INSTRUCTIONS FOR THE INSERTION OF REVISION 2
TO THE STEAM GENERATOR REPAIR REPORT

REMOVE	INSERT
Record of Revisions dated November 4, 1986	Record of Revisions dated July 24, 1987
Page ii, Revision 1	Page ii, Revision 2
Page v, Revision 0	Page v, Revision 2
Pages vii & viia, Revision 1	Pages vii & viia, Revision 2
Page 44, Revision 0	Page 44, Revision 2
Page 49a thru 49j, Revision 1	Pages 49a thru 49q, Revision 2
Page 55, Revision 0	Pages 55, 55a, & 55b, Revision 2
Pages 61, Revision 1 & 62, Revision 0	Pages 61 & 62, Revision 2
Page 67, Revision 1	Page 67, Revision 2
Page 75 & 76, Revision 1	Pages 75 & 76, Revision 2
Page 82a, Revision 1	Page 82a, Revision 2
Page 144, Revision 0	Page 144, Revision 2
Pages 163, 164, & 165, Revision 0	Pages 163, 163a, 164, & 165, Revision 2
Pages 170 & 171, Revision 0	Pages 170 & 171, Revision 2
Page 173, Revision 0	Page 173, Revision 2
Page 183, Revision 0	Page 183, Revision 2
Pages 202 & 203, Revision 0	Pages 202 & 203, Revision 2
Page 1-2, Revision 1	Page 1-2, Revision 2
Page 1-4, Revision 1	Page 1-4, Revision 2
Pages 1-6 thru 1-19, Revision 1	Pages 1-6 thru 1-20, Revision 2

RECORD OF REVISIONS

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

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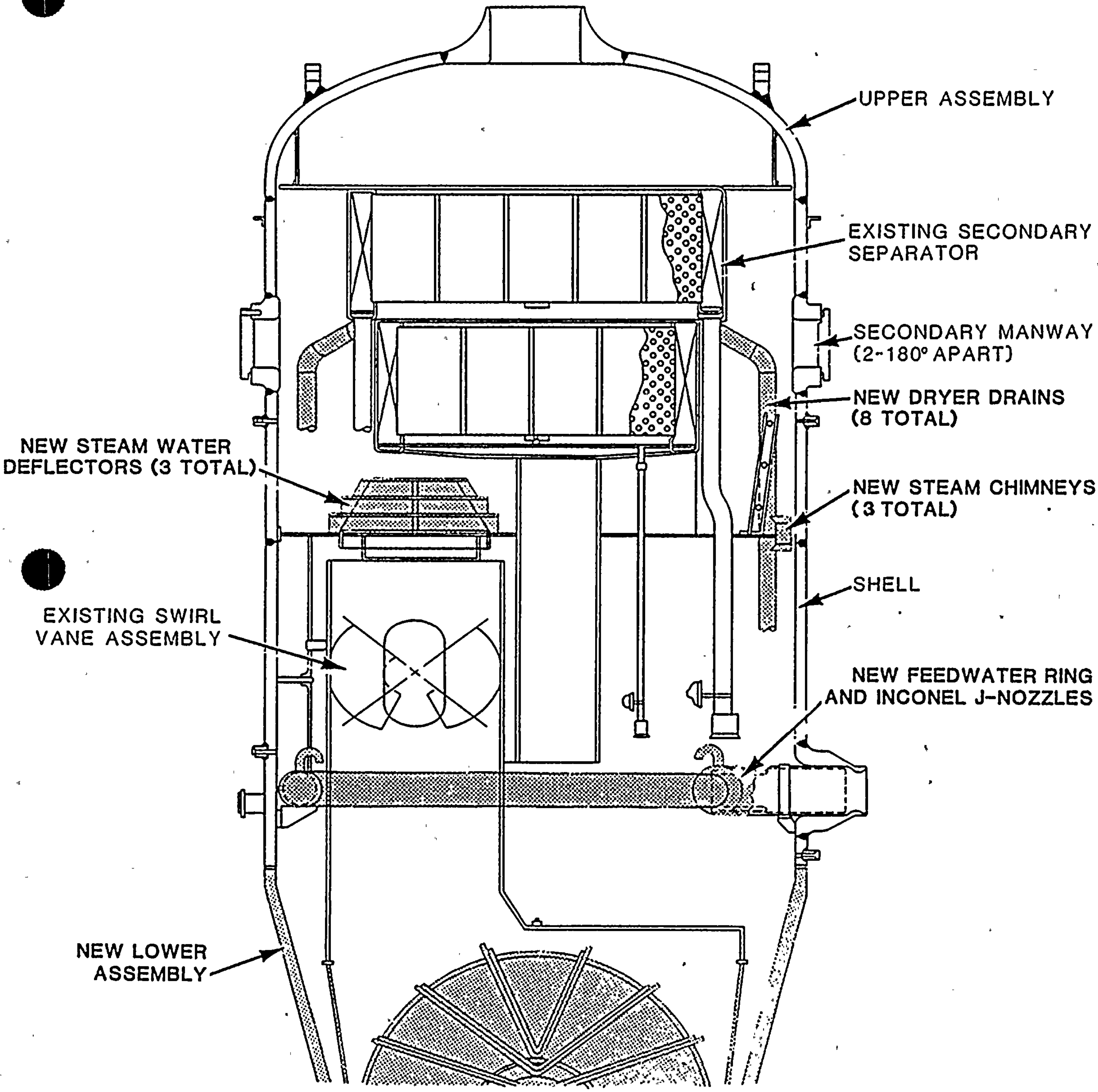
SECTION 7 - ENVIRONMENTAL REPORT

LIST OF TABLES

2

LIST OF TABLES cont'd.

FIGURE 2.2-2
MODIFICATIONS TO UPPER ASSEMBLY INTERNALS



| 2

o Although there will be no fuel in the Unit 2 core, Unit 2 will be considered to be in Mode 6 during the Steam Generator Repair Project. Unit 2 Technical Specifications will be adhered to with the exception of those Technical Specifications listed in Table 3.2-3. The Technical Specifications listed in Table 3.2-3 will not be applicable during the Steam Generator Repair Project. For purposes of Technical Specification applicability, the Steam Generator Repair Project will begin when the last fuel assembly from the Unit 2 core is placed in the spent fuel pool and will end when the first fuel assembly is removed from the spent fuel pool to refuel the Unit 2 core. 2

TABLE [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."	<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."
ACI 318-83, "Building Code Requirements for Reinforced Concrete, Chapters 3, 4, and 5."	<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.
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 3.2-1 (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, 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	

TABLE 3.2-1 (Continued)

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. 2 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".	Exceptions: - 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
	for 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".	

TABLE 3.2-1 (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.

TABLE 3.2-1 (Continued)

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".	Exception: - 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 3.2-2 (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.
1.74	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)	<u>Exceptions</u>: - "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. <u>Exception</u>: Sister splices will be substituted for production splice required for tensile testing under Section 4.9 of ANSI N45.2.5-1974. 2

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.

2

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 2

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. 2
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. 2
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.

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.

3.3.3 Modification to Auxiliary Building Structural Steel

To handle the loads associated with the Steam Generator Repair Project, the existing Auxiliary Building overhead bridge crane will be upgraded to single-failure-proof status and a second 150/20 ton single-failure-proof overhead bridge crane will be installed in the Auxiliary Building. Both cranes will travel on the existing rails, which extend the length of the auxiliary building, while carrying loads approaching 250 tons (see Section 6.2.1 and Supplement 1 of this report for a detailed description of the cranes and the load handling methodologies).

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Each crane rail is supported by a crane rail girder which in turn transfers the crane load to the auxiliary building structural steel columns. An analysis was performed to ensure the integrity of the existing auxiliary building structural steel elements which support the crane loads. The analysis was performed assuming both cranes operating in tandem while moving a 300 ton load. The results of the analysis shows that the existing auxiliary building structural steel is adequate to support the crane loads with minor modifications.

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3.3.4 Polar Crane Power Circuit Relocation

Approximately 200 feet of the polar crane power supply cable is located in the cut area of a Unit 2 steam generator doghouse enclosure wall. To eliminate 2 this cable as a cut interference and at the same time provide maximum availability of the polar crane, the cable will be permanently relocated prior to the start of the steam generator repair project. The entire cable, from the containment penetration connection up to the crane, will be replaced to avoid splicing. The rerouted new cable is of approximately the same length as the existing cable and therefore will not significantly increase the permanent combustible fire loading in the containment building. The rerouted cable will be mounted to the walls per Seismic Class I requirements.

3.4 Post Shutdown Activities

3.4.1 Containment Preparations

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:

- o Fire Detection
- o Communication
- o Steam Generator Process Instrumentation
- o Containment Ventilation
- o Fuel Handling
- o Hydrogen Recombiner
- o 600 V Non-Ess Dist. & 120/208 V Lighting
- o 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 insulation shall be removed, cleaned, wrapped in plastic bags and stored in strong tight containers. These containers will be stored outside containment 2
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 Steam Generator Cutting Methods and Locations

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
will be attached to the steam generator lower assembly and it will be lowered 2
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 lower assembly has passed by the southwest corner of the 2
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.[1] IN.	WALL IN.	JOINT	PROCESS[2]	FILLER[3]	MINIMUM PREHEAT °F	POSTHEAT °F	WELD FINISH	NDE[4]	
Feedwater											
Nozzle to elbow	SA508,C1-2 SA234,WPB	16	0.843	Single V 35-40° without backing ring (flat root)	GTAW-r SMAW-f	ER70S-2 E7018	250	1100-1200 1 hr Above 600 heat & cool 400/hr	Grind to remove weld ripple	RT 1/3 fill plus RT final & MT	
Elbow to reducer or	SA234,WPB	16	0.843	Single V with backing ring	SMAW GTAW cap	E7018 ER70S-2	50	1100-1200 1 hr Above 600 heat & cool 400/hr	As welded	RT, MT	
Reducer to Pipe	SA234,WPB SA106,B	14	0.705	Single V with backing ring	SMAW GTAW cap	E7018 ER70S-2	50	None	As welded	RT, MT	
Liner	SA106,B	12-3/4	0.5	Single V	GTAW-r SMAW-f	ER70S-2 E7018	50	None	As welded	VT-1, RT	2
SG Vessel											
Transition cone to plate	SA508,C1-3 to SA533, A, C1-1	175-3/4	3.62	Double V modified backgouge	SMAW-r &SAW, GMAW or FCAW	E9018-M or E8018-C3 & matching wire E81Ni1	250	1100-1200 2 hr 30 m Above 800 heat & cool 110/hr	Grind for UT exam	RT,UT, & PT or MT	
Wrapper plate and misc. non press comp	SA285,C	124.25	3/8	Single V w/o backing (flat root)	SMAW or GTAW	E7018 ER70S-2	50	None	Grind flush	MT	
Misc Connections o Blowdown	SA508,C1-1a to SA105, red. to SA206,B pipe	2.5 to 2	>1-1/4	Socket	SMAW or GTAW	E7018 ER70S-2	50	None	As welded	MT	
o Drain	SA106,B	1									
o Level	SA106,B	3/4									

TABLE 3.6-1 (cont'd.)
STEAM GENERATOR REPAIR WELDS

WELD	MATERIAL	OUTSIDE DIA.[1] IN.	WALL IN.	JOINT	PROCESS[2]	FILLER[3]	MINIMUM PREHEAT °F	POSTHEAT °F	WELD FINISH	NDE[4]
<u>Main Steam</u>										
o Nozzle to elbow or elbow to elbow &	SA508, C1-2 to SA234, WPB	32	1-1/8	Single V 35-40° backing ring	SMAW B/R with GTAW cap	E7018	250	1100-1200 1 hr 15 m Above 600 heat & cool	As welded	RT, MT
o Reducer to elbow or pipe	SA234, WPB or SA155, C1-1, Gr-KC70	32/30	1-1/8 1	Single V 35-40° backing ring	SMAW B/R with GTAW cap	E7018	250	1100-1200 1 hr 15 m Above 600 heat & cool 350/hr		RT, MT
<u>Reactor Coolant</u>										
Elbow to SG nozzle	SA351, CF8M to E308 weld overlay on carbon steel	31 ID	2.88	Single V flat root	GTAW-r SMAW-f or Auto GTAW/GMAW-f	ER308 E308 ER308	50	None	Grind & polish with 360 grit or finer	RT,UT,PT

1. Outside diameter except as noted.
2. Welds shall be made and qualified in accordance with the requirements of ASME Code Sections III and IX.
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, NB2420.
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 the 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. 2

- 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.

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 reach 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 Fire Protection Evaluation

The effect of a Unit 2 construction fire was evaluated by assuming that the equipment in the Unit 2 containment and Auxiliary Building fire areas directly affected by construction activities would be damaged. Loss of all equipment in the combined fire areas would not cause loss of Unit 1 safe shutdown capability. 2

The fixed combustible loading of these fire areas will not be significantly affected by construction activities. Transient combustible loading in the construction areas will increase beyond the levels assessed in the Fire Hazards Analysis for normal conditions.

The Safe Shutdown Capability Assessment exemption requests and fire barrier evaluations for the affected fire areas were reviewed to assess the impact of increased fire loadings and fire hazards due to construction activities. Construction activities were determined not to impact the validity of these evaluations with respect to Unit 1 shutdown capability provided there is no continuity of combustibles, such as wooden temporary stairs and trash chutes, between the Crane Bay and the 650' elevation of the Auxiliary Building which could promote rapid fire spread. Temporary stairs and other structures connecting these elevations will be made primarily of non-combustible materials or compensatory measures will be provided. 2

The repair contractor will operate under existing plant procedures and administrative controls with the exception of areas turned over to his direct control for construction, access, and laydown. The repair contractor will prepare a fire protection program to govern work activities in the areas under his control which will be designed to minimize construction fire hazards.

6.4 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.

6.4.1 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.

6.4.2 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.

6.4.3 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 is anticipated that there will be no changes to the plant NPDES permit. 2

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 or the operation of the repaired steam generators will impact the aquatic ecology associated with the plant site. 2

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
1986	336

2

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

2

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 disturbed by construction activities. If areas not previously disturbed | 2
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 of oil and polluting materials will be | 2
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. The AEPSC Radiological Support Section will be responsible for obtaining the required permits. 2

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. This point-by-point comparison 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

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 the same as the design rated load. 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. 2

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 in a tandem configuration. 2

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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. The ranges of temperature, pressure, and humidity anticipated for crane usage are as follows:

Temperature: Ambient temperature inside the auxiliary building with seasonal variations between winter and summer.

