the tubes in Steam Generator 23.

It was noted that during the course of this complete eddy current inspection, defects and distorted indications at hot leg support plate intersections were found. Since the condition of the tubes at support plate intersections could impact any future decision to sleeve the tubesheet region, we elected to remove tube samples to assess the condition of the tubes at the support plates. Five tubes were removed, three including the first to third support plate intersections and two including the first to fifth support plate intersections.

Final results of hot leg eddy current indications found during the three forced outages resulted in a total of 147 tubes being plugged. The plugging criteria implemented for this eddy current inspection included not only the 40% and greater through wall Technical Specification plugging limit for defective tubes but additional conservatism of an administrative plugging limit for tubes with indications at the top of the tubesheet and in the crevice region.

Of the total tubes examined, 93 tubes were plugged to comply with Technical Specification requirements. The others were plugged for conservatism to prevent further deterioration of tubes at the top of the tubesheet and in the crevice region, since these could impact future decisions regarding remedial actions of the type discussed above.

UNIT 2 TUBES PLUGGED

Following are the tubes plugged during the first Unit 2 Emergency Outage:

Steam Generator No. 23

Row	Column	% of Indication
15	55	96%
16	56	96%

UNIT 2 TUBES PLUGGED Con't

Following are the tubes plugged during the second Unit 2 Emergency Outage:

Steam Generator No. 23

Row	Column	% of Indication
4	19	46%
7	19	SQR
11	20	63%
10	24	87%
5	27	DI
7	28	94%
8	30	SQR
7	31	87%
7	32	88%
23	32	87%
25	34	SQR
11	40	94%
9	42	SQR
14	48	58%
16	48	59%
22	49	61%
11	50	DI
18	51	DI
17	52	SQR
13	56	62%
13	57	SQR
16	57	27%**
11	60	47%
15	60	SQR
12	61	SQR
15	61	SQR
13	62	36%**
14	62	SQR
22	65	50%
14	70	99%
11	71	93%
11	72	48%
9	74	84%
8	76	75%
9	77	64%

Following is a list of tubes plugged during the third Unit 2 Emergency Outage:

Steam Generator No. 21

Row	Column	% of Indication
20	13	54%
13	14	43%
11	15	40%
7	18	73%
36	19	63%

UNIT 2 TUBES PLUGGED Con't

Steam Generator No. 21 Con't

Row	Column	% of Indication
25	21	57%
16	32	41%
17	34	55%
16	35	DI
10	36	DI
16	41	57%
15	45	DI
21	45	43%
11	49	41%
5	71	50%

Steam Generator No. 22

Row	Column	% of Indication
16	10	DI
3	15	71%
4	19	62%
16	29	82%
20	31	85%
19	33	68%
45	37	43%
14	41	96%
18	46	93%
19	46	81%
16	47	79%
28	50	46%
25	51	SQR
27	54	47%
11	55	62%
12	57	42%
13	57	70%
16	57	83%
11	59	86%
11	61	47%
14	61	37%**
15	61	31%**
6	67	45%
17	71	DI
18	71	52%
11	25	ADS*
7	38	39%*
6	40	NDD*
12	42	80%*
18	77	34%*

*Denotes tubes which were removed from the steam generator for metallurgical analysis. These voids in the tubesheet were plugged utilizing the welded plug technique.

UNIT 2 TUBES PLUGGED Con't

Steam Generator No. 23

Row	Column	% of Indication
6	8	36%**
4	11	DI
28	19	43%
3	20	DI
32	20	30%**
31	22	39%**
38	23	DI
41	27	54%
37	30	82%
40	34	41%
24	35	59%
41	46	DI

Steam Generator No. 24

Row	Column	% of Indication
20	11	SQR
6	19	56%
14	22	30%**
3	24	78%
33	28	-20%**
4	49	66%
19	52	97%
8	53	43%
15	53	DI
16	54	44%
30	54	69%
12	55	47%
13	55	DI
17	55	71%
10	56	DI
19	56	61%
23	56	44%
10	57	43%
11	57	78%
12	57	56%
14	57	79%
19	58	69%
22	59	81%
23	59	86%
16	60	32%**
19	60	83%
18	61	59%
19	61	90%
18	62	32%**
13	63	30%**
16	63	NDD*
18	63	42%
19	63	49%
20	63	45%

UNIT 2 TUBES PLUGGED Con't

Steam Generator No. 24 Con't

Row	Column	% of Indication
19	64	50%
9	65	56%
16	65	34%**
17	65	32%**
20	65	31%**
16	66	31%**
18	66	37%**
20	66	45%
21	66	41%
18	67	38%**
19	67	DI
20	67	55%
34	67	41%
35	67	54%
18	70	91%
21	72	49%
19	75	49%
19	81	49%
7	94	96%

*This tube was plugged inadvertantly. The tube that should have been plugged was Row 15 Column 63 which is still in service with a 31% indication, 3.1" above the tubesheet. The indication did not exceed Technical Specification plugging limits.

NOTE:

** - Administratively Plugged
DI - Distorted Indication
SQR - Squirrel - Unquantifiable signal within a tubesheet
NDD - No detectable defect
ADS - Absolute Drift Signal

PORV'S

Following is a summary of tests results and maintenance performed on the Pressurizer Power Operated Relief Valves during the calendar year 1985.

Unit 1

NRV-151

Date Tested	Stroke Time Open	Stroke Time Close
01/21/85	4.7	1.6
	4.8	1.5
	4.8	1.5
01/25/85	4.8	1.4
08/30/85	5.0	1.5
	5.0	1.7
	5.0	1.7
09/05/85	5.0	1.7
09/20/85	4.7	1.3
11/27/85	4.8	1.9

NOTE:

On January 18, 1985, Supplemental Job Order No. 87069X1 was sent to the C&I Section to investigate a problem with NRV-151 leaking by. The investigation consisted of adjusting the valve stem length and pressure in accordance with C&I IMP.030. Kept 2/3's of the NRV's blocked open at all times and verified closing and opening times were within specification.

NRV-152

Date Tested	Stroke Time Open	Stroke Time Close
01/14/85	4.0	N/A
01/25/85	4.4	1.5
08/30/85	4.3	1.5
	4.4.	1.6
	4.3	1.5
09/05/85	4.2	1.5
09/19/85	4.5	1.3
09/20/85	3.8	1.9
10/24/85	4.0	1.9
11/27/85	3.9	1.5

NOTE:

On September 19, 1985, the valve stroke cycle was done using emergency air only.

