NRC FORM 366 (7-77)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

CONTROL BLOCK: | | | | | | | (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

Field	Value
0 1 Licensee Code (2)	M I D C C 2
License Number (3)	0 0 0 0 0 0 0 0 0 0 0
License Type (4)	4 1 1 1 1
CAT (5)	
CON'T 0 1 Report Source (6)	L
Docket Number (7)	0 5 0 0 0 3 1 6
Event Date (8)	1 2 2 3 7 8
Report Date (9)	0 1 0 5 7 9

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 WITH THE UNIT IN MODE 1 OPERATION, BOTH CONTROL ROOM PRESSURIZATION FANS WERE

0 3 DETERMINED TO BE INOPERABLE CONTRARY TO THE REQUIREMENTS OF TECH. SPEC. 3.7.5.1.

0 4 ONE FAN WAS RETURNED TO OPERABLE STATUS IN 56 MINUTES AND THE OTHER FAN IN 62

0 5 MINUTES. NO PROBABLE CONSEQUENCES TO THE HEALTH AND SAFETY OF THE PUBLIC.

0 6

0 7

0 8

0 9	System Code (11)	Cause Code (12)	Cause Subcode (13)	Component Code (14)	Comp. Subcode (15)	Valve Subcode (16)
	A A	A	E	C K T B R K	E	Z

LER/RO Report Number (17)	Event Year		Sequential Report No.		Occurrence Code	Report Type		Revision No.
	7 8	—	1 0 1	/	0 1	T	—	0

Action Taken (18)	Future Action (19)	Effect on Plant (20)	Shutdown Method (21)	Hours (22)	Attachment Submitted (23)	NPRD-4 Form Sub. (24)	Prime Comp. Supplier (25)	Component Manufacturer (26)
B	H	Z	Z	0 0 0 0	Y	N	Z	A 1 9 8

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 WATER SEEPED INTO THE LOCAL ACTUATION SWITCH OF FIRE PROTECTION WATER SPRAY FOR

1 1 THE CHARCOAL FILTER ASSOCIATED WITH THE CONTROL ROOM PRESSURIZATION FANS. THIS

1 2 CAUSED ACTUATION OF THE FIRE SPRAY SYSTEM AND LOCKOUT OF THE FANS. THE SPRAY

1 3 ACTUATION VALVED OFF BUT THE FANS WOULD NOT RESET UNTIL THE LOCAL CONTROL BOX WAS

1 4 DRAINED AND DRYED OUT. THE WATER SEEPAGE INTO THE CONTROL BOX (CONTINUED NEXT PG.)

1 5	Facility Status (28)	% Power (29)	Other Status (30)	Method of Discovery (31)	Discovery Description (32)
	E	1 0 0	NA	A	ALARM OF FIRE SPRAY OPERATIONS

1 6	Activity Released (33)	Content of Release (34)	Amount of Activity (35)	Location of Release (36)
	Z	Z	NA	NA

PERSONNEL EXPOSURES

1 7	Number (37)	Type (38)	Description (39)
	0 0 0	Z	NA

PERSONNEL INJURIES

1 8	Number (40)	Description (41)
	0 0 0	NA

LOSS OF OR DAMAGE TO FACILITY (43)

1 9	Type (42)	Description
	Z	NA

PUBLICITY

2 0	Issued (44)	Description (45)	NRC USE ONLY
	N	NA	

7901150092

NAME OF PREPARER R. S. Lease PHONE: (616) 465-5901 X-313

GPO 917-926

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (CONTINUED):

WAS DIRECTLY TRACED TO A CONCRETE CORE DRILLING OPERATION BY CONSTRUCTION PERSONNEL. THE CONSTRUCTION MANAGER HAS ISSUED A LETTER TO ALL CONSTRUCTION CONTRACTORS AND CONSTRUCTION ENGINEERING SECTION HEADS DIRECTING THEM THAT WATER RUN OFF FROM ALL CONSTRUCTION JOBS MUST BE DIRECTED AND CONTROLLED.

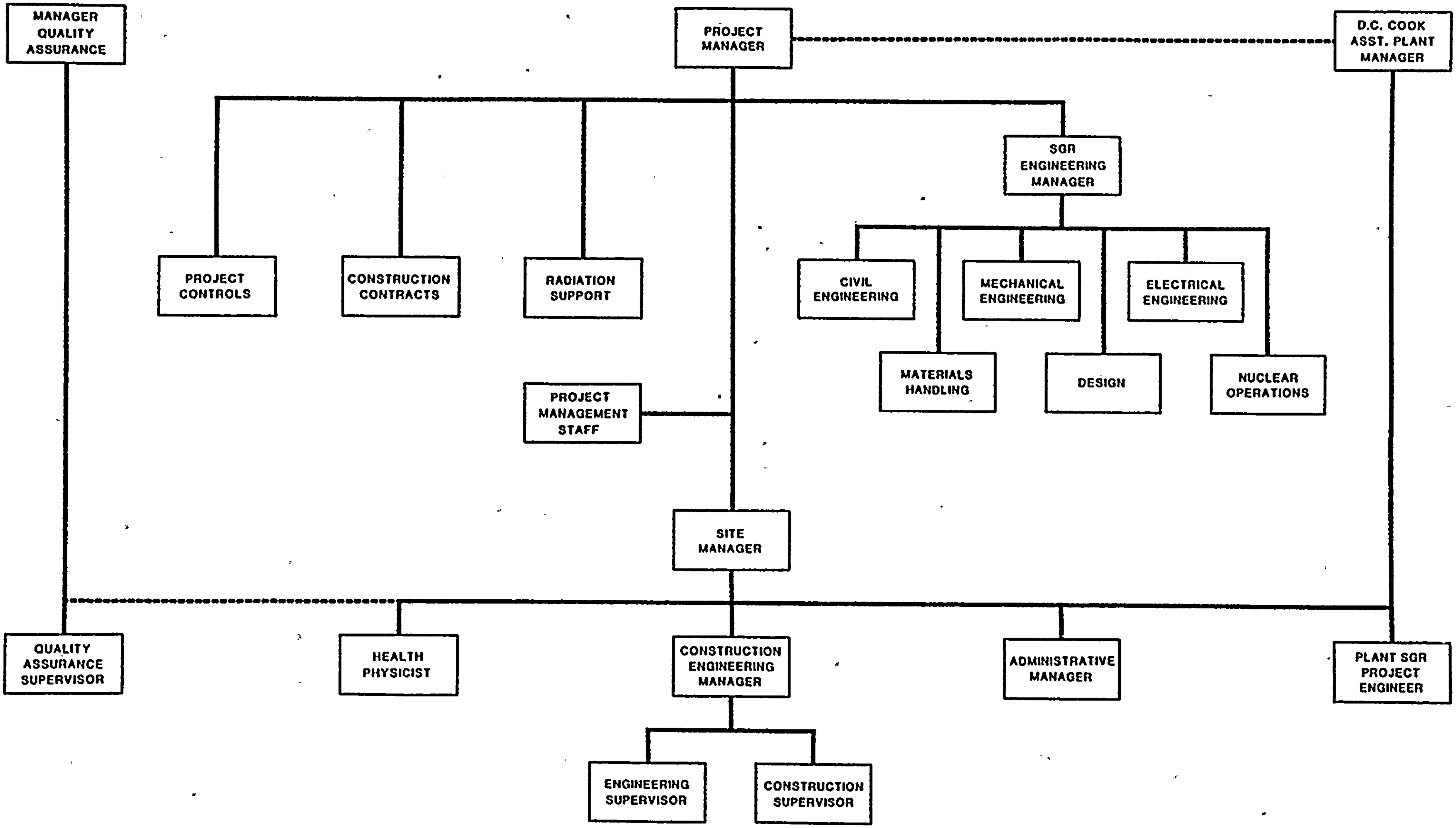
RECORD OF REVISIONS

Superseded per Rev 4 to Steam Generator Repair Rpt dtd 10/20/87

REVISION NO.	REVISION DATE	DATE ISSUED	ISSUED BY
0	November 4, 1986	November 4, 1986	T. G. Harshbarger
1	March 30, 1987	March 30, 1987	T. G. Harshbarger
2	July 24, 1987	July 24, 1987	T. G. Harshbarger
3	September 25, 1987	September 25, 1987	T. G. Harshbarger
4	October 20, 1987	October 20, 1987	T. G. Harshbarger

LIST OF SUPPLEMENTS

FIGURE 1.3-2
DONALD C. COOK UNIT 2
STEAM GENERATOR REPAIR PROJECT
ORGANIZATION CHART



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TA[illegible]-1

INDUSTRY CODES AND STANDARDS APPLICABLE TO THE STEAM GENERATOR REPAIR PROJECT

CODE OR STANDARD	ADDITIONAL INFORMATION/EXCEPTION
ACI 301-84, "Specifications for Structural Concrete Buildings, Chapters 2 and 3."	Exception: Mix proportions shall be selected (1) utilizing laboratory or field trial batches, (2) previous satisfactory performance on similar work using the same or similar materials, or (3) prior experience with these or similar materials to provide concrete of the required strength, durability, workability, economy, etc. . .
ACI 304-85, "Recommended Practices for Measuring, Mixing, Transporting, and Placing Concrete."	
ACI 315-80, "Details and Detailing of Concrete Reinforcement."	
ACI 308-81, "Recommended Practice for Curing Concrete."	Exception: Curing shall be for a period of seven (7) days or until standard cured cylinders reach a comprehensive strength of 3500 PSI, whichever is first. Adherence to this criteria shall be sufficient to preclude testing for "Evaluation of Procedures," "Curing Criteria Effectiveness" or "Maturity Factor Basis."
ACI 318-83, "Building Code Requirements for Reinforced Concrete, Chapters 3, 4, and 5."	Exception: Mix proportions shall be selected (1) utilizing laboratory or field trial batches, (2) previous satisfactory performance on similar work using the same or similar materials, or (3) prior experience with these or similar materials to provide concrete of the required strength, durability, workability, economy, etc.
American Welding Society D.1.1-1986, "Structural Welding Code Steel."	
American Welding Society D.1.3.-1981, "Structural Welding Code, Sheet Steel."	
ASME Boiler and Pressure Vessel Code, Section II, "Material Specifications," edition and addenda in use at time of material procurement.	