Pressure: Ambient pressure except during refueling outage activities, when slightly negative pressure ($\geq$ 1/8 inch w.g.) will be maintained as required by Technical Specification 4.9.12.d.4. 2

Humidity: This could range from a minimum of 0% to a maximum 100%.

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. The allowable stress range for the fatigue design of this crane is higher than the normal 2
design allowables of Crane Manufacturers Association of America (CMAA) Specification No. 70-1975. As a result, a fatigue analysis will not be performed, since it is not a governing factor in design of the crane.

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." Further,
low-alloy materials will not be used in the main load 2
support structure.

3. SAFETY FEATURES

3.2 Auxiliary Systems

The auxiliary 20 ton hoist is of single-failure-proof 2
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 radio remote control unit that 2
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. The maximum load (including static and inertia forces) on each individual wire rope in the dual reeving system with the MCL attached 2
will not exceed 10% of the manufacturer's published breaking strength. 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.

Protection against excessive wire rope wear and fatigue
damage will be ensured through periodic inspection and 2
maintenance.

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 of approximately 4.5
FPM is less than the suggested operating speed in the 2
"slow" column of Figure 70-6 of CMAA specification #70.

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

The lifting beams and other devices attached to the crane hook block will be designed to have factors of safety based on guidelines noted in NUREG-0612 and NUREG-0554. Each device will be able to support a load of three times the load (static and dynamic) being handled without permanent deformation as recommended in Section 4.6 of NUREG-0554. 2

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, emergency power disconnect switches will be located at operating floor 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. 2

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 radio remote control unit. In addition, as stated earlier emergency power disconnect switches will be located at operating floor level to interrupt power to the crane independent of the radio remote control unit. 2

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.

Proof-testing of a subcomponent is an independent verification of the subcomponent's ability to perform. The main hook block and eye bolt of the hook block assembly will be tested at 200% of the design-rated load (DRL). Before and after this test, the hook and eye bolt will be subject to nondestructive examinations. The wire rope supplier will test a section of wire rope by subjecting it to an overload condition until breaking occurs. No other components of the crane shall be proof-tested. Upon successful completion of the above proof tests, the overhead crane will be tested at 125% of the DRL. This test will ensure the ability of the crane and its subcomponents to perform their intended function.

8.2 <u>Static and Dynamic Load Tests</u>

The overhead crane will be tested after installation by means of a no-load test and a 125% capacity load test. The no-load test consists of operating each crane motion to its extreme travel limit without a load on the hook. During the no-load test, the crane bridge shall travel the entire length of the runway, the top-running trolley shall traverse the crane bridge, and the hook block shall be operated through its complete vertical travel limits. Upon successful completion of the no-load test, the 125% capacity DRL test will be conducted. Each crane motion shall be engaged with the 125% DRL test load suspended 2
from the hook. However, due to the physical restrictions of the plant, each motion will not be operated to its full travel limit during the 125% DRL load test.

8.3 <u>Two-Block Test</u>

Although the hoist is equipped with an overload sensing device, load-anchor testing is not recommended by the crane manufacturer (Whiting Corporation). Since Whiting customers have followed the recommendation, there is no available information on past load-anchor tests. The overload-sensing device will be preset and tested using a load higher than the preset load. The last sentence of Section 8.3 of NUREG-0554 states: "The crane manufacturer may suggest additional or substitute test procedures that will ensure the proper functioning of protective overload devices." Based on that provision, and per crane manufacturers' recommendations, we are planning to perform the overload testing rather than the load-anchor test.

8.4 <u>Operation Tests</u>

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

The operating manual supplied by the crane manufacturer will comply with Section 9.0 in its entirety, including details on preventive maintenance program items noted in the first paragraph of Section 9.0 of NUREG-0554. The existing plant procedures on the preventive maintenance program will be revised to address the above-noted items. 2

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 wculd 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.

1.2.6 Monorail Hoist

Due to the configuration of the two cranes in the auxiliary building there will be some areas in the auxiliary building that cannot be reached by either crane. To provide access to these areas the new crane will be equipped with a 2,500 lb. capacity, fully electric hoist mounted on a fixed monorail suspended from the idler girder of the new crane. The hoist will weigh approximately 1,200 lbs. and will have a vertical lift of approximately 122 feet. Control of the hoist will be by radio remote control. The hoist is being designed to ANSI/ASME HST-4M-1985, "Performance Standards for Overhead Electric Wire Rope Hoist." 2

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.

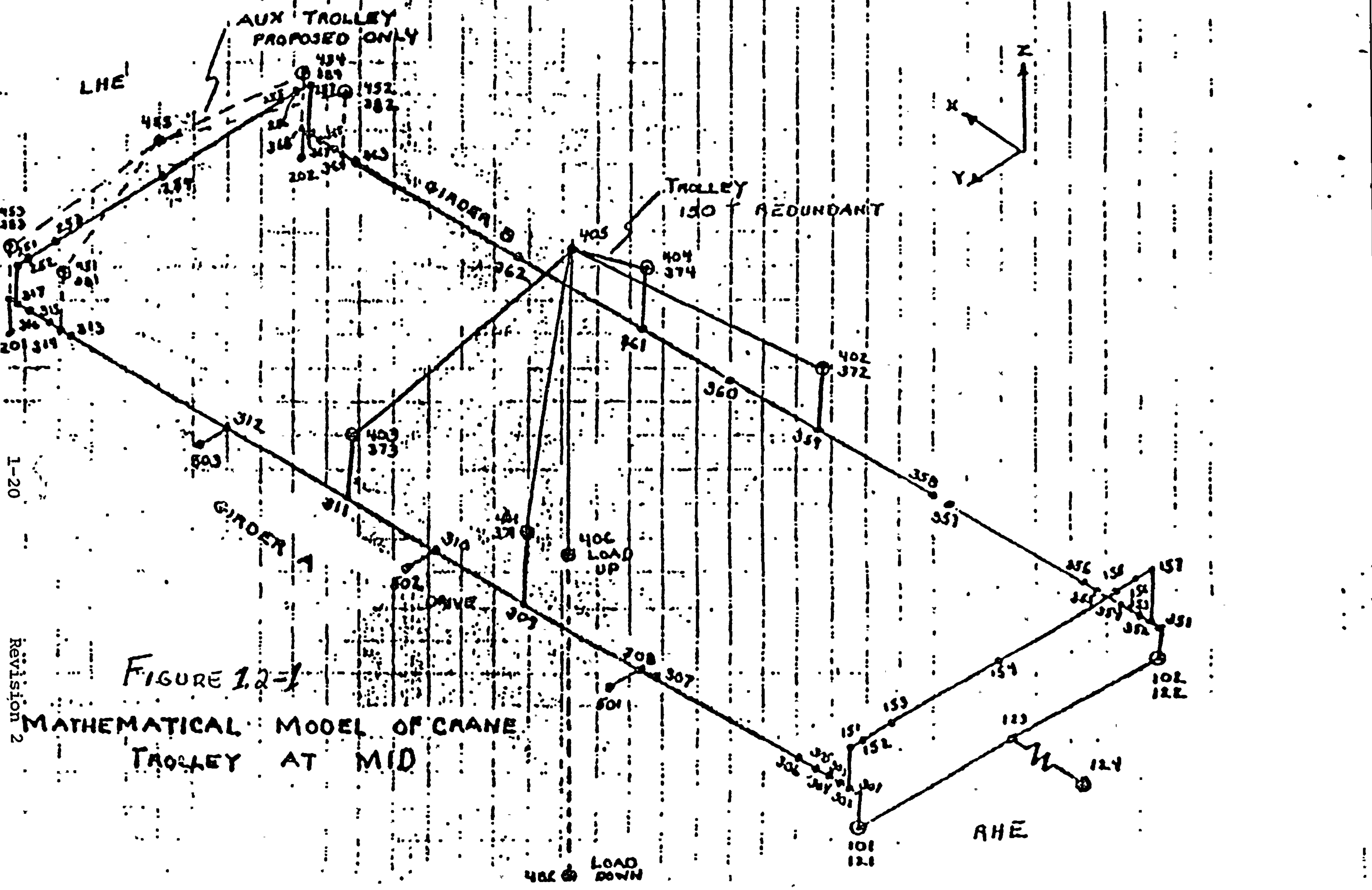

FIGURE 1.2-1
MATHEMATICAL MODEL OF CRANE
TROLLEY AT MID

ATTACHMENT NO. 3 TO AEP:NRC:0894L

Letter Report MT/SMART-090(89)

APPROVED
IN GENERAL
ENGINEERING DEPARTMENT
AMERICAN ELECTRIC POWER SERVICE CORP.
PER ______ DATE Sept. 6, 1989

D. C. COOK UNIT 2

REACTOR VESSEL HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION

April 1989

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ATTACHMENT NO. 3

TO

AEP:NRC:0894L

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HEATUP AND COOLDOWN LIMIT CURVES
FOR NORMAL OPERATION

1.0 INTRODUCTION

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} (reference nil-ductility temperature) for the reactor vessel. The most limiting RT_{NDT} of the material in the core region of the reactor vessel is determined by using the preservice reactor vessel material fracture toughness properties and estimating the radiation-induced ΔRT_{NDT}. RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99 Rev. 2 (Radiation Embrittlement of Reactor Vessel Materials)[1].

2.0 FRACTURE TOUGHNESS PROPERTIES

The fracture-toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the NRC Regulatory Standard Review Plan[2]. The pre-irradiation fracture-toughness properties for the materials in the D. C. Cook Unit 2 reactor vessel are presented in table 1.

3.0 ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99 Rev. 2 [1] the adjusted reference temperature (ART) for each material in the beltline is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \qquad (1)$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta RT_{NDT} = [CF]f^{(0.28-0.10 \log f)} = [CF]\,[ff] \qquad (2)$$

The value, "f", used in equation (2) is the calculated value of the neutron fluence at the location in the vessel at the location of the postulated defect, n/cm^2 ($E > 1$ MeV) divided by 10^{19}. The fluence factor, "ff" is shown in figure 1.

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth.

$$f_{(\text{depth } X)} = f_{\text{surface}}(e^{-.24x}) \qquad (3)$$

where x (in inches) is the depth into the vessel wall measured from the vessel inner (wetted) surface. The attenuated fluence is then used in equation (2) to calculate ΔRT_{NDT} at the specific depth.

CF (°F) is the chemistry factor, obtained from reference 1 for the beltline region materials of the D. C. Cook Unit 2 reactor pressure vessel. The

limiting material was found to be the intermediate shell plate C5556-2 for D. C. Cook Unit 2 for 12 EFPY and 32 EFPY. The calculation of ART for this limiting material is shown in table 2. The ART values at 1/4T and 3/4T locations will be used to develop the reactor pressure vessel heatup and cooldown curves as described in the following sections.

4.0 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code[3]. The K_{IR} curve is given by the following equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145 (T-RT_{NDT}^{*} + 160)] \qquad (4)$$

where

K_{IR} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}*

Therefore, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code[3] as follows:

$$C\, K_{IM} + K_{IT} \leq K_{IR} \qquad (5)$$

*NOTE: RT_{NDT} as used in the ASME Code [3] is in fact the adjusted reference temperature (ART) as defined in NRC Regulatory Guide 1.99, Rev. 2 [1] and calculated in section 3.0.

where

K_{IM} = stress intensity factor caused by membrane (pressure) stress

K_{IT} = stress intensity factor caused by the thermal gradients

K_{IR} = function of temperature relative to the RT_{NDT} of the material

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{IT}, for the reference flaw are computed. From equation (5), the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G to the ASME Code [3] is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw.

During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel inside surface. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4 T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{IR} exceeds K_{IT}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4 T location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4 T defect at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4 T crack during heatup is lower than the K_{IR} for the 1/4 T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{IR}'s do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4 T flaw is considered. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a 1/4 T deep outside

surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside, and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the 1983 Amendment to 10CFR50[4] has a rule which addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure.

5.0 HEATUP AND COOLDOWN LIMIT CURVES

Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated using the methods described in section 4.0, and the Westinghouse procedure of reference 5.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in figures 2

3, 6 and 7 for 12 EFPY and in figures 4, 5, 8 and 9 for 32 EFPY. Figures 2 through 5 do not have any margins for instrumentation error and figures 6 through 9 contain margins for instrumentation error. This is in addition to other criteria which must be met before the reactor is made critical.

The leak limit curve shown in figures 2, 4, 6 and 8 represent the minimum temperature requirements at the leak test pressure specified by applicable codes[2,3]. The leak test limit curves were determined by the methods of references 2 and 4.

Finally, table 1 indicates that the limiting flange RT_{NDT} of 30°F occurs in the vessel flange so the minimum allowable temperature of this region is 150°F per reference 4. These limits are less restrictive than the limits shown on figures 2 through 9.

Figures 2 through 9 define the limits for ensuring prevention of nonductile failure for the D. C. Cook Unit 2 Primary Reactor Coolant System.

6.0 REFERENCES

1. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May, 1988.

2. "Fracture Toughness Requirements," Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

3. ASME Boiler and Pressure Vessel Code, Section III, Division 1 - Appendixes, "Rules for Construction of Nuclear Power Plant Components, Appendix G, Protection Against Nonductile Failure," pp. 558-563, 1986 Edition, American Society of Mechanical Engineers, New York, 1986.

4. Code of Federal Regulations, 10CFR50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Vol. 48 No. 104, May 27, 1983.

5. "Procedure for Developing Heatup and Cooldown Curves," Westinghouse Electric Corporation, Generation Technology Systems Division Procedure GTSD-A-1.12 (Rev. 0), July 13, 1988.

6. "Assessment of Regulatory Guide 1.99, Revision 2 Upon D. C. Cook Units 1 and 2 Adjusted Reference Temperatures and Pressure-Temperature Curves" SWRI Project 17-2544.