PORV'S Con't

Unit 1 Con't

NRV-153

Date Tested	Stroke Time Open	Close
01/14/85	3.7	N/A
01/21/85	4.5	1.6
	4.5	1.5
	4.5	1.4
01/25/85	4.5	1.5
08/30/85	4.3	1.5
	4.4	1.5
	4.3	1.4
09/05/85	4.4	1.6
09/18/85	4.0	1.5
09/20/85	3.8	1.4
11/27/85	3.8	1.6

NOTE:

On September 17, 1985, Emergency Job Order No. 14158 was sent to the Maintenance Department to replace air bottles and to tighten the fittings on the tubing. This work was completed and signed off on September 19, 1985.

UNIT 2

NRV-151

Date Tested	Stroke Time Open	Close
07/18/85	N/A	1.1
07/21/85	4.6	N/A
07/26/85	4.6	1.4
07/27/85	4.6	1.3
07/30/85	4.6	N/A
10/09/85	4.3	1.4

NOTE:

On July 18, 1985, Supplemental Job Order No. 033852X2 was issued to the C&I Section to set the stroke on NRV-151. This was accomplished and the Job Order was closed out on July 27, 1985.

PORV'S Con't

Unit 2 Con't

NRV-152

Date Tested	Stroke Time Open	Close
07/18/85	N/A	1.0
07/21/85	4.0	N/A
07/27/85	3.9	1.5
07/30/85	3.9	N/A
09/04/85	3.6	1.8
09/06/85	3.5	1.5
10/09/85	3.6	1.5
	3.5	1.3
10/10/85	3.6	1.6

NOTE:

On July 18, 1985, Supplemental Job Order No. 33852X2 was sent to the C&I Section to investigate NRV-152 leaking by. The Maintenance Department replaced the diaphragm and set the stroke. The Job Order was closed out on July 27, 1985.

On October 9, 1985, Job Order No. 40335 was sent to the Maintenance Department to investigate and/or repair leak on diaphragm. The investigation showed that there was an air leak around the flange and therefore the bolting was tightened in order to stop leakage. The Job Order was closed out on October 10, 1985.

NRV-153

Date Tested	Stroke Time Open	Close
07/18/85	N/A	1.1
07/21/85	3.3	N/A
07/26/85	3.0	1.7
07/27/85	3.0	1.8
07/30/85	3.0	N/A
10/19/85	3.5	1.5
	3.2	1.5

NOTE:

On July 18, 1985, Supplemental Job Order No. 33852X2 was sent to the Technical Department to investigate NRV-153 leaking by. The valve had the setpoints verified and returned to operation on July 27, 1985.

Unit 2 Con't

PORV'S Con't

NRV-153 Con't

On October 9, 1985, Emergency Job Order No. 14091 was sent to the Technical Department to investigate the cause of NRV-153 not giving full closure indications. The rod was tightened up and the bolts positioned properly to allow proper signal annunciation. The Job Order was closed out on October 9, 1985.

During the last year there were three challenges to the Pressurizer Power Operated Relief Valves in Unit 2. Condition Reports relative to these challenges were submitted and satisfactorily closed. These Condition Reports are on file in the D.C. Cook Plant Master Plant File.

CHANGES TO FACILITY

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).

DC-12-1742 (Unit 1 Only)

Brief Description

An air operated containment isolation valve (PCR-40) and a check valve were installed on the containment penetration for the Plant Air System of Unit 1 of the Donald C. Cook Nuclear Plant. The air operated valve which is located outside containment and the check valve which is located inside containment fulfills the requirements for double isolation for containment penetrations. The air operated valve can be operated from the Control Room and is designed to close upon a containment Phase A isolation signal.

Prior to the modification, the system was required to be isolated by the installation of a blind flange prior to the Reactor Coolant System entering Mode 4 (Hot Shutdown). This modification will allow remaining maintenance activities requiring the use of the Plant Air System to continue during primary system heat-up following a cold shutdown or refueling outage.

Safety Evaluation

This RFC is considered safety related because it involves modification of a containment penetration and its corresponding isolation system. The plant service air line is presently a Class E containment isolation system which includes a closed manual valve and a membrane barrier (such as a blind flange). The proposed system which is comprised of a check valve and an automatic valve will be a Class A containment isolation system as defined in FSAR Chapter 5, Section 5.4. This is consistent with the licensing basis of the Cook Plant which differs somewhat from current NRC criteria such as GDC-56. We are not required at the present time to comply with GDC-56 and as such this RFC is acceptable.

This safety review is conducted on the RFC compliance of containment integrity under various plant conditions and it indicates that the proposed changes do not create substantial safety hazard nor involve an unreviewed safety question as defined in 10 CFR 50.59.

DC-01-1811

Brief Description

RFC-01-1811 Revision 0 was to restore operation of the 65 Incore Thermocouples by replacement and/or recovery of the inoperable T/Cs. Revision 1 altered the scope to include a complete upgrade of the Incore Thermocouple system to meet, as closely as possible, the requirements of Reg. Guide 1.97, Rev. 3 and NUREG 0737. It was discovered during efforts on Revision 0 that replacement of the inoperable T/Cs was not required if electrical isolation between the T/Cs and the plant computers were provided.

Sixty-three (63) of the sixty-five (65) thermocouples are divided into two (2) electrically independent channels. Each channel is energized from a Class 1E power source, and physically separated, except at the Reactor Vessel Head Area, up to and including the isolation devices. These thermocouples are divided as evenly as possible to provide adequate coverage of each core quadrant.

Environmentally and seismically qualified connectors and mineral insulated cabling was installed from the Core Exit Thermocouples (CET) nozzles up to a location past the missile shield wall, where a transition to qualified organic cabling took place via qualified splices. From this point the qualified organic thermocouple cabling proceeded through penetrations to terminate at new signal processing equipment.

Signal processing equipment consisted of two (2) racks of hardware capable of covering the low level thermocouple signals into high level (4-20 or 10-50 ma) signals. The two (2) racks are Class 1E and qualified to the appropriate environmental and seismic requirements of IEEE Std 323-1974 and 344-1975. Isolation and separation between 1E and non-1E signals was provided.

Cold junction compensation is done automatically by the equipment. The two (2) racks are physically separated using the guidance of Regulatory Guide 1.75 including the 1E signals and non-1E signals. Thirty-three (33) signals are processed on one (1) rack and thirty-two (32) on the other. These racks are each powered by an 1E power source. The outputs of these racks is transmitted to the normal plant process and Technical Support Center computers, two (2) backup displays, and the Saturation Margin Monitor (SMMs) incore thermocouple inputs (two [2] per quadrant for a total of eight [8]). Isolation between 1E and non-1E equipment is provided in the electronics of the various pieces of equipment. The primary display is the plant process computer which has direct readout and hard copy capabilities of all thermocouples. The range will be 200°F to 2300°F. Trending capabilities also exist which can show the time history of Core Exit Thermocouples temperatures on demand.

