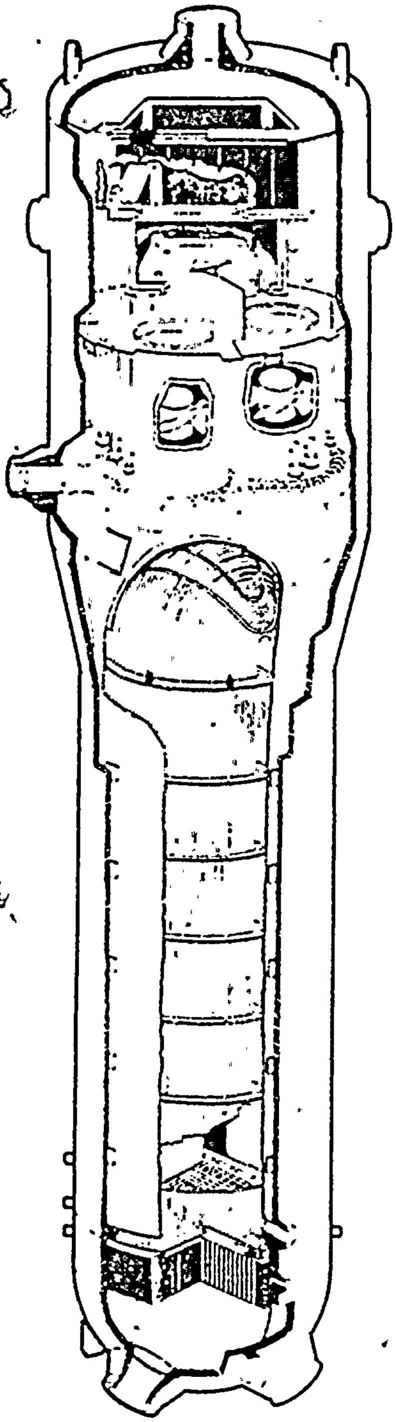
REVISION 1

OF THE

STEAM GENERATOR REPAIR REPORT

FOR THE

DONALD C. COOK NUCLEAR PLANT UNIT NO. 2



steam generator repair report

Donald C. Cook Nuclear Plant

Unit No. 2

U.S. Nuclear Regulatory Commission

Docket No. 50-316

License No. DPR-74

Indiana & Michigan Electric Company AEP

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INSTRUCTIONS FOR THE INSERTION OF REVISION 1
TO THE STEAM GENERATOR REPAIR REPORT

REMOVE	INSERT
Page ii, Revision 0	Page ii, Revision 1
- - -	Page via, Revision 1
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- - -	Supplement 1 Tab
- - -	Supplement 1, Single-Failure-Proof Crane Information, Pages 1-1 through 1-19, Revision 1

RECORD OF REVISIONS

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0	November 4, 1986	November 4, 1986	T. G. Harshbarger
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D. C. COOK PLANT UNIT NO. 2

STEAM GENERATOR REPAIR REPORT

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SECTION 1 - INTRODUCTION

1.1 History of Donald C. Cook Unit 2 Steam Generator Problems and Present Status

Donald C. Cook Unit 2 incorporates a nuclear steam supply system manufactured by Westinghouse, and is licensed for 3411 MWt. Initial criticality occurred on March 10, 1978. The unit completed its fifth fuel cycle on February 28, 1986, at which time about 5.3 effective full power years of operation had been accrued.

Donald C. Cook Unit 2 has four Westinghouse Series 51 steam generators, each having 3388 tubes, 0.875 inch O.D. by 0.050 inch thick; tubing material is Inconel 600 in the mill annealed condition. The tubes are hardrolled for a distance of approximately 2.25 inches above the bottom of the tubesheet, leaving an open annular crevice of approximately 18.75 inch depth and 7 to 9 mil radial gap. Tube support plates are carbon steel with drilled tube holes having a radial clearance of approximately 8 mils.

All volatile treatment using hydrazine and ammonia has been utilized for secondary water chemistry control throughout the life of Donald C. Cook Unit 2, including all preservice testing. The evolution of the Donald C. Cook Plant secondary water chemistry specifications from 1974 to the present, indicative of increased awareness of the need for good water chemistry, is included as Table 1.1-1.

From initial start-up until November 1983, there were no indications of steam generator secondary side corrosion except for minor tube

Table 1.1-1

D. C. COOK NUCLEAR PLANT
SECONDARY SIDE WATER
CHEMISTRY SPECIFICATION HISTORY
STEAM GENERATOR

	AUGUST 1985	OCTOBER 1984	NOVEMBER 1983		SEPTEMBER 1974	
	Cook Limit	Cook Limit	Cook Normal	Cook Limit	Cook Normal	Cook Limit
Cat Cond (umho)	.8*	1.1	<1.5	1.5	2.0	7.0
Sodium (ppb)	20	20	<20	20	100	500
Chloride (ppb)	30	50	<50	50	150	500
Sulfate (ppb)	30	--	N/A	N/A	N/A	N/A
pH	7.5	--	8.6-9.2	8.6-9.2	<8.8	8.5
OH as $CaCO_3$ (ppb)	--	150	<150	--	<150	1000
Silica (ppb)	500	500	<500	500		
Boron (ppm)	5-10	--	--	--		

* CORRECTED FOR BORON

denting at the hot leg tubesheet surface. The only significant tube degradation experienced during that time was primary side cracking of the Row 1 U-bends; preventive plugging of all Row 1 tubes in 1984 resolved that issue. The other, although minor, degradation mechanism noted during the first five calendar years was wear at anti-vibration bar (AVB) intersections; the total number of tubes involved is small, and subsequent inspections have shown little if any growth rate on tubes with previous AVB wear.

The first significant indication of secondary side corrosion of Donald C. Cook Unit 2 came in late 1983. During start-up on November 7, 1983 following an outage to plug leaking Row 1 tubes, there were immediate indications of primary-to-secondary leakage in steam generator 21, and the unit was removed from service. Visual inspection of the primary side of the tubesheet under a static head of water showed the hot leg of tube R16C40 to be leaking. Subsequent eddy current testing (ECT) of about 725 tubes in steam generator 21 revealed the defect in R16C40 to be just above the secondary face of the tubesheet, and indicated two additional tubes, R14C40 and R14C41, with similar tube degradation. In addition, ECT of over 500 tubes in steam generator 22 was performed; no degradation was found. The unit was restarted on November 22, 1983, and ran until March 10, 1984, when it was removed from service for refueling.

Steam generator activities during that refueling outage included complete ECT of all four steam generators and removal of sections of

seven tubes for metallurgical analysis. ECT resulted in plugging 61 tubes due to indications at or just above the tubesheet surface and 5 tubes due to indications in the tubesheet crevice region. Metallography confirmed that degradation was due to intergranular attack/stress corrosion cracking (IGA/SCC), probably caused by caustic environment. The unit was restarted on July 7, 1984 and ran without steam generator related problems until it was removed from service on July 15, 1985, with an indicated leak of 0.22 gpm in S/G 23.

Visual inspection under a static head of water showed one leaking tube (R16C56) in steam generator 23. Helium leak detection revealed no other leakage. ECT of the leaking tube indicated a defect approximately one inch below the top of the tubesheet. Additional ECT of a block of 24 tubes around tube R16C56 revealed tube R15C55 to have a similar defect. Re-analysis of ECT data taken in 1984 showed that tube R15C55 had a 20 percent through-wall indication that was not identified at the time. Tubes R15C55 and R16C56 were plugged.