TABLE 1

D. C. COOK UNIT 2 REACTOR VESSEL TOUGHNESS PROPERTIES (UNIRRADIATED)

Component	Code No.	Material Type	Cu (%)	Ni (%)	T_{NDT} (^oF)	50 ft-lb 35 Mil Temp(a) (^oF)	RT_{NDT} (^oF)	USE MWD (ft-lb)	USE NMWD(a) (ft-lb)
Closure Head Dome	B0048-2	A533B Cl. 1	NA	0.64	-20	30	-20	148	96
Closure Head Segment	B9883-2	A533B Cl. 1	NA	0.66	-20	- 3	-20	143.5	93
Closure Head Segment	A5189-2	A533B Cl. 1	NA	0.63	10	72	12	140.5	91
Closure Head Flange	4437-V-1	A508 Cl. 2	NA	0.70	-20	5	-20	239	155
Vessel Flange	4436-V-2	A508 Cl. 2	NA	0.70	30	15	30	161	105
Inlet Nozzle	269T-2	A508 Cl. 2	NA	0.85	-20	-15	-20	201.5	131
Inlet Nozzle	270T-1	A508 Cl. 2	NA	0.91	-20	- 3	-20	239.5	156
Inlet Nozzle	269T-1	A508 Cl. 2	NA	NA	-10	NA	-10	NA	NA
Inlet Nozzle	270T-2	A508 Cl. 2	NA	NA	-10	NA	-10	NA	NA
Outlet Nozzle	271T-1	A508 Cl. 2	NA	0.80	0	12	0	>179	NA
Outlet Nozzle	271T-2	A508 Cl. 2	NA	0.80	0	-15	0	181	117.5
Outlet Nozzle	272T-1	A508 Cl. 2	NA	NA	-10	NA	-10	NA	NA
Outlet Nozzle	272T-2	A508 Cl. 2	NA	NA	0	NA	0	NA	NA
Upper Shell	C5518-2	A533B Cl. 1	0.12	0.61	10	88	28	107.5	70
Upper Shell	C5521-1	A533B Cl. 1	0.14	0.59	0	93	33	112	73
Upper Shell	C5518-1	A533B Cl. 1	0.12	0.57	10	66	10	> 82.5	NA
Inter. Shell	C5556-2	A533B Cl. 1	0.15	0.57	0	118(b)	58(b)	109.5	90(b)
Inter Shell	C5521-2	A533B Cl. 1	0.14	0.58	10	98(b)	38(b)	111.5	86(b)
Lower Shell	C5540-2	A533B Cl. 1	0.11	0.64	-20	35(b)	-20(b)	113	110(b)
Lower Shell	C5592-1	A533B Cl. 1	0.14	0.59	-20	25(b)	-20(b)	107	103(b)
Bottom Head Segment	C5823-2	A533B Cl. 1	NA	0.57	-10	45	-10	129	84
Bottom Head Segment	A4957-3	A533B Cl. 1	NA	0.51	-10	20	-10	149	97
Bottom Head Dome	B0018-18	A533B Cl. 1	NA	0.61	-50	9	-50	177	115
Inter. & Lower Shell Long. and Girth Weld Seam	(HT S3986 & SAW Linde 124 Flux Lot No. 0934)		0.06	0.97	-40	25	-35(b)	NA	97(b)

a) Estimated per NRC Standard Review Plan
b) Actual values
NA - Not available or not applicable, as appropriate

MWD - Major Working Direction
NMWD - Normal to MWD

TABLE 2

CALCULATION OF ADJUSTED REFERENCE TEMPERATURES FOR LIMITING D. C. COOK UNIT 2 REACTOR VESSEL MATERIAL - INTERMEDIATE SHELL PLATE C5556-2

	Regulatory Guide 1.99 - Revision 2			
	12 EFPY		32 EFPY	
Parameter	1/4 T	3/4 T	1/4 T	3/4 T
Chemistry Factor, CF (°F)	108.4	108.4	108.4	108.4
Fluence, f (10^{19} n/cm^2)[a]	0.47	0.17	1.24	0.44
Fluence Factor, ff	0.79	0.53	1.06	0.78

****

ΔRT_{NDT} = CF x ff (°F)	86	58	115	85
Initial RT_{NDT}, I (°F)	58	58	58	58
Margin, M (°F)[b]	34	34	34	34

****

Revision 2 to Regulatory Guide 1.99

Adjusted Reference Temperature, ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin	178	150	207	177

****

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1 Mev) = 2.1 at 32 EFPY [6] at vessel inner surface. The D. C. Cook Unit 2 reactor vessel wall thickness is 8.63 inches at the beltline region.

(b) Margin, M = $2[\sigma_I^2 + \sigma_\Delta^2]^{0.5}$, σ_I is the standard deviation for initial RT_{NDT} and σ_Δ is the standard deviation for ΔRT_{NDT}. σ_I is assumed to be 0°F for measured values of initial RT_{NDT} and σ_Δ is 17°F for base metal [1].

TABLE 3

CALCULATION OF ADJUSTED REFERENCE TEMPERATURES FOR D. C. COOK UNIT 2 REACTOR VESSEL MATERIAL - INTERMEDIATE SHELL PLATE C5521-2

	Regulatory Guide 1.99 - Revision 2			
	12 EFPY		32 EFPY	
Parameter	1/4 T	3/4 T	1/4 T	3/4 T
Chemistry Factor, CF (°F)[a]	110	110	110	110
Fluence, f (10^{19} n/cm^2)[b]	0.47	0.17	1.24	0.44
Fluence Factor, ff	0.79	0.53	1.06	0.78

****

ΔRT_{NDT} = CF x ff (°F)	87	58	117	86
Initial RT_{NDT}, I (°F)	38	38	38	38
Margin, M (°F)[c]	17	17	17	17

****

Revision 2 to Regulatory Guide 1.99

Adjusted Reference Temperature, ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin	142	113	172	141

****

(a) Based on surveillance capsule data.

(b) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1 Mev) = 2.1 at 32 EFPY [6] at vessel inner surface. The D. C. Cook Unit 2 reactor vessel wall thickness is 8.63 inches at the beltline region.

(c) Margin, $M = 2[\sigma_I^2 + \sigma_\Delta^2]^{.5}$, σ_I is the standard deviation for initial RT_{NDT} and σ_Δ is the standard deviation for ΔRT_{NDT}. σ_I is assumed to be 0°F for measured values of initial RT_{NDT} and σ_Δ is 17°F for base metal. σ_Δ is cut in half to 8.5°F, since surveillance capsule data is used [1].

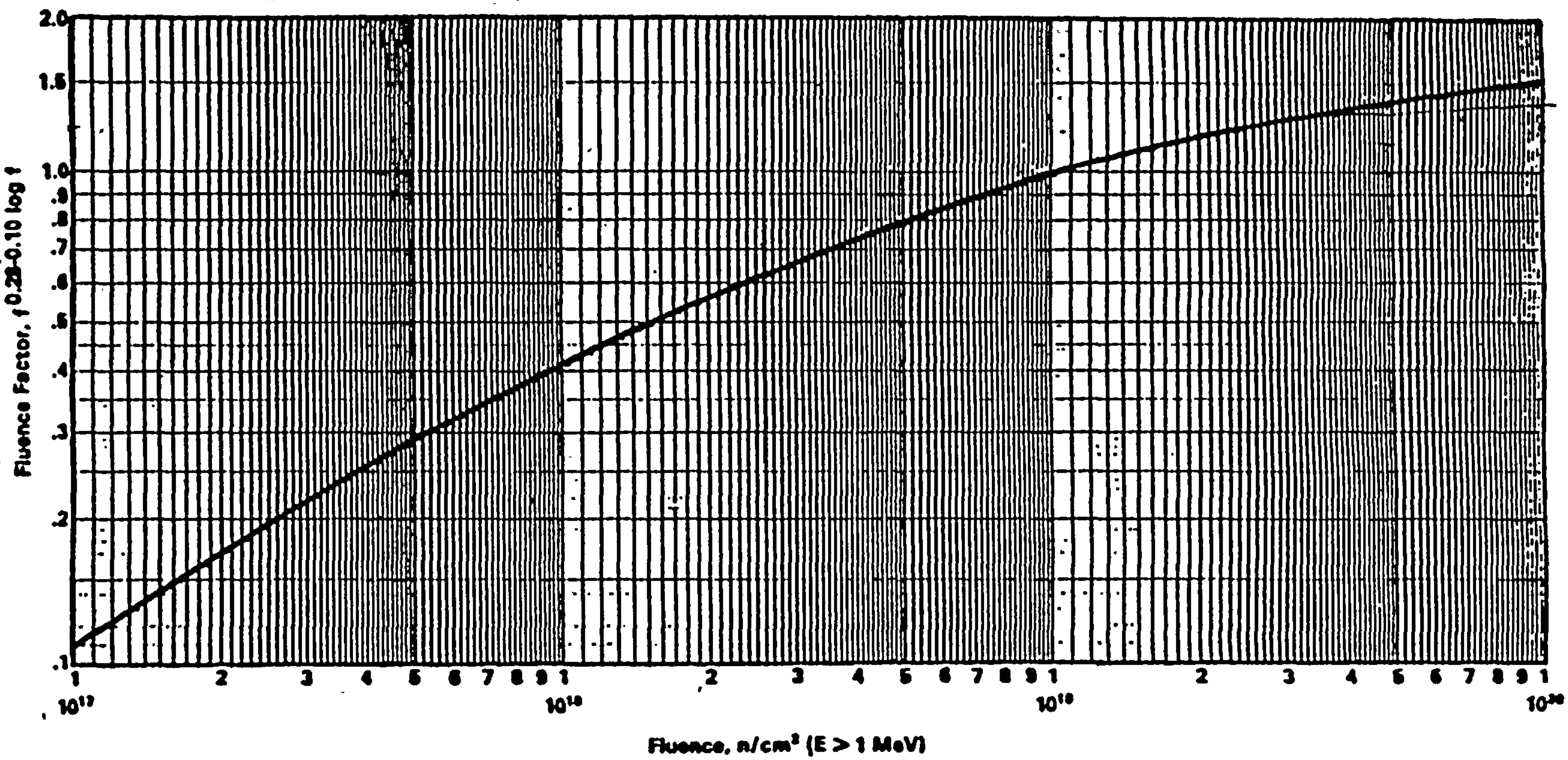

Figure 1. Fluence Factor for Use in the Expression for ΔRT_{NDT}

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE C5556-2

RT_{NDT} AFTER 12 EFPY: 1/4T, 178°F

3/4T, 150°F

CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 12 EFPY. DOES NOT CONTAIN MARGIN FOR POSSIBLE INSTRUMENT ERRORS.

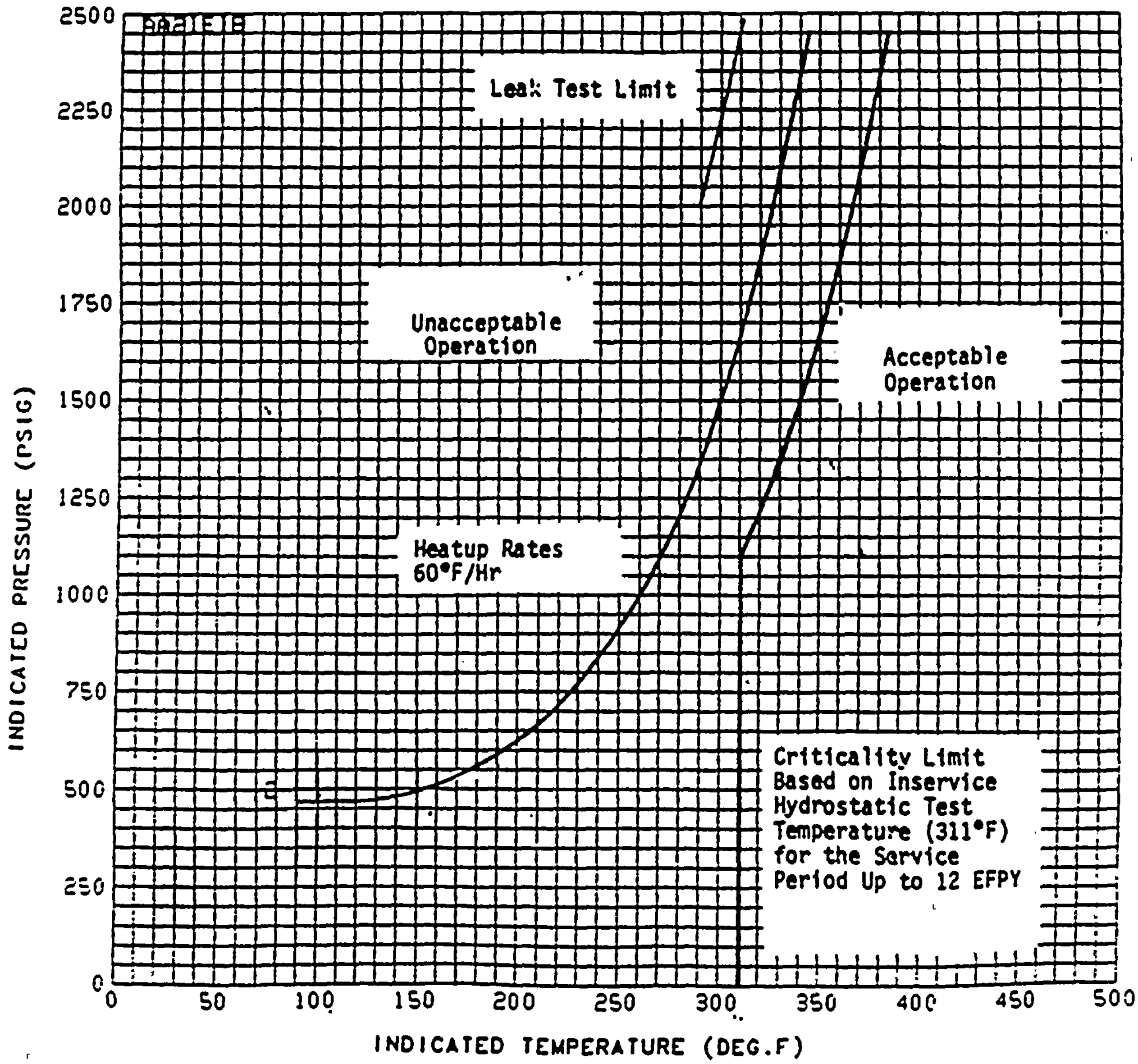

Figure 2. D. C. Cook Unit 2 Reactor Coolant System Heatup Limitations Applicable for the First 12 EFPY (Without Margins)

[illegible]

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE C5556-2

RT_{NDT} AFTER 12 EFPY: 1/4T, 178°F

3/4T, 150°F

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 12 EFPY. DOES NOT CONTAIN MARGIN FOR POSSIBLE INSTRUMENT ERRORS.