Alarm capabilities presently exist and are consistent with Emergency Operating Procedures. The operator display devices were reviewed, for human factors.

A spatially oriented core map, available on demand, showing the temperature at each CET location is available from the Technical Support Center Control Terminal (CRT) located in the Control Room. The Technical Support Center computer is not designated as the primary operator display. Therefore, this method of providing a core map is in variance with NUREG-0737. However, it is our opinion that the Technical Support Center computer man-machine interface, together with operator training in use of the upgraded Emergency Operating Procedures, provides adequate compensatory methods for this deviation. It is therefore concluded that this deviation from NUREG-0737 is not significant and should be an acceptable method of providing a core map.

A Class 1E backup display is provided for each channel with the capability for the selective reading of a minimum of sixteen (16) operable thermocouples (four [4] per quadrant).

The present design specifies manual switching between each thermocouple. It is expected that the switching can be completed between all CETs within a time interval of six (6) minutes. The displayed temperature range will be 200°F to 2300°F.

Safety Evaluation

NS&L review indicates that the upgraded incore thermocouple system must meet, to the extent possible, the requirements of NUREG-0737 Item II.F.2, Attachment 1, "Design and Qualification Criteria for PWR Incore Thermocouples," which endorses Appendix B to NUREG-0737 and NRC Generic Letter 82-28. Where these requirements cannot be explicitly met, these deviations are justified.

This change 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 general public.

DC-12-2448 Revision 7

Brief Description

RFC DC-12-2448 Revision 7 authorized installation of high-range Noble gas measuring assemblies on the steam jet air ejector and gland steam leakoff Noble gas monitors.

RFD DC-12-2448 was issued to install an upgraded radiation monitoring system. The purpose of Revision 7 to RFC DC-12-2448 is to extend the range of the Noble gas monitors from 1000 μci/cc to 100,000 μci/cc, as indicated in NUREG-0578 Section 2.1.8.b, and as required in NUREG-0737 Item II.F.1.

Safety Evaluation

RFC DC-12-2448 Revision 7 is classified as "safety interface". This is because the system is designed to provide important information to Plant staff during both accident and postaccident conditions (i.e., a steam generator tube rupture).

The Nuclear Safety and Licensing Section (NS&L) has reviewed RFC DC-12-2448 Revision 7 under the provisions of Section Procedure No. 7, "Safety Review of Design Changes". Details of the review are presented below:

1. NS&L understands that the new assemblies will function over the range of 10^0 to 10^5 μci/cc. NUREG-0737 Item II.F.1 requirements include:

   i. A maximum range of 10^5 μci/cc for noble gas monitors functioning on undiluted PWR condenser air removal system exhausts, and

   ii. A range overlap of a factor of 10 for individual monitors.

   The new high range assemblies will overlap the present intermediate range assemblies by a factor of 10^3, since the presently installed system has a maximum range of 10^3 μci/cc.

2. Section 11.3.3.1 of the updated FSAR states the "control interlocks (on the Process Radiation Monitoring System) fail in the 'high radiation' position upon instrument failure and must be manually reset. Instrument failure alarms are initiated upon failure of the radiation monitor, loss of detector signal or loss of power".

NS&L understands that these safety mechanisms are unaffected by the proposed change. It is to be noted that the monitors in question do not have any trip functions associated with them. They do, however generate an alarm upon a high radiation level.

3. Table 11.3-1 of the FSAR will need to be revised in accordance with AEPSC General Procedure 5.8.

4. AEP:NRC:0586 Table 1 listed the range of the Gland Steam Condenser Vent and Steam Jet Air Ejector Noble gas monitors as 10^{-7} to 10^{3} µci/cc. The RFC Lead Engineer is responsible for interfacing with NS&L in order to update this submittal upon implementation of the RFC Revision.

5. The proposed changes appear to have no impact on current Technical Specifications. They may, however, affect changes to the Technical Specifications related to NUREG-0737. A copy of this memo will be directed to the NS&L Senior Licensing Engineer for consideration in future proposed Technical Specification changes.

Based on the review detailed above, NS&L concludes that the proposed changes involve no additional risk to the health and safety of the general public, and that there are no open items with regard to RFC DC-12-2448 Revision 7.

The purpose of RFC DC-12-2448 Revision 7 is to authorize the installation of high-range Noble gas measuring assemblies on the steam jet air ejector and gland steam leakoff Noble gas monitors. It is concluded that RFC DC-12-2448 Revision 7 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 general public.

<u>DC-12-2448 Revision 8</u>

Brief Description

RFC DC-12-2448 Revision 8 authorized installing the capability to safely obtain and analyze grab samples of containment atmosphere in a post-accident environment.

It has been determined that RFC DC-12-2448 should be enhanced to provide for ALARA concerns and improved sampling methodology use for NUREG-0737 Items II.B3 Criteria 1,3,6,11 and II.F1 Criterion 2. The Revision 8 Design Change requested below are to rectify these possible operational and radiation exposure problems presented by the original design.

The original design used the Eberline SPING 4s (ERS-1400 & 2400) for Post Accident Grab Samples. This presented two problems; 1) the post-accident containment pressure can damage the detectors inside the SPINGs; 2) the post-accident radiation level is too high for the SPING's pump flow rate causing damage to the pump, and greatly increasing radiation exposure to personnel.

To resolve this, Revision 8 granted approval to purchase two pump/grab sample pallets (accident range monitors AXM-1s supplied by Eberline Instrument Corp.) and low flow pumps (6L/min.).

The Eberline AXM-1 Monitor is an accident range noble gas monitor with collection of particulate and iodine samples.

The AXM-1 comprises four major assemblies; a noble gas pallet (NGP-1), a grab sample pallet (GSP-1), a bulk filter assembly (BFA-1) and a data acquisition module (DAM-4-6). Each assembly is described in brief to educate the reader on the overall workings of the Accident Range Monitor.

1. Grab Sample Pallet (GSP-1)

   The Grab Sample Pallet is an assembly designed to be located in an inhabitable location during the accident condition. It contains all necessary elements to perform the following functions:

   a) Collect particulate and iodine samples at a rate of 1/60 of the normal sampler flow rate.

   b) Purge the assembly of radioactive gasses.

   c) Remove safely the collected particulate and iodine samples to a lab for analysis.

2. Bulk Filter Assembly (BFA-1)

   The Bulk Filter Assembly is designed to be located in an area which may be normally uninhabited. Its location should be chosen to limit personnel exposures from the radioactive particulate and iodines which are filtered out by the cartridges in the BFA-1. Shielding of the filter may be required. The sample, now entirely gaseous in composition, enters the NGP-1.