TABLE 3.2-[illegible] (Continued)

CODE OR STANDARD	ADDITIONAL INFORMATION/EXCEPTION
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. The original Construction code for D. C. Cook Unit 2 nuclear vessels is Section III, 1968 Edition plus Addenda through Winter 1968, and for piping components is ANSI B31.1-1967 and ANSI B31.7-1969. As allowed by ASME Section XI, Subarticle IWA-7210, selected portions of the original Construction Codes dealing with installation and testing will be updated to applicable portions of Section III, 1983 Edition plus Addenda through Summer 1984.	<u>Exceptions</u>: - Consistent with the plant design basis, fracture toughness requirements will not apply. - N-stamping of fabricated piping components will not be required. 2
ASME Boiler and Pressure Vessel Code, Section IX, "Welding and Brazing Qualifications," edition and addenda in use at time of procedure qualification.	
ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1983 Edition plus Addenda through Summer 1983.	<u>Exception</u>: - Consistent with the plant design basis, fracture toughness requirements will not apply.
ANSI B31.1, "Power Piping", edition and addenda in use at time of contract award for field piping services.	<u>Exception</u>: - This code applies only to power piping not classified under ASME Section III, Division 1.
ANSI N45.2 - 1977 Quality Assurance Program Requirements for Nuclear Facilities	2
USAS (ANSI) B31.1-1967, "Power Piping". USAS (ANSI) B31.7-1969, "Nuclear Power Piping".	<u>Exception</u>: - As noted under ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Vessels/Rules for Construction of Nuclear Power Plant Components" above, these codes represent the original Construction Code for

CODE OR STANDARD	ADDITIONAL INFORMATION/EXCEPTION
	nuclear piping components. Portions dealing with materials and fabrication for new nuclear pressure retaining components, and installation and testing of all nuclear pressure retaining components, will be updated to ASME Section III, with the exception that fracture toughness requirements will not apply. The piping design basis and any additional design activities relating to nuclear piping systems will be in accordance with USAS (ANSI) B31.1-1967.
ASTM C31 "Standard Method of Making and Curing Concrete Specimens in the Field".	
ASTM C33 "Standard Specification for Coarse Aggregates".	<u>Exceptions</u>: - The average fineness modulus of the fine aggregate may be between 2.5 and 3.0, however individual samples shall not vary more than 0.20 from the average. - Compliance with gradation and fineness modulus requirements for fine aggregate shall consist of 4 out of 5 successive test results meeting the specifications. - Coarse aggregate gradation shall be Number 57, 1 inch x #4. - Coarse aggregate sodium sulfate soundness loss shall be a 10 percent maximum at 5 cycles. - Coarse aggregate Los Angeles Abrasion loss shall be a maximum of 40 percent at 500 revolutions.
ASTM C39 "Test Method for Compressive Strength of Cylindrical Specimens".	
ASTM C40 "Test Method for Organic Impurities in Fine Aggregates for Concrete".	

CODE OR STANDARD	ADDITIONAL INFORMATION/EXCEPTION
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ASTM C39 "Test Method for Compressive Strength of Cylindrical Specimens".	
ASTM C40 "Test Method for Organic Impurities in Fine Aggregates for Concrete".	

TABLE 3.2 (Continued)

CODE OR STANDARD	ADDITIONAL INFORMATION/EXCEPTION
ASTM C88 "Test Method for Soundness of Aggregates by Use of Sodium Sulfate or Magnesium Sulfate".	
ASTM C94 "Standard Specification for Ready Mix Concrete".	
ASTM C117 "Test Method for Materials Finer Than No. 200 Sieve in Mineral Aggregates by Washing".	
ASTM C123 "Test Method for Lightweight Pieces in Aggregate".	
ASTM C127 "Test Method for Specific Gravity and Adsorption of Coarse Aggregate".	
ASTM C128 "Test Method for Specific Gravity and Adsorption for Fine Aggregate".	
ASTM C131 "Test Method of Resistance to Degradation of Small-Size Coarse Aggregate by Abrasion and Impact in the Los Angeles Machine".	
ASTM C136 "Method for Sieve Analysis of Fine and Coarse Aggregates".	
ASTM C138 "Test Method for Unit Weight, Yield, and Air Content (Gravimetric) of Concrete".	<u>Exceptions</u>: - Except strike off bar utilized in lieu of glass plate for unit weight determination. - Except "Yield" and "Air Content (Gravimetric)" portions will not be utilized.
ASTM C142 "Test Method for Clay Lumps and Friable Particles in Aggregate".	
ASTM C143 "Test Method for Slump of Portland Cement Concrete".	
ASTM C150 "Specification for Portland Cement".	<u>Exceptions</u>: - Except cement shall be free of false set when tested in accordance with ASTM C451.

CODE OR STANDARD	ADDITIONAL INFORMATION/EXCEPTION
	- Except total alkalies shall not exceed 0.60 percent by weight when calculated as the percentage of Na_2O plus 0.658 times the percentage of K_2O.
ASTM C172 "Method of Sampling Freshly Mixed Concrete".	
ASTM C231 "Test Method for Air Content of Freshly Mixed Concrete by the Pressure Method".	<u>Exception</u>: - Only the Type B Apparatus shall be utilized.
ASTM C260 "Specifications for Air-Entrained Admixtures for Concrete".	
ASTM C289 "Test Method for Potential Reactivity of Aggregates (Chemical Method)".	
ASTM C309 "Specifications for Liquid Membrane-Forming Compounds for Curing Concrete".	
ASTM C311 "Methods of Sampling and Testing Fly Ash or Natural Pozzolans for Use as a Mineral Admixture in Portland Cement Concrete".	
ASTM C494 "Specification for Chemical Admixtures in Concrete".	
ASTM C566 "Test Method for Total Moisture Content of Aggregate by Drying".	
ASTM C617 "Practice for Capping Cylindrical Concrete Specimens".	
ASTM C618 "Specification for Fly Ash and Rain or Calcined Natural Pozzolan for Use as a Mineral Admixture in Portland Cement Concrete".	
ASTM C702 "Methods for Reducing Field Samples of Aggregate to Testing Size".	

TABLE 3.2-1 (Continued)

CODE OR STANDARD	ADDITIONAL INFORMATION/EXCEPTION
SSPC-SP1 through SP10 - 1982 Steel Structures Painting Council Specifications for Surface Preparation of Steel Surfaces	

Note: 1) All ASTMs are latest edition.

TABLE 3[illegible]2

USNRC REGULATORY GUIDES APPLICABLE TO THE
STEAM GENERATOR REPAIR PROJECT FIELD WORK

REGULATORY GUIDE NUMBER	REGULATORY GUIDE TITLE	REGULATORY GUIDE REVISION	ADDITIONAL INFORMATION/EXCEPTIONS
1.8	Personnel Selection and Training	1-R (9/75)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.
1.26	Quality Group Classification and Standards for Water, Steam, and Rad-waste Containing Components of Nuclear Power Plants	3 (2/76)	Classification of Class 2 and 3 components for the purpose of implementing ASME Section XI requirements was made in accordance with this guide.
Safety Guide 30	Quality Assurance Requirements for Installation, Inspection and Testing of Instrumentation and Electrical Equipment	(8/72)	Committed to in UFSAR, Section 1.7, "QAPD", Appndix. A.
1.31	Control of Ferrite Content in Stainless Steel Weld Metal	3 (4/78)	The requirements of this guide are now covered by ASME Section III. Field work relating to the steam generator repair project will be in compliance with this regulatory guide.
Safety Guide 33	Quality Assurance Program Requirements (Operational)	(11/72)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants	0 (3/73)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.

TABLE 3.2-2 (Continued)

REGULATORY GUIDE NUMBER	REGULATORY GUIDE TITLE	REGULATORY GUIDE REVISION	ADDITIONAL INFORMATION/EXCEPTIONS
1.38	Quality Assurance Requirements for Packing, Shipping, Receiving Storage, and Handling of Items for Water-Cooled Nuclear Power Plants	1 (10/76)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants	1 (10/76)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.
1.44	Control of Sensitized Stainless Steel	0 (5/73)	If applicable to this repair project, the field work will comply to this guide.
1.48	Design Limits and Loading Combinations for Seismic Category I Fluid System Components		This regulatory guide was withdrawn 3/4/85 (see 50FR9732).
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel	0 (5/73)	Project repair work will be performed in compliance with this regulatory guide.
1.54	Quality Assurance Requirements for Protective Coatings Applied	0 (6/73)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A. Exception: Committed only to ANSI N101.4-1972.
1.58	Qualification of Nuclear Power Plant Inspection Examination and Testing Personnel	1 (9/80)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.

TABLE 3.2-[illegible]tinued)

REGULATORY GUIDE NUMBER	REGULATORY GUIDE TITLE	REGULATORY GUIDE REVISION	ADDITIONAL INFORMATION/EXCEPTIONS
1.64	Quality Assurance Requirements for the Design of Nuclear Power Plants	0 (10/73)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.
1.68	Initial Test Program for Water-Cooled Nuclear Power Plants	2 (8/78)	This regulatory guide will be used only for guidance in developing a test program for those components and systems affected by the Steam Generator Repair Project.
1.71	Welder Qualifications for Areas of Limited Accessibility	0 (12/73)	Welders making welds in areas of restricted accessibility will be required to practice and qualify on a similar configuration to the weld being made.
1.74	Quality Assurance Terms and Definitions	0 (2/74)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.
1.88	Collection, Storage, and Maintenance of Nuclear Power Plants Quality Assurance Records	2 (10/76)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.
1.89	Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants	1 (7/84)	Project repair work will be performed in accordance with this regulatory guide.

TABLE 3.2 (Continued)

REGULATORY GUIDE NUMBER	REGULATORY GUIDE TITLE	REGULATORY GUIDE REVISION	ADDITIONAL INFORMATION/EXCEPTIONS
1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants	1 (4/76)	Exceptions: - "Grout testing" (ASTM C109) included in Table B of ANSI N45.2.5-1974 is inappropriate for field testing as it is a sophisticated laboratory test utilized for cement evaluation. In lieu of daily tests, pre-packaged non-shrink grouts shall be accepted for use on the basis of manufacturer's certification or compressive strength tests made in the field. Confirmation compressive strength tests shall be made during the first day's production and thereafter on a basis of either once per day of every one-hundred (100) bags used, whichever is least. - Water and ice shall be sampled and tested to ensure either potability or certified to contain not more than 2,000 parts per million of chlorides as Cl, nor more than 1,500 parts per million of sulfates as SO_4. Acceptability of this water or ice shall be per this certification and preclude the ASTM's referenced in Table B of ANSI N45.2.5-1974. - The reference, in Table B of ANSI N45.2.5-1974, to soft fragment testing per ASTM changed designations to ASTM C851 which was deleted in 1985. No testing for soft fragments is intended. Exception: Sister splices will be substituted for production splice required for tensile testing under Section 4.9 of ANSI N45.2.5-1974.

TABLE 3.2-[illegible] (Continued)

REGULATORY GUIDE NUMBER	REGULATORY GUIDE TITLE	REGULATORY GUIDE REVISION	ADDITIONAL INFORMATION/EXCEPTIONS
1.100	Seismic Qualification of Electric Equipment Important to Safety for Nuclear Power Plants	1 (8/77)	Project repair work will be performed in accordance with this regulatory guide.
1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems.	0-R (5/77)	Exception: Committed to ANSI N45.2.8 (1975), "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants" per UFSAR, Section 1.7, "QAPD", Appendix A. Not committed to this regulatory guide.
1.123	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Plants	1 (7/77)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.
1.131	Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants	0 (8/77)	Project field work will be performed in accordance with this regulatory guide.
1.144	Auditing of Quality Assurance Programs for Nuclear Power Plants	0 (1/79)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.
1.146	Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants	0 (8/80)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.

TABLE 3.2-3

D. C. COOK UNIT 2 TECHNICAL SPECIFICATIONS NOT APPLICABLE DURING THE STEAM GENERATOR REPAIR PROJECT

TECHNICAL SPECIFICATION NUMBER	TITLE	ADDITIONAL INFORMATION/COMMENT
3.1.1.3	Reactivity Control Systems - Boron Dilution	This Technical Specification ensures adequate mixing of coolant with the low boron concentration stream being introduced into the system. This mixing prevents a large concentration gradient in the core which would cause localized power excursions. With no fuel in the reactor vessel, there is no concern about decay heat removal or boron mixing.
3.1.2.1	Reactivity Control Systems - Boration Systems - Flow Paths - Shutdown	This Technical Specification requires that one boron injection flow path remains operable. This ensures that negative reactivity control is available. With no fuel in the reactor vessel there is no need for negative reactivity control.
3.1.2.5	Reactivity Control Systems - Boric Acid Transfer Pumps - Shutdown	This Technical Specification requires that at least one boric acid transfer pump remain operable. This ensures that negative reactivity control is available. With no fuel in the reactor vessel there is no need for negative reactivity control.
3.3.3.9	Instrumentation - Radioactive Liquid Effluent Instrumentation, specifically the following surveillance requirements: 4.3.3.9.2, 1b Steam Generator Blowdown Line (2-R-19) 4.3.3.9.2, 1c Steam Generator Blowdown Treatment Effluent (2-R-24)	Because there will be no steam or steam generators these two monitors will not be maintained operable.