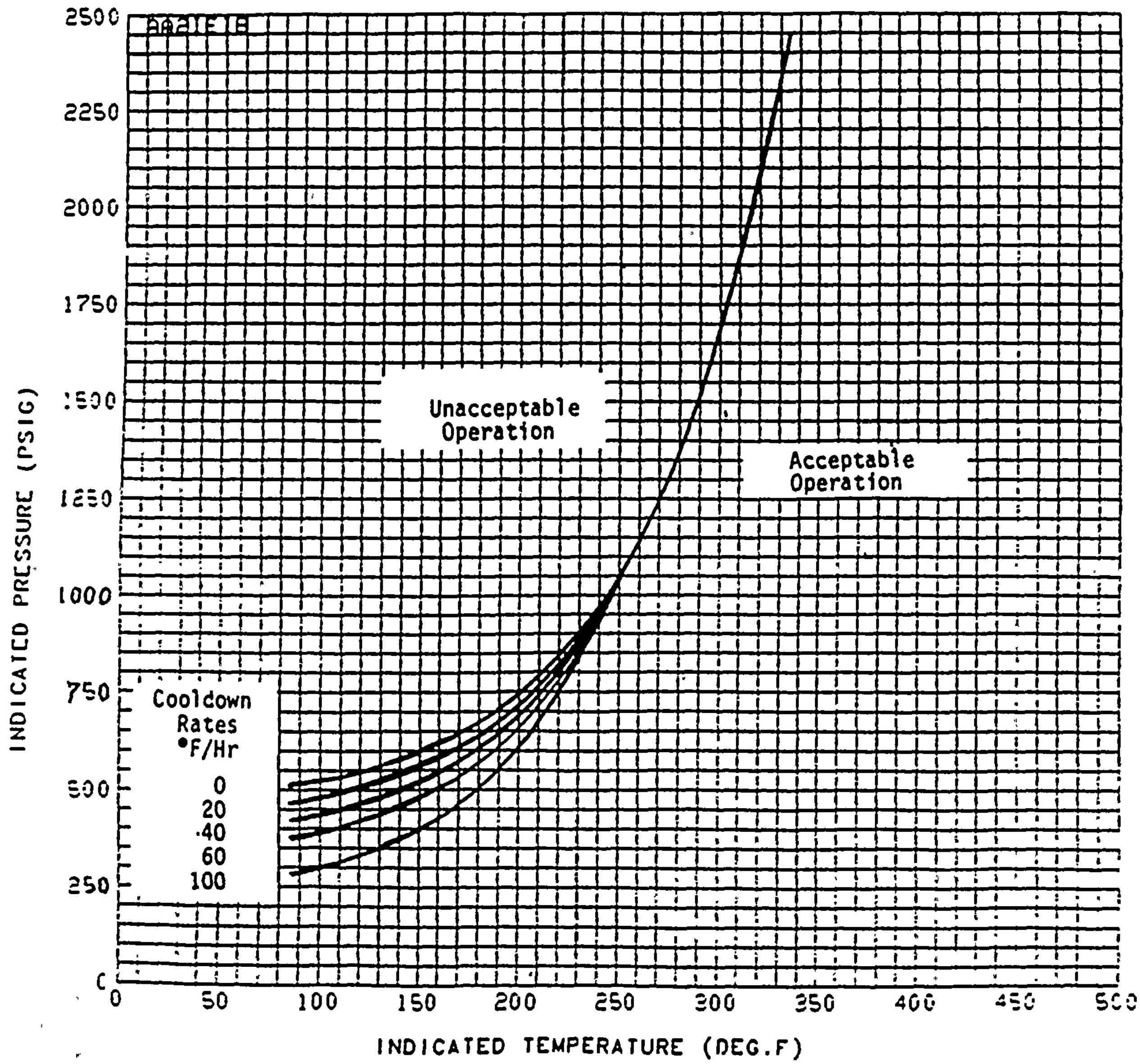

Figure 3. D. C. Cook Unit 2 Reactor Coolant System Cooldown Limitations Applicable for the First 12 EFPY (Without Margins)

[illegible]

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE C5556-2

RT_{NDT} AFTER 32 EFPY: 1/4T, 207°F

3/4T, 177°F

CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 32 EFPY. DOES NOT CONTAIN MARGIN FOR POSSIBLE INSTRUMENT ERRORS.

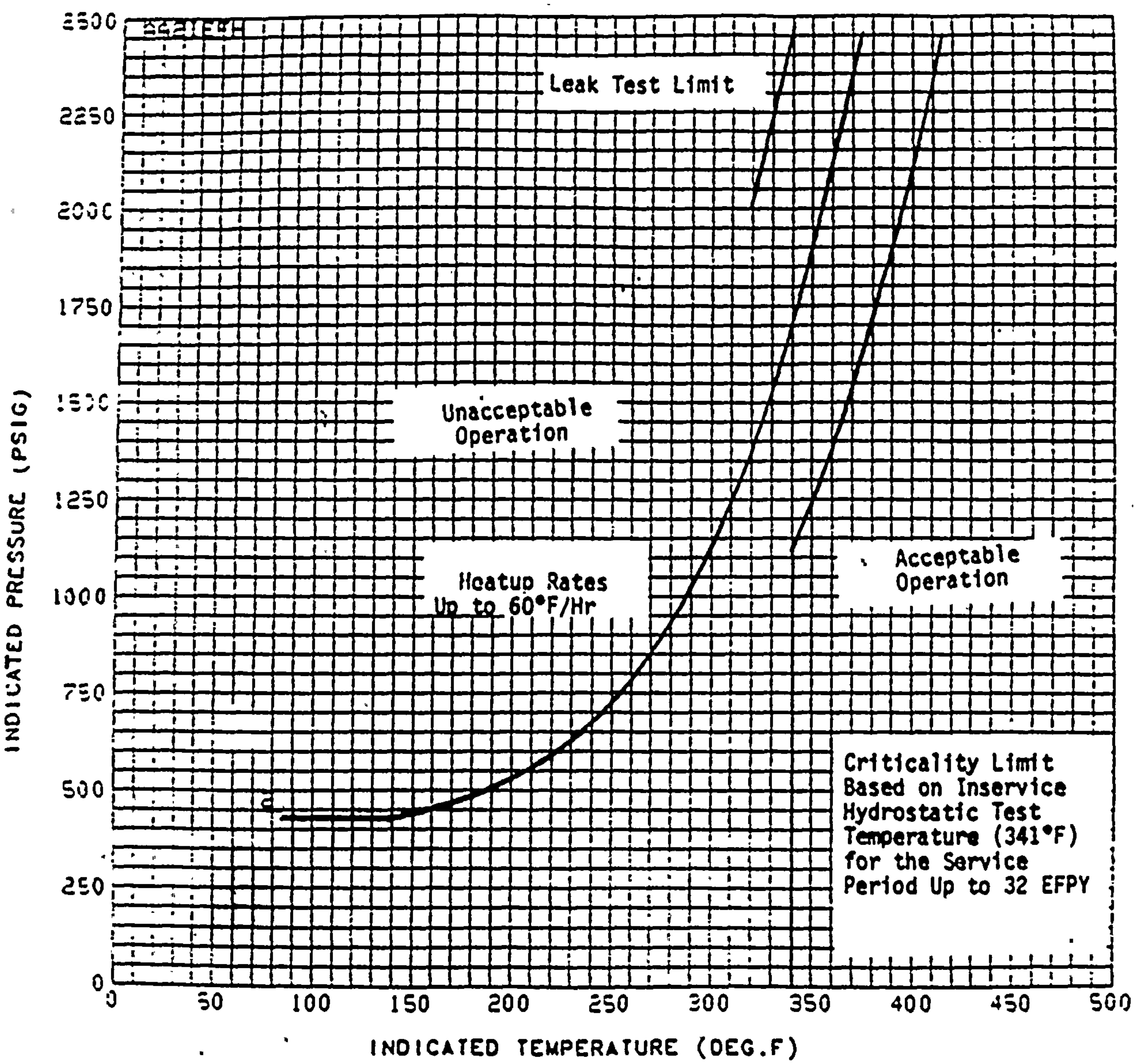

Figure 4. D. C. Cook Unit 2 Reactor Coolant System Heatup Limitations Applicable for the First 32 EFPY (Without Margins)

[illegible]

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE C5556-2

RT_{NDT} AFTER 32 EFPY: 1/4T, 207°F

3/4T, 177°F

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 32 EFPY. DOES NOT CONTAIN MARGIN FOR POSSIBLE INSTRUMENT ERRORS.

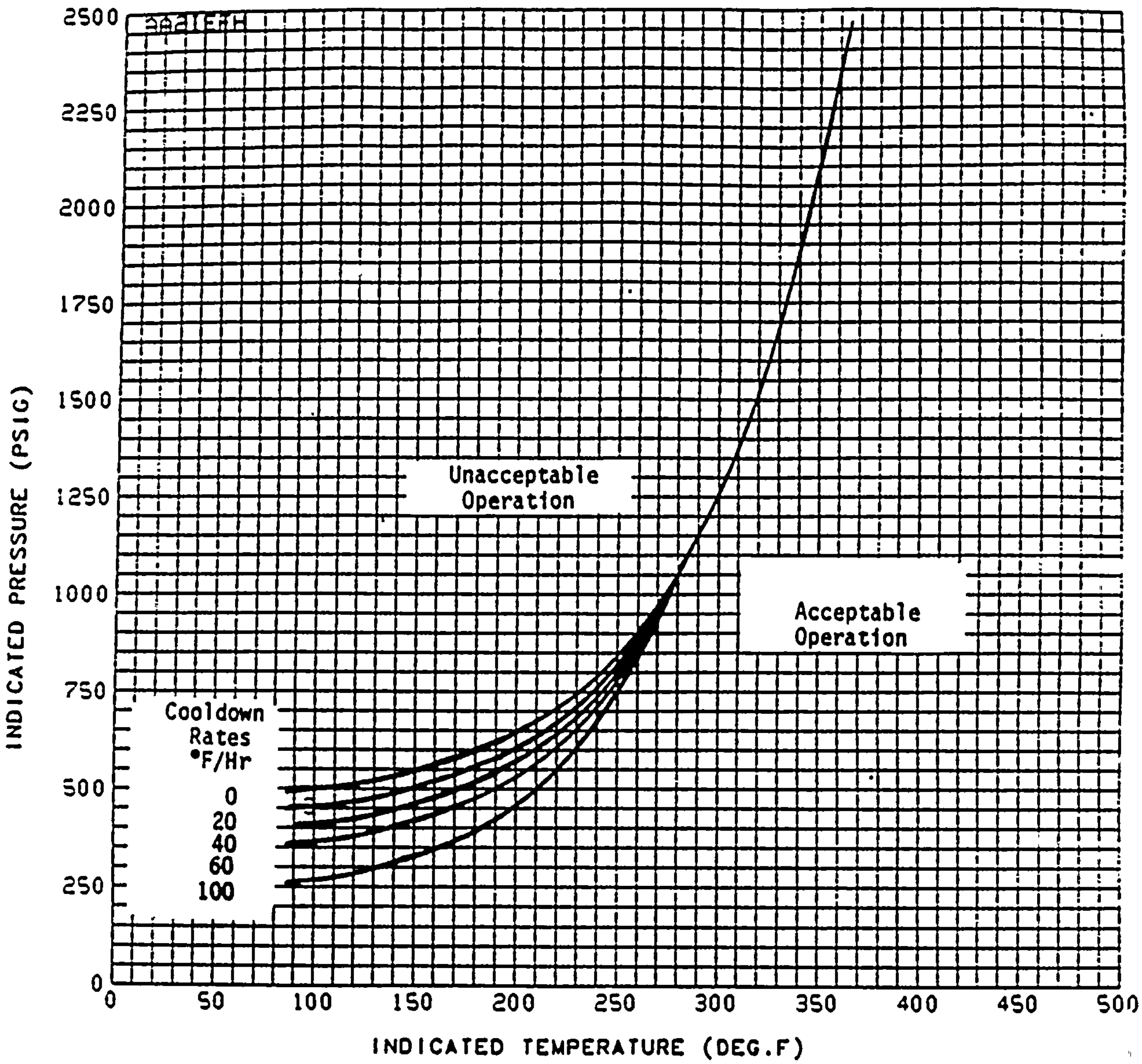

Figure 5. D. C. Cook Unit 2 Reactor Coolant System Cooldown Limitations Applicable for the First 32 EFPY (Without Margins)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE C5556-2

RT_{NDT} AFTER 12 EFPY: 1/4T, 178°F

3/4T, 150°F

CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 12 EFPY. CONTAIN MARGIN OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