Donald C. Cook Unit 2 was restarted on August 2. During start-up, radiation monitors on the condenser air ejectors and samples of steam generator blowdown indicated slight additional leakage in steam generator 23. The unit was again removed from service. ECT of approximately 1500 tubes in the sludge pile region of the tubesheet was conducted. As a result, 35 tubes were plugged, many of which had no previous indication of degradation. Due to concern over the apparent pervasiveness of IGA/SCC in the tubesheet region,

a boric acid soak was performed at 30 percent power with 40 to 55 ppm boron concentration, followed by on-line boric acid addition to maintain a 5 to 10 ppm boron concentration. Addition of boric acid was intended to neutralize the alkaline environment and thus possibly slow the rate of IGA/SCC.

Donald C. Cook Unit 2 was again restarted, but on August 23, during a hold at 30 percent power for a boric acid soak, a 0.20 gpm steam generator leak was detected. All four steam generators were opened and visually inspected under a static head of water. Two leaking tubes were identified in steam generators 22 (R14C41) and 24 (R19C52). ECT was expanded to include all the tubes in all steam generators. This inspection resulted in plugging an additional 110 tubes. For the first time, evidence of tube corrosion at hot leg support plate intersections was indicated by ECT. Because the condition of the tubes at support plate intersections could influence any future decision to repair the tubesheet region by sleeving, five tube samples were removed to assess the condition of the tubes at support plates. Analysis of the tube samples confirmed the presence of axially oriented intergranular stress corrosion cracks at support plate locations.

Following the complete ECT, Donald C. Cook Unit 2 was successfully returned to service in October 1985, and ran without significant steam generator related problems until removed from service for refueling on February 28, 1986. At the time of shutdown, steam generator primary-to-secondary leakage was in the range of .001 to .04 gpm;

hydrostatic testing at 600 psig revealed one leaking tube (R16C45) in steam generator 22. Subsequent ECT showed the defect to be located about one inch below the hot leg tubesheet. ECT of all affected areas of all four steam generators resulted in plugging 151 tubes.

To date the total number of tubes plugged in each steam generator is as follows: steam generator 21, 142 tubes; steam generator 22, 210 tubes; steam generator 23, 210 tubes; and steam generator 24, 201 tubes.

1.2 Reasons for the Steam Generator Repair

The steam generators at the Donald C. Cook Unit 2 have experienced corrosion-related phenomena, as discussed in Section 1.1, which have required periodic examination and plugging of steam generator tubes to ensure continued plant operation. At present, Donald C. Cook Unit 2 is continuing to receive on-line boric acid treatment and is being administratively limited to 80% power to retard the rate of steam generator tube degradation. However, the need to plug additional tubes may continue, and the Technical Specification plugging limit may be reached in the future. This could lead to relicensing efforts and possibly result in permanent power limitations.

As a result of continued tube degradation, the associated penalty of reduced generating capacity, and the lack of practical repair techniques, I&MECo is proposing to replace the Donald C. Cook Unit 2 steam generator lower assemblies. This replacement would repair the steam generators and allow the unit to operate at its design capacity.

1.3 Repair Project Summary

1.3.1 Purpose of the Steam Generator Repair Report

This document discusses the steam generator repair project which will be implemented to restore the reliability and performance of the Donald C. Cook Unit 2 steam generators. The discussion presented herein demonstrates that the repair work and subsequent operation can be conducted without undue risk to the health and safety of the general public and personnel engaged in the repair work.

The information contained herein is not intended to supplant the information in the Donald C. Cook Nuclear Plant Updated Final Safety Analysis Report (FSAR), but is intended to identify significant changes that may result from the repair project. The Donald C. Cook Nuclear Plant FSAR can be consulted for specific details about referenced existing equipment, systems, or components.

The information presented in this document reflects the most current design information at the time of preparation of this document.

1.3.2 Scope of Repair Activities

The steam generator repair project proposed by the Indiana & Michigan Electric Company for the Donald C. Cook Unit 2 is similar to the steam generator repairs completed by the Virginia Electric and Power Company for the Surry Power Station and by Wisconsin Electric Power Company for the Point Beach Nuclear Plant Unit 1. Both projects utilized the reactor coolant system pipe-cutting process to remove the steam generator lower assemblies as proposed by the I&MECo for the Donald C. Cook Unit 2.

This repair will be done in accordance with the ASME Boiler & Pressure Vessel Code, Section XI, 1983 Edition and Addenda through the Summer of 1983.

Repair of the Donald C. Cook Unit 2 steam generators will involve the complete replacement of the steam generator lower assemblies. An opening will be cut in the reinforced concrete doghouses surrounding the steam generators to provide access. To facilitate removal, each steam generator will be cut on the upper assembly shell plate just above the transition cone girth weld and at the inlet and outlet reactor coolant piping nozzles. The steam line piping and feedwater piping will be cut and sections removed. The steam generator upper assembly will be lifted off and removed from containment to undergo internal modifications to the moisture separation and feedring equipment. The steam generator lower assemblies will then be lifted from their supports and transported out of containment to the temporary on-site steam generator storage facility. The removal route

for both the upper and the lower assemblies will be through the equipment hatch at the 650′ elevation and through the auxiliary building to the railroad bay. The containment equipment hatch is sized to accommodate steam generator replacement without containment modification. The existing polar crane will be utilized for all heavy lifting inside of the containment building. The replacement steam generator lower assemblies will be transported and installed in a similar manner. The original upper and new lower assemblies as well as all associated piping will be welded together in the field.

All components and piping will be reinstalled to meet the original design configurations and installation requirements, thus eliminating any design modifications which would require changes to the original design analysis.

1.3.3 Steam Generator Lower Assembly Characteristics

Westinghouse will fabricate replacement steam generator lower assemblies. The design of the replacement lower assemblies will match the design performance of the original lower assemblies. However, several design features that do not alter mechanical performance are included in the design. These design features will provide improved thermal hydraulic performance and improved access to the tube bundle, and will reduce the potential for secondary side corrosion. These design modifications are discussed in detail in Section 2.0.

1.3.4 Schedule of Repair Activities

The tentative planning date to begin the steam generator repair project is March 1989. This date is based on estimated fabrication schedules and considerations for seasonal system demands. It is estimated the steam generator repair project will require a 12-month outage -- 9 months for the actual repair and 3 months for associated fuel handling activities. However, licensing, engineering, design, and planning activities are proceeding on an expedited basis in order to support an earlier start date, if required. The earliest date this repair project could commence is mid-1988.

1.3.5 Identification of Principle Agents and Contractors

Indiana & Michigan Electric Company (I&MECo) is a corporation duly organized under the laws of the State of Indiana with its principal place of business at One Summit Square, Ft. Wayne, Indiana. I&MECo is the sole owner and operator of the Donald C. Cook Nuclear Plant, Bridgman, Michigan.

American Electric Power Service Corporation (AEPSC) is a corporation duly organized under the laws of the State of New York with its principal place of business at One Riverside Plaza, Columbus, Ohio. I&MECo and AEPSC are both subsidiaries of American Electric Power Company, Incorporated (AEP).

Westinghouse manufactured the existing steam generators and will provide the replacement steam generator lower assemblies. Westinghouse's expertise will be utilized as appropriate to assist in developing the engineering and construction procedures and in providing site support during the repair project. 1

AEPSC, which will have overall responsibility for establishing the technical requirements for the repair project, has been actively engaged in nuclear power operations with the construction, operation, and maintenance of the Donald C. Cook Nuclear Plant Units 1 and 2. This involvement represents a total operating experience of approximately 19 years.

AEPSC has established a Steam Generator Repair Project Management Organization to manage and direct the repair of the Unit 2 Steam Generators. As shown in Figure 1.3-1, the Steam Generator Repair Project Management Organization will report to the AEPSC Assistant Vice President - Project Management who in turn reports to the AEPSC Vice Chairman - Engineering and Construction.