TABLE 3.2-3
(Continued)

TECHNICAL SPECIFICATION NUMBER	TITLE	ADDITIONAL INFORMATION/COMMENT
3.3.3.10	Instrumentation - Radioactive Gaseous Process and Effluent Monitoring Instrumentation, Specifically the following surveillance requirements: 4.3.3.10.2, 2a Condenser Evacuation System Noble Gas Activity Monitor (SRA-2905) 4.3.3.10.2, 2b Condenser Evacuation System Effluent Flow Rate (SFR-401, 2-MR-054, SRA-2910) 4.3.3.10.2, 6a Gland Seal Exhaust Noble Gas Activity (SRA-2805) 4.3.3.10.2, 6b System Effluent Flow Rate (SFR-201, 2-MR-054, SRA-2810)	Because there will be no steam or steam generators these eight monitors will not be maintained operable.
3.4.7	Reactor Coolant System - Chemistry	This technical specification provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. During the steam generator repair there will be a period of approximately six months when the Reactor Coolant System will be drained to half-loop, the reactor vessel head will be in place and the Residual Heat Removal Pumps will be shutdown. During this portion of the outage it will not be possible to obtain a chemistry sample from the Reactor Coolant System. Therefore the Reactor Coolant System will be placed within specification limits prior to this shutdown and isolation period. Once sampling can be reestablished following the steam generator repair it will be verified that the Reactor Coolant System is still within the chemistry limits. If the Reactor Coolant System is not within the chemistry limits, the system will be cleaned up prior to reloading fuel into the reactor. Our engineering

TABLE 3.2-3
(Continued)

TECHNICAL SPECIFICATION NUMBER	TITLE	ADDITIONAL INFORMATION/COMMENT
		evaluation has determined that the structural integrity of the Reactor Coolant System will not be diminished by an unlikely increase in chlorides or fluorides above the technical specification limits of 0.16 ppm. This is based on the Reactor Coolant System being at ambient temperature during this period and that stress corrosion cracking (SCC) does not occur below 80°F and rarely at less than 145°F. Also, SCC does not occur until the concentration of chloride and fluoride reaches several orders of magnitude above the technical specification limit of 0.15 ppm; the level below which the Reactor Coolant System will be left at during the period of shutdown and isolation.
3.9.1	Refueling Operations - Boron Concentration	Since there will be no fuel in the reactor vessel limitations on reactivity conditions in the reactor vessel are no longer a concern.
3.9.2	Refueling Operations - Instrumentation	Since there will be no fuel in the reactor vessel there will be no change in the reactivity condition of the core, therefore, the source range neutron flux monitors are not needed.
3.9.8.1	Refueling Operations - Residual Heat Removal and Coolant Circulation	With no fuel in the reactor vessel there will be no residual heat to remove. Therefore, there is no need to maintain an operational residual heat removal loop.
3.9.8.2	Refueling Operations - Low Water Level	With no fuel in the reactor vessel there will be no residual heat to remove. Therefore, there is no need to maintain an operational residual heat removal loop.

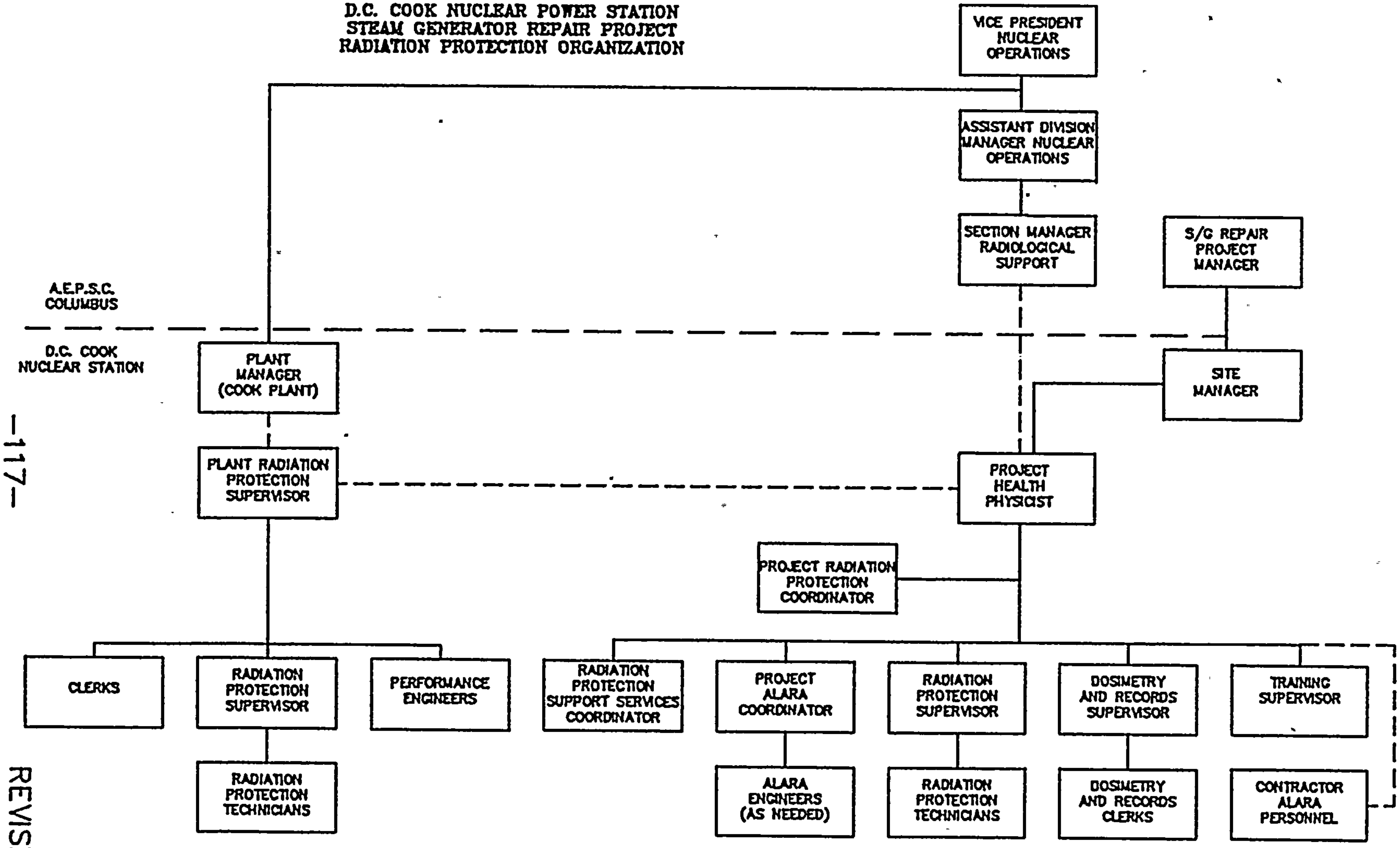
TABLE 3.2-3
(Continued)

TECHNICAL SPECIFICATION NUMBER	TITLE	ADDITIONAL INFORMATION/COMMENT
6.5.1.6(a)	Administrative Controls - Plant Nuclear Safety Review Committee - Responsibilities	The PNSRC will review the following steam generator repair project documents: 1. The Steam Generator Repair Report 2. The Steam Generator Repair Quality Assurance Program 3. Procedures covering return to service testing.
6.8.2	Administrative Controls - Procedures	The PNSRC will review the procedures written covering return to service testing.
6.8.3	Administrative Controls - Procedures	Temporary changes made to procedures covering return to service testing provided items a, b, and c of technical specification 6.8.3 are satisfied.
6.12.2	Administrative Controls - High Radiation Area	The keys to those high radiation areas turned over to the steam generator project team shall be maintained under the administrative control of the Project Health Physicist.

2

Figure 3.8.1

D.C. COOK NUCLEAR POWER STATION
STEAM GENERATOR REPAIR PROJECT
RADIATION PROTECTION ORGANIZATION



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Superseded per Rev 2 pg 6 SGRR 7/24/87
T & Repair Rpt
Generator Com
50-316

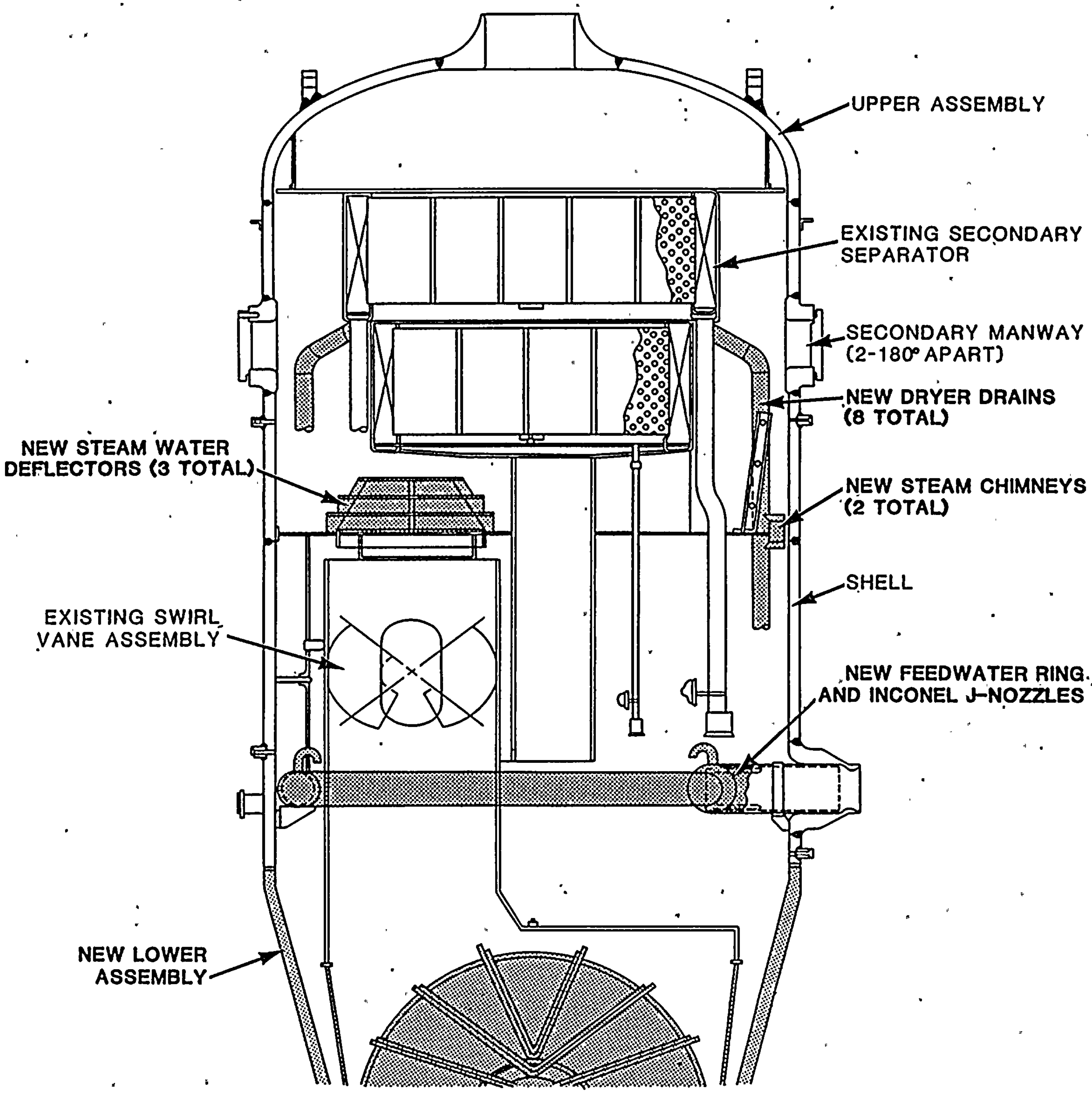
SECTION 7 - ENVIRONMENTAL REPORT

LIST OF TABLES

LIST OF TABLES cont'd.