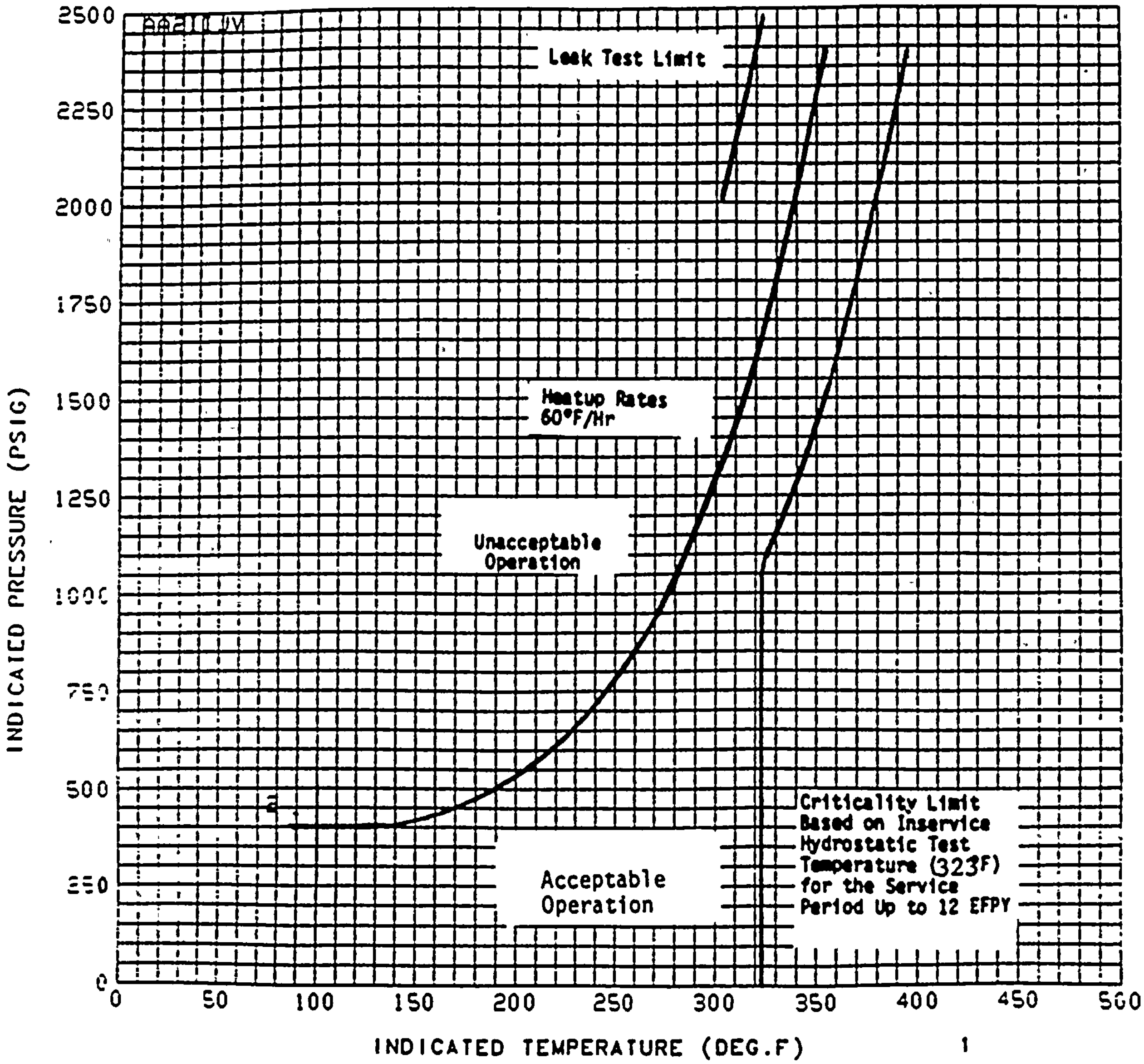

Figure 6. D. C. Cook Unit 2 Reactor Coolant System Heatup Limitations Applicable for the First 12 EFPY (With Margins)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE C5556-2

RT_{NDT} AFTER 12 EFPY: 1/4T, 178°F

3/4T, 150°F

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 12 EFPY. CONTAIN MARGIN OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

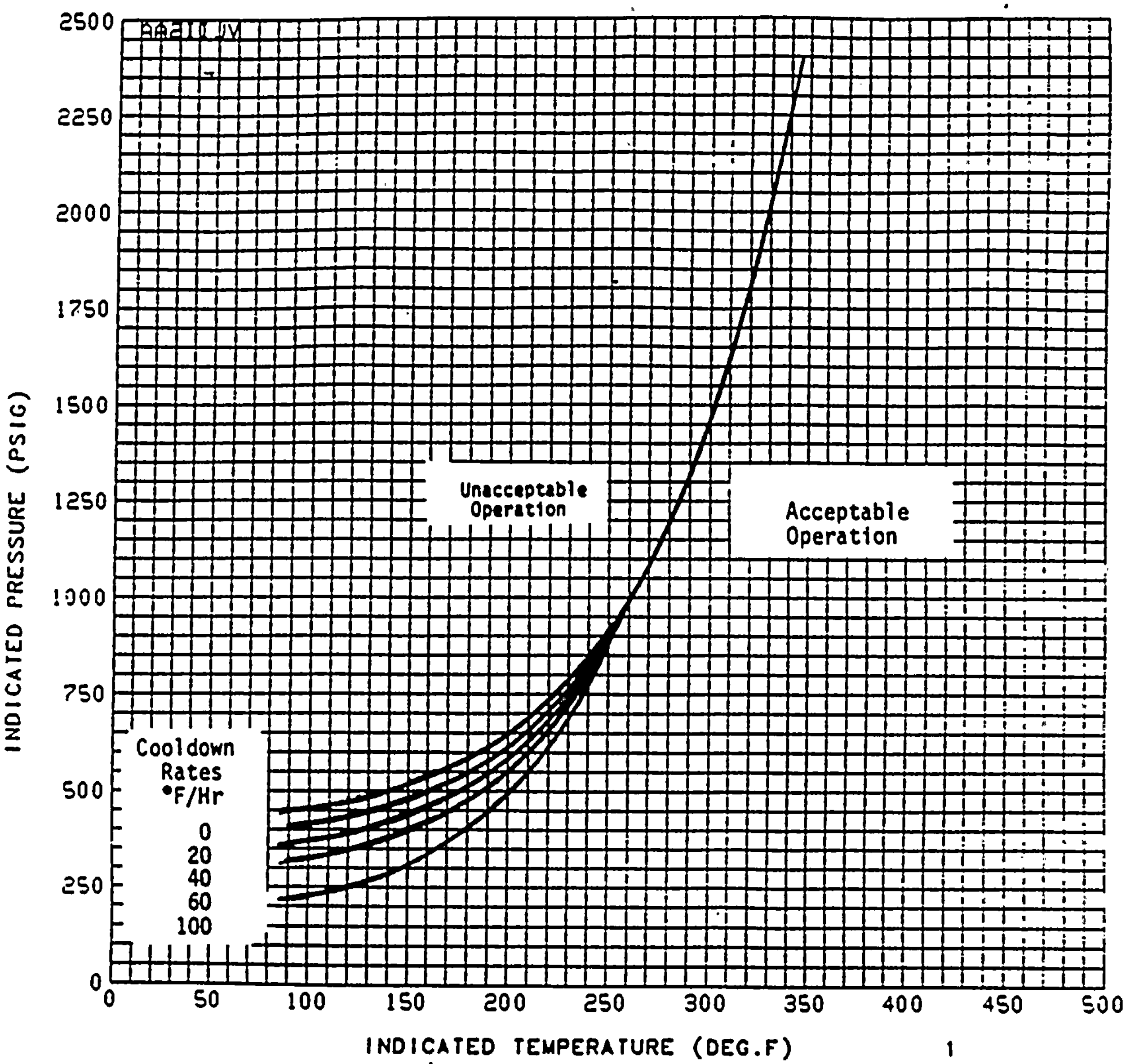

Figure 7. D. C. Cook Unit 2 Reactor Coolant System Cooldown Limitations Applicable for the First 12 EFPY (With Margins)

3716s/041989:10

[illegible]

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE C5556-2

RT_{NDT} AFTER 32 EFPY: 1/4T, 207°F

3/4T, 177°F

CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 32 EFPY. CONTAIN MARGIN OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

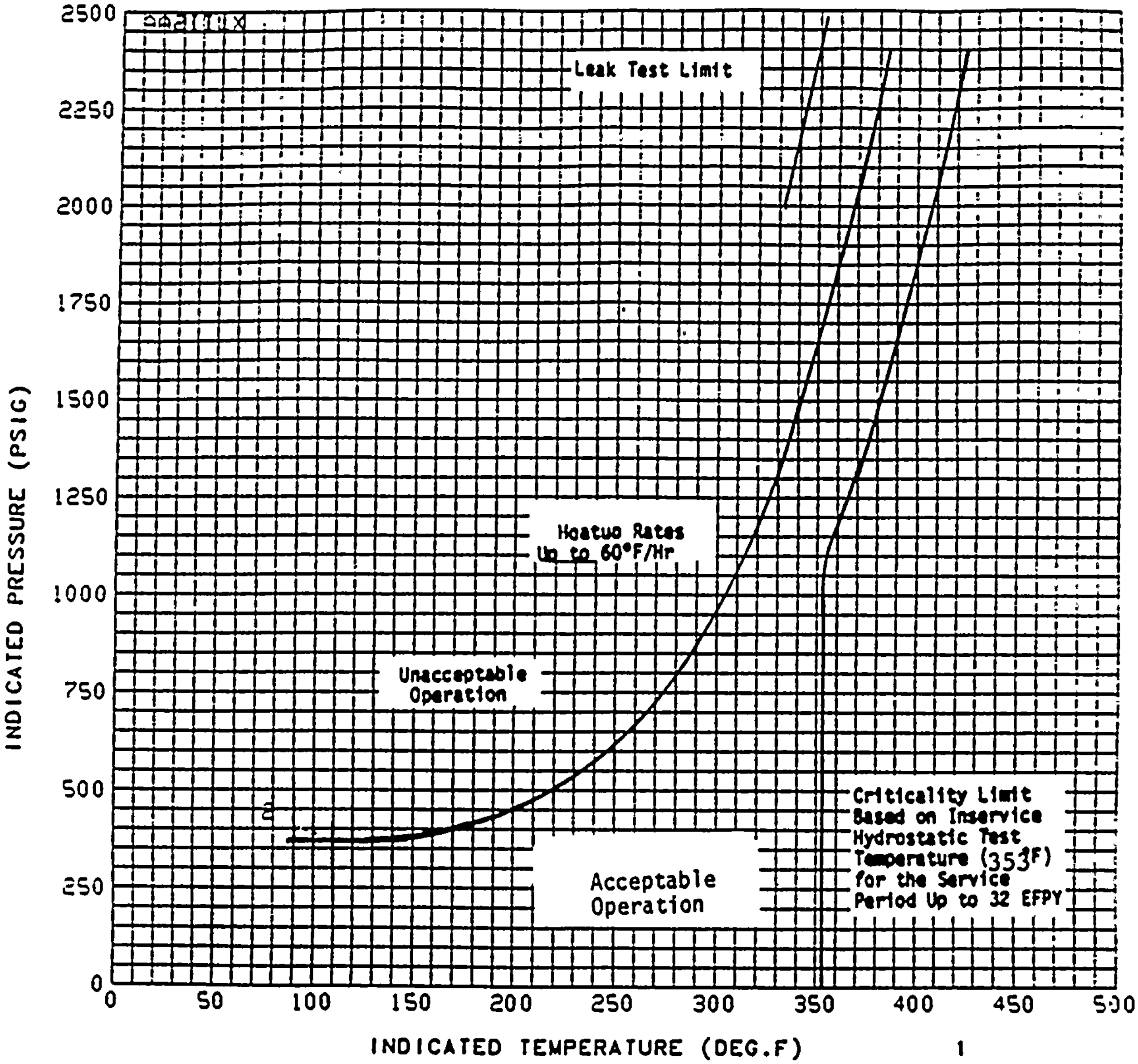

Figure 8. D. C. Cook Unit 2 Reactor Coolant System Heatup Limitations Applicable for the First 32 EFPY (With Margins)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE C5556-2

RT_{NDT} AFTER 32 EFPY: 1/4T, 207°F

3/4T, 177°F

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 32 EFPY. CONTAIN MARGIN OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

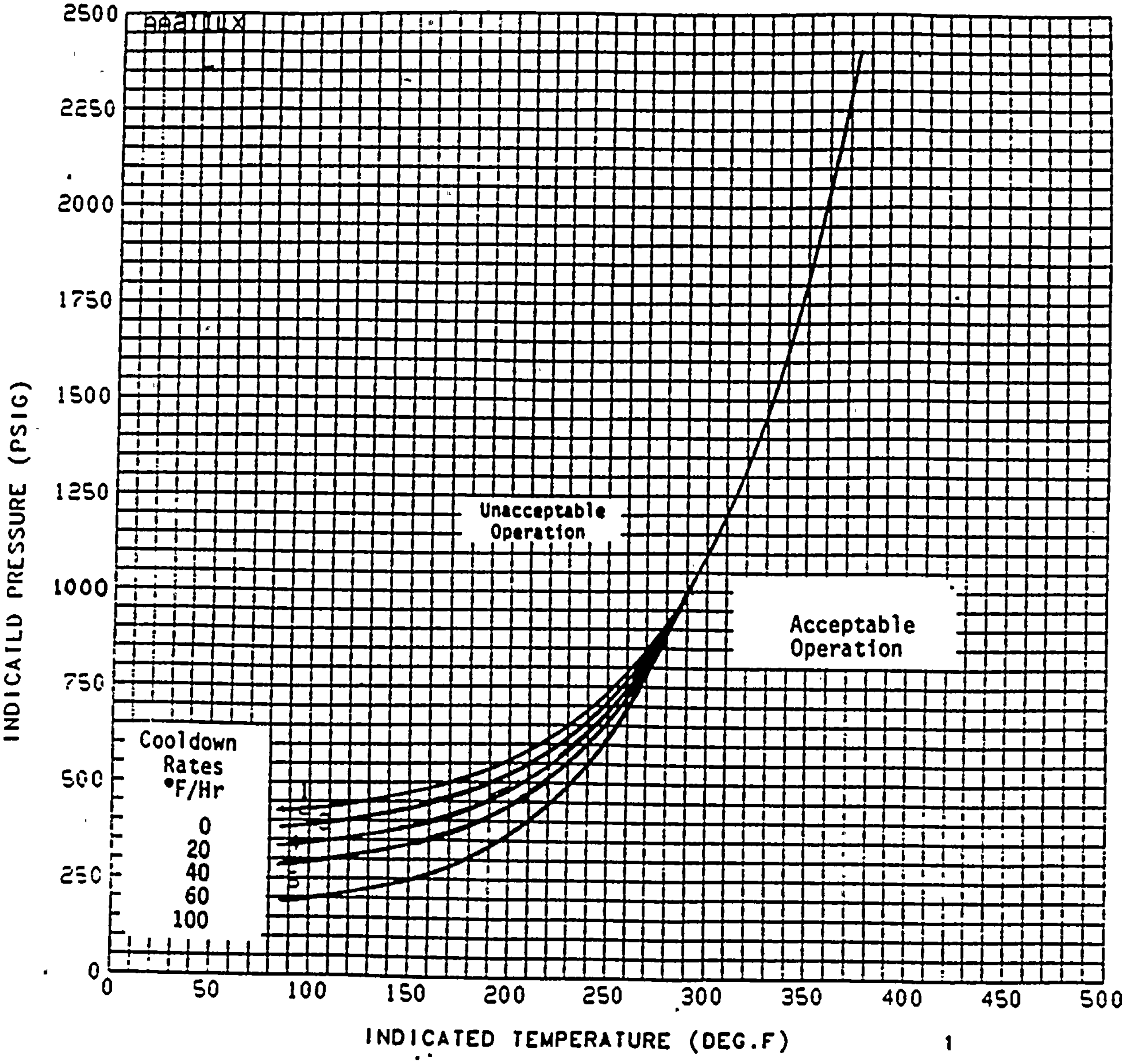

Figure 9. D. C. Cook Unit 2 Reactor Coolant System Cooldown Limitations Applicable for the First 32 EFPY (With Margins)