3. Noble Gas Pallet (NGP-1)

   The Noble Gas Pallet is an assembly designed to be located in an uninhabited location during the accident condition. It contains all the necessary elements to perform the following functions:

a) Move the Sample through the system, including exit of the system.

b) Monitor the noble gas concentration of the sample flow.

c) Determine when a low flow conditions exists.

4. Data Acquisition Module (DAM-4)

The detectors on the GSP-1 and NGP-1 have interfacing electronics (IBs) mounted on their associated assemblies. These electronics then are connected to the data acquisition module (DAM-4) which contains the microcomputer which is the heart of the system. The programs (software) which control the system are stored in read-only memory (ROM) and, therefore, are fixed. Only the parameters of the system can be varied.

The microcomputer performs the tasks of data acquisition, history file management, operational status check and alarm determination. In addition, it communicates with man through a Central Control Unit (Control Terminal).

A digital display is provided, which indicates the numerical data (the display values) and status (both physical status and alarm status) of the channel to which it is selected.

Detectors on the noble gas pallet and on the grab sample pallet detect radiation and the signals are processed by the interface boxes (IB-4X-HT-CCs). The output signals from the interface boxes are input to the detector input-output boards of the microcomputer located in the DAM-4. These signals are converted to count rate by the microcomputer which then performs all mathematical calculations and control functions.

The DAM-4 program for the AXM-1 recognized four active channels. They are:

Channel 1 (SA-16) Particulate and Iodine Filter Gamma Activity

Channel 2 (SA-15) Noble Gas Channel Background Subtraction

Channel 3 (SA-15) High Range Noble Gas

Channel 4 (SA-14) Intermediate Range Noble Gas

The Model AXM-1 has undergone qualification testing for Class 1E equipment in accordance with IEEE 323-1974 and IEEE 344-1975.

Safety Evaluation

RFC DC-12-2448 Revision 8 is classified "safety interface" because it involves the accurate analysis of post-accident conditions inside containment. The RFC requires seismic mounting due to connections with safety related valves and its proximity to safety related equipment, i.e., the containment spray pumps.

The Nuclear Safety and Licensing Section (NS&L) has reviewed RFC DC-12-2448 Revision 8 under the provisions of NS&L Section Procedure No. 7, "Safety Review of Design Changes". This review addresses concerns of post-accident environment fail safe design, NUREG-0737 requirements, and activation of heat tracing. The items identified as concerns are listed followed by an operational brief.

The post-accident environment at the atmosphere sample inlet in lower containment can be extremely harsh concerning conditions of temperature, humidity, pressure, radiation, and hydrogen burns. The grab sample piping and hardware exposed to containment atmosphere is capable of withstanding, without any leaks, the most severe accident environment at the time and place where the sample is taken. It is assumed that the sample is taken as soon as Phase B isolation can be reset to open valves ECR-33, 35 and 36. The sampling equipment (AXM-1) is designed to withstand pressures and temperatures of 125 psig and 250°F. To ensure the integrity of the system during sampling, the system should not be opened if these conditions are exceeded.

The heat tracing should eliminate problems due to humidity. Condensation in the system would be minimal and have negligible effects, since most of the containment atmosphere is returned to containment when the system is purged.

The radiation exposure concerns are addressed in the ALARA review. The ALARA review states that a sample can be safely taken from AXM-1 with proper control of the system at any time during or after an accident.

Since procedure and design are intertwined, this safety review must address procedural concerns. The procedure applying this equipment must address concerns of ALARA and off-site dose listed in this review and the ALARA review. These concerns include:

1) minimizing time spent near the containment spray pump rooms and the sample.

2) isolation and purging of the lines prior to disconnecting the sample.

3) minimize the time containment isolation valves are open.

4) immediately close the containment isolation valves should any leak be detected.

As recommended in the ALARA review the procedure should be tested and timed to minimize risk and exposure. The results of the test and procedure will then be analyzed by the Radiological Support Section for comment and approval.

To avoid problems associated with hydrogen burns, procedures must include verification that hydrogen concentration be below the flammability limit in lower containment prior to opening valves ECR-33, 35 and 36. The procedural requirements are summarized later.

Assuming either a single active or passive failure of the equipment the sampling equipment (AXM-1) is designed to prevent any release of contaminated gas outside containment. The valves to containment close on a Phase B isolation signal. If power in a train is lost while a sample is being taken, the other safety train will be able to close the system. The grab sample system will then be inoperable, but the containment integrity will be maintained. The AXM-1 is designed so that any credible failure is not possible if the AXM-1 is used properly. To use the system the Eberline SPING 4s (ERS-1400 & 2400) must be isolated and grab sample pallet connected prior to opening the containment isolation valves. Isolation valves in the gas sample pallet are opened to provide the grab sample. Once the grab sample has been collected, the isolation valves in the AXM-1 are closed and the lines purged with clean air so that the grab sample can be safely removed and another pallet connected. A G-M detector is located in the pallet to ensure that the lines have been properly purged and isolated before removal of the sample thus minimizing any possibility of release. Plant procedures must assure the proper sequence of events to maintain a closed system. The possibility of a single passive failure due to a seismic event resulting in dose exceeding 10 CFR part 100 is not considered credible because of the relatively short time the system will be open. This is consistent with the NUREG -0737 requirements for the balance of the post-accident sampling system. RFC DC-12-2448 Revision 8 must satisfy the requirements of NUREG-0737 Items II.B.3 Criteria 1, 3, 6, 8, 11 and II.F.1 Criterion 2. Criterion 8, which requires the ability after an accident to take daily samples the first week and weekly thereafter, the Lead Engineer agrees that Criterion 8 should be included in this review and has concurred that RFC DC-12-2448 Revision 8 will fulfill all the aforementioned NUREG-0737 requirements.

The line from the containment wall to the grab sample is heat traced in RFC DC-12-2448 to simulate containment temperatures to avoid iodine plateout. The heat tracing will be energized upon a Phase A isolation signal and will remain energized as long as grab samples are desired. Procedures must be revised to deactivate the heat tracing after spurious safety injection signals.

The procedural requirements of this safety review are summarized below. Since most of these items are not under direct control of the Lead Engineer, they are listed as concerns resulting from this safety review.

1. The Radiological Support Section should review and approve the final design, procedures, and results of testing to assure all the concerns of the ALARA review have been addressed.

2. The emergency operating procedures must be written to include verification that hydrogen concentration be below the flammability limit and that pressure and temperature are below the design limits of the AXM-1 (125 psig and 250 F) in lower containment prior to opening valves ECR-33, 35 and 36.

3. The Lead Engineer must review the final emergency operating procedures to ensure that the sequence and use of the equipment is in accordance with the design intent.