1

FIGURE 2.2-2
MODIFICATIONS TO UPPER ASSEMBLY INTERNALS



TABL[illegible].2-1

INDUSTRY CODES AND STANDARDS APPLICABLE TO THE STEAM GENERATOR REPAIR PROJECT

CODE OR STANDARD	ADDITIONAL INFORMATION/EXCEPTION
ACI 301-84, "Specifications for Structural Concrete Buildings, Chapters 2 and 3."	<u>Exception</u>: Mix proportions shall be selected (1) utilizing laboratory or field trial batches, (2) previous satisfactory performance on similar work using the same or similar materials, or (3) prior experience with these or similar materials to provide concrete of the required strength, durability, workability, economy, etc. . .
ACI 304-85, "Recommended Practices for Measuring, Mixing, Transporting, and Placing Concrete."	
ACI 315-80, "Details and Detailing of Concrete Reinforcement."	
ACI 308-81, "Recommended Practice for Curing Concrete."	<u>Exception</u>: Curing shall be for a period of seven (7) days or until standard cured cylinders reach a comprehensive strength of 3500 PSI, whichever is first. Adherence to this criteria shall be sufficient to preclude testing for "Evaluation of Procedures," "Curing Criteria Effectiveness" or "Maturity Factor Basis."
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TABLE 3.2-1 (Continued)

CODE OR STANDARD	ADDITIONAL INFORMATION/EXCEPTION
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. The original Construction code for D. C. Cook Unit 2 nuclear vessels is Section III, 1968 Edition plus Addenda through Winter 1968, and for piping components is ANSI B31.1-1967 and ANSI B31.7-1969. As allowed by ASME Section XI, Subarticle IWA-7210, portions of the original Construction Codes dealing with installation and testing will be updated to applicable portions of Section III, 1983 Edition plus Addenda through Summer 1984.	<u>Exceptions</u>: - Consistent with the plant design basis, fracture toughness requirements will not apply. - N-stamping of fabricated piping components will not be required.
ASME Boiler and Pressure Vessel Code, Section IX, "Welding and Brazing Qualifications," edition and addenda in use at time of procedure qualification.	
ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1983 Edition plus Addenda through Summer 1983.	<u>Exception</u>: - Consistent with the plant design basis, fracture toughness requirements will not apply.
ANSI B31.1, "Power Piping", edition and addenda in use at time of contract aware for field piping services.	<u>Exception</u>: - This code applies only to power piping not classified under ASME Section III, Division 1.
USAS (ANSI) B31.1-1967, "Power Piping". USAS (ANSI) B31.7-1969, "Nuclear Power Piping".	<u>Exception</u>: - As noted under ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Vessels/Rules for Construction of Nuclear Power Plant Components" above, these codes represent the original Construction Code for for nuclear piping components. Portions dealing with materials and fabrication for new nuclear piping components, and installation and testing of all nuclear piping components, will be updated to ASME Section III, with the exception that fracture toughness requirements will not apply.

TABLE 3.2-[illegible] (Continued)

CODE OR STANDARD	ADDITIONAL INFORMATION/EXCEPTION
	The piping design basis and any additional design activities relating to nuclear piping systems will be in accordance with USAS (ANSI) B31.1-1967.
ASTM C31 "Standard Method of Making and Curing Concrete Specimens in the Field".	
ASTM C33 "Standard Specification for Coarse Aggregates".	<u>Exceptions</u>: - The average fineness modulus of the fine aggregate may be between 2.5 and 3.0, however individual samples shall not vary more than 0.20 from the average. - Compliance with gradation and fineness modulus requirements for fine aggregate shall consist of 4 out of 5 successive test results meeting the specifications. - Coarse aggregate gradation shall be Number 57, 1 inch x #4. - Coarse aggregate sodium sulfate soundness loss shall be a 10 percent maximum at 5 cycles. - Coarse aggregate Los Angeles Abrasion loss shall be a maximum of 40 percent at 500 revolutions.
ASTM C39 "Test Method for Compressive Strength of Cylindrical Specimens".	
ASTM C40 "Test Method for Organic Impurities in Fine Aggregates for Concrete".	
ASTM C88 "Test Method for Soundness of Aggregates by Use of Sodium Sulfate or Magnesium Sulfate".	
ASTM C94 "Standard Specification for Ready Mix Concrete".	
ASTM C117 "Test Method for Materials Finer Than No. 200 Sieve in Mineral Aggregates by Washing".	

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TABLE 3.2- (Continued)

CODE OR STANDARD	ADDITIONAL INFORMATION/EXCEPTION
ASTM C123 "Test Method for Lightweight Pieces in Aggregate".	
ASTM C127 "Test Method for Specific Gravity and Adsorption of Coarse Aggregate".	
ASTM C128 "Test Method for Specific Gravity and Adsorption for Fine Aggregate".	
ASTM C131 "Test Method of Resistance to Degradation of Small-Size Coarse Aggregate by Abrasion and Impact in the Los Angeles Machine".	
ASTM C136 "Method for Sieve Analysis of Fine and Coarse Aggregates".	
ASTM C138 "Test Method for Unit Weight, Yield, and Air Content (Gravimetric) of Concrete".	<u>Exceptions</u>: - Except strike off bar utilized in lieu of glass plate for unit weight determination. - Except "Yield" and "Air Content (Gravimetric)" portions will not be utilized.
ASTM C142 "Test Method for Clay Lumps and Friable Particles in Aggregate".	
ASTM C143 "Test Method for Slump of Portland Cement Concrete".	
ASTM C150 "Specification for Portland Cement".	<u>Exceptions</u>: - Except cement shall be free of false set when tested in accordance with ASTM C451. - Except total alkalies shall not exceed 0.60 percent by weight when calculated as the percentage of Na_2O plus 0.658 times the percentage of K_2O.
ASTM C172 "Method of Sampling Freshly Mixed Concrete".	

TABLE 3.2 (Continued)

CODE OR STANDARD	ADDITIONAL INFORMATION/EXCEPTION
ASTM C231 "Test Method for Air Content of Freshly Mixed Concrete by the Pressure Method".	<u>Exception</u>: - Only the Type B Apparatus shall be utilized.
ASTM C260 "Specifications for Air-Entrained Admixtures for Concrete".	
ASTM C289 "Test Method for Potential Reactivity of Aggregates (Chemical Method)".	
ASTM C309 "Specifications for Liquid Membrane-Forming Compounds for Curing Concrete".	
ASTM C311 "Methods of Sampling and Testing Fly Ash or Natural Pozzolans for Use as a Mineral Admixture in Portland Cement Concrete".	
ASTM C494 "Specification for Chemical Admixtures in Concrete".	
ASTM C566 "Test Method for Total Moisture Content of Aggregate by Drying".	
ASTM C617 "Practice for Capping Cylindrical Concrete Specimens".	
ASTM C618 "Specification for Fly Ash and Rain or Calcined Natural Pozzolan for Use as a Mineral Admixture in Portland Cement Concrete".	
ASTM C702 "Methods for Reducing Field Samples of Aggregate to Testing Size".	
SSPC-SP1 through SP10 - 1982 Steel Structures Painting Council Specifications for Surface Preparation of Steel Surfaces	

Notes: 1) All ASTMs are latest edition.

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TABLE [illegible]2-2

USNRC REGULATORY GUIDES APPLICABLE TO THE
STEAM GENERATOR REPAIR PROJECT FIELD WORK

REGULATORY GUIDE NUMBER	REGULATORY GUIDE TITLE	REGULATORY GUIDE REVISION	ADDITIONAL INFORMATION/EXCEPTIONS
1.8	Personnel Selection and Training	1-R (9/75)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.
1.26	Quality Group Classification and Standards for Water, Steam, and Rad-waste Containing Components of Nuclear Power Plants	3 (2/76)	Classification of Class 2 and 3 components for the purpose of implementing ASME Section XI requirements was made in accordance with this guide.
Safety Guide 30	Quality Assurance Requirements for Installation, Inspection and Testing of Instrumentation and Electrical Equipment	(8/72)	Committed to in UFSAR, Section 1.7, "QAPD", Appndix A.
1.31	Control of Ferrite Content in Stainless Steel Weld Metal	3 (4/78)	The requirements of this guide are now covered by ASME Section III. Field work relating to the steam generator repair project will be in compliance with this regulatory guide.
Safety Guide 33	Quality Assurance Program Requirements (Operational)	(11/72)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants	0 (3/73)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.

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REGULATORY GUIDE NUMBER	REGULATORY GUIDE TITLE	REGULATORY GUIDE REVISION	ADDITIONAL INFORMATION/EXCEPTIONS
1.38	Quality Assurance Requirements for Packing, Shipping, Receiving Storage, and Handling of Items for Water-Cooled Nuclear Power Plants	1 (10/76)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants	1 (10/76)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.
1.44	Control of Sensitized Stainless Steel	0 (5/73)	If applicable to this repair project, the field work will comply to this guide. 1
1.48	Design Limits and Loading Combinations for Seismic Category I Fluid System Components		This regulatory guide was withdrawn 3/4/85 (see 50FR9732).
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel	0 (5/73)	Project repair work will be performed in compliance with this regulatory guide.
1.54	Quality Assurance Requirements for Protective Coatings Applied	0 (6/73)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A. Exception: Committed only to ANSI N101.4-1972.
1.58	Qualification of Nuclear Power Plant Inspection Examination and Testing Personnel	1 (9/80)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.

TABLE 3.2-2 (Continued)

REGULATORY GUIDE NUMBER	REGULATORY GUIDE TITLE	REGULATORY GUIDE REVISION	ADDITIONAL INFORMATION/EXCEPTIONS
1.64	Quality Assurance Requirements for the Design of Nuclear Power Plants	0 (10/73)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.
1.68	Initial Test Program for Water-Cooled Nuclear Power Plants	2 (8/78)	This regulatory guide will be used only for guidance in developing a test program for those components and systems affected by the Steam Generator Repair Project.
1.71	Welder Qualifications for Areas of Limited Accessibility	0 (12/73)	Welders making welds in areas of restricted accessibility will be required to practice and qualify on a similar configuration to the weld being made. 1
1.74	Quality Assurance Terms and Definitions	0 (2/74)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.
1.88	Collection, Storage, and Maintenance of Nuclear Power Plants Quality Assurance Records	2 (10/76)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.
1.89	Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants	1 (7/84)	Project repair work will be performed in accordance with this regulatory guide.