APPENDIX A
HEATUP AND COOLDOWN DATA
WITHOUT MARGINS FOR INSTRUMENTATION ERROR

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/17/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 1 (STEADY-STATE COOLDOWN)

IRRADIATION PERIOD = 12.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	509.21	18	170.000	642.13	35	255.000	1095.62
2	90.000	513.34	19	175.000	656.27	36	260.000	1143.47
3	95.000	517.78	20	180.000	671.48	37	265.000	1194.84
4	100.000	522.55	21	185.000	687.65	38	270.000	1249.75
5	105.000	527.68	22	190.000	705.22	39	275.000	1309.08
6	110.000	533.19	23	195.000	724.07	40	280.000	1372.63
7	115.000	539.02	24	200.000	744.22	41	285.000	1440.59
8	120.000	545.39	25	205.000	766.03	42	290.000	1513.87
9	125.000	552.24	26	210.000	789.30	43	295.000	1592.21
10	130.000	559.61	27	215.000	814.47	44	300.000	1675.99
11	135.000	567.54	28	220.000	841.39	45	305.000	1765.86
12	140.000	576.05	29	225.000	870.44	46	310.000	1862.27
13	145.000	585.09	30	230.000	901.57	47	315.000	1965.54
14	150.000	594.93	31	235.000	934.98	48	320.000	2075.96
15	155.000	605.51	32	240.000	970.84	49	325.000	2193.76
16	160.000	616.89	33	245.000	1009.61	50	330.000	2319.71
17	165.000	629.12	34	250.000	1051.09	51	335.000	2464.07

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/17/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 2 (20 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 12.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	466.48	13	145.000	545.60	25	205.000	738.03
2	90.000	470.69	14	150.000	555.98	26	210.000	763.11
3	95.000	475.26	15	155.000	567.17	27	215.000	789.84
4	100.000	480.17	16	160.000	579.20	28	220.000	818.93
5	105.000	485.48	17	165.000	592.06	29	225.000	850.03
6	110.000	491.19	18	170.000	606.00	30	230.000	883.38
7	115.000	497.28	19	175.000	621.04	31	235.000	919.27
8	120.000	503.91	20	180.000	637.05	32	240.000	958.08
9	125.000	511.08	21	185.000	654.47	33	245.000	999.63
10	130.000	518.79	22	190.000	673.17	34	250.000	1044.25
11	135.000	527.11	23	195.000	693.19	35	255.000	1092.25
12	140.000	536.05	24	200.000	714.87			

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/17/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 3 (40 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 12.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	422.80	13	145.000	505.83	25	205.000	711.06
2	90.000	427.14	14	150.000	516.61	26	210.000	737.79
3	95.000	431.85	15	155.000	528.48	27	215.000	766.80
4	100.000	436.93	16	160.000	541.15	28	220.000	797.81
5	105.000	442.38	17	165.000	554.94	29	225.000	831.21
6	110.000	448.32	18	170.000	569.78	30	230.000	867.30
7	115.000	454.76	19	175.000	585.68	31	235.000	906.02
8	120.000	461.70	20	180.000	602.92	32	240.000	947.64
9	125.000	469.22	21	185.000	621.52	33	245.000	992.43
10	130.000	477.32	22	190.000	641.39	34	250.000	1040.58
11	135.000	486.08	23	195.000	662.97	35	255.000	1092.38
12	140.000	495.43	24	200.000	686.02			

A4

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/17/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 4 (60 DEG-F / HR COOLDOWN.)

IRRADIATION PERIOD = 12.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	378.13	13	145.000	465.16	24	200.000	658.17
2	90.000	382.60	14	150.000	476.82	25	205.000	684.90
3	95.000	387.49	15	155.000	489.45	26	210.000	713.87
4	100.000	392.77	16	160.000	502.96	27	215.000	744.91
5	105.000	398.52	17	165.000	517.67	28	220.000	778.32
6	110.000	404.73	18	170.000	533.51	29	225.000	814.52
7	115.000	411.43	19	175.000	550.53	30	230.000	853.34
8	120.000	418.72	20	180.000	568.97	31	235.000	895.11
9	125.000	426.63	21	185.000	588.77	32	240.000	940.06
10	130.000	435.17	22	190.000	610.22	33	245.000	988.47
11	135.000	444.36	23	195.000	633.23	34	250.000	1040.52
12	140.000	454.35						

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/17/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 5 (100 DEG-F/HR COOLDOWN)

IRRADIATION PERIOD = 12.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	285.51	12	140.000	370.53	23	195.000	576.78
2	90.000	290.35	13	145.000	382.83	24	200.000	605.64
3	95.000	295.67	14	150.000	396.12	25	205.000	636.83
4	100.000	301.46	15	155.000	410.49	26	210.000	670.59
5	105.000	307.80	16	160.000	426.07	27	215.000	706.88
6	110.000	314.67	17	165.000	442.89	28	220.000	745.98
7	115.000	322.16	18	170.000	461.12	29	225.000	788.17
8	120.000	330.30	19	175.000	480.87	30	230.000	833.60
9	125.000	339.19	20	180.000	502.08	31	235.000	882.61
10	130.000	348.80	21	185.000	525.14	32	240.000	935.38
11	135.000	359.27	22	190.000	549.89	33	245.000	992.23

AMP 60F/HR HEATUP CURVE REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/17/89

THE FOLLOWING DATA WERE CALCULATEDFOR THE INSERVICE HYDROSTATIC LEAK TEST.

MINIMUM INSERVICE LEAK TEST TEMPERATURE (12.000 EFPY)

PRESSURE (PSI)	TEMPERATURE (DEG.F)
2000	290
2485	311

PRESSURE (PSI)	PRESSURE STRESS (PSI)	1.5 K1M (PSI SQ.RT.IN.)
2000	21444	89745
2485	26645	112505

A7

AMP 60F/HR HEATUP CURVE REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/17/89

COMPOSITE CURVE PLOTTED FOR HEATUP PROFILE 2 HEATUP RATE(S) (DEG.F/HR) = 60.0

IRRADIATION PERIOD = 12.000 EFP YEARS
FLAW DEPTH = (1-AOWIN)T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	~~481.20~~	19	175.000	545.24	37	265.000	1044.56
2	90.000	~~471.71~~	20	180.000	558.93	38	270.000	1095.79
3	95.000	~~465.04~~ 457.04	21	185.000	573.79	39	275.000	1150.75
4	100.000	~~460.48~~	22	190.000	589.71	40	280.000	1209.57
5	105.000	~~457.88~~	23	195.000	607.06	41	285.000	1272.94
6	110.000	457.04	24	200.000	625.76	42	290.000	1340.67
7	115.000	457.64	25	205.000	645.78	43	295.000	1413.27
8	120.000	459.42	26	210.000	667.48	44	300.000	1490.95
9	125.000	462.44	27	215.000	690.70	45	305.000	1574.43
10	130.000	466.44	28	220.000	715.85	46	310.000	1663.52
11	135.000	471.51	29	225.000	742.72	47	315.000	1759.00
12	140.000	477.47	30	230.000	771.78	48	320.000	1860.76
13	145.000	484.40	31	235.000	802.89	49	325.000	1970.00
14	150.000	492.19	32	240.000	836.27	50	330.000	2086.46
15	155.000	500.85	33	245.000	872.14	51	335.000	2210.89
16	160.000	510.50	34	250.000	910.88	52	340.000	2329.54
17	165.000	521.12	35	255.000	952.33	53	345.000	2452.78
18	170.000	532.71	36	260.000	996.79			

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2

04/17/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 1 (STEADY-STATE COOLDOWN)

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	490.18	20	180.000	595.92	39	275.000	1013.49
2	90.000	492.78	21	185.000	606.58	40	280.000	1055.25
3	95.000	495.68	22	190.000	618.04	41	285.000	1100.10
4	100.000	498.79	23	195.000	630.35	42	290.000	1148.28
5	105.000	502.13	24	200.000	643.46	43	295.000	1199.86
6	110.000	505.73	25	205.000	657.70	44	300.000	1255.36
7	115.000	509.60	26	210.000	672.99	45	305.000	1315.11
8	120.000	513.75	27	215.000	689.29	46	310.000	1379.04
9	125.000	518.22	28	220.000	706.99	47	315.000	1447.54
10	130.000	523.03	29	225.000	725.97	48	320.000	1521.18
11	135.000	528.19	30	230.000	746.26	49	325.000	1600.05
12	140.000	533.75	31	235.000	768.20	50	330.000	1684.61
13	145.000	539.61	32	240.000	791.66	51	335.000	1775.12
14	150.000	546.03	33	245.000	817.01	52	340.000	1872.08
15	155.000	552.83	34	250.000	844.12	53	345.000	1975.92
16	160.000	560.35	35	255.000	873.20	54	350.000	2086.92
17	165.000	568.33	36	260.000	904.72	55	355.000	2205.68
18	170.000	576.91	37	265.000	938.36	56	360.000	2332.27
19	175.000	586.01	38	270.000	974.48	57	365.000	2467.28

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/17/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 2 (20 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	446.03	15	155.000	510.72	29	225.000	694.37
2	90.000	448.71	16	160.000	518.49	30	230.000	716.23
3	95.000	451.62	17	165.000	526.88	31	235.000	739.59
4	100.000	454.74	18	170.000	535.90	32	240.000	764.88
5	105.000	458.14	19	175.000	545.53	33	245.000	791.94
6	110.000	461.79	20	180.000	555.99	34	250.000	821.18
7	115.000	465.74	21	185.000	567.28	35	255.000	852.55
8	120.000	469.99	22	190.000	579.41	36	260.000	886.18
9	125.000	474.60	23	195.000	592.37	37	265.000	922.38
10	130.000	479.55	24	200.000	606.44	38	270.000	961.48
11	135.000	484.91	25	205.000	621.60	39	275.000	1003.38
12	140.000	490.66	26	210.000	637.75	40	280.000	1048.39
13	145.000	496.80	27	215.000	655.32	41	285.000	1096.79
14	150.000	503.49	28	220.000	674.18			

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/17/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 3 (40 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	400.88	15	155.000	467.79	29	225.000	663.47
2	90.000	403.64	16	160.000	475.96	30	230.000	686.75
3	95.000	406.52	17	165.000	484.82	31	235.000	712.03
4	100.000	409.68	18	170.000	494.25	32	240.000	739.04
5	105.000	413.13	19	175.000	504.56	33	245.000	768.31
6	110.000	416.86	20	180.000	515.64	34	250.000	799.66
7	115.000	420.92	21	185.000	527.63	35	255.000	833.40
8	120.000	425.29	22	190.000	540.42	36	260.000	869.82
9	125.000	430.05	23	195.000	554.35	37	265.000	908.86
10	130.000	435.18	24	200.000	569.34	38	270.000	950.97
11	135.000	440.68	25	205.000	585.40	39	275.000	996.26
12	140.000	446.68	26	210.000	602.81	40	280.000	1044.85
13	145.000	453.19	27	215.000	621.60	41	285.000	1097.17
14	150.000	460.20	28	220.000	641.66			

FORM F906

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/17/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 4 (60 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	354.69	15	155.000	424.11	28	220.000	609.81
2	90.000	357.38	16	160.000	432.74	29	225.000	633.21
3	95.000	360.35	17	165.000	442.04	30	230.000	658.32
4	100.000	363.52	18	170.000	452.14	31	235.000	685.37
5	105.000	367.05	19	175.000	463.08	32	240.000	714.67
6	110.000	370.87	20	180.000	474.87	33	245.000	746.08
7	115.000	375.06	21	185.000	487.64	34	250.000	779.88
8	120.000	379.58	22	190.000	501.31	35	255.000	816.51
9	125.000	384.52	23	195.000	516.19	36	260.000	855.78
10	130.000	389.86	24	200.000	532.21	37	265.000	898.05
11	135.000	395.68	25	205.000	549.43	38	270.000	943.52
12	140.000	401.96	26	210.000	568.09	39	275.000	992.47
13	145.000	408.73	27	215.000	588.11	40	280.000	1045.12
14	150.000	416.10						

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/17/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 5 (100 DEG-F/HR COOLDOWN)

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	258.40	14	150.000	325.47	27	215.000	523.34
2	90.000	261.17	15	155.000	334.49	28	220.000	548.47
3	95.000	264.26	16	160.000	344.25	29	225.000	575.78
4	100.000	267.64	17	165.000	354.88	30	230.000	605.08
5	105.000	271.40	18	170.000	366.32	31	235.000	636.74
6	110.000	275.49	19	175.000	378.82	32	240.000	671.01
7	115.000	280.01	20	180.000	392.31	33	245.000	707.88
8	120.000	284.90	21	185.000	406.91	34	250.000	747.53
9	125.000	290.31	22	190.000	422.73	35	255.000	790.35
10	130.000	296.18	23	195.000	439.81	36	260.000	836.45
11	135.000	302.62	24	200.000	458.32	37	265.000	886.17
12	140.000	309.60	25	205.000	478.38	38	270.000	939.69
13	145.000	317.24	26	210.000	499.92	39	275.000	997.32

AMP 60F/HR HEATUP CURVE REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/17/89

THE FOLLOWING DATA WERE CALCULATEDFOR THE INSERVICE HYDROSTATIC LEAK TEST.