4. The operating procedures must be revised to deactivate the heat tracing after spurious safety injection signals.

5. The emergency operating procedures for the AXM-1 must be written to ensure that containment isolation valves ECR-33, 35 and 36 are kept open in an accident only while taking a sample, and that the containment isolation valves be immediately closed should any leak be detected in the sampling system.

The purpose of RFC DC-12-2448 Revision 8 is to authorize the design, procurement and installation of equipment and piping necessary to obtain containment atmosphere grab samples after an accident. Provided the conditions and concerns included in this safety review are completed, it is concluded that RFC DC-12-2448 Revision 8 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 general public.

DC-12-2461 (Unit 1 Only)

Brief Description

RFC DC-12-2461 connected 350kW of pressurizer heater capacity to the emergency diesel generator safety buses such that during a loss of off-site power Reactor Coolant System (RCS) pressure control can be maintained. Pressure control is necessary to achieve and maintain adequate subcooling (50°F margin) and natural circulation conditions in the RCS. This modification is being made in accordance with the requirements of NUREG-0578 Item 2.1.1a as amended by NUREG-0737.

Safety Evaluation

This change is considered safety related since it involves the emergency power distribution system.

Westinghouse analyses have shown that a capacity of 150 kW of pressurizer heaters is needed for RCS pressure control to maintain adequate natural circulation conditions. In order to meet the single failure criteria a redundant capacity of 150 kW is required from the opposite emergency diesel generator safety bus in conformance with NRC General Design Criteria - 34. The operator must have full control of this capacity of heaters in the main Control Room and hotshut down panel in conformance to NRC General Design Criteria - 19.

NRC Branch Technical Position RSB 5-1 when applied to the pressurizer heater requirements of NUREG-0578, mandates the use of emergency power to control RCS pressure in the transition to cold shutdown by natural circulation. The NRC is requiring that plant procedures and training be instituted to cover RCS cooldown on natural circulation. RFC DC-12-2461 is consistent with these requirements.

In performing this modification careful consideration must be given to:

1. Diesel Generator load capacity, shedding, restoration and sequencing.
2. Physical separation requirements.
3. Interface requirements with non-Class 1E portions of system.
4. Environmental and seismic qualification for safety grade portions of system. (Heaters are non-Class 1E and are not required to be upgraded).

RFC DC-12-2461 does not create a substantial safety hazard nor does it constitute an unreviewed safety question as defined in 10 CFR 50.59.

DC-12-2610

Brief Description

RFC DC-12-2610 provides for the installation of volume dampers in the 150 ice condenser duct panels of the air distribution system of each unit. The previous air distribution systems of each unit's ice condenser did not have a means of balancing the supply of chilled air to the 150 ice condenser duct panels. This uneven air distribution caused temperature gradients which in turn increase ice sublimation. To minimize this problem and supply a balanced flow of air to each duct panel, this RFC installed balanced volume dampers in the inlet ducts to the 150 ice condenser duct panels.

Safety Evaluation

This RFC was declared safety interface after being reviewed according to NS&L Procedure No. 7. During the review the following two concerns were identified:

> First, it must be assured that if the volume dampers are not balanced at the time of initial installation, adequate means of support are provided on an interim basis or the initial installation be limited to the supporting angles and some form of sufficient insulation. Second, it is also requested that a copy of the calculations that are performed to support the balanced positions of the volume dampers be included in the RFC packet.
>
> With the above considerations in mind, the Nuclear Safety and Licensing Section does not have reason to object to the changes proposed by this RFC. RFC DC-12-2610 does not involve an unreviewed safety question as defined in 10 CFR 50.59 nor does it compromise the health and safety of the general public.

DC-12-2639 (Unit 1 Only)

Brief Description

Frequent CRID II failures caused an initiation of steam dump to occur. It was not the design intent of the protective circuitry to initiate a steam dump due to a CRID II failure.

This RFC installed a voltage sensing relay to monitor the CRID II output. Using a contact from the relay, as an additional interlock to permissive C-9 (i.e. permissive C-9 is blocked by CRID II failure). This installation prevents steam dump initiation in the event CRID II is lost.

Safety Evaluation

This RFC has been classified as safety related because it affects the Reactor Protective System.

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, it was decided that the relay would be seismically qualified.

The purpose of this review is for procurement, design, and installation. With this in mind, it is concluded that this RFC 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.

DC-12-2651 (Unit 1 Only)

Brief Description

The minimum flow lines of the SI pumps were modified. The following changes were made:

The first change installed a flow metering device in each of the minimum flow lines to allow for metering via an associated differential pressure indicator.

The addition of individual flow indicators to each of the two (2) affected lines allows for improved monitoring and timely identification of abnormal operating conditions It will enable plant personnel to take immediate actions to prevent pump damage from overheating (such overheating may occur if a minimum flow rate of about thirty [30] gpm is not available to remove the energy output of a single pump motor). The rotometer (Plant ID No. IFI-265) previously installed in the common portion of the minimum flow line (downstream of the SI minimum flow isolation valves IMO-262 and IMO-263) had its internals removed to eliminate an unnecessary flow restriction. The rotometer body, however, has been left with the exception of the top-mounted site glass. This site glass has been removed and the connection capped.

The second change involved the replacement of four (4) piston check valves (Plant ID No. CS297E, CS297W, SI104N & SI104S) located on the minimum flow piping at the discharge of the safety injection (SI) and centrifugal charging pumps (CCP) with swing check valves. This proposed design change stems from the August, 1982 SI pump failure caused by sticking of the piston check valve on the Unit No. 1 North SI pump minimum flow line. Such sticking blocks minimum flow, thereby removing the capability to cool the pump while Reactor Coolant System (RCS) pressure is above the shutoff head for SI flow to the core.

Swing check valves have been chosen as a replacement for the piston check valves because it is believed that the swing type valves are less likely to stick closed as compared to piston type valves. Furthermore, the size of the swing check valves on the SI minimum flow piping will be increased to 1½" from 3/4" since 3/4" swing check valves of adequate quality for this service are not commercially available. The 3/4" diameter pipe section between the minimum flow orifice assemblies (Plant ID No. RO-104) and the check valves will thus be replaced with a 1½" Schedule 80 section to accommodate the new check valves. No change in check valve size has been made to the CC pump minimum flow piping check valves.

Additionally, in order to prevent the possibility of water jets leaving the new minimum flow orifice assemblies (see Change #3 discussion below) from damaging the swing check valves, the relative location of the orifice assemblies and the swing check valves will be switched (i.e., the new check valves will be located upstream of the minimum flow orifice assemblies). Although the old piston check valves and the new swing check valves will effectively be subjected to the same average minimum flow velocity, this change has been requested because it is believed that the plates internal to the swing check valves (which do not exist in piston check valves) are more likely to be damaged from the water jets present at the exits of the orifice assemblies.