TABLE 3.2-2 (Continued)

REGULATORY GUIDE NUMBER	REGULATORY GUIDE TITLE	REGULATORY GUIDE REVISION	ADDITIONAL INFORMATION/EXCEPTIONS
1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants	1 (4/76)	Exceptions: - "Grout testing" (ASTM C109) included in Table B of ANSI N45.2.5-1974 is inappropriate for field testing as it is a sophisticated laboratory test utilized for cement evaluation. In lieu of daily tests, pre-packaged non-shrink grouts shall be accepted for use on the basis of manufacturer's certification or compressive strength tests made in the field. Confirmation compressive strength tests shall be made during the first day's production and thereafter on a basis of either once per day of every one-hundred (100) bags used, whichever is least. - Water and ice shall be sampled and tested to ensure either potability or certified to contain not more than 2,000 parts per million of chlorides as Cl, nor more than 1,500 parts per million of sulfates as SO_4. Acceptability of this water or ice shall be per this certification and preclude the ASTM's referenced in Table B of ANSI N45.2.5-1974. - The reference, in Table B of ANSI N45.2.5-1974, to soft fragment testing per ASTM changed designations to ASTM C851 which was deleted in 1985. No testing for soft fragments is intended.

TABLE 3.2-2 (Continued)

REGULATORY GUIDE NUMBER	REGULATORY GUIDE TITLE	REGULATORY GUIDE REVISION	ADDITIONAL INFORMATION/EXCEPTIONS
1.100	Seismic Qualification of Electric Equipment Important to Safety for Nuclear Power Plants	1 (8/77)	Project repair work will be performed in accordance with this regulatory guide.
1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems.	0-R (5/77)	Exception: Committed to ANSI N45.2.8 (1975), "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants" per UFSAR, Section 1.7, "QAPD", Appendix A. Not committed to this regulatory guide. 1
1.123	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Plants	1 (7/77)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.
1.131	Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants	0 (8/77)	Project field work will be performed in accordance with this regulatory guide.
1.144	Auditing of Quality Assurance Programs for Nuclear Power Plants	0 (1/79)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.
1.146	Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants	0 (8/80)	Committed to in UFSAR, Section 1.7, "QAPD", Appendix A.

The replacement lower assemblies will be transported to the Donald C. Cook Plant by barge/railroad combination. They will be barged to Mt. Vernon, Indiana, where they will be transferred to railroad cars for transportation by rail to the plant. The lower assemblies will be drained, dried and sealed prior to shipment. A nitrogen blanket will be maintained on the primary and secondary side during shipment and storage. During transportation the assemblies will be supported on the barge/car deck on specially fabricated saddles, tied down by cables and restrained by end braces secured to the deck.

3.4 Post Shutdown Activities

3.4.1 <u>Containment Preparations</u>

3.4.1.1 Reactor Vessel

Prior to the start of repair project the reactor will be defueled. The upper internals will be returned to the reactor vessel and the reactor vessel head reinstalled. The missile shields will be reinstalled and a heavy steel work platform will be assembled over the refueling cavity. Lay-up procedures to insure reactor vessel cleanliness, prevent foreign objects from entering the reactor vessel, and minimize corrosion of the reactor coolant system will be developed.

3.4.1.2 Polar Crane

The polar crane is equipped with a 250-ton capacity main hoist and 35-ton auxiliary hoist mounted on a single trolley. The polar crane possesses sufficient capacity to handle all major lifting requirements for the steam generator project inside containment and can be rerated to a higher capacity as required; however, rerating of the hoists is not anticipated.

Some circuits of the following systems will be temporarily disconnected and/or removed:

- Fire Detection
- Communication
- Steam Generator Process Instrumentation
- Containment Ventilation
- Fuel Handling
- Hydrogen Recombiner
- 600 V Non-Ess Dist. & 120/208 V Lighting
- Seismic Instrumentation

Equipment determined to be essential during the Steam Generator Repair Project will be relocated, and/or its cable, conduit, and cable trays will be re-routed as required to maintain the equipment in proper operating condition.

3.4.2.7 Heating, Ventilation and Air Conditioning Ductwork

Ductwork in the removal pathway will be removed or temporary relocated. Duct pieces removed will be cleaned, marked and placed in temporary storage outside containment until needed for reinstallation.

3.4.2.8 Steam Generator Insulation

The existing steam generator metallic insulation will be reused. The outer dimensions of the replacement steam generators duplicates the original steam generators, although some insulation sections will require modifications to accommodate the additional hand holes and inspection ports. Sections of 1
insulation shall be removed, cleaned, wrapped in plastic bags and stored in wooden crates. Storage crates will be stored outside containment off the

ground and protected from the weather. Sequence of removal and storage location will be documented to facilitate installation. Those sections requiring modifications will be stored separately to allow rework prior to installation. The original equipment supplier, Diamond Power Speciality Corp., will provide procedures and technical supervision for insulation removal, storage, modifications and installation.

3.4.2.9 Seismic Restraints Removal

The steam generator snubbers will be removed to provide access for handling and movement of the steam generators. In addition, the pipe whip restraint at the main steam pipe will also be removed.

Removal and storage of the snubbers and restraints will be in accordance with approved procedures and/or specifications. Snubbers are periodically removed for ISI testing and off-site disassembly and inspection by an independent laboratory. Removal and reinstallation procedures will be similar to those established for the periodic inspections.

3.4.2.10 Fire Sensors

Thermistor cable tray fire sensors will be pulled back where they extend beyond removed cable tray sections. These sensor circuits will remain in service during the steam generator project and will be reinstalled in accordance with approved procedures.

3.5 **Steam Generator Removal Activities**

3.5.1 <u>Steam Generator Cutting Methods and Locations</u>

3.5.1.1 Feedwater and Main Steam Line Piping Cuts

The feedwater and main steam lines will be mechanically cut in two places. The location of the cuts, the equipment to be used, and the method of cutting

After the lifting assembly is installed, the crane shall take the weight of the lower assembly while the lower assembly is still supported by the temporary lateral support and the steam generator support columns. The temporary lateral support will be removed and the lower assembly then lifted slightly off its support columns.

The lower assembly shall be raised until the lifting assembly is approximately 2'-0" below the underside of the steam generator doghouse enclosure roof and then moved horizontally until it is within approximately 6 inches of the opening in the steam generator doghouse enclosure wall. It will be lifted again until the bottom of the lower assembly clears the horizontal wall cut. It will then be moved horizontally out of the steam generator enclosure. After clearing the steam generator doghouse enclosure a downending fixture 1
will be attached to the steam generator lower assembly audit will be lowered onto a set of low profile saddles. After the lower assembly has been secured to the saddles and the saddles have been placed on rollers, the upper assembly will be winched through the equipment hatch.

Once the lower assembly is through the Unit 2 equipment hatch and resting on the transport deck in the auxiliary building between the Unit 1 and Unit 2 equipment hatches, it will be attached to the tandem auxiliary building bridge cranes. The lower assembly will then be lifted, rotated and moved in a southeast direction until it has passed the southwest corner of the spent fuel pool. After the upper assembly has passed by the southwest corner of the spent fuel pool it will be oriented in an east-west direction and moved to the eastern edge of the elevation 650' floor. At the eastern edge of the elevation 650' floor, the lower assembly will be moved out into the railroad bay and oriented in a north-south direction, lowered to the 609' elevation and secured to a wheeled transporter. The lower assembly will then be transported

TABLE 3.6-1

STEAM GENERATOR REPAIR WELDS

WELD	MATERIAL	OUTSIDE Dia.[2]	WALL in.	JOINT	PROCESS[1]	FILLER[3]	MINIMUM PREHEAT °F	POSTHEAT °F	WELD FINISH	NDE[4]
Feedwater Reducer to Pipe or	SA-105 to SA-106,Gr-B	16"	.843"	Single V 35-40° with backing ring	SMAW with GTAW Cap	E7018 E70S-2	50	1100-1200 1 hour Above 600 heat & cool 400/hr	As welded	RT,MT
Pipe to Pipe	SA-106,Gr-B	14"	.705"	As above	As above	E7018	50	Not req.		RT,MT
Feedwater Nozzle to pipe	SA508,Cl-2 to SA106,Gr-B	16"	.843"	Single V 35-40° without backing ring (flat root)	GTAW root SMAW fill GTAW cap	E70S-2 E7018 E70S-2	175	1100-1200 1 hour Above 600 heat & cool 400/hr.	Grind to remove weld ripple	RT root RFT final MT
Level Inst. Tap to Inst. Pipe	SA508,Cl-1a to SA106,Gr-B	2" or less	varies	Socket	SMAW or GTAW	E7018 or E70S-2	50	Not req.	As welded	MT
Pressure Inst. Tap to Inst. Pipe	SA508,Cl-1a to SA106,Gr-B	2" or less	varies	Socket	SMAW or GTAW	E7018 or E70S-2	50	Not req.	As welded	MT
Blowdown/Drain Nozzle to Pipe	SA508,Cl-1a to SA106,Gr-B	2" or less	varies	Socket	SMAW or GTAW	E7018 or E70S-2	50	Not req.	As welded	MT
Blowdown Pipe to Pipe	SA-106,Gr-B									
Main Steam Pipe to Pipe	SA-155,Cl-1 Gr KC70 (SA691,Cl-32 Gr CMSH)	32"	1 1/8"	Single V 35-40" with backing ring	SMAW with GTAW cap	E7018	50	1100-1200 2 hours Above 600 heat & cool 350/hr.	As welded	RT,MT

1
1
1
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Revision 1

TABLE 3.6-1 (cont.)

STEAM GENERATOR REPAIR WELDS

WELD	MATERIAL	OUTSIDE Dia.[2]	WALL in.	JOINT	PROCESS[1]	FILLER[3]	MINIMUM PREHEAT °F	POSTHEAT °F	WELD FINISH	NDE[4]
Main Steam Nozzle to pipe	SA508,Cl-2 to SA155,Cl-1 Gr KC70 (SA691,Cl-32 Gr CMSH)	32"	1 1/8"	Single V 35-40° & 10-15°	SMAW with GTAW cap	E7018 E70S-2	175	1100-1200 2 hours Above 600 heat & cool 350/hr.	As welded	RT,MT
Reactor Coolant Pipe to Steam generator nozzle	SA-451,CPF8M (316) to 308 weld overlay	31 ID	2.88	Single U flat root	GTAW root SMAW	ER316 & E316	50	Not req.	Grind&polish with 360 grit or finer	RT,UT,PT
Shell Transition cone to plate	SA508,Cl-3 to SA533,Gr-A Cl-1	175 3/4"	3.62"	Single U backgouge or Single U with backing Remove backing backgouge	SMAW or SAW	E9018 B3 Erector's Choice for SAW	250	1100-1200 2 hr 30 min Above 800 heat & cool 110/hr	Grind for UT exam	RT,UT,PT or MT
Wrapper Plate and Misc. Internal None Pressure Components	SA-285,Gr-C	124.25"	3/8"	Single V flat root	SMAW	E7018	50	Not req.	Grind, Repair imperfections	MT

1 Welds shall be performed and qualified in accordance with the requirements of ASME Code Sections III and IX.
2 Outside diameter except as noted.
3 Weld filler metals and electrodes to be ordered in accordance with ASME Code Section II, Part C. Austenitic stainless steel to meet delta ferrite requirements in ASME Code Section III,NB-2433. Covered electrodes to meet analysis tests of ASME Code Section III, NB-2420.
4 NDE to be in accordance with ASME Section V with acceptance standards in accordance with ASME Code Section III.

In addition, a Plant/Project interface document shall be implemented to define areas of responsibility, communications, control, and interface between teh Project Radiation Protection/ALARA Group and the Plant Radiation Protection Section. Regular meetings between members of these two groups will be held to insure adequate communications and dissemination of information. 1

- No changes are expected due to differences in initial conditions (zero load steam temperature and pressure are identical for the unit with repaired steam generators). The no load steam generator mass decreases insignificantly (~2.0 percent).