MINIMUM INSERVICE LEAK TEST TEMPERATURE (32.000 EFPY)

PRESSURE (PSI)	TEMPERATURE (DEG.F)
2000	319
2485	341

PRESSURE (PSI)	PRESSURE STRESS (PSI)	1.5 K1M (PSI SQ.RT.IN.)
2000	21444	89745
2485	26645	112505

AMP 60F/HR HEATUP CURVE REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/17/89

COMPOSITE CURVE PLOTTED FOR HEATUP PROFILE 2 HEATUP RATE(S) (DEG.F/HR) = 60.0

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = (1-AOWIN)T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)			INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	~~459.12~~		21	185.000	499.94	41	285.000	972.05
2	90.000	~~448.89~~		22	190.000	510.60	42	290.000	1018.19
3	95.000	~~441.83~~	428.65	23	195.000	522.16	43	295.000	1067.73
4	100.000	~~435.78~~		24	200.000	534.64	44	300.000	1120.91
5	105.000	~~432.04~~		25	205.000	548.02	45	305.000	1177.86
6	110.000	~~429.65~~		26	210.000	562.55	46	310.000	1239.07
7	115.000	428.65		27	215.000	578.23	47	315.000	1304.48
8	120.000	428.66		28	220.000	594.99	48	320.000	1374.83
9	125.000	429.72		29	225.000	613.17	49	325.000	1450.01
10	130.000	431.59		30	230.000	632.71	50	330.000	1530.61
11	135.000	434.32		31	235.000	653.64	51	335.000	1616.85
12	140.000	437.74		32	240.000	676.27	52	340.000	1709.26
13	145.000	441.85		33	245.000	700.47	53	345.000	1807.94
14	150.000	446.67		34	250.000	726.59	54	350.000	1913.47
15	155.000	452.21		35	255.000	754.61	55	355.000	2012.67
16	160.000	458.39		36	260.000	784.63	56	360.000	2113.98
17	165.000	465.27		37	265.000	817.10	57	365.000	2222.23
18	170.000	472.84		38	270.000	851.82	58	370.000	2337.74
19	175.000	481.14		39	275.000	889.08	59	375.000	2461.04
20	180.000	490.18		40	280.000	929.09			

APPENDIX B
HEATUP AND COOLDOWN DATA
WITH MARGINS FOR INSTRUMENTATION ERROR

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/26/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 1 (STEADY-STATE COOLDOWN)

IRRADIATION PERIOD = 12.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	441.80	19	175.000	569.12	37	265.000	1035.62
2	90.000	445.37	20	180.000	582.13	38	270.000	1083.47
3	95.000	449.21	21	185.000	596.27	39	275.000	1134.84
4	100.000	453.34	22	190.000	611.48	40	280.000	1189.75
5	105.000	457.78	23	195.000	627.65	41	285.000	1249.08
6	110.000	462.55	24	200.000	645.22	42	290.000	1312.63
7	115.000	467.68	25	205.000	664.07	43	295.000	1380.59
8	120.000	473.19	26	210.000	684.22	44	300.000	1453.87
9	125.000	479.02	27	215.000	706.03	45	305.000	1532.21
10	130.000	485.39	28	220.000	729.30	46	310.000	1615.99
11	135.000	492.24	29	225.000	754.47	47	315.000	1705.86
12	140.000	499.61	30	230.000	781.39	48	320.000	1802.27
13	145.000	507.54	31	235.000	810.44	49	325.000	1905.54
14	150.000	516.05	32	240.000	841.57	50	330.000	2015.96
15	155.000	525.09	33	245.000	874.98	51	335.000	2133.76
16	160.000	534.93	34	250.000	910.84	52	340.000	2259.71
17	165.000	545.51	35	255.000	949.61	53	345.000	2394.07
18	170.000	556.89	36	260.000	991.09			

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/26/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 2 (20 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 12.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	398.83	14	150.000	476.05	26	210.000	654.87
2	90.000	402.55	15	155.000	485.60	27	215.000	678.03
3	95.000	406.48	16	160.000	495.98	28	220.000	703.11
4	100.000	410.69	17	165.000	507.17	29	225.000	729.94
5	105.000	415.28	18	170.000	519.20	30	230.000	758.93
6	110.000	420.17	19	175.000	532.06	31	235.000	790.03
7	115.000	425.48	20	180.000	546.00	32	240.000	823.38
8	120.000	431.19	21	185.000	561.04	33	245.000	859.27
9	125.000	437.28	22	190.000	577.05	34	250.000	898.08
10	130.000	443.91	23	195.000	594.47	35	255.000	939.63
11	135.000	451.08	24	200.000	613.17	36	260.000	984.28
12	140.000	458.79	25	205.000	633.19	37	265.000	1032.25
13	145.000	467.11						

FORM P506

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/26/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 3 (40 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 12.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	355.09	14	150.000	435.43	26	210.000	626.02
2	90.000	358.78	15	155.000	445.63	27	215.000	651.06
3	95.000	362.80	16	160.000	456.61	28	220.000	677.79
4	100.000	367.14	17	165.000	468.48	29	225.000	706.80
5	105.000	371.85	18	170.000	481.15	30	230.000	737.81
6	110.000	376.93	19	175.000	494.94	31	235.000	771.21
7	115.000	382.38	20	180.000	509.78	32	240.000	807.30
8	120.000	388.32	21	185.000	525.68	33	245.000	846.02
9	125.000	394.78	22	190.000	542.92	34	250.000	887.64
10	130.000	401.70	23	195.000	561.52	35	255.000	932.43
11	135.000	409.22	24	200.000	581.39	36	260.000	980.58
12	140.000	417.32	25	205.000	602.97	37	265.000	1032.38
13	145.000	426.08						

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/26/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 4 (60 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 12.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	310.21	13	145.000	384.36	25	205.000	573.23
2	90.000	313.89	14	150.000	394.35	26	210.000	598.17
3	95.000	318.13	15	155.000	405.16	27	215.000	624.80
4	100.000	322.60	16	160.000	416.82	28	220.000	653.87
5	105.000	327.49	17	165.000	429.45	29	225.000	684.91
6	110.000	332.77	18	170.000	442.96	30	230.000	718.32
7	115.000	338.52	19	175.000	457.67	31	235.000	754.52
8	120.000	344.73	20	180.000	473.51	32	240.000	793.34
9	125.000	351.43	21	185.000	490.53	33	245.000	835.11
10	130.000	358.72	22	190.000	508.97	34	250.000	880.06
11	135.000	366.63	23	195.000	528.77	35	255.000	928.47
12	140.000	375.17	24	200.000	550.22	36	260.000	980.52

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/26/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 5 (100 DEG-F/HR COOLDOWN)

IRRADIATION PERIOD = 12.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	217.05	13	145.000	298.27	25	205.000	516.78
2	90.000	221.08	14	150.000	310.53	26	210.000	545.64
3	95.000	225.51	15	155.000	322.83	27	215.000	576.83
4	100.000	230.35	16	160.000	336.12	28	220.000	610.59
5	105.000	235.67	17	165.000	350.49	29	225.000	646.88
6	110.000	241.46	18	170.000	366.07	30	230.000	685.98
7	115.000	247.80	19	175.000	382.89	31	235.000	728.17
8	120.000	254.67	20	180.000	401.12	32	240.000	773.60
9	125.000	262.16	21	185.000	420.87	33	245.000	822.61
10	130.000	270.30	22	190.000	442.08	34	250.000	875.38
11	135.000	279.19	23	195.000	465.14	35	255.000	932.23
12	140.000	288.80	24	200.000	489.89			

AMP 60F/HR HEATUP CURVE REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/26/89

THE FOLLOWING DATA WERE CALCULATEDFOR THE INSERVICE HYDROSTATIC LEAK TEST.

MINIMUM INSERVICE LEAK TEST TEMPERATURE (12.000 EFPY)

PRESSURE (PSI)	TEMPERATURE (DEG.F)
2000	303
2485	323

PRESSURE (PSI)	PRESSURE STRESS (PSI)	1.5 K1M (PSI SQ RT IN.)
2000	22088	92529
2485	27288	115366

AMP 60F/HR HEATUP CURVE REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/26/89

COMPOSITE CURVE PLOTTED FOR HEATUP PROFILE 2 HEATUP RATE(S) (DEG.F/HR) = 60.0

IRRADIATION PERIOD = 12.000 EFP YEARS
FLAW DEPTH = (1-AOWIN)T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	~~441.86~~	20	180.000	472.71	38	270.000	936.79
2	90.000	~~433.77~~	21	185.000	485.24	39	275.000	984.56
3	95.000	~~421.48~~	22	190.000	498.93	40	280.000	1035.79
4	100.000	~~411.71~~ 397.04	23	195.000	513.79	41	285.000	1090.75
5	105.000	~~405.04~~	24	200.000	529.71	42	290.000	1149.57
6	110.000	~~400.49~~	25	205.000	547.06	43	295.000	1212.94
7	115.000	~~397.90~~	26	210.000	565.76	44	300.000	1280.67
8	120.000	397.04	27	215.000	585.78	45	305.000	1353.27
9	125.000	397.64	28	220.000	607.48	46	310.000	1430.85
10	130.000	399.42	29	225.000	630.70	47	315.000	1514.43
11	135.000	402.44	30	230.000	655.85	48	320.000	1603.52
12	140.000	406.44	31	235.000	682.72	49	325.000	1699.00
13	145.000	411.51	32	240.000	711.78	50	330.000	1800.76
14	150.000	417.47	33	245.000	742.89	51	335.000	1910.00
15	155.000	424.40	34	250.000	776.27	52	340.000	2026.46
16	160.000	432.19	35	255.000	812.14	53	345.000	2150.89
17	165.000	440.55	36	260.000	850.88	54	350.000	2269.54
18	170.000	450.50	37	265.000	892.33	55	355.000	2392.78
19	175.000	461.12						

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 1 (STEADY-STATE COOLDOWN)

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)
1	85.000	425.35	21	185.000	526.01	41	285.000	953.49
2	90.000	427.68	22	190.000	535.92	42	290.000	995.25
3	95.000	430.18	23	195.000	546.58	43	295.000	1040.10
4	100.000	432.78	24	200.000	558.04	44	300.000	1088.28
5	105.000	435.68	25	205.000	570.35	45	305.000	1139.86
6	110.000	438.79	26	210.000	583.46	46	310.000	1195.36
7	115.000	442.13	27	215.000	597.70	47	315.000	1255.11
8	120.000	445.73	28	220.000	612.99	48	320.000	1319.04
9	125.000	449.60	29	225.000	628.29	49	325.000	1387.54
10	130.000	453.75	30	230.000	646.99	50	330.000	1461.18
11	135.000	458.22	31	235.000	665.97	51	335.000	1540.05
12	140.000	463.03	32	240.000	686.26	52	340.000	1624.61
13	145.000	468.19	33	245.000	708.20	53	345.000	1715.12
14	150.000	473.75	34	250.000	731.66	54	350.000	1812.08
15	155.000	479.61	35	255.000	757.01	55	355.000	1915.92
16	160.000	486.03	36	260.000	784.12	56	360.000	2026.92
17	165.000	492.93	37	265.000	813.20	57	365.000	2145.68
18	170.000	500.35	38	270.000	844.72	58	370.000	2272.27
19	175.000	508.33	39	275.000	878.36	59	375.000	2407.28
20	180.000	516.91	40	280.000	914.48			

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/26/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 2 (20 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	381.28	16	160.000	443.49	30	230.000	614.18
2	90.000	383.54	17	165.000	450.72	31	235.000	634.37
3	95.000	386.03	18	170.000	458.49	32	240.000	656.23
4	100.000	388.71	19	175.000	466.88	33	245.000	679.59
5	105.000	391.62	20	180.000	475.90	34	250.000	704.88
6	110.000	394.74	21	185.000	485.53	35	255.000	731.94
7	115.000	398.14	22	190.000	495.99	36	260.000	761.18
8	120.000	401.79	23	195.000	507.28	37	265.000	792.55
9	125.000	405.74	24	200.000	519.41	38	270.000	826.18
10	130.000	409.99	25	205.000	532.37	39	275.000	862.38
11	135.000	414.60	26	210.000	546.44	40	280.000	901.48
12	140.000	419.55	27	215.000	561.60	41	285.000	943.38
13	145.000	424.81	28	220.000	577.75	42	290.000	988.39
14	150.000	430.66	29	225.000	595.32	43	295.000	1036.79
15	155.000	436.80						

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/26/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 3 (40 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	336.22	16	160.000	400.20	30	230.000	581.66
2	90.000	338.48	17	165.000	407.79	31	235.000	603.47
3	95.000	340.86	18	170.000	415.96	32	240.000	626.75
4	100.000	343.64	19	175.000	424.82	33	245.000	652.03
5	105.000	346.52	20	180.000	434.25	34	250.000	679.04
6	110.000	349.68	21	185.000	444.56	35	255.000	708.31
7	115.000	353.13	22	190.000	455.64	36	260.000	739.66
8	120.000	356.86	23	195.000	467.63	37	265.000	773.40
9	125.000	360.82	24	200.000	480.42	38	270.000	809.82
10	130.000	365.29	25	205.000	494.35	39	275.000	848.96
11	135.000	370.05	26	210.000	509.34	40	280.000	890.87
12	140.000	375.18	27	215.000	525.40	41	285.000	936.26
13	145.000	380.68	28	220.000	542.81	42	290.000	984.85
14	150.000	386.68	29	225.000	561.60	43	295.000	1037.17
15	155.000	393.19						

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/26/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 4 (60 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	289.86	15	155.000	348.73	29	225.000	528.11
2	90.000	292.20	16	160.000	356.10	30	230.000	549.81
3	95.000	294.69	17	165.000	364.11	31	235.000	573.21
4	100.000	297.38	18	170.000	372.74	32	240.000	598.32
5	105.000	300.35	19	175.000	382.04	33	245.000	625.37
6	110.000	303.52	20	180.000	392.14	34	250.000	654.67
7	115.000	307.05	21	185.000	403.08	35	255.000	686.08
8	120.000	310.87	22	190.000	414.87	36	260.000	719.88
9	125.000	315.06	23	195.000	427.64	37	265.000	756.51
10	130.000	319.58	24	200.000	441.31	38	270.000	795.75
11	135.000	324.52	25	205.000	456.19	39	275.000	838.05
12	140.000	329.86	26	210.000	472.21	40	280.000	883.52
13	145.000	335.68	27	215.000	489.43	41	285.000	932.47
14	150.000	341.96	28	220.000	508.09	42	290.000	985.12

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/26/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 5 (100 DEG-F/HR COOLDOWN)

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	193.64	15	155.000	257.24	29	225.000	463.34
2	90.000	195.87	16	160.000	265.47	30	230.000	488.47
3	95.000	198.40	17	165.000	274.48	31	235.000	515.78
4	100.000	201.17	18	170.000	284.25	32	240.000	545.08
5	105.000	204.26	19	175.000	294.88	33	245.000	576.74
6	110.000	207.64	20	180.000	306.32	34	250.000	611.01
7	115.000	211.40	21	185.000	318.82	35	255.000	647.85
8	120.000	215.49	22	190.000	332.31	36	260.000	687.53
9	125.000	220.01	23	195.000	346.91	37	265.000	730.38
10	130.000	224.90	24	200.000	362.73	38	270.000	776.45
11	135.000	230.31	25	205.000	379.81	39	275.000	826.17
12	140.000	236.18	26	210.000	398.32	40	280.000	879.69
13	145.000	242.62	27	215.000	418.38	41	285.000	937.32
14	150.000	249.60	28	220.000	439.92			

AMP 60F/HR HEATUP CURVE REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/26/89

THE FOLLOWING DATA WERE CALCULATEDFOR THE INSERVICE HYDROSTATIC LEAK TEST.