The third and final change made was the installation of new SI minimum flow orifice assemblies designed to increase the minimum flow rate from the present thirty (30) gpm to sixty (60) gpm. This change is intended to preclude damage to the SI pumps by decreasing the probability of occurrence of low flow conditions.

Safety Evaluation

The subject RFC is classified "safety related" because: a) the ECCS is a seismic Class I Engineered Safety Feature (ESF) which must function during various Donald C. Cook Nuclear Plant design basis accidents; b) the flow rate of emergency coolant to the core will be decreased during Loss-of-Coolant Accidents (LOCAs); and c) environmental qualifications of the new local flow indicators is required.

This change will provide for indication of low flow conditions in the vicinity of the SI pumps only, and not in the Control Rooms. Thus, the new instrumentation will not be relied upon by the Plant operators in the performance of safety related activities, such as the tripping of an SI pump during a LOCA. It is also understood that no electrical work (i.e., routing of safety grade cables, etc.) will be performed as part of this Design Change. It should also be noted that switching the relative position of the orifice assemblies and the check valves is acceptable since the primary function of the valves (i.e., preventing backwash into an inactive pump if another pump is started up) should be fulfilled.

The following items had to be considered in the Engineering and Design of this RFC:

ITEM 1

During the recirculation phase of LOCA, radioactivity deposited in containment sump water is expected to be present in SI system piping. This Design Change then may result in an additional radiation source (due to contaminated fluid in the sample taps or differential pressure transmitter) in the SI pump rooms after a LOCA.

Therefore, additional radiological hazards created by the new instrument system had to be considered.

ITEM 2

The minimum flow lines from which the sample flow is drawn to the indicators are Seismic Class I. Therefore, the sample lines and flow indicators must be qualified Seismic Class I, using the appropriate response spectra. All sample line hangers and indicator mounts must restrain the system sufficiently to ensure that failure of the ECCS boundary does not occur.

ITEM 3

Due to additional combustibles the fire loading characteristics in the SI pump room were reviewed.

ITEM 4

In accordance with good flow measuring techniques a 4' length of minimum flow piping upstream and a 1' length of minimum flow piping downstream of the flow metering orifices will be increased in size from 3/4" to 1½". Seismic considerations for this system have been met.

NOTE: The following items are directed towards the previously described Design Change 2.

ITEM 5

The seismic qualification required for the new minimum flow check valves, considered more than dead weight (i.e., system boundary integrity during vibration). The analysis considered the combination of all design changes. That is, since the orifice assemblies will be changed out (SI system only), a section of pipe increased in diameter (SI system only), and the check valves replaced, the new minimum flow piping system resulting from this RFC could respond in a significantly different way to postulated seismic events than the presently installed minimum flow piping system.

ITEM 6

Attention was given to the functional capability of the new swing check valves during postulated seismic events. This matter involves the potential for check valves failing closed during an earthquake. If this were to happen prior to or during the early (i.e., high RCS pressure) phase of a small break LOCA, it is conceivable that cooling to both SI and/or CC pumps could be lost and/or one or both sets of pumps could be deadheaded. This potential common mode failure could result in irretrievable loss of one or both sets of pumps and consequential violation of present Final Safety Analysis Report (FSAR) analyses.

NOTE: The following items are directed toward the previously described Design Change 3.

ITEM 7

This Design Change would necessarily result in decreased SI flow to the reactor core during design basis accidents, thus impacting upon analyses presented in the Donald C. Cook Nuclear Plant FSAR. Furthermore, since the SI system is a part of the ECCs, this Design Change must be reviewed for conformance to GDC 35 (Emergency Core Cooling), 36 (Inspection of Emergency Core Cooling System), and 37 (Testing of Emergency Core Cooling System) of 10 CFR 50 Appendix A, to 10 CFR 50.46 (Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors), and to 10 CFR 50 Appendix K (ECCS Evaluation Models).

ITEM 8

Plant Technical Specification 3/4.5.2 effectively limits the SI minimum flow rate to 50 gpm during the performance of flow balance tests. It is expected that a design minimum flow rate of sixty (60) gpm will lead to an actual minimum flow rate of approximately forty-four (44) gpm during the flow balance tests. Therefore a Technical Specification change regarding the revised SI pump flow characteristics has been approved by NRC staff.

ITEM 9

A conservative reading of applicable ECCs regulations (i.e., 10 CFR 50.46, 10 CFR 50 Appendix K, GDC 35 of 10 CFR 50 Appendix A, etc.) indicates that the impact of the requested Design Change upon Donald C. Cook Nuclear Plant small break LOCA analyses must be assessed.

ITEM 10

The Design Change was evaluated against non-LOCA design basis accidents. As a minimum, the impact of reduced SI flow must be evaluated for the Donald C. Cook Nuclear Plant steam line break analysis (see FSAR Section 14.2.5 [Unit Nos. 1 and 2], FSAR Appendix B [Unit Nos. 1 and 2], and FSAR Appendix C [Unit No. 2]). Of particular concern is that boric acid delivered by the ECCs (assuming one [1] operable charging pump) is needed to shut down the reactor, assuming the most pessimistic combination of circumstances. Reduced SI flow will tend to increase the power spike, however, thus making it more difficult to shut down.

ITEM 11

Also associated with NRC approval of this RFC is the effect of the Design Change upon calculated containment pressures for design basis accidents. More specifically, during the post-reflood phase of a LOCA, a reduction in SI flow will result in a reduction in the amount of steam condensed in the RCS cold legs. Thus, more steam may be released through the break and into containment, thereby raising containment pressure and affecting the ice condenser melt-out time.

ITEM 12

With regard to the effect of SI flow containment pressure during a LOCA, it is noted that SI flow is not significant during a large break LOCA (indeed, small changes in SI flow have no effect Peak Clad Temperature during such a scenario). Since containment design pressure is 12.0 psig and the peak containment pressure resulting from a large break LOCA is given by the Unit No. 1 FSAR Update Section 14.3.4.3.7 as 11.5 psig it is concluded that the small decrease in SI flow will not result in exceeding containment design capacity. (Additionally, it is noted that the referenced FSAR analysis did not assume energy absorption to containment structural heat sinks -- this provides additional margin for any assumed energy added to the containment atmosphere).

For the Small Break LOCA (SBLOCA) scenario, it is possible that decreased SI flow will result in decreased steam condensation in the Reactor Coolant System loops, and thus increased steam release into containment. The containment peak pressures which can be achieved during SBLOCAs are, however, bounded by the peak pressures resulting from large break LOCAs, and hence there is no reason to believe that the pressures high enough to threaten containment integrity. In any event, the containment spray system may be actuated (either manually or through automatic initiation logic) to limit the partial pressure of steam within containment, thereby limiting the containment pressure rise. No further review is required by NS&L from a conceptual design standpoint.