Therefore the conclusions of the existing steam line break analyses remain valid for the repaired steam generators.

6.1.2.5 Steam System Piping Failures

Refer to Section 6.1.2.4 for discussion that applies to this accident as well.

6.1.2.6 Loss of External Load

Donald C. Cook Unit 2 is designed to have full load rejection capability, and a reactor trip may not occur following a loss of external load. It is expected that steam dump valves would open in such a load rejection, dumping steam directly to the condenser. Reactor coolant temperature and pressure do not significantly increase if the turbine bypass system and pressurizer pressure control system are functioning properly. If the steam dump valves do not operate, the reactor will trip due to high pressurizer signal, high pressurizer level signal, or overtemperature Δ T signal. Primarily to show the adequacy of the pressure-relieving devices and to demonstrate core protection margins, the Donald C. Cook FSAR and analysis of record analyze cases where the steam dump valves do not operate, and there is no direct reactor trip due to a turbine trip. It is shown in the FSAR and the analysis of record that the accident criteria on system pressure and DNB are not violated in any of the loss-of-load cases.

An accident involving the dropping or tipping of the steam generators during the removal process is considered highly unlikely because of the strict controls which will be placed on the movement process. In the unlikely event that an accident involving the steam generators does occur, our reviews have determined that the only potential interactions with shared systems of significant concern involve the spent fuel pool cooling equipment located in the vicinity of the load path. However, the slight potential for damaging spent fuel pool cooling equipment is not considered to represent an unreviewed safety question as defined in 10 CFR 50.59. This conclusion is based on the various malfunction analyses presented in Chapter 9.4 of the FSAR. These analyses conclude that it is not possible for a piping failure to cause drainage of the pool below the top of the stored fuel elements. In the event all cooling for the pool is lost, it would take a minimum of 8 hours for the temperature in the pool to each 180°F (which still allows 32°F margin to boiling). Thus, sufficient time exists to either restore cooling capability or replace water which could be lost through boiloff to prevent damage to the stored fuel elements.

6.3 Analysis of Significant Hazards Consideration

This section presents, pursuant to 10 CFR 50.91, the analysis which sets forth the determination that the Steam Generator Repair Project does not involve any Significant Hazard Consideration as defined by 10 CFR 50.92.

In addition to the appraisal on the significant hazards issue using the standards in 10 CFR 50.92, which are presented below, it is important to note that the Steam Generator Repair Project proposed by I&MECo involves practices that have been successfully implemented at two other commercial nuclear power plants, namely, the steam generator repairs completed by the Virginia Electric

and Power Company for the Surry Power Station and by the Wisconsin Electric Power Company for the Point Beach Nuclear Plant, Unit 1. The repair project is also similar to the repair projects conducted by the Carolina Power and Light Company for the H. B. Robinson Steam Electric Plant, Unit No. 2 and by the Florida Power and Light Company for the Turkey Point Plant Units 3 and 4.

6.3.1 Criterion 1

Involve a significant increase in the probability or consequences of an accident.

The Steam Generator Repair Project does not affect the probability or consequence of an accident. The probability or consequence of an accident is determined by the design and operation of plant systems. The repair project involves the replacement of the Donald C. Cook Unit 2 Steam Generator Lower Assemblies. Due to the almost identical design of the replacement lower assemblies the repair of the Donald C. Cook Unit 2 steam generators is a replacement in kind and will not change the design or operation of plant systems. Thus, this repair does not involve a significant increase in the probability or consequences of an accident previously evaluated.

6.3.2 Criterion 2

Create the possibility of a new or different kind of accident from any accident previously evaluated.

The possibility of a new or different kind of accident is not created by the repair to the Donald C. Cook Unit 2 steam generators. All components and piping will be reinstalled to meet the original design and configurations and installation requirements. Therefore, because there will be no changes to the plant and plant systems design no new or different accidents are created.

6.3.3 Criterion 3

Involve a significant reduction in a margin of safety.

Section 2.2 of this report illustrates that, although certain design enhancements have been made, the steam generator repair will result in very little change to the original operating parameters. Therefore, the impact on the accident analysis, as shown in Section 6.1 will be insignificant and there will be no significant resolution in the margin of safety.

7.3.2 Regional Historic, Archeological, Architectural, Scenic, Cultural, and Natural Features

No known historic, archeological, architectural or natural resources exist on the portion of the plant site affected by the Steam Generator Repair Project.

The access road used during plant construction parallels the beach and will be used for light construction traffic during the repair project. This traffic may pose an aesthetic impact to individuals using the beach for recreation, however, this is a temporary impact that will end with the completion of the repair project.

7.3.3 Hydrology

7.3.3.1 Ground Water

No impact to the site ground water is expected to occur as a result of the Steam Generator Repair Project.

7.3.3.2 Surface Water

No impact to the surface water associated with the plant site is expected to occur as a result of the construction phase of the Steam Generator Repair Project. In addition, the repaired steam generators will have essentially the same amount of blowdown discharged during operation as do the original steam generators and it anticipates that there will be no changes to the plant NPDES permit.

7.3.4 Geology

There will be no geological impacts as the result of the Steam Generator Repair Project. Excavation, grading, and compaction will occur in limited amounts and these actions will occur in areas previously disturbed (i.e. parking lots, roadways, and laydown areas).

7.3.5 Ecology

7.3.5.1 Terrestrial Ecology

There will be no impacts to the terrestrial ecology surrounding the plant site for the following reasons:

- No habitat will be removed as a result of the Steam Generator Repair Project since all activities related to the repair project will occur on previously disturbed area (i.e. existing access roads, parking lots, laydown area.

- Since the area affected is already subjected to the intrusion of man and machinery (i.e. security patrols, existing security lights, and normal plant operations), animals residing in the areas adjacent to the construction related activities should not be disturbed by the increased activity.

7.3.5.2 Aquatic Ecology

As discussed in Section 7.3.3.2 neither the construction phase of the Steam Generator Repair Program and the operation of the repaired steam generators will not impact the aquatic ecology associated with the plant site.

TABLE 7.4-1
DONALD C. COOK PER UNIT AVERAGE ANNUAL MAN-REM EXPENDITURES

YEAR	Exposure (Man-rem)
1980	246
1981	327
1982	321
1983	283
1984	344
1985	448

TABLE 7.4-6

COMPARISON OF GASEOUS EFFLUENT RELEASES
FROM DONALD C. COOK NUCLEAR PLANT

Radioactive Species	Average 1985 Release/Unit (Ci)	Estimated Release During the SG Repair Effort (Ci)
Noble gases	2.47×10^{-3}	Negligible
Iodines	6.46×10^{-2}	6.9×10^{-6}(1)
Particulates	3.72×10^{-2}	2.92×10^{-4}
Tritium	10.8	Negligible

Notes

(1) Estimated from Surry Unit 2 Data.

7.9 Environmental Controls

The following environmental controls shall be utilized to minimize the environmental impacts associated with the steam generator repair program. These environmental controls shall be reviewed by the contractor prior to the start of work. In addition, it is recommended that these environmental controls be included as part of the contractor work specifications.

7.9.1 Noise

To reduce the impact of noise on the surrounding community, the majority of the construction activities involving the use of heavy machinery will take place only during the day shift. If second shift construction activity involving heavy machinery must occur, it will end by 9:00 p.m. Noise from internal combustion engines will be controlled by the use of exhaust mufflers.

7.9.2 Limitations of Machinery Movement

No machinery will be allowed to operate in areas not previously distributed by construction activities. If areas not previously disturbed are inadvertently impacted by machinery, it will be the responsibility of the contractor operating the machinery to restore the disturbed area to its original state.

7.9.3 Handling and Storage of Oil and Polluting Materials

The handling and storage and oil and polluting materials will be conducted in accordance with the D. C. Cook, "Oil Spill Prevention Control and Countermeasure Plan," and the D. C. Cook, "Pollution Incident Prevention Plan."

7.9.4 Environmental Monitoring

Periodic inspections of the construction activities will be conducted. If any of the construction activities appear to be causing significant environmental impacts, appropriate actions will be taken.

7.9.5 Permits

A list of State and local permits needed to begin construction activities at D. C. Cook will be developed by the D. C. Cook Environmental Section and the AEPSC Radiological Support Section. However, it will be the responsibility of the contractor to obtain the required permits.

7.10 **Conclusion**

It is concluded that with the proper mitigation practices as outlined in the Environmental Controls Section of this report, no significant adverse environmental impact will result from the proposed activity, that there are no preferable alternatives to the proposed action and that the impacts associated with the repair program are outweighed by its benefits.

It is further concluded that the site preparation work, as described in Section 3, does not involve an unreviewed environmental question pursuant to Part II, Section 3.1 of the Donald C. Cook Plant Environmental Technical Specifications.

D. C. COOK PLANT UNIT NO. 2
STEAM GENERATOR REPAIR REPORT

SUPPLEMENT 1

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design, fabrication, inspection, testing and operation as delineated in NUREG-0554 and supplemented by NUREG-0612. This evaluation is presented in the form of a point-by-point comparison to NUREG-0554 which was developed by AEPSC and Whiting Corporation. The new crane will meet all applicable sections of CMAA Specification #70, Revision 75 and ANSI B30.2.0 - 1967. For ease in making a point-by-point comparison the following section numbers correspond to the section numbers in NUREG-0554:

2. SPECIFICATION AND DESIGN CRITERIA

2.1 Construction and Operating Periods

Since the Donald C. Cook Nuclear Plant is an operating plant, the construction portion of this section is not applicable. For the repair project and subsequent operating period the new crane will be designed per CMAA #70, Revision 75. Dynamic loads are considered due to load accelerations associated with a 150-ton load but not seismic loadings. Simultaneous static and dynamic loading will not stress the equipment beyond the material yield.

2.2 Maximum Critical Load

Since the new crane will be operating indoors, degradation due to exposure will not be considered a factor in the crane design. However, items subject to wear will have an additional design factor applied to them (see Table 2.2-1 of this supplement).

2.2 Maximum Critical Loads (cont'd.)

The crane is being designed per CMAA #70, Revision 75 for dynamic loads due to the load accelerations associated with 150 ton load. Considering dynamic loads due only to load accelerations, the maximum critical load is 150 tons. However, as presented in the preliminary seismic analysis discussion, Section 1.2.3, when dynamic loads due to a seismic event (safe shutdown earthquake) are applied to the crane the maximum critical load is 60 tons.

A maximum critical load of 60 tons is sufficient for all but 24 lifts associated with the repair project.

Because these 24 lifts are one time only special lifts the provisions of NUREG-0612 Section 5.1.1(4) will apply. This section states that for special lifts, loads imposed by the safe shutdown earthquake need not be included in the dynamic loads imposed on the lifting device. Therefore, for these 24 special lifts the maximum critical load will be the same as the design rated load of 150 tons. The design rated load and the maximum critical load will be marked on the crane.

TABLE 2.2-2

STEAM GENERATOR REPAIR PROJECT
AUXILIARY BUILDING CRANE LIFTS
OVER 60 TONS

Item	Est. Wt. (Tons)	Number Lifts
Steam Generator Concrete Doghouse Front Roof Section	70	4
Steam Generator Concrete Doghouse Back Roof Section	60	4
Old Steam Generator Upper Assembly	112	4
Old Steam Generator * Lower Assembly	247	4
New Steam Generator * Lower Assembly	240	4
Refurbished Steam Generator Upper Assembly	112	4
		24 Total

* These lifts will be made using the upgraded existing crane and the new crane tandem.