The Steam Generator Repair Project Management Organization is responsible for the direction and coordination of all Steam Generator
Repair Project activities including engineering, licensing, radiation 1
protection, design, and related testing. The repair field work will also be directed by the Steam Generator Repair Project Management Organization utilizing a composite work force of I&MECo and AEPSC personnel, construction craftsman, and selected specialty contractors who have proven expertise in certain phases of the work.

The Steam Generator Repair Project Management Organization will interface with the Assistant Plant Manager - Technical Support, the Plant Steam Generator Repair Project Engineer, and the AEPSC Vice President - Nuclear Operations (see Figure 1.3-2). Overall administrative control of the Unit 2 containment and systems shall remain with the D. C. Cook Plant Manager. However, after fuel is removed from the Unit 2 reactor and unit layup is completed by the plant staff, the containment and tagged out systems will be released to the Repair Project Management Organization to commence repair activities.

The Repair Project Management Organization shall establish procedures
to ensure that only authorized work is performed and that all such
work is fully documented in approved work packages or procedures. The
work packages or procedures will: identify the system, component, 1
and/or structure to be worked on; describe in detail the work to be
performed; identify procedures to be used, inspection hold points,
related pre/post work activity testing and required documentation.

1.3.6 Quality Assurance Program

American Electric Power Company, Inc., recognizes the fundamental importance of controlling the design, modification and operation of Indiana & Michigan Electric Company's Donald C. Cook Nuclear Plant by implementing a planned and documented Quality Assurance Program. The Donald C. Cook Unit 2 Steam Generator Repair Project will have a Quality Assurance Program that supports the goals of maintaining the safety and reliability of the Donald C. Cook Nuclear Plant at the highest level, and conducting safety-related activities in compliance with applicable regulations, codes, standards and established corporate policies and practices.

The Donald C. Cook Nuclear Plant FSAR, Chapter 1.7, "Quality Assurance" (also referred to as the "Updated Quality Assurance Program Description") supplemented with a Steam Generator Repair Project Quality Assurance Program Description Supplement (see Section 5 of this report) will administer the regulations, codes, standards and established corporate policies and practices as applicable to the Donald C. Cook Unit 2 Steam Generator Repair Project.

1.3.7 Radiological Protection Program (ALARA)

It is the policy of AEP and its subsidiary operating utility I&MECo to conduct the Steam Generator Repair Project in such a manner that exposures to both on-site and off-site personnel are maintained at levels that are As Low As Reasonably Achievable (ALARA), that environmental contamination is held to a minimum, and that loss of equipment due to radioactive contamination is kept acceptably low.

In order to achieve this goal the requirements of 10 CFR 20, "Standards for Protection Against Radiation," the guidelines contained in Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposure at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," and the guidance given in the "Donald C. Cook Nuclear Plant Radiation Protection Manual," will be followed as applicable to the Steam Generator Repair Project. As such, the entire repair project will be preplanned to the extent necessary. Mockups and training will be used as appropriate to minimize outage time and radiation exposure. Decontamination and other exposure saving techniques will be used where their benefit in achieving a level of protection equals or exceeds the cost of implementation. Scaffolding and other components will be prefabricated to the extent practical to minimize radiation exposure and outage time. Details of the Radiological Protection Program can be found in Section 3.8.1, "Project Radiation Protection Plan" of this report.

1.3.8 Off-Site Radiological Considerations

Controls will be in effect during all phases of the Steam Generator Repair Project to limit doses and the spread of radioactive contamination. As a result of these controls, both physical and administrative, evaluations of projected liquid and gaseous releases generated by the Steam Generator Repair Project indicate that these releases should be less than those during comparable periods of normal operations. After repair, normal releases should be reduced as a result of enhanced steam generator integrity. A detailed discussion of potential off-site radiological releases is contained in Section 7.4.

1.3.9 Safety Aspects

The potential impact of the repaired steam generators on each appropriate accident analyzed in the FSAR has been evaluated (see Section 6). The essentially identical design of the replacement components with improved thermal-hydraulics, corrosion resistance, and maintainability, does not result in any adverse changes in the plant operating conditions discussed in the FSAR. The FSAR accident analyses remain valid for the repaired steam generators. Therefore, no unreviewed safety questions exist due to operation of the plant with the repaired steam generators.

In addition to the FSAR accidents, analyses of the safety aspects of heavy loads and shared systems have been conducted (see Section 6.2). All analyses and reviews conducted to date conclude that the repair project and subsequent operation can be accomplished without undue risk to the health and safety of the general public or to personnel engaged in the repair project and will not involve an unreviewed safety question.

1.3.10 Steam Generator Lower Assembly Disposal

The repair project and ultimate disposal of the existing lower assemblies are separable issues. Section 7.7.5 discusses the disposal options evaluated and demonstrates the feasibility of disposal. The steam generator lower assemblies will be stored in a temporary on-site steam generator storage facility until shipment to a burial facility is practical or until unit decommissioning.

1.3.11 Environmental Impacts

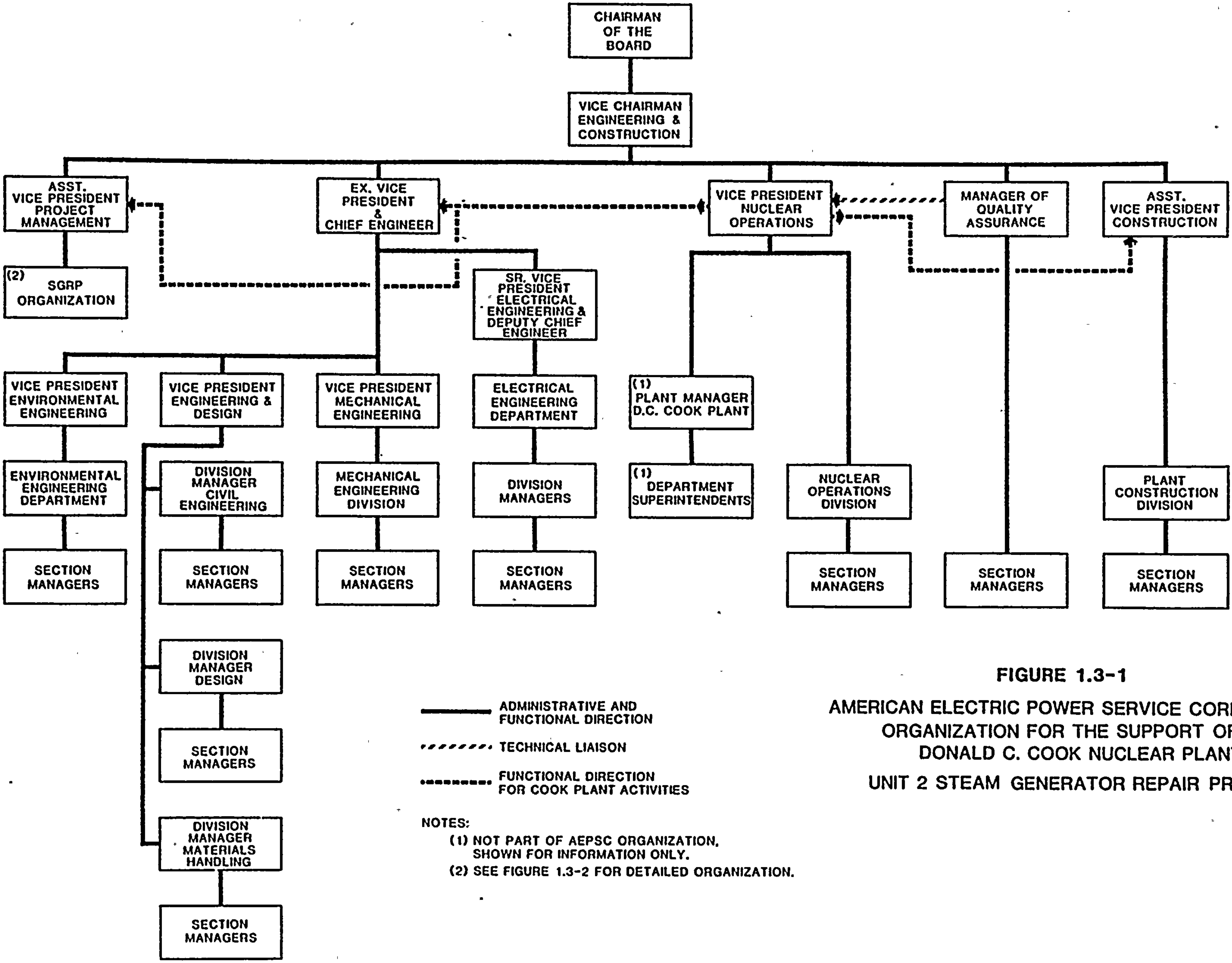
Section 7 addresses the effects the project will have on the environment, and includes considered alternatives to the proposed activity. The most significant environmental impact is the occupational radiation exposure associated with the proposed repair project. No construction activities taking place in areas not previously disturbed.