Due to resolution of the above items RFC-2651 does not constitute an unreviewed safety question as defined in 10 CFR 50.59, nor will it create a substantial hazard to the health and safety of the general public.

DC-12-2663

Brief Description

This RFC modified the trip control of the reactor trip breakers and the reactor trip bypass breakers control so that a Reactor Protection System (RPS) automatic trip signal will activate both the undervoltage and shunt trip coils. This was required by Generic Letter 83-28 which was issued by NRC July 8, 1983 indicating actions to be taken by licenses based on the generic implication of the Salem ATWS events. The modification was made to improve reliability of the reactor trip system.

Safety Evaluation

This RFC has been classified as safety related because it affects the safe shutdown and isolation system of the reactor and the mitigation of design basis accidents.

Nuclear Safety and Licensing has reviewed the proposed change as per the review criteria in NS&L Procedure No. 7. The result of this review identified thirteen (13) design concerns which had been submitted to reviewed and approved by the NRC prior to implementation. NRC approval was granted by Docket Nos. 50-315 and 50-316 dated January 23, 1985 with two requirements which required resolution and subsequently were fulfilled.

With this mind, it is concluded that, the RFC 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.

DC-01-2667

Brief Description

This RFC covers the design, procurement and installation of the following items:

1. Local Indicating Pressure Transmitters for S/G Pressure. An additional tap off of the steam generator lines upstream of the stop valves was made. This pressure signal was forwarded to the Local Shutdown Indication (LSI) panel by electrical pressure transmitter.

2. Temperature hot and temperature cold for RCS Loops has been made available at 1-LSI-4 panel to provide RCS information to the LSI system.

3. A new source range monitoring channel was added in an existing spare excore monitor well to provide indication at 1-LSI-4 in accordance with 10 CFR 50 Appendix R, Section III G.2. Criteria. This will ensure that source range monitoring information is available to support alternate shutdown.

4. A new local shutdown panel 1-LSI-4 was installed to provide a centralized control and communication point for all emergency activities outside the Control Room. All required safe shutdown instrumentation which presently exist on panels 1-LSI-1 and 1-LSI-2 is also on 1-LSI-4.

Safety Evaluation

This RFC has been classified as safety related because:

1) The importance of the LSI panels in certain alternate shutdown scenarios and
2) Class I seismic systems are involved.

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 the following design concerns were addressed:

CONCERN 1

The seismic analysis of the steam line between the steam generator and its respective stop valve (e.g., S/G and MRV-210) should be reviewed to establish that the results have not significantly changed.

CONCERN 2

The relevant portions of the FSAR need to be revised.

CONCERN 3

Since the actual cable routing is not finalized, the Lead Engineer should ensure that the installation satisfies the separation criteria required by Section III.G.2 of Appendix R. For dual RTDs, the leads should be insulated and separated as soon practicable.

CONCERN 4

It should be noted that the RTDs associated with Loops 110, 120, 130, 140, 210, 220, 230 and 240 are part of our Environmental Qualification program which we have submitted to the NRC to meet the provisions of 10 CFR 50.49. As per our understanding, these new RTDs are yet to be qualified, they should meet IEE 323-1974. Part of our submission to the NRC requires that components not in our stores as of February 22, 1983 will have to meet IEE 323-1974. Qualification data and test results will be entered into the centralized EQ file when available, but prior to installation in the Cook Plant. This data will be reviewed by NS&L for adequacy.

CONCERN 5

The electrical loads and cabling will be environmentally qualified for both safety and non-safety portions of the loop from the RTDs up to the LSIs. This is to avoid any QA errors and also to take into account any interaction of non-safety and safety cables.

After having addressed the above design concerns it is concluded that this RFC 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.

DC-01-2668

Brief Description

This RFC modified the Essential Service Water (ESW) and Component Cooling Water (CCW) control circuits. The specific modifications are as follows: 1) modified Unit 2 East and West ESW pump control circuit by isolating the pressure switches shared by both units ESW pumps; 2) removed Unit 1 SI signal contact from each of the Unit 2 East and West ESW pump control circuit; and 3) modified the Unit 2 East and West CCW pump control circuit by isolating pressure switches to Unit 2 hot shutdown panel. These modifications will contribute to bringing the ESW and CCW pump control circuits into compliance with Appendix R., Section III G.

Safety Evaluation

This RFC is classified as safety related because the changes involve Class 1E electrical control circuits critical to the function of design-engineered features of the plant.

The following items had to be considered in the Engineering and Design of this RFC:

ITEM 1

New isolation relays should be seismically qualified by either testing or analysis and installed to Seismic Class I standards.

ITEM 2

Any changes to electrical safety cable routings must be reviewed for conformance to the separation criteria of Specification #DCCEE-130-QCN.

ITEM 3

Electrical safety cable routing changes should be reviewed for protection from missiles and high energy line break interactions.

ITEM 4

Isolation relay failure modes should be considered for impact on system function (i.e., is the failure a "safe mode" failure?).

ITEM 5

Conduct a circuit trace analysis to ensure that the proposed changes will not affect the designed sequencing and/or operation of Engineered Safety Features Activation System (ESFAS) as stipulated in the FSAR and ESW System Description SD-DCC-HP102 (File #13A) and CCW System Description SD-DCC-HP103 (File #35A).

This RFC does not constitute an unreviewed safety question as defined in 10 CFR 50.59, nor does it create a substantial safety hazard or adversely affect the health and safety of the public.

<u>DC-01-2764</u>

Brief Description

RFC DC-01-2764 replaced the system Train A and B 250 volt batteries and battery racks, relocated the ventilation system in the 1-CD battery room, and seismically qualified the mounting of the 1-AB and 1-CD battery racks and all ventilation duct modifications to the 1-CD battery room.

The new battery differs in some respect from the battery it replaces and these differences should be noted. The new battery has lead-calcium grids instead of lead-antimony. The float and equalize voltage of the cells is higher than for lead-antimony cells. The number of cells has been reduced to 116 instead of the previous 120. The net result is that, although the cell voltage is higher, the terminal voltage for both the float and equalize charge is the same as for the lead antimony grid battery. The cells have been sized to meet the service requirements when discharged to a cell voltage of 1.812 volts. This will result in a terminal voltage of 210 which is the same as the previously installed battery.

Safety Evaluation

RFC DC-01-2764 is classified safety related because it involves replacement of safety Trains A and B of a Class 1E engineering safeguards system.

The Nuclear Safety and Licensing Section (NS&L) has reviewed RFC DC-01-2764 under the provisions of NS&L Section Procedure No. 7, "Safety Review of Design Changes". As a result of this review special attention has been focused on seismic criteria, plant security and changes in the plant technical specifications.