TABLE 2.2-2

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Old Steam Generator * Lower Assembly	240	4
New Steam Generator * Lower Assembly	247	4
Refurbished Steam Generator Upper Assembly	112	4
		24 Total

* These lifts will be made using the upgraded existing crane and the new crane in a tandem configuration. 2

2.3 Operating Environment

Since the crane will be operated in the auxiliary building the crane will not be subjected to design basis accident type changes in pressure, temperature, humidity or exposed to corrosive or hazardous conditions. Therefore, such considerations have not been included in the design of the crane. As discussed in the following section, a minimum operating temperature will be determined.

2.4 Material Properties

In addition to impact testing requirements on the main hook, structural members essential to structural integrity and greater in thickness than 5/8 inches are fabricated of impact tested material in accordance with the Section III of the ASME code. The minimum operating temperature of the crane will be established by the crane manufacturer. Any necessary steps to prevent operation of the crane below the minimum operating temperature will be taken. In addition, low alloy steels are not used in the fabrication of the crane, and cast iron is restricted to non-load bearing components.

2.5 Seismic Design

See Section 1.2.3.

2.6 Lamellar Tearing

The main bridge girders and structural load support members of the trolley, specifically those members supporting the critical load, are fabricated from structural plate. Welded, rolled structural shapes are not used for these members. Moreover, weld joints associated with the structural members within the main hoist load path are typically oriented such that the induced stresses will not be manifested in lamellar tearing at the weld zone. All weld joints whose failure could result in the drop of a critical load will be nondestructively examined. If any of these weld joint geometrics would be susceptible to lamellar tearing, the base metal at the joints will be nondestructively examined.

2.7 Structural Fatigue

As stated in Section 2.1, the crane will not be used for plant construction lifts. A fatigue analysis will not be performed on the structures of this crane nor does it seem reasonable that the results of such an investigation would prove meaningful. Designing for endurance in consideration of cyclic loading and material fatigue

limits has generally not proven to be governing in overhead crane design. Moreover, the fatigue stress level of materials is typically beyond normal design stress allowables.

2.8 Welding Procedures

Welding, welding procedures (pre heat, post weld heat treatments), and welder qualifications are in accordance with AWS D1.1 "Structural Welding Code."

3. SAFETY FEATURES

3.2 Auxiliary Systems

The auxiliary hoist is of single-failure-proof design.

Where dual components are not provided within either hoist mechanical load path, redundancy is provided through an increased design factor on such components as required per NUREG-0612.

3.3 Electric Control Systems

Limit controls are incorporated to minimize the likelihood of inflicting damage to the hoisting drive machinery and structure that otherwise might occur through inattentive and/or unskilled operator action. An emergency stop button will be added to the control pendant that will interrupt the power supply to the crane and stop all crane motion.

3.4 Emergency Repairs

This crane is designed so that, should a malfunction or failure of controls or components occur, it will be able to hold the load while repairs and adjustments are made.

4. HOISTING MACHINERY

4.1 Reeving System

The static-inertia design factor of the wire rope, with all parts in the dual system supporting the DRL is 11 to 1. Such conservative design more than surpasses requirements to sustain the dynamic effects of load transfer due to the loss of one of the two independent rope systems with an ample design margin remaining in the six parts supporting the load. Compliance to this

recommendation requires high alloy rope. By definition, reverse bends do not exist in the reeving system of the main hoist. Studies have been conducted to establish the effects of reverse bend on fatigue life. In consideration for the geometry of wire rope (helix) construction, unless the distance between the sheaves in the load block and head block are under one lead of the wire rope, a reverse bend cycle is not incurred. Moreover, the ratio of rope to sheave diameter in the only qualifying area of the hoist mechanism is related to the drum, which is 30 to 1; 125% of minimum requirement per CMAA Spec. #70, Rev. 75.

The pitch diameter of running sheaves and drums shall be in accordance with CMAA Spec. #70, Rev. 75. All fleet angles within the main hoist reeving are within the recommended 3 1/2 degrees. The crane is equipped with an equalizer beam/fixed sheave arrangement that provides two separate and complete reeving systems.

4.2 Drum Support

The indicated drum support provisions are included in the design which, as required, would insure against disengagement of the drum from its braking control system.

4.3 Head and Load Blocks

Both reeving systems associated with this crane are designed with dual reeving. This design will ensure the vertical load balance is maintained.

Each load-attaching point (sister hook and eye bolt) is amply designed to sustain 200% of the 150-ton DRL. The overhead crane shall be load tested at 125% of the 150-ton DRL.

Nondestructive examination of the sister hook and eye bolt will be performed. After successful completion of the load test, a complete inspection of the crane, including a nondestructive examination of the sister hook and eye bolt, will be performed.

4.4 Hoisting Speed

The main hoist full rated load speed at 4.5 FPM is considered to be "slow" for this rated load. Further, the rope line speed at the drum at approximately 27 FPM is considered to be conservative.

4.5 Design Against Two-Blocking

The main hoist is equipped with two independent travel limit control devices in addition to a load sensing system, as suggested, to insure against two-blocking. Actuation of hoist travel limit switches or load sensing devices will deenergize the hoist drive. In addition,

the mechanical holding brake will have the capability to withstand the maximum torque of the driving motor.

4.6 Lifting Device

Lifting devices for attachment to the main hook will meet or exceed these specified requirements.

4.7 Wire Rope Protection

Operation of the hoist is only to be attempted with the trolley and block aligned over the center of the load for a vertical lift.

4.8 Machinery Alignment

The provisions of this paragraph are incorporated in the design of the overhead crane.

4.9 Hoist Braking System

The provisions of this paragraph are incorporated in the design of the overhead crane.

5. BRIDGE AND TROLLEY

5.1 Braking Capacity

The bridge and trolley drives will each be provided with an appropriately sized electric holding brake which, upon interruption of power, is applied whether through operator action or violation of travel limit provisions on the trolley and restrict area limit controls for the bridge. Further, these brakes are capable of being operated manually.

The AC induction-motors and magnetic controls utilized for these drives are not prone to an overspeed condition, which is attributed to inherent operating characteristics. Therefore, overspeed limit controls for the bridge and trolley motion equipped with this type of drive would represent a needless feature. Moreover, the motor controls are provided with adequate overload protection.

The mechanical drive components are designed to sustain maximum peak loadings capable of being transmitted by either the motor or brake under all attitudes of normal crane operation.

All other recommendations of this section are compatible with the design of the crane.

5.2 Safety Stops

As stated in Section 5.1, an overspeed condition considering the type of drive used for the bridge and trolley is not a concern with this equipment. Appropriately designed and sized bumpers and stops are provided in accordance with CMAA Spec. #70 Rev. 75 and are adequate to absorb the energy of the trolley and bridge in the event of limit switch malfunction.

6. DRIVERS AND CONTROLS

6.1 Driver Selection

The main hoist motor was selected on the basis of hoisting the design-rated load (150 tons) at the design hoisting speed. Further, all proper and due consideration was given to the design of related mechanical and structural components to adequately resist peak torques transmitted by this motor within normal design limits.

Hoist overspeed and overload sensing-limit control provisions have been incorporated to guard against such occurrences. Additionally, the hoist holding brakes are capable of controlling the design rated load within the 3 inches (8 cm) specified stopping distance. In addition, an emergency stop button will be located at ground level to interrupt power to the crane independent of the crane controls. Since the MCL is less than the DRL, administrative controls will be established to reset the overloading sensing device.

6.2 Driver Control Systems

The design considerations discussed in this section have been addressed and incorporated as appropriate except for the restriction of simultaneous operation of motions. The crane is not used to handle spent fuel assemblies.

6.3 Malfunction Protection

Features to sense, respond to, and secure the load in the event of hoist overspeed, overcurrent, overload, over travel, and loss of one rope of the dual reeving system have been incorporated.

6.4 Slow Speed Drives

Features recommended in this paragraph will be incorporated as part of the motion control circuitry.

6.5 Safety Devices

Each hoist is equipped with two independent hoist overtravel limit controls.

6.6 Control Stations

Since this crane is not equipped with a cab, the complete operating control system and emergency controls for the crane will be located on a pendant control. In addition, as stated earlier an emergency stop button will be located at ground level to interrupt power to the crane independent of the pendant control.

Since the design rated load is greater than the maximum critical load, administrative controls will be established to ensure that the resetting of the overload sensing device is properly conducted.

7. INSTALLATION INSTRUCTIONS

7.1 General

Complete operation, maintenance, installation and testing instructions will be provided for the overhead crane by the crane manufacturer.

7.2 Construction and Operating Periods

As discussed in Section 2.1 this crane will not be used for plant construction. The crane will be designed for Class A-1 service as defined in CMAA Specification #70, Revision 75. The allowable design stress limits will not be exceeded during the repair project.

During and after installation of the crane, the proper assembly of electrical and structural components should be verified.

8. TESTING AND PREVENTIVE MAINTENANCE

8.1 General

A complete check will be made of all the crane's mechanical and electrical systems to verify the proper installation and to prepare the crane for testing.

The only components that will have been proof-tested at the time of installation are the main hook, eye bolt and wire rope.

8.2 Static and Dynamic Load Tests

The crane will be load tested at 125% of the design rated load. The design rated load of this crane is 150 tons. During the 125% load test, the crane motions shall be limited due to the physical restrictions of the auxiliary building.

During the no-load test, however, each crane motion shall be operated to its full travel limit.

8.3 Two-Block Test

Although the hoist is equipped with an overload sensing device, under no circumstances should such a test be conducted for the mere purpose of demonstrating design adequacy. The purpose in providing numerous limit control devices is to ensure against such an occurrence. The two load travel limit control switches will be checked prior to lifting a load. The overload sensing device can be operationally checked within the design rated load of the crane without the need to secure the hoist to a fixed anchor for the purpose of generating an excessive load.

8.4 Operation Tests

Whiting's standard procedures require a no-load running test before shipment. Calibration and adjustments for hoist overload and overspeed will be done after installation.

8.5 Maintenance

A maintenance program including periodic inspections of the crane will be developed. This maintenance program will ensure that the crane is maintained at the design rated load. Both the maximum critical load and the design rated load will be plainly marked on each side of the crane.

9. OPERATING MANUAL

Whiting's standard Operations and Maintenance Manual, which is to be provided for the overhead crane, will provide sufficient information in the proper operation of the overhead crane, lubrication instructions, parts ordering information, and periodic inspection points.

10. QUALITY ASSURANCE

The Whiting Corporation is on the Donald C. Cook Nuclear Plant Qualified Suppliers List for spare and replacement crane parts. Whiting has a QA program that complies with ANSI N.45.2-1971/NRC Regulatory Guide 1.28. This program applies also to the fabrication of new cranes for nuclear power plants. Whiting will be audited for QSL recertification in April 1987.

Donald C. Cook Nuclear Procedure MHI 2071, "Qualification and Training of Crane Operators," covers qualification requirements of crane operators and will be revised as necessary to reflect the single-failure-proof features of the new crane.

1.2.3 Seismic Analysis

This section presents the preliminary seismic analysis conducted to demonstrate the largest load the new crane

can stop and hold during a safe shutdown earthquake. The following information provides a description of the method of analysis, the assumptions used, and the mathematical model evaluated in the analysis.

1.2.3.1 Analysis Description

The crane was analyzed to determine the effect of seismic excitations. For this analysis, the matrix displacement method was used based upon finite element techniques. The crane was mathematically modeled as a system of node points interconnected by various finite elements representing straight beams. All masses and inertias were distributed among the nodes whose degrees of freedom characterize the response of the structure. The interconnecting finite elements were assigned stiffness equivalent to that of the actual structure.