1.3.12 Security

Security for the plant during the repair project will be governed by the "Modified Amended Security Plan" (MASP). Requirements specified in the MASP and applicable NRC requirements will be strictly adhered to during the repair project. Some changes and/or additions to the security facilities may occur to facilitate the ingress and egress of the increased work force required during the repair project. Additional discussion of plant security during the repair project can be found in Section 4, "Plant Security."

1.3.13 Significant Hazards Considerations

It is concluded, as discussed in Section 6.3, "Analysis of Significant Hazards Consideration," that the proposed repair to the Donald C. Cook Unit 2 steam generators does not involve significant hazards considerations. This conclusion is based on two important aspects of the proposed repair project. First, other than enhancements in the design of the lower assemblies to preclude future tube degradation, no significant changes will be made to the plant or plant systems as the result of the repair. Second, the repair project will be conducted using technologies that are well developed and previously demonstrated. Therefore, as stated above, it is concluded that there are no significant hazards considerations involved in the Steam Generator Repair Project.



NOTES:

(1) NOT PART OF AEPSC ORGANIZATION, SHOWN FOR INFORMATION ONLY.

(2) SEE FIGURE 1.3-2 FOR DETAILED ORGANIZATION.

FIGURE 1.3-1

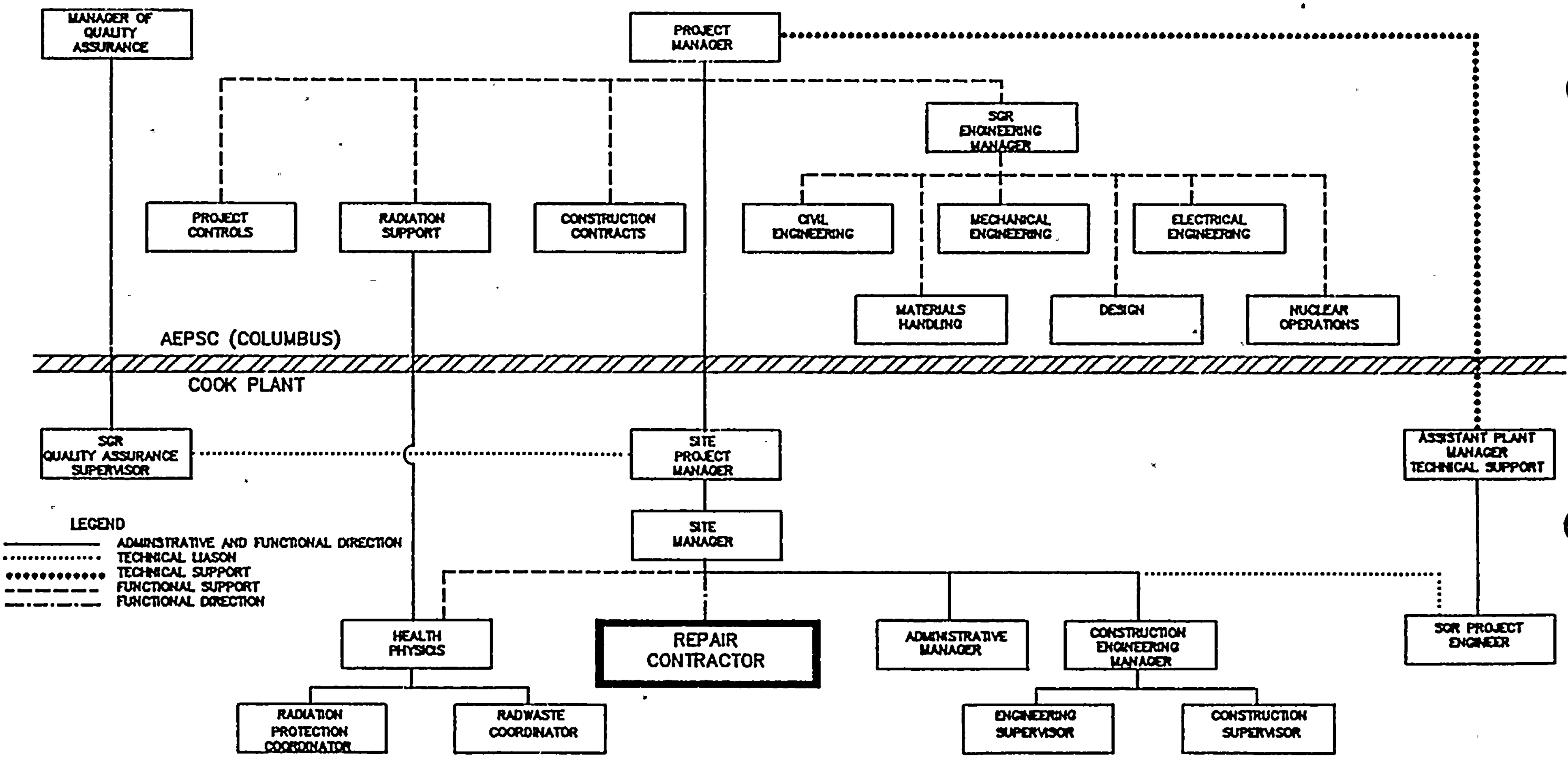
AMERICAN ELECTRIC POWER SERVICE CORPORATION ORGANIZATION FOR THE SUPPORT OF THE DONALD C. COOK NUCLEAR PLANT

UNIT 2 STEAM GENERATOR REPAIR PROJECT

1

FIGURE 1.3-2

DONALD C. COOK UNIT 2 STEAM GENERATOR REPAIR PROJECT ORGANIZATION



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SECTION 2 - REPLACEMENT COMPONENT DESIGN

2.1 General Description

The repaired steam generators are similar in design and are functionally the same as the original steam generators. The steam generators are vertical, shell and U-tube heat exchangers with integral moisture separating equipment as described in Section 4.2. of the FSAR. Certain design changes have been made in the repaired steam generators which address the operating experience of the original steam generators as described in sections 1.1 and 1.2 of this report, and which enhance the overall reliability and maintainability of the steam generators. The major changes are summarized below.

Only the lower assembly, up to and including the transition cone, will actually be replaced. The upper assembly, which includes the steam separating equipment, will be reused after refurbishing with a new feedring with Inconel 600 J-nozzles and upgrading the separators to enhance performance. Refurbishment of the upper assembly will be performed in the field. 1

The tubes in the lower bundle are made of thermally treated Inconel 690, which has superior resistance to intergranular corrosion and cracking. Increased heat transfer area is provided by an increase in the number of tubes, achieved by reducing the pitch of the tubes. The entire length of each tube in the tubesheet is hydraulically expanded to minimize the crevice which might otherwise accumulate corrosive materials. The inside eight rows of tubes, because of their tight radius, are heat treated to relieve the residual stresses.