The purpose of the RFC DC-01-2764 review is to authorize the design and procurement of batteries and racks for the 1-AB and 1-CD battery rooms and installation of the 1-AB batteries. This RFC also authorizes design, procurement and installation of ventilation ducts in the 1-CD battery room. With regard to each of the above items it is concluded that RFC DC-01-2764 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 general public.

DC-01-2766

Brief Description

The purpose of this RFC is to replace the current Critical Reactor Instrument Distribution (CRIDs) with a more reliable inverter and alternate power source. The new inverter includes a bumpless transfer switch, manual bypass switch and forced air cooling package. The inverter has extra capacity to handle future load increases and a higher ambient temperature rating (0-50°C). The existing 3∴ 600vac input to the CRID inverters will be removed.

The existing 600vac source will be used to feed a regulating transformer (600 to 120vac) to be used as an alternate source.

The inverters were purchased Class 1E from the manufacturer with seismic and environmental qualification.

Safety Evaluation

This RFC is classified as safety related because the inverters supply power to the critical reactor instrumentation distribution system.

NS&L has reviewed these design changes as per the review criteria in NS&L Section Procedure No. 7. As a result of this review, the following items had to be considered during the engineering and design of this RFC:

1. Changes to the FSAR are necessary to reflect installation of the transformer and the change in power source. This change will be submitted later for incorporation in the FSAR update.

2. Page 3/4 8-10 of the Technical Specification footnote wording must be clarified. This clarification does not involve a licensing issue or an unreviewed safety question. The footnote reference to this CRID will be removed upon updating the T/S.

3. Train separation of the isolimiters should be resolved and properly annotated in the final design package.

4. Securing the isolimiters and the CRID cabinets must conform to the appropriate seismic criteria.

The purpose of this review is for procurement, design and installation. With this in mind, it is concluded that this RFC 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.

<u>DC-12-2839 (Unit 1 Only)</u>

Brief Description

RFC DC-12-2839 calls for the replacement of the Reactor Coolant System (RCS) narrow range Resistance Temperature Detectors (RTDs) with RTDs manufactured by the RdF Corporation. The terminations for the RdF Corporation RTDs were modified to ensure the environmental qualifications of the installation.

Safety Evaluation

RFC DC-12-2839 is classified safety related because the narrow range RTDs are Class 1E devices which provide an input to the Reactor Protection System.

The Nuclear Safety & Licensing Section (NS&L) has reviewed RFC DC-12-2839 under the provisions of NS&L Section Procedure No. 7, "Safety Review of Design Changes". As a result of this review special attention was focused on seismic and environmental qualification.

With regard to environmental qualification, the RdF Corporation RTDs will meet the requirements of 10 CFR 50.49, including qualification to IEEE Std. 323-1974. Documentation in support of the qualification must be retained in accordance with 10 CFR 50.49 and AEPSC General Procedure No. 42. Seismic qualification of the new RTDs has been addressed by qualification to IEEE Std. 344-1975.

NS&L also notes that the RdF Corporation RTDs should be added to the 10 CFR 50.49 equipment list presented in Section 14.4 of the Updated FSAR. This revision should be effected as per AEPSC General Procedure No. 5.8.1 during the next annual update review process.

It is concluded that RFC DC-12-2839 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 general public.

<u>DC-12-2852 (Unit 1 Only)</u>

Brief Description

RFC DC-12-2852 authorized, procured and installed materials necessary to satisfy commitments made to the NRC under Licensee Event Report 1-85-007. The materials were used to make the following modifications in both units:

1. Replace the present normal HVAC intake dampers with ANSI N509 certified bubble tight dampers. This will eliminate unfiltered outdoor air intake, which will substantially reduce operator thyroid dose rates following a design base accident,

2. Replace the return register from the Control Room Mechanical Equipment Room with a heavy duty damper. This will allow more precise flow adjustment to facilitate easier balancing for ensuring that the

Mechanical Equipment Room is at a positive pressure in the emergency mode,

3. Install an air flow measuring station in the Cleanup/Pressurization System. This will allow precise Tech. Spec. surveillance readings to be taken with substantially reduced time and effort, and will allow quick and accurate resetting of system flow when required,

Safety Evaluation

RFC DC-12-2852 is classified as safety related. This is because modifications will be performed on Seismic Class I systems.

The Nuclear Safety and Licensing Section has reviewed RFC DC-12-2852 under the provisions of Section Procedure No. 7 "Safety Review of Design Changes". Details of the review of each of the items listed in the "Description" section follows:

ITEM 1

NS&L understands that the dampers will be brought and installed to Seismic Class I specifications. Our review, therefore, finds nothing to preclude installation of these dampers.

ITEM 2

As in Item 1, NS&L understands that the dampers will be procured and installed as Seismic Class I. Our review raises no concerns with this item.

ITEM 3

NS&L notes that the flow measurement obtained from the new measuring station will be used for Technical Specification surveillance requirements. Based on conversations with the HVAC Section, we understand that calibration of the gauges can be adequately addressed by procedures of the cognizant divisions.

Additionally, the station will be brought and installed as Seismic Class I. Our review, therefore, identifies no open items with respect to this section.

The purpose of the subject RFC is to request procurement and installation of items committed to in LER 1-85-007. It is concluded that the proposed items 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 general public.

Gouf 3-25 F

American Electric Power System
INDIANA & MICHIGAN ELECTRIC COMPANY
DONALD C. COOK NUCLEAR PLANT
P.O. Box 458, Bridgman, Michigan 49106
(616) 465-5901

February 28, 1986

Mr. J.G. Keppler, Regional Administrator
United States Nuclear Regulatory Commission
Region III
799 Roosevelt Road
Glen Ellyn, IL 60137

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

Dear Mr. Keppler:

Two copies of the 1985 Annual Operating Report for the Donald C. Cook Nuclear Plant are being transmitted to you under this cover letter. The information contained in this report covers the activities delineated in Appendix A (Section 6.9.1.5) of the Donald C. Cook Nuclear Plant Technical Specifications, and the requirements of 10 CFR 50.59.

Additional copies of this report have been transmitted to the Office of Inspection and Enforcement and the Office of Management Information and Program Control of the United States Nuclear Regulatory Commission as specified in Regulatory Guide 10.1.

Respectfully,

A. Alan Blind for

W.G. Smith, Jr.
Plant Manager

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/jas

Attachments

cc: John E. Dolan
M.P. Alexich
R.W. Jurgensen
R.F. Kroeger
J.G. Feinstein
G. Charnoff, Esq.
R.C. Callen, MPSC
D. Hahn
R.O. Bruggee, EPRI
J.F. Stietzel
Dir., IE
Dir., MIPC

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