The mathematical model represents as accurately as possible the flexibility of the bridge girders, hoist rope, and girder end connection. The trolley, the drive units and the bridge trucks were represented as rigid bodies.

The crane was analyzed with the trolley positioned at mid-span. This was done with loads of 50 and 60 tons in the down position. Preliminary calculations showed that this condition would produce the maximum girder stress for a given load.

The dynamic analysis was of the mode frequency (MODAL) type, solving for the resonant frequencies and the mode shapes that characterize the crane. The modes with meaningful participation in a given direction are directly expanded by the computer program to yield the expanded mode shapes, the element stresses and the reaction values. This type of analysis is linear and plastic deformation, sliding, friction, and slack rope are not taken into account.

The normal mode approach was employed for the analysis of the components. All significant eigen-values and eigen-vectors were extracted, and these modes were combined by the method specified by the U. S. Nuclear Regulatory Commission, Regulatory Guide 1.29, Rev. 1, Section 1.2.2 (Combination of Modal Responses with Closely Spaced Modes by the 10% Method). Those modes with mode coefficient ratios less than 1% in the x direction or 0.5% in the y and z directions were dropped because their contribution is proportionally small when compared to the largest mode coefficient of the related directional excitation. The results of the three orthogonal dynamic excitations were combined by the square root of the sum of the squares method (SRSS) and then absolutely added to the results of the static condition.

Because the y reaction exceeds the frictional resistance of those bridge wheels that are braked, slip will occur. The maximum acceleration in the y direction will be reduced from that predicted by the modal analysis. The primary y mode was therefore reduced by a scale factor such that the resulting y reaction approaches the maximum that could be sustained before slip. The results were then resummed as previously described.

In order to assure structural integrity, the job specification requires that the maximum stresses not exceed the minimum yield strength of the material divided by 1.5 for the OBE and 1.1 for the SSE.

The crane is constructed of ASTM A36 structural steel except for components which are specifically noted in the report. A36 material has a specified minimum yield strength of 36 ksi. The combined bending and axial stresses are limited to 24 ksi for the OBE and 32.7 ksi for the SSE.

The actual properties of the specified materials show a great deal of variation and are generally considerably higher than the minimum required by the material specification. Also the maximum stresses occur only at a point on a section and cannot be themselves be indicative of the tendency of the section to permanently deform, especially when the nominal stresses on the extreme fibers of the adjoining faces are significantly lower. It is therefore conservative to compare the combined bending and axial stresses at the corners with the specified allowables to assure structural integrity.

Impact factors for wheel flange to rail contact, etc., have been consider negligible. The state of the art is such that these impacts cannot rigorously be studied; however, independent time history analyses have been run in many cases, all indicating slow relative motion between the rail and the wheel. This is because of the time dependency of the forcing function coming from the building into the crane. Note that the only coupling through which these forces can be transmitted is dynamic friction. Upon reaching the rail the wheel will first rise through the corner radius and then contact the rail. During this period, the structure is starting to deflect as the end of the crane in this direction is flexible.

The computer analysis was performed using ANSYS, a large scale finite element program.

1.2.3.2 Summary of Results

The crane was mathematically modeled using finite elements. On the basis of preliminary runs, the number of degrees of freedom and the significance criteria for modal expansion were adjusted. Static and three load step

reduced modal runs were made and the results summed. Because slip occurs, the y excitation was proportioned and these results resummed.

The crane was analyzed with the main trolley at mid-span (see Figure 1.2-1). For this position the analysis was done with 50 and 60 ton loads on the main hook in the low position. From preliminary studies, the load case considered should yield the maximum stresses in the girders.

Because of the seismic acceleration a slack rope condition was found to exist under certain conditions. This cannot be truly simulated with a linear modal analysis. However our experience with time history analyses shows that a modal analysis tends to produce conservative results. The rope load predicated by the modal analysis is well below the allowable rope load.

When the excess dynamic rope load (that which produces a slack rope) is deducted, a small upkick is produced by the loading conditions examined. When the wheel loads parallel to the runway are compared with the vertical wheel load times the coefficient of friction, it is found that the crane bridge will tend to slide under certain loading conditions examined. This sliding is oscillatory in nature and the loadings predicted by a modal analysis are conservative. The wheel loads have been adjusted to account for frictional effects.

Although some non-linearities are produced by the specified excitations the specified linear analysis will conservatively predict the behavior of the crane during a seismic excitation.

The crane was found to meet the requirements for a seismic excitation with a 60 ton load on the main hook.

1.2.4 Lifting Beams

Stress levels of all load-bearing members of the lifting beam will not exceed 6,000 psi under rated load. This low stress level meets requirements of NUREG-0612 and ANSI N14.6 specifications for increased design factors for single-load-path components. Further, this design stress level qualifies for material test exemptions per Paragraph AM 218 of the ASME Boiler and Pressure Vessel Code, Section III, Division 2, as referenced in Paragraph 3.3.6 of ANSI N14.6-1978.

Proposed lifting beam will not be subject to high amounts of radiation, 200 mili-rem/hour maximum, nor will it be submerged at any time. Based on this criteria the proposed lifting beam design will not be subject to any sections of ANSI N14.6-1978 which refers to submerged duty, decontamination or radiation degradation.

Application of any coating system onto the lifting beam must not violate E.P.A. codes.

Under Section 6 of ANSI N14.6-1978 the main beam section and the hooks swivel are single path designed with stress levels below 6,000 psi. Since the materials for these items will have mill certification and that 100% of critical welds will undergo nondestructive examination to ensure structural integrity, these two items will not be subject to load test of three times their rated capacity. These two items will however be subjected to a 150% load test.

1.2.5 Interfacing Lift Points

Interfacing lift points will be dual-load-path and will be designed to shear stress levels not to exceed 4,500 psi under rated load. This design stress levels qualifies for material test exemptions per Paragraph AM 218 of the ASME Boiler and Pressure Vessel Code, Section III, Division 2 as referenced in Paragraph 3.2.6 of ANSI N14.6-1978.

1.3 CONCLUSION

The new crane being purchased by the Indiana & Michigan Electric Company for use during the Steam Generator Repair Project has been evaluated against the criteria of NUREG-0554 and NUREG-0612. Results of this evaluation have shown that the crane being purchased meets the guidelines and criteria of NUREG-0554 and NUREG-0612 and therefore will be classified and used as a single-failure-proof crane.

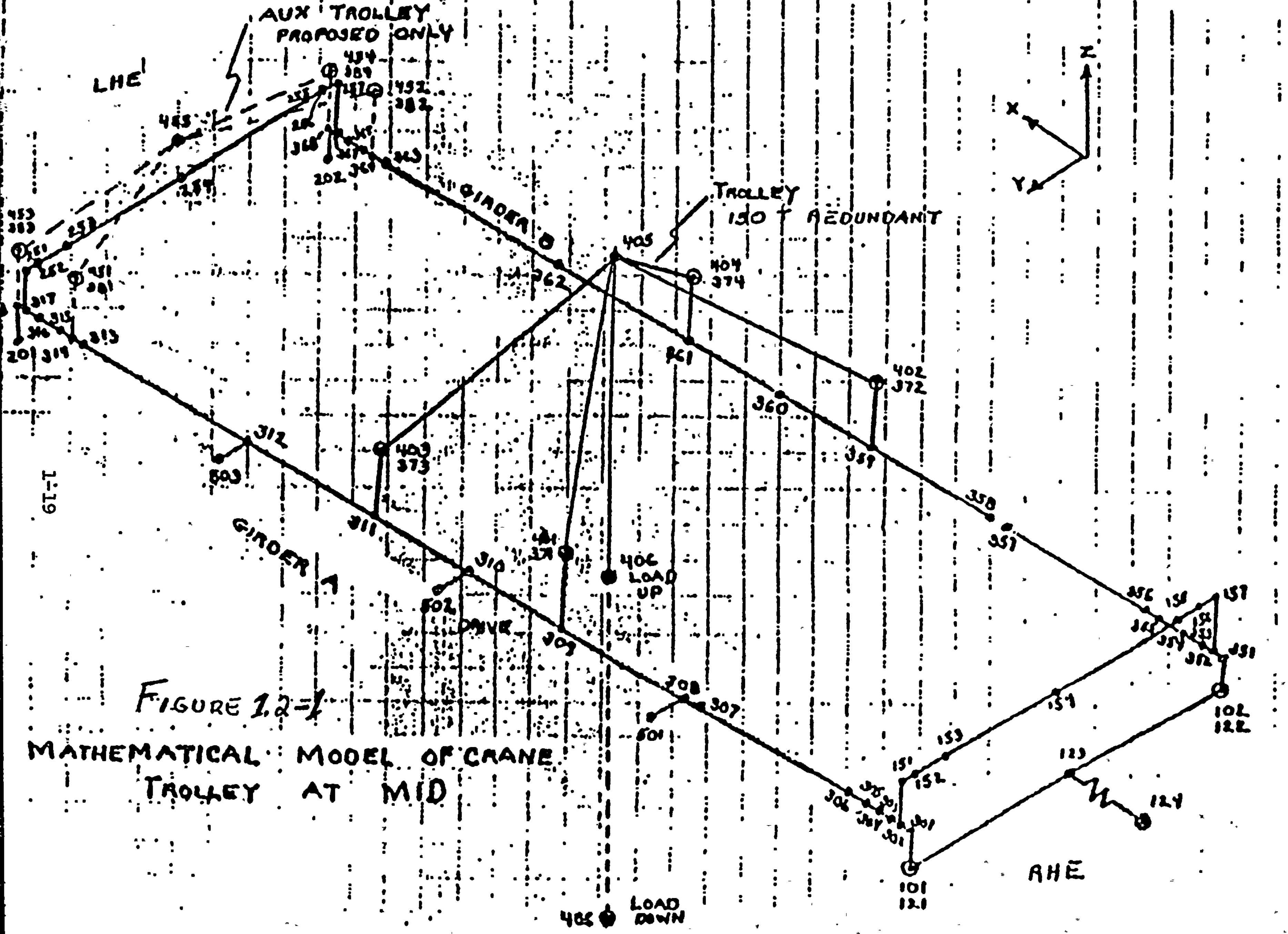

FIGURE 1.2-1

MATHEMATICAL MODEL OF CRANE
TROLLEY AT MID

- No changes are expected due to differences in initial conditions (zero load steam temperature and pressure are identical for the unit with repaired steam generators). The no load steam generator mass decreases insignificantly (~2.0 percent).

Therefore the conclusions of the existing steam line break analyses remain valid for the repaired steam generators.

6.1.2.5 Steam System Piping Failures

Refer to Section 6.1.2.4 for discussion that applies to this accident as well.

6.1.2.6 Loss of External Load

Donald C. Cook Unit 2 is designed to have full load rejection capability, and a reactor trip may not occur following a loss of external load. It is expected that steam dump valves would open in such a load rejection, dumping steam directly to the condenser. Reactor coolant temperature and pressure do not significantly increase if the turbine bypass system and pressurizer pressure control system are functioning properly. If the steam dump valves do not operate, the reactor will trip due to high pressurizer pressure signal, 2
high pressurizer level signal, or overtemperature T signal. Primarily to show the adequacy of the pressure-relieving devices and to demonstrate core protection margins, the Donald C. Cook FSAR and analysis of record analyze cases where the steam dump valves do not operate, and there is no direct reactor trip due to a turbine trip. It is shown in the FSAR and the analysis of record that the accident criteria on system pressure and DNB are not violated in any of the loss-of-load cases.

[illegible]