The tubes in the bundle are supported by stainless steel support plates with quatrefoil tube holes. The plates are resistant to corrosion, and the quatrefoil holes minimize accumulation of corrosives between the tubes and the support plates due to dryout. A flow distribution baffle is provided above the tubesheet, which together with tube lane flow blocks, controls the fluid velocities to minimize sludge deposition on the tubesheet. The tube holes in the flow distribution baffle are of octafoil design to minimize sludge accumulation between the tubes and the baffle.

Three sets of anti-vibration bars (AVBs) are installed between adjacent columns of tubes to enhance the support of the tubes in the U-bend region. The tubes and AVBs are dimensionally controlled to provide adequate support of the tubes against fluid induced motion which can lead to wear. The material of the AVBs is type 405 stainless steel, which results in reduced wear of the contacting tubes. In addition, the AVBs are wider than the original design to provide greater contact area between the tubes and AVBs.

Additional features are provided which enhance the overall maintainability of the steam generators. Six handholes are provided: four are below the flow distribution baffle but above the tubesheet (two centered on the tubelane, and two 90 degrees removed from the tubelane), and two of the handholes are located above the flow distribution baffle. The grid location of the tubes is permanently marked on the primary side tubesheet.

Overall, the design of the repaired steam generators provides the same heat transfer performance with an increased margin compared to the original steam generators. This design also provides features which enhance resistance to known degradation mechanisms, and which enhance reliability and maintainability of the repaired steam generators.

2.2 Comparison With Existing Component Design

This section provides a comparison of the replacement lower assemblies with the existing lower assemblies, and shows that the replacements are essentially identical, both physically and functionally, to the existing assemblies.

2.2.1 Physical Comparison

The replacement steam generator lower assemblies are designed and fabricated to be physical duplicates of the original lower assemblies. All major dimensions and orientation angles for both the existing and replacement components are identical. Dry weight, wet weight, and center of gravity will remain essentially the same. Therefore, changes to the existing supports and other plant components will not be necessary.. Comparison of the key items such as outer dimensions and nozzle orientations are provided in Table 2.2-1. Sketches of the repaired steam generator are shown in Figures 2.2-1 and 2.2-2.

Table 2.2-1

COMPARISON BETWEEN THE ORIGINAL AND
REPAIRED STEAM GENERATORS

Item Description	Original	Repaired
Overall Length, in.	521.75	*
Lower Shell Diameter, in.	135.2-136.0	135.4-136.0
Transition Cone Diameter, Max., in.	175.75	*
Primary Nozzle Diameters, ID., in.	31.00	*
Primary Manway Diameters, ID., in.	16.00	*
Blowdown Connections, ID., in.	1.7	2.1
Support Pad Dimensions, in.	9 x 16	*
Primary Nozzles, orientation	**	*
Primary Manways, orientation	**	*
Blowdown Connections, orientation	**	*
Support Pads, orientation	**	*
Dry Weight, lbs. (entire steam generator)	662,000	***
Wet Weight, lbs. (entire steam generator, flooded)	1,094,000	***
Center of Gravity, in. From Steam Generator Support Pads (entire steam generator, dry)	29.2	***

* No change

** All major orientation dimensions and angles for both the original and repaired components are essentially the same as noted on Figure 2.2-3.

*** The dry weight, wet weight, and center of gravity for both the original and repaired components are essentially the same.

2.2.2 Parametric Comparison

The repaired steam generators will have physical, mechanical and thermal characteristics consistent with the original design and safety analysis presented in the FSAR.

The design data comparison for the original steam generators and the repaired steam generators is shown in Table 2.2-2. The thermal performance data for the repaired steam generator will remain essentially the same as the original steam generators.

2.2.3 Materials Comparison

Materials used in the fabrication of the replacement lower assemblies will be identical to those used in the original lower assemblies with the following exceptions:

- o Transition cone and stub barrel material has been changed from ASME SA-533, Grade A Class 1 to ASME SA-508, Class 3;
- o Tube sheet forging material has been changed from ASME SA-508, Class 2 to ASME SA-508, Class 2A.
- o Support plate material has been changed from ASME SA-285, Class C to ASME SA-240, Type 405.
- o The steam generator tube material has been changed from ASME SB-163 Alloy 600 to ASME SB-163 Alloy 690 (Code Case N-20)

These material changes will not compromise the performance of the steam generators. Additional information concerning material changes is provided in Table 2.2-3.

Table 2.2-2

COMPARISON OF DESIGN DATA BETWEEN
THE ORIGINAL AND REPAIRED STEAM GENERATORS
(DESIGN DATA PER STEAM GENERATOR)

	Original	Repaired
Design Pressure, Primary/Secondary, psig	2485/1085	*
Design Temperature, Primary/Secondary, °F	650/600	*
Hydrostatic Test Pressure, Primary, psig	3106	*
Hydrostatic Test Pressure, Secondary, psig	1356	*
Overall Height, ft-in.	67-8	*
Shell OD, lower/upper, in.	135.2-136.0/ 175.9	135.4 - 136.0/*
Shell Thickness, lower/upper, in.	3.25/2.82	*
U-tube OD, in.	0.875	*
Tube Wall Thickness, (nominal) in.	0.050	*
Number of U-Tubes	3388	3592
Number of Manways/ID, in.	4/16	*
Number of Handholes/ID, in.	2/6	6/6
Tube Length, average effective length, ft-in.	66-4	66-3
Total Heat Transfer Surface Area, ft^2	51,500	54,500
Reactor Coolant Water Volume, ft^3	1080	1133
Secondary Side Volume, ft^3	5730	5666
Primary Side Conditions at 100% Load:		
Heat Transfer Rate, BTU/hr	2903×10^6	2910×10^6
Coolant Inlet Temperature, °F	606.4	*
Coolant Outlet Temperature, °F	541.0	*
Flow Rate, lb/hr	33.7×10^6	*
Pressure Loss, psi	30.4	26.24
Fluid Heat Content, BTU	28.2×10^6	29.6×10^6

Table 2.2-2 (cont.)

	Original	Repaired
Secondary Side Conditions at 100% Load:		
Steam Temperature, ^{o}F	521	*
Steam Flow, lb/hr	3.685×10^{6}	*
Steam Pressure, psig	820	*
Maximum Moisture Carryover, wt%	0.25	0.15
Fouling Factor, hr.-ft^{2}-^{o}F/BTU	0.00005	*
Fluid Heat Content, BTU	6.17×10^{7}	5.83×10^{7}
Feedwater Temperature, 100% Load, ^{o}F	431	*
Secondary Side Water Volume, No Load, Ft^{3}	3403	3338
Secondary Side Water Volume 100% Load, Ft^{3}	2228	2080

* No Change

Table 2.2-3

COMPARISON OF MATERIALS OF CONSTRUCTION BETWEEN THE ORIGINAL AND REPAIRED STEAM GENERATORS

	Original	Repaired
Plates (shell courses)	ASME SA-533 Grade A Class 1	ASME SA-533, Grade A Class 2
Transition Cone	ASME SA-533, Grade A Class 1	ASME SA-508 Class 3
Stub Barrel	ASME SA-533, Grade A Class 1	ASME SA-508 Class 3
Tube Sheet Forging	ASME SA-508, Class 2	ASME SA-508 Class 2A
Channel Head Casting	ASME SA-216, Grade WCC	*
Support Plates	ASME SA-285, Grade C	ASME SA-240, Type 405
Channel Head Cladding	ASME SFA-5.9, Class ER 309L	*
Tubesheet Cladding	ASME SFA-5.14, Class ERNiCr-3	*
Tubes	ASME SB-163, Alloy 600	ASME SB-163, Alloy 690

* No Change

2.3 **Component Design Improvements**

2.3.1 Design Improvements to Minimize Potential for Tube Degradation

In order to place the discussion of design features in the proper context, it is helpful to first review problem areas in the existing design. These problem areas are illustrated in Figure 2.3-1 and are discussed below.

- Intergranular Corrosion (IGC) in the Tubesheet Region and at Support Plate Intersections IGC is a broad term for dissolution of grain boundaries in the microstructure of metal alloys due to a combination chemical attack and stress. Several features of the Donald C. Cook Unit 2 original steam generators make them susceptible to IGC damage:

    - The tubing, mill annealed Inconel 600, has a microstructure that can experience IGC in an alkaline environment.

    - The tube-to-tubesheet joint is sealed only in the bottom 2.5 inches of the 21 inch thick tubesheet, leaving an annular crevice about 18.5 inches deep, with a nominal radial gap of 8 mils. This open crevice around each tube provides a vessel for alternate wetting/drying action to concentrate any impurities in the feedwater, and can form an aggressive alkaline solution which comes in contact with the tube.

- The 0.75 inch thick carbon steel support plates have round tube holes with a nominal radial clearance of 8 mils. As with the tubesheet crevice, this support plate crevice allows concentration of impurities and provides a localized aggressive environment at the tube-to-tube support plate intersections. The presence of flow holes in the plates tends to minimize flow around the tubes, and allows the aggressive environment to exist. Additionally, corrosion of the carbon steel plates completely fills the gap in some locations and squeezes the tube, thereby adding stress and contributing to IGC.

- Thermalhydraulic characteristics result in dryout and sludge accumulation on the tubesheet, thus providing another localized aggressive environment just above the tubesheet surface.

IGC in the tubesheet region and at support plate intersections is a pervasive problem in the Donald C. Cook Unit 2 original steam generators, and is the primary reason for repairing the steam generators.

o <u>Mechanical Wear at Tube-To-Antivibration Bar (AVB) Intersections</u>

Excessive tube vibration in the U-bend region is prevented by sets of square AVBs inserted between columns of tubes. Some fretting wear at tube-to-AVB intersections has occurred for several reasons:

- The chrome-plated Inconel used for the Donald C. Cook Unit 2 AVBs does not provide the best wear couple with the Inconel tubes.

- The free span length of tubing resulting from the spacing of AVBs is quite long in some of the larger radius U-bends, making those tubes more susceptible to vibration.

- The design and manufacturing tolerances employed result in excessive gaps in some locations.

o <u>Primary Side Stress Cracking of Tight Radius U-Bends</u> Cracking at the tangent points and the apex of Row 1 tubes was a problem until all the Donald C. Cook Unit 2 Row 1 tubes were plugged as a preventive measure in 1984. This has been a common problem in the industry, and is caused by two factors:

  - The mill annealed Inconel 600 is susceptible to primary water stress corrosion cracking.

  - The very tight bend radius of the Row 1 and 2 tubes results in high residual stresses due to the bending operation.

Design improvements which address these problem areas are: 1) better tube material, 2) sealed tube-to-tubesheet joint, 3) stainless steel quatrefoil support plates, 4) optimized AVB installation, 5) flow distribution baffle, and 6) stress reduction in tight radius U-bends. Each improvement is discussed below.

2.3.1.1 Better Tube Material

Tube material selected for the Donald C. Cook Unit 2 replacement steam generators is SB-163 Alloy 690, C.C.N-20 or Inconel 690, thermally treated (I-690 TT). I-690 TT with optimum thermal treatment (annealing at about 1900^{o}F, followed by chromium replenishment at 1320-1350^{o}F for 10 hours) has been shown to be several orders of magnitude more corrosion-resistant than mill annealed Inconel 600. It is slightly more resistant to caustic induced IGC than the other state-of-the-art material, thermally treated Inconel 600 (I-600 TT), and is much more resistant to primary water stress corrosion cracking than I-600 TT.

2.3.1.2 Sealed Tube-to-Tubesheet Joint

Following the typical tube welding sequence consisting of tube leg insertion into tubesheet tube holes, tube leg tack expansion, and tube end welding, the tube welds of the Donald C. Cook Unit 2 replacement lower assemblies will be helium gas leak tested and the tube legs then hydraulically expanded to the tube hole diameter for essentially the full thickness of the tubesheet. Full-depth expansion minimizes the potential for crevice boiling and reduces the potential for the buildup of impurities in the tube-to-tubesheet hole crevice region.

In addition, hydraulic expansion minimizes residual tube stresses in the transition zone between expanded and unexpanded sections of the tube. The original steam generator tubes were partially expanded in the lower end of the tubesheet holes using a mechanical rolling process.

2.3.1.3 Stainless Steel Quatrefoil Support Plates

The tube support plate material will be ferritic stainless steel, which is more resistant to corrosion than carbon steel. Patch plates will not be used in the replacement lower assemblies' support plates.

The quatrefoil tube support plate hole design features four flow lobes and four support lands. The lands support the tubing and the lobes provide a path for water to flow adjacent to the tube. The quatrefoil design directs the flow along the tubes in a way that minimizes steam formation and chemical concentration at tube-to-tube support plate intersections. The quatrefoil support plate results in higher average velocities adjacent to the tubes than the original lower assembly support plates, which feature circulation flow holes between tube holes. The quatrefoil support plate minimizes sludge deposition. The combination of high velocities in the support plate region and corrosion resistant material should minimize the potential for tube corrosion in the vicinity of the support plates.

2.3.1.4 Optimized AVB Installation

Three sets of AVBs are used in the replacement steam generator tube bundle U-bend region. The AVBs are fabricated from SA-240 Type 405 stainless steel which features enhanced wear characteristics when compared to the original chrome plated two-set AVB assembly. The AVB design and tube fabrication controls combine to minimize tube-to-AVB clearances. This, together with the material change and an increase in the width of the AVBs, will minimize the wear potential between the AVBs and tubes.

2.3.1.5 Flow Distribution Baffle

A flow distribution baffle (FDB) of ferritic stainless steel is located approximately 23 inches above the tubesheet. The FDB has octafoil broached tube holes surrounding an open central cutout. The FDB directs flow radially across the tubesheet and up the center of the bundle through the central cutout, thus increasing lateral flow velocities across the tubesheet. This design causes any sludge to deposit near the center of the bundle at the blowdown offtake. In addition, the octafoil hole design reduces the potential for sludge accumulation at the FDB.

2.3.1.6 Stress Reduction in Tight Radius U-Bends

The Donald C. Cook Unit 2 replacement lower assemblies feature a minimum U-bend radius of 3.14 inches; the original assemblies' minimum bend radius of 2.19 inches resulted in higher residual stresses from bending. Additionally, the U-bend regions of the eight innermost tube rows of the replacement lower assemblies are stress relieved after bending to further reduce residual stresses.

2.3.2 Design Improvements to Increase Performance

Several modification and refinements have been incorporated into recent steam generator designs in order to enhance their thermal/hydraulic characteristics. These refinements have evolved from knowledge gained through operating experience and ongoing research and development programs. Performance enhancements included in the Donald C. Cook Unit 2 Model 51F replacement design are discussed below.

2.3.2.1 Tube End Condition

The replacement lower assemblies' tubes will be flush with the tube hole opening when welded to the tubesheet cladding. By eliminating the protruding tube ends and tube fillet welds of the original design, entry pressure losses are reduced, resulting in a lower pressure drop in the primary loop.

2.3.2.2 Controlled Tube Wall Thickness

To allow optimal thermal hydraulic performance, the heat transfer tubing is procured with a wall thickness control. A maximum average wall thickness of 0.050 inches will be provided for each replacement lower assembly tube set.

2.3.2.3 Upper Assembly Internals Modifications

Additional secondary separator drains and middle deck plate relief openings will be added to enhance the drainage of separated water in the steam drum. Also, steam vents will be installed inside the existing access openings in the middle deck plates to minimize the re-entrainment of separated water. These modifications will increase the quality of the steam exiting the steam nozzle.

2.3.3 <u>Design Improvements to Enhance Maintainability and Reliability</u>

Operating experience has resulted in certain changes in design that are directed towards enhancing the maintainability and reliability of the repaired steam generators. Each improvement is discussed below.

2.3.3.1 Replacement Feedring with Inconel J-Nozzles

New feedrings will be installed which incorporate Inconel J-nozzles. Inconel J-nozzles reduce the potential for erosion of the nozzles and enhance the reliability of the steam generators.

2.3.3.2 Tube End Welds and Tubesheet Marking

The tube ends are welded flush with the tube plate cladding, thereby minimizing locations for crud buildup. The tube end locations will be marked on a 2 x 2 row by column pattern to facilitate tube identification.

2.3.3.3 Access Ports

The replacement lower assemblies will include additional access ports. Four 6-inch access ports will be located slightly above the tubesheet approximately 90 degrees apart, with two located on the tube lane. Two additional 6-inch ports will be located between the FDB and the first tube support plate, bringing the total number of access ports to 6. This access port arrangement will enhance inspections and facilitate sludge lancing.

2.3.3.4 Inspection Ports

Two 4-inch inspection ports are located on the transition cone lower shell upstand at an elevation slightly above the top tube support plate. The ports, which are located on the tube lane centerline, will facilitate inspection of the top tube support plate and the tubing U-bend area. Row 1 tubes are directly observable through these inspection ports.

2.3.3.5 Electropolishing and Passivation

To determine the acceptability of electropolishing/passivation, various industry papers were reviewed to determine the state-of-the-art and experience with respect to both methods. Additionally, the steam generator manufacturer was consulted. It has been determined from these information sources that we cannot make a definite recommendation to perform either method.

The inability of the manufacturer to perform these processes in the shop has precluded electropolishing/passivation prior to steam generator shipment. Application of field techniques are still being evaluated. Initial assessments indicate that these processes are not fully qualified, not economically justified and not endorsed by the vessel manufacturer. 4

2.4 Codes and Standards

The original steam generators were designed, fabricated, inspected and tested as Class A component in conformance with the 1968 ASME Boiler and Pressure Vessel Code, Section III, plus Addenda through Winter 1968 and Code Cases 1401 and 1498. All pressure boundary materials and weld filler material welded conformed to specifications set forth by Section III of the ASME code. Non-pressure retaining parts on the secondary side were in accordance with applicable ASTM or ASME material specifications.

The design, material, fabrication, inspection, examination, and testing of replacement steam generator components or assemblies furnished by Westinghouse are in accordance with the codes and standards, including all applicable addenda, referenced herein.

The original "N" stamp will be maintained for the repaired steam generators. The replacement lower assemblies will be "NPT" stamped. Field closure welds will be "NA" stamped.

Field work performed as part of the Steam Generator Repair Project will be in accordance with the applicable Industry Codes and Standards listed in Table 3.2-1 and the applicable USNRC Regulatory Guides listed in Table 3.2-2.

2.4.1 Industry Codes and Standards

- ASME Boiler and Pressure Vessel Code, Section II, "Material Specifications," 1983 Edition plus Addenda through Summer 1984.

o ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Vessels/Rules for Construction of Nuclear Power Plant Components," edition and addenda as discussed below.

Westinghouse Nuclear Components Division, acting as I&MECo's (Owner's) Designee, has designed for manufacture components for replacement of the steam generator lower assemblies in D. C. Cook Unit 2. The replacement components are designed and are being fabricated to meet Section III, Division 1 - Subsection NB (Class 1 Components), 1983 Edition plus Addenda through Summer 1984. Manufacture to this Code
edition represents an enhancement to the requirements of the 1
original Construction Code - Section III, 1968 Edition plus Addenda through Winter 1968.

To be consistent with reuse of some existing steam generator components (specifically the upper assemblies), the Code-required Stress Report will consider the repaired vessel as a whole and will conform to the original Stress Report, which was prepared, reviewed, and certified in accordance with Section III, 1968 Edition plus Addenda through Winter 1968, with portions updated to consider requirements of Section III, 1971 Edition. In general, material strength properties from Section III, 1968 Edition plus Addenda through Winter 1968 will be used in the Stress Report. For materials whose strength properties do not exist in the earlier edition, the material strength

properties used in the Stress Report shall be determined from Section III, 1983 Edition plus Addenda through Summer 1984. Additionally, in cases where the later edition specifies more conservative material strength properties than the earlier edition, the more conservative values shall be used in the Stress Report.

- ASME Boiler and Pressure Vessel Code, Section IX, "Welding and Brazing Qualification," edition and addenda in effect at time of procedure qualification and welding procedure specification generation. 1

2.4.2 USNRC Regulatory Guides

- Regulatory Guide 1.31 - Control of Ferrite Content in Stainless Steel Weld Metal (Rev. 3, April 1978).

  The requirements of this guide are now covered by ASME Section III. All manufacturing activities relating to the steam generator repair project will be in compliance with Regulatory Guide 1.31.

- Regulatory Guide 1.37 - Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (March, 1973).

  Westinghouse complies with this guide, and the requirements of ANSI N45.2.1 referenced in the guide.

- USNRC Regulatory Guide 1.50, Rev. 0(5/73), "Control of Preheat Temperature for Welding Low-Alloy Steel."

  <u>Exception</u> - Westinghouse does not comply with Regulatory Position No. 2, which states "preheat temperature should be maintained until a post-weld heat treatment has been performed." Due to the size and weight of component assemblies and subassemblies and the configuration of the post-weld heating furnace, maintaining preheat until post-weld treatment begins is not possible. In lieu of this practice, Westinghouse procedures require a "hydrogen bake" cycle (raising preheat to minimum 400°F and holding for minimum 4 hours) prior to lowering temperature to ambient. 1

- <u>Regulatory Guide 1.84</u> - Design and Fabrication Code Case Acceptability - ASME Section III Division I (Rev. 24, July 1986).

  <u>Exception</u> - Westinghouse has not applied, and does not expect to apply, any code cases in the fabrication effort.

- <u>Regulatory Guide 1.85</u> - Material Code Case Acceptability - ASME Section III Division I (Rev. 24, July 1986).

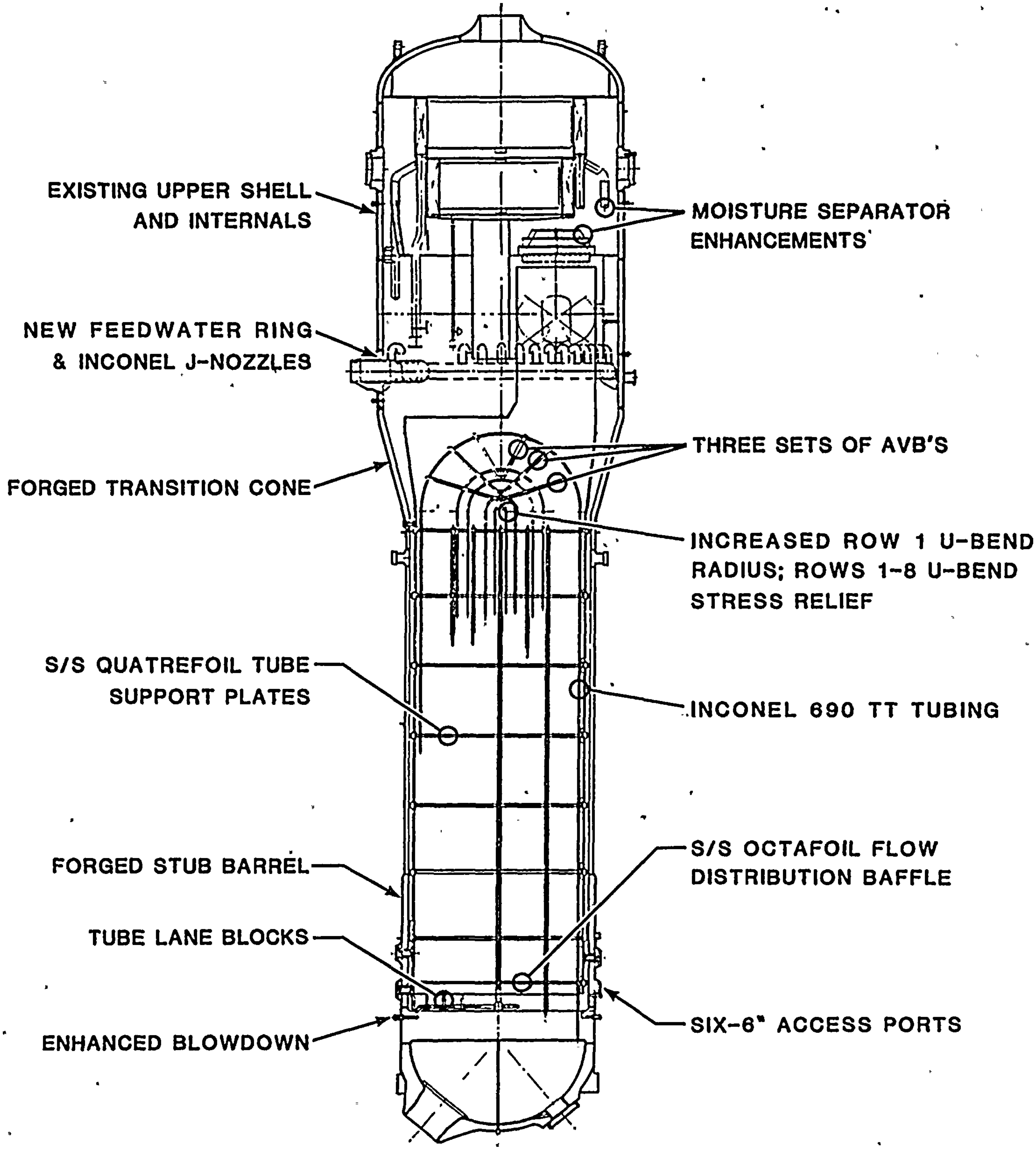
  <u>Exception</u> - Code Case N-20, which covers the use of Inconel 690 tubing and which is accepted as part of Regulatory Position No. 1.A(2), will be applied. No other code case is expected to apply.

2.5 Shop Tests and Inspections

The tests and inspections required by the ASME Code, Section III will 1
be conducted during the fabrication of the replacement steam generator
lower assemblies. In addition to these ASME requirements, further
tests and inspections will be conducted at the fabrication facility.
After the tubing installation is completed a gas leak test will be
performed to demonstrate the integrity of the tube-to-tubesheet welds.
The primary side of the steam generator will be hydrotested at the
shop in accordance with approved procedures. I&MECo will arrange to
perform audit and surveillance functions related to fabrication and 1
shop testing.

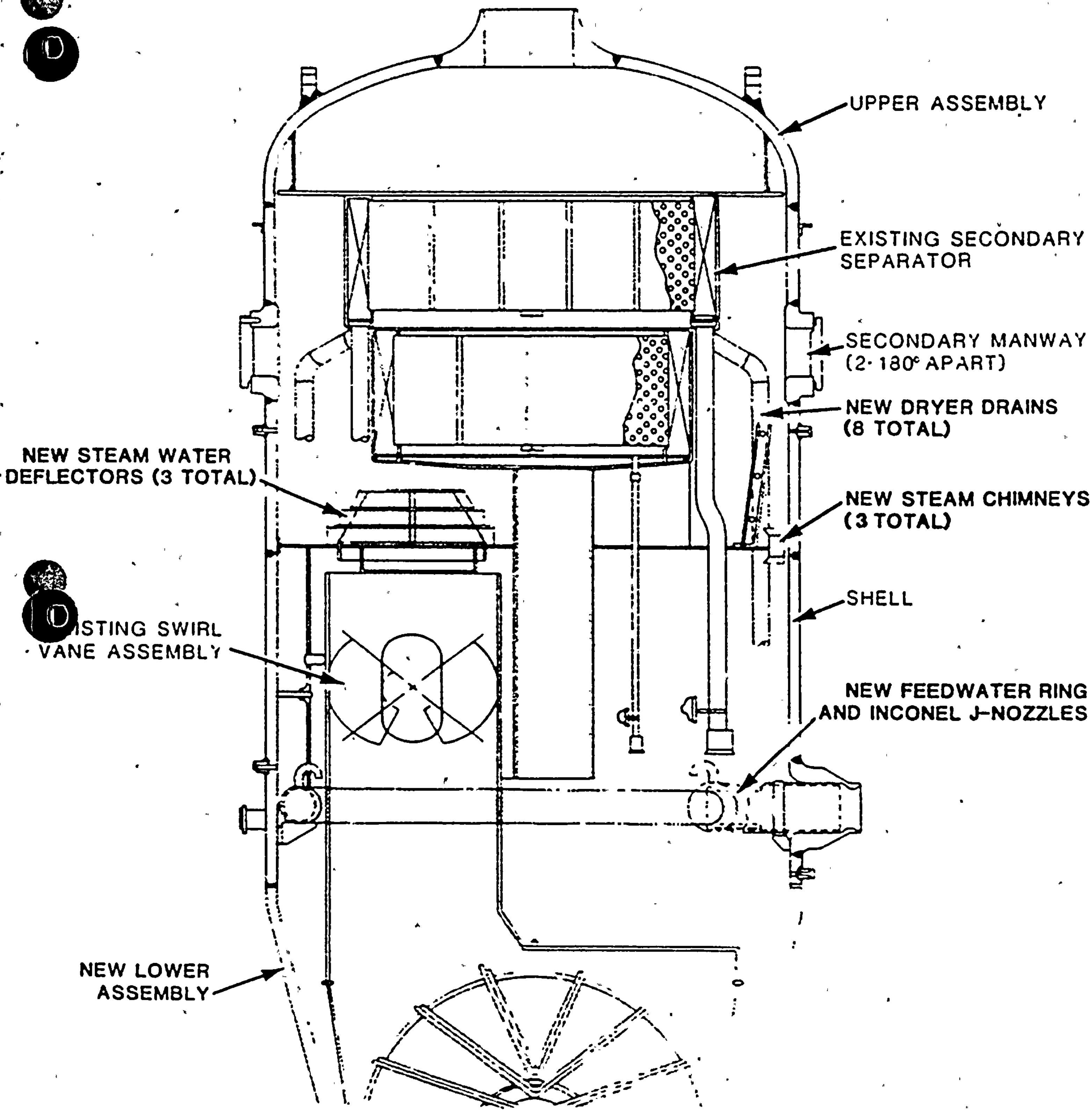
FIGURE 2.2-1

REPAIRED STEAM GENERATOR GENERAL ARRANGEMENT



MODEL 51F STEAM GENERATOR

FIGURE 2.2-2
MODIFICATIONS TO UPPER ASSEMBLY INTERNALS



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