MINIMUM INSERVICE LEAK TEST TEMPERATURE (32.000 EFPY)

PRESSURE (PSI)	TEMPERATURE (DEG.F)
2000	332
2485	353

PRESSURE (PSI)	PRESSURE STRESS (PSI)	1.5 K1M (PSI SQ.RT.IN.)
2000	22088	92529
2485	27288	115366

AMP 60F/HR HEATUP CURVE REG. GUIDE 1.99,REV.2 FOR PLATE C5556-2 04/26/89

COMPOSITE CURVE PLOTTED FOR HEATUP PROFILE 2 HEATUP RATE(S) (DEG.F/HR) = 60.0

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = (1-AOWIN)T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	~~420.03~~	22	190.000	430.18	42	290.000	869.09
2	90.000	~~412.18~~	23	195.000	439.94	43	295.000	912.05
3	95.000	~~399.12~~	24	200.000	450.60	44	300.000	958.19
4	100.000	~~388.89~~ 368.65	25	205.000	462.16	45	305.000	1007.73
5	105.000	~~381.03~~	26	210.000	474.64	46	310.000	1060.91
6	110.000	~~375.78~~	27	215.000	488.02	47	315.000	1117.96
7	115.000	~~372.04~~	28	220.000	502.55	48	320.000	1179.07
8	120.000	~~369.65~~	29	225.000	518.23	49	325.000	1244.48
9	125.000	368.65	30	230.000	534.99	50	330.000	1314.83
10	130.000	368.66	31	235.000	553.17	51	335.000	1390.01
11	135.000	369.72	32	240.000	572.71	52	340.000	1470.61
12	140.000	371.59	33	245.000	593.64	53	345.000	1556.85
13	145.000	374.32	34	250.000	616.27	54	350.000	1649.26
14	150.000	377.74	35	255.000	640.47	55	355.000	1747.94
15	155.000	381.85	36	260.000	666.89	56	360.000	1853.47
16	160.000	386.67	37	265.000	694.61	57	365.000	1952.67
17	165.000	392.21	38	270.000	724.63	58	370.000	2053.88
18	170.000	398.39	39	275.000	757.10	59	375.000	2162.23
19	175.000	405.27	40	280.000	791.82	60	380.000	2277.74
20	180.000	412.84	41	285.000	829.08	61	385.000	2401.04
21	185.000	421.14						

Attachment 1 TO AEP:NRC:1077A

DESCRIPTION AND 10 CFR 50.92 ANALYSES FOR CHANGES
TO THE DONALD C. COOK NUCLEAR PLANT UNIT 2
TECHNICAL SPECIFICATIONS

A. DESCRIPTION OF AMENDMENT REQUEST

In this application to amend the Donald C. Cook Nuclear Plant Unit 2 license, we propose to revise the Technical Specifications by implementing a Core Operating Limits Report (COLR) for Unit 2 in accordance with Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications." The proposed Technical Specification (T/S) changes are as follows:

1. Definition 1.39, "CORE OPERATING LIMITS REPORT," was added in accordance with Generic Letter 88-16

2. T/S 3.1.1.4, "MODERATOR TEMPERATURE COEFFICIENT," was revised by removing the cycle-specific moderator temperature coefficient limit for end of life and placing it in the COLR in accordance with Generic Letter 88-16.

3. T/S 4.1.1.4.2.b, a surveillance requirement for T/S 3.1.1.4, was revised by referencing the 300 ppm surveillance limit in the COLR.

4. T/S 3.1.3.1, "MOVABLE CONTROL ASSEMBLIES GROUP HEIGHT," was revised by removing the reference to Figure 3.1-1, "ROD BANK STEP POSITION AS A FUNCTION OF POWER," and referencing the COLR.

5. T/S 3.1.3.4, "ROD DROP TIME," was revised by removing the value for the fully withdrawn position and placing it in the COLR in accordance with Generic Letter 88-16.

6. T/S 3.1.3.5, "SHUTDOWN ROD INSERTION LIMIT," was revised by removing the value for the insertion limit and placing it in the COLR in accordance with Generic Letter 88-16. The T/S was also revised by changing the phrase "fully withdrawn" to a phrase that references the insertion limit.

7. T/S 3.1.3.6 was revised by removing the reference to Figure 3.1-1, "ROD BANK STEP POSITION AS A FUNCTION OF POWER," and referencing the COLR, and by moving Figure 3.1-1, page 3/4 1-26, from the T/Ss to the COLR in accordance with Generic Letter 88-16.

8. T/S 3.2.1, "AXIAL FLUX DIFFERENCE," was revised by removing the cycle-specific target band limits from the T/S and referencing the COLR, and by removing Figure 3.2-1, "ALLOWABLE DEVIATION FROM TARGET FLUX DIFFERENCE," page 3/4 2-4, and placing it in the COLR in accordance with Generic Letter 88-16. Surveillance requirement 4.2.1.4 was also revised by removing the values for the axial flux difference target band and referencing the COLR.

9. T/S 3.2.2, "HEAT FLUX HOT CHANNEL FACTOR $F_Q(Z)$," was revised by moving the cycle-specific F_Q limit from the T/S to the COLR, and moving the associated figures (Figures 3.2-2, page 3/4 2-8, and 3.2-2(a), page 3/4 2-8a) into the COLR in accordance with Generic Letter 88-16.

10. T/S 3.2.3, "NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$," was revised by moving the cycle-specific $F^N_{\Delta H}$ and its power factor multiplier from the T/S to the COLR in accordance with Generic Letter 88-16.

11. T/S 3.2.6, "ALLOWABLE POWER LEVEL - APL", was revised by moving Figure 3.2-3, page 3/4 2-8b, V(Z) AS A FUNCTION OF CORE HEIGHT," from the T/S to the COLR, and referencing the cycle-specific F_Q limit in the COLR in accordance with Generic Letter 88-16. Starting in Cycle 8, a cycle-specific V(Z) function will replace the current generic V(Z) calculated by Advanced Nuclear Fuels.

12. T/S 6.9.1.11, "CORE OPERATING LIMITS REPORT," was added to the T/Ss in accordance with Generic Letter 88-16.

13. In addition corresponding changes have been made to the appropriate Bases sections.

B. BACKGROUND

Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," provides guidance for relocating cycle-specific parameters from the T/Ss so that future reload designs can be implemented in accordance with 10 CFR 50.59. The generic letter requires that the cycle-specific parameters be maintained in a COLR and the NRC be informed of changes to the COLR. Any changes to the operating limits will made using approved NRC methodologies for Cook Nuclear Plant. This change will not affect the operation or safety of Cook Nuclear Plant Unit 2 since the actions required to be completed should a limit be exceeded will not be removed from the T/Ss.

The proposed T/S changes include the removal of the moderator temperature coefficient, the shutdown rod insertion limit, the control rod insertion limits (Figure 3.1-1), the axial flux difference operational limits (Figure 3.2-1), the heat flux hot channel factor limit and associated factors (Figures 3.2-2 and 3.2-2(a)), the nuclear enthalpy hot channel factor limit and associated factors, and the allowable power level factors F_Q, K(Z), and V(Z) from the T/Ss. The proposed T/Ss retain the Limiting Condition for Operation (LCO) wording and surveillance requirements by referring to the specific limits which are provided to the NRC in a COLR in accordance with the proposed T/S 6.9.1.11. Relocation of these cycle-specific limits is consistent with the guidance provided

by the NRC in Generic Letter 88-16. Removal of the limits from the T/Ss will allow cycle-specific limits to be implemented in accordance with approved methodology in a timely and cost effective manner.

C. JUSTIFICATION

These T/S changes comply with the NRC guidance given in Generic Letter 88-16. Implementation of a COLR for each unit will allow for cycle-specific parameter limit changes in accordance with the referenced approved methodology of Section 6.9.1.11. The relocation of the cycle-specific parameter limits to the COLR will allow for greater flexibility in optimizing designs to enhance the economics for each cycle.

D. SAFETY EVALUATION

The relocation of the cycle-specific parameters does not affect the operation of Cook Nuclear Plant Unit 2. The revised T/S will require the same actions to be taken when a limit is exceeded as required by the present T/S. Revisions to the COLR will be made in accordance with 10 CFR 50.59 and the NRC will be notified of all revisions in accordance with proposed T/S 6.9.1.11. All revisions to the COLR will be based on NRC-approved methodologies.

The proposed revision to the T/Ss simply revises the method by which changes to the cycle-specific parameter limits can be requested and implemented. The parameters and methodologies associated with calculating them are presented below:

1) Revisions to the Moderator Temperature Coefficient, Rod Drop Time Rod Insertion, Shutdown Rod Insertion, and Control Rod Insertion Limits will be based on the Westinghouse methodology previously reviewed and approved by the NRC and described in WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology."

2) The Axial Flux Difference Limits will be determined in accordance with Westinghouse methodology previously reviewed and approved by the NRC and described in WCAP-10216-P-A, Part B, "Relaxation of Constant Axial Offset Control/F_Q Surveillance Technical Specification;" WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology;" and WCAP-8385, "Power Distribution Control and Load Following Procedures."

3) The Heat Flux Hot Channel Factor $F_Q(Z)$ will be determined in accordance with Westinghouse methodology previously reviewed and approved by the NRC and described in WCAP-10266-P-A, Rev. 2, "The 1981 Version of Westinghouse Evaluation Model Using BASH Code;" WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology;" and WCAP-10216-P-A, Part B, "Relaxation of Constant Axial Offset Control/F_Q Surveillance Technical Specification."

4) The Nuclear Enthalpy Rise Hot Channel Factor $F^{N}_{\Delta H}$ will be determined in accordance with Westinghouse methodology previously reviewed and approved by the NRC and described in WCAP-10266-P-A, Rev. 2, "The 1981 Version of Westinghouse Evaluation Model Using BASH Code"; and WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology."

Calculating the cycle-specific parameter limits in accordance with an NRC-approved methodology ensures the limits are consistent with the applicable safety analysis addressed in the Updated Cook Nuclear Plant FSAR. In addition, the proposed T/S 6.9.1.11 is written to require that the limits be maintained in accordance with the approved methodology.

As discussed in this submittal, there is reasonable assurance that the T/S changes associated with relocating cycle-specific parameters out of the T/S will not adversely affect the health and safety of the public.

E. NO SIGNIFICANT HAZARDS EVALUATION

The changes presented in this amendment request are purely editorial in nature. Per 10 CFR 50.92, a proposed amendment will involve a no significant hazards consideration if the proposed amendment does not:

(1) Involve a significant increase in the probability or consequence of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

The following is provided for the three categories of the significant hazards consideration standards.

<u>Criterion 1</u>

Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The moderator temperature coefficient limit, rod drop time rod insertion limit, shutdown rod insertion limit, control rod insertion limit, axial flux difference operational limits, heat flux hot channel factor limit, and nuclear enthalpy rise hot channel factor limit are cycle-specific parameters. The removal of the cycle-specific parameters from the T/Ss has no influence or impact on the probability or consequences of a previously evaluated accident. The cycle-specific parameter

limits, although not in the T/Ss, will be maintained in the COLR and referenced in the Cook Nuclear Plant T/Ss. The proposed amendment still requires the same action be taken if limits are exceeded as is required by current T/Ss. Future reloads will be evaluated using NRC-approved methodologies and will be examined per the requirements of 10 CFR 50.59. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2

Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

There is no physical alteration to any plant system, nor is there a change in the method by which any safety related system performs its function. As stated above, the proposed change is administrative in nature. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3

Does the change involve a significant reduction in a margin of safety?

The margin of safety is not affected by the removal of cycle-specific parameter limits from the T/Ss. The proposed amendment still requires operation within the core limits as determined from the NRC-approved reload design methodologies. Appropriate actions will continue to be taken if limits are violated. The development of the limits for future reloads will continue to conform to those methods described in NRC-approved documentation. In addition, each future reload will involve a 10 CFR 50.59 review. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The first of these examples refers to changes that are purely administrative in nature: for example, changes to achieve consistency throughout the T/Ss, correction of an error, or a change in nomenclature. As these changes are purely editorial and do not impact safety in any way, we believe the Federal Register example cited is applicable and that the changes involve no significant hazards consideration.

F. ENVIRONMENTAL EVALUATION

I&M has evaluated the proposed changes and determined that the changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increases in the amounts of any effluents that may be released off site, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed changes is